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Book of Abstracts

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Contents

OV

Overview

OV/1-1 · Overview progress and future plan of EAST Project

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Abstract: EAST is a full superconducting tokamak with advanced configuration. The physic and engineering design; the R&D for the fabrication and testing of CICC and all magnets, for the key components such as insulators, high T_c current leads, as well as the most of important sub-systems such as the 2 kW cryogenic system, the PF and TF power supply, the control and data acquisition system have been completed since 1998. The final assembly has completed at end of 2005. Basing on above progress the first engineering commissioning of EAST has been done successfully. The all magnets were cooling down to 4.5–5 K stably and then charged successfully. 8200 A (2.0 T) with the 5000 S charge duration for TF system has been achieved. The all subsystems has been also tested and showed satisfying the design requirements. It is planning to obtain the first plasma around July∼August. The future experimental plan, especially for diverter experiments with long pulse discharge will begin soon. All of progress above and future plans of EAST project will be given in this paper with the individual reference papers in detail.

OV/1-2 · Overview of JT-60U Results for Development of Steady-State Advanced Tokamak Scenario

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Abstract: Towards development of a steady-state advanced tokamak scenario, JT-60U experiments recently focus on sustainment of high performance plasmas exceeding the time scale of related key issues, such as MHD activity, confinement and transport, current diffusion and plasma wall interaction. This paper presents recent JT-60U experimental results since the last IAEA conference with the emphasis on phenomena dominated in various time scales. Installation of ferritic steel tiles enables to access new regimes due to increase in net heating power and confinement improvement attributed to the reduction of fast ion loss. High β_N exceeding no wall ideal limit is sustained for ∼120 ms longer than the resistive diffusion time of the wall in the positive shear plasma. The plasma rotation retards growth of resistive wall mode due to its stabilizing effect. High $\beta_N H_{98}$ of 2.2 with $\beta_N \sim 2.3$ and $H_{98} \sim 1$ is sustained for 23.1 s significantly longer than the current diffusion time ($\sim 12\tau_R$) in the positive shear plasma. In this discharge, neoclassical tearing mode is suppressed by pressure profile control in the early phase and current diffusion does not affect the MHD activity. However, confinement degrades in the latter phase due to increase in the electron density caused by enhancement of recycling. In the reversed shear plasma with a high bootstrap current fraction of ∼75%, the evolution of inductive field is found to be largely affected by the change in bootstrap current, indicating strong linkage between pressure and current profiles. In this linkage, the plasma disrupts due to ideal MHD limit at $q_{min} = 4$ in heating duration of ∼4 s without pressure profile control for degradation of the internal transport barrier. The real time control for current and pressure profiles based on real time q_{min} calculation using motional stark effect diagnostics are developed. With high divertor pumping rate enhanced by adjusting the strike points to the pumping slots, the density is successfully controlled for ∼30 s by the active divertor pumping in high-density ELMy H-mode plasmas with wall saturation and even with outgas from the wall. Both hydrogen retention and carbon deposition are smaller than those observed in other carbon machines.

OV/1-3 · Overview of JET Results

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Abstract: The development for ITER of integrated operating scenarios with acceptable first wall power loadings and high fusion power has gained urgency in view of JET's intent to install an ITER-like beryllium wall, tungsten divertor and major heating system upgrades. Understanding key issues has progressed with recent analyses of past results and the 2006 campaigns will exploit new enhancements, extending operation to highly shaped, high power plasmas with improved diagnostics. Issues include ELM transient heat loads, edge plasma flows, tritium retention, density peaking, impurity accumulation and performance improvement by q profile control. ELMs cause first wall erosion and damage (0.6–3.4 MJ/m^2 deposited on ITER divertor targets), strong temporal variations at targets (as in kinetic models; otherwise heat fluxes underestimated by 4 for 1 MJ JET ELMs), $T_i/T_e > 2$ and ion energy in far SOL ∼50% pedestal (as in transient ELM model), and increased energy fraction to wall with ELM size. Edge plasma models and measurements show strongly intermittent turbulent transport, fluxes significantly higher than collisional,

steep plasma profiles in SOL near separatrix, and large parallel flows. Post-mortem analyses show strong net deposition on inner targets, in agreement with injected C flowing from outer to inner target. Inner target fuel retention is similar in tile castellations and on surfaces, raising concern about tritium inventory in ITER. A clear dependence of density peaking on collisionality is confirmed, predicting peaking ~1.5 in ITER. High Z transport is anomalous, Ni accumulating with i-heating (as in gyrokinetic calculations of ITG turbulence) and slightly outwards with e-heating (as in TEM turbulence driven by R/L_{Te}), offering impurity control with α heating. q profile control improves performance by ICCD increasing magnetic shear near $q \sim 1$ and destabilising sawteeth (which otherwise destabilise NTMs) and inductive current/LHCD producing $q_0 > 1$ sawteeth-free hybrid modes (confinement higher than ITER scaling; sometimes with steep T_i gradients) and ITBs (well-localised narrow layers with low heat diffusivity near shear reversal point due to $E \times B$ shear stabilisation of turbulence with measured poloidal rotation well above neoclassical). The no-wall pressure limit can be exceeded with RWMs stabilised above a critical rotation at $q = 2$, which may be marginal in ITER.

OV/1-4 · Development in the DIII-D Tokamak of Advanced Operating Scenarios and Associated Control Techniques for ITER

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Abstract: The DIII-D research program is focused on providing solutions to issues critical to the future success of ITER, both in achieving its basic mission goal of $Q = 10$ operation and in enhancing the ITER physics program through development of operating regimes capable of sustaining higher Q values. Significant progress has been made in the ability to control key plasma features and using such control to expand the operational limits of stationary and steady-state tokamak operation. Recent experiments have demonstrated the capability to suppress the key plasma instabilities of concern for ITER (including ELMs, neoclassical tearing modes, and resistive wall modes) by external means, techniques for mitigating the effects of disruptions, and control of the current profile evolution. The use of these techniques has allowed an expansion of the envelope of viable, stationary tokamak operation, highlighted by the demonstration of sustained (~2 s) operation with normalized $\beta \sim 4$ (50% above the no-wall stability limit) as well as fully noninductive operation with toroidal $\beta \sim 3.5\%$. This developmental research is supported by a vigorous basic physics program, which also addresses several key ITER issues. These include edge carbon transport and tritium co-deposition on plasma facing surfaces, fast-ion instabilities and their effects on the fast ion population, and identification of the underlying mechanisms responsible for transport in a tokamak plasma. Highlights of recent research in these areas will be presented. In addition, experiments in the coming year will seek to take advantage of extensive upgrades recently made to the DIII-D facility. These include the reorientation of a neutral beam to allow counter- and low-rotation plasmas, a new lower divertor for density control in double null plasmas, and increased EC power. It is anticipated that these upgrades will increase the flexibility in controlling and optimizing of plasma operation in DIII-D.
* Work supported by the US DOE under DE-FC02-04ER54698.

OV/2-1 · Extended Steady-State and High-Beta Regimes of Net-Current Free Heliotron Plasmas in the Large Helical Device

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Abstract: The performance of net-current free heliotron plasmas has been developed by an upgrade of the heating power and the pumping/fueling capability in the Large Helical Device (LHD) and understanding of the confinement physics of net-current free plasmas has been deepened. LHD is a superconducting magnetic confinement device employing a heliotron configuration. The heating capability is 15 MW of NBI, 2.9 MW of ICRF and 2.1 MW of ECH. The operational regime has been extended with regard to the pulse length and high beta. Achievements of a volume averaged beta of 4.5% and discharge duration of 54 min with a total input energy of 1.6 GJ are highlighted. These two major achievements emphasize the fundamental advantage of net-current free heliotron plasmas. Long pulse operation has been pursued by ICRF heating and a dynamic control of heat load on the divertor plate. A plasma with an ion temperature of 2 keV and a density of 8×10^{18} m^{-3} was created and maintained for more than 30 min. An essential reason for this success is that highly energetic trapped ions are well confined due to drift optimization. The high ion energy tail up to 1.6 MeV has been observed simultaneously. A current control of the helical coil which consists of 3 independent layers has enabled us to scan the plasma aspect ratio. Large aspect ratio reduces the Shafranov shift and is preferable to the MHD equilibrium beta limit as well as power

deposition of the NBI heating while MHD stability is violated due to a suppression of the spontaneous magnetic well. The effect of MHD instabilities is still mitigated at larger aspect ratio compared to the regular operation. Consequently, the highest beta value of 4.5% has been achieved at aspect ratio of 6.6 and the magnetic field of 0.425 T. This high beta state is maintained for more than 10 times the energy confinement time. The beta value still increases with the heating power. An internal diffusion barrier has realized a super dense core as high as 5×10^{20} m^{-3}. Complete detached plasma with the line-averaged density of 2×10^{20} m^{-3} has been maintained in quasi-steady state. New findings also enable us to extend the envelope of explorable physical parameter space. Diversified studies in LHD have elucidated the broad scope of steady-state high temperature plasmas.

OV/2-2 · Overview of ASDEX Upgrade Results

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Abstract: The programme of ASDEX Upgrade, a divertor tokamak with an ITER-like plasma and coil configuration, supports the physics base for ITER operation in both the foreseen standard scenario as well as scenarii with improved performance (improved H-mode, non-inductive current drive). Moreover, physics issues with impact beyond ITER are already identified and addressed. The highlights of the recent progress are presented. For anomalous transport (energy, hydrogen, impurities) a multi-faceted picture of mode dominance in different plasma parameter regimes of ITG, TEM and ETG turbulence was developed including collisions. The active control of MHD instabilities (sawteeth, NTMs) concentrates on ECCD as proposed for ITER. NTMs were completely stabilized with very localized deposition of dc ECCD, while for deposition widths larger than the marginal island size modulated injection phased with the island O-point was demonstrated to be advantageous. The structure and dynamics of natural and pellet induced, mitigated type I ELMs are similar starting from field-aligned helical structures and developing to outward drifting filaments. A small, high-frequency "grassy" ELM regime is compared to the type II regime. Also in view of ITER, our versatile heating system allows to decouple the effects of bulk plasma and fast ion population. For the first time it was possible to identify directly the phase correlation of fast ions losses with MHD activity (TAEs, NTMs, ELMs). The coverage of the vessel interior with tungsten was further extended up to 85%, where the highest erosion occurs at the LFS poloidal limiters and is dominated by fast particles from NBI as well as impurities accelerated by ICRF. The W concentration could usually be kept acceptable low using ELM pace-making (pellets) and tailored central electron heating. The stationary improved H-mode, discovered at ASDEX Upgrade in 1998, is best suited for a ITER hybrid scenario and could extend ITER operation beyond its standard H-mode performance. It promises either higher fusion performance ($Q > 30$ at full current) or longer pulses of up to 1 hr at reduced current. The operational range of this regime extends from ITER collisionalities up to high, divertor relevant edge densities. Besides peaked density profiles enhanced edge pressure gradients and consequently higher pedestal top pressures contribute to confinement improvement.

OV/2-3 · Overview of Physics Results from MAST

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Abstract: Important advances have been made on MAST, aided by substantial developments to plasma control, diagnostics and heating systems. The parameter range of the MAST confinement database has been extended and it now also includes pellet-fuelled discharges. Co-ordinated studies on MAST and DIII-D provide a strong link between the aspect ratio and beta scaling of H-mode energy confinement, consistent with that obtained when MAST data were merged with a subset of the ITPA database. Efficient pellet fuelling has been observed in H-mode discharges and post-pellet losses are low. Electron and ion ITBs are readily formed and their evolution has been investigated. Electron and ion thermal diffusivities have been reduced to values close to the ion neoclassical level. Non-linear GS2 calculations predict transport from the ETG mode at mid-radius in MAST H-mode comparable with experimental values. Error field correction coils have been used to determine the locked mode threshold scaling which is comparable with that in conventional tokamaks. The impact of plasma rotation on sawteeth has been investigated with co- and counter-NBI and the results have been well-modelled using the MISHKA-F code. The supra-Alfvénic ion population in MAST leads to a rich variety of fast particle driven instabilities. Their characteristics, beta dependence and impact on the fast ion population have been investigated. Off-axis NBCD and heating has been studied. Measurements are consistent with classical fast ion modelling and indicate

efficient heating and significant driven current. Electron Bernstein wave heating has been observed via the O-X-B mode conversion process. Further advances in non-solenoid start-up techniques have been made. High pedestal temperature plasmas have been produced with collisionalities one order of magnitude lower than in previous MAST experiments. Pedestal widths in these plasmas agree better with banana orbit scalings and ELM losses are increased, consistent with the broad mode structures predicted by stability analyses. New measurements clearly show that ELM filaments persist for ~200 μs during which time their toroidal rotation slows down and they accelerate radially outwards. SOL flows have been studied using a Gundestrup probe and 2D imaging of toroidally symmetric impurity gas puffing and compared with predictions from the B2SOLPS5.0 code.

OV/2-4 · Recent Physics Results from the National Spherical Torus Experiment

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Abstract: The National Spherical Torus Experiment (NSTX) has made considerable progress in advancing the scientific understanding of high performance long-pulse plasmas needed for low-aspect-ratio Spherical Torus (ST) concepts and for ITER. Plasma durations up to 1.5 s (approximately 5 current redistribution times) have been achieved at plasma currents of 0.7 MA with non-inductive current fractions approaching 70% while achieving toroidal and normalized beta values of 16% and 5.7% (mT/MA), respectively. Newly available Motional Stark Effect data has confirmed a strong correlation between improved electron energy confinement and degree of magnetic shear reversal, and the low aspect ratio and wide range of achievable beta in NSTX are providing unique data for confinement scaling studies. In MHD research, six mid-plane ex-vessel radial field coils have been utilized to infer and correct intrinsic error fields, investigate locked tearing mode thresholds, provide robust rotation control with non-resonant $n = 3$ field ripple, and measure the resonant field amplification spectrum of rotationally-stabilized resistive wall modes (RWMs) and the RWM critical rotation frequency. Advanced boundary shape control has been utilized to study the role of magnetic balance on the H-mode access threshold and ELM stability. Optimal ELM characteristics are typically obtained in a shape with negative bias, i.e. toward lower single null, while balanced double-null shapes maintained during the current ramp have resulted in H-mode regimes with pedestal temperatures nearly twice those of typical H-modes and the first evidence of current-hole formation in the plasma core. In the area of energetic particle research, cyclic neutron rate drops have been associated with the destabilization of multiple large Toroidal Alfvén Eigenmodes (TAEs) similar to the "sea-of-TAEs" predicted for ITER, albeit at lower TAE toroidal mode number $n = 1$–6. Finally, non-inductive plasma start-up research is particularly important for the ST concept, and Coaxial Helicity Injection has now produced 60kA of persistent current on closed flux surfaces in NSTX.

OV/3-1 · Integration of High Power, Long Pulse Operation in Tore Supra in Preparation for ITER

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Abstract: The Tore Supra tokamak 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. During the last two years, its scientific programme has entered a new phase toward higher power long pulse. Such a programme not only requires developing sound technologies, but also a strong modelling activity in order to integrate the various constraints through new innovative control schemes, pioneering the integration work that will be required on ITER. Discharges lasting several resistive times, with combined LHCD and ICRH at high power level, in plasmas close to the Greenwald limit and T_i close to T_e have been obtained. When pre-forming the current profile, improved confinement has been maintained together with q(0) ~ 1.5 for duration up to 20 s. Injecting high power in Tore Supra where all PFC are actively cooled raises the question of safe and reliable operation while optimising plasma performance. A typical example is the maximisation of injected power while maintaining the RF launchers surface temperature in the safe domain and controlling the LH-power deposition width. An advanced scheme for the physics and operational integration of plasma scenarios has been developed, and discharges at 7 MW total injected power have been controlled in an MHD stable region for 60 s. Such sophisticated feedback algorithms rely on a good understanding of the involved phenomena and accurate modelling. RF induced hot spots (sheath rectification, electron acceleration in the LHCD launcher near field) are now quantitatively modelled. A new LH wave absorption tool has been developed, using a 3-D bounce-averaged relativistic electron drift-kinetic solver. More fundamental modelling is also performed in the field of turbulence and MHD as exemplified by the upgrading to a 5D version of the gyrokinetic code GYSELA. An innovative, non destructive testing

device has been developed for PFC qualification after manufacturing. It has also been applied to check, in situ, the toroidal pump limiter elements of Tore Supra, between experimental campaigns. Such IR-based techniques are part of the continuous effort, to derive rigorous acceptance protocols allowing to reach the high reliability required for ITER PFC manufacturing.

OV/3-2 · Overview of Alcator C-Mod Research Program

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Abstract: Alcator C-Mod has compared plasma performance with all-metal plasma facing components (PFCs) to PFCs coated with boron to assess projections of performance from current experiments to next-generation burning tokamak plasmas. Low-Z coatings reduce metallic impurity influx and diminish radiative losses leading to higher H-mode pedestal pressure that increases temperature and pressure globally through profile stiffness. The beneficial effects of wall conditioning persist for a duration (1–50 shots) that increases in rough proportion to the quantity of boron deposited. Understanding the mechanisms responsible for regulating the H-mode pedestal height is also crucial for projecting performance in ITER. Pressure profile measurements in L-mode Ohmic plasmas suggest that the controlling mechanism in the near scrape-off (SOL) layer is a critical gradient phenomenon, similar to the scaling of edge pressure gradient in H-mode plasmas observed previously. The cross-field transport coefficients decrease markedly as I_p is increased, corresponding to both higher pedestal density and density gradient. Disruption mitigation through massive gas-jet impurity puffing has been extended to significantly higher plasma pressures and shorter disruption times. The fraction of total plasma energy radiated increases with Z of the impurity gas, reaching ∼90% for krypton. Impurity puffing reduces disruption halo currents by ∼50% and reduces divertor surface heating, indicating this technique has promise for implementation on ITER. Nonlinear gyrokinetic simulations with the GS2 code that incorporate a synthetic diagnostic to compute the density fluctuation spectrum measured by phase contrast imaging are in good agreement with experimental results, providing additional evidence for the role of TEM turbulence in plasmas with an internal transport barrier. Fast framing camera images of intermittent turbulent structures show they travel coherently through the entire SOL. The peak of the radial velocity distribution is about 1% of the ion sound speed and is unaffected by the connection length along the magnetic field to material surfaces. The 'chirping' evolution of Alfvén cascades during current ramps with intense ICRF heating measured by PCI agrees well with calculations of the NOVA-K code that includes geodesic deformation of the Alfvén continuum.

OV/3-3 · Overview of TCV Results

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Abstract: The *Tokamak à Configuration Variable*, TCV, addresses scientific questions to improve our understanding of magnetically confined plasmas and our ability to control them in ITER relevant scenarios, and explores avenues to improve the plasma performance on the way to a conceptual fusion power plant that cannot necessarily be investigated in ITER. The unique flexibility of its shaping and control systems is matched by that of its Electron Cyclotron Heating (ECH) and current drive (ECCD) systems. These include 3 MW from six 82.7 GHz gyrotrons used at the second harmonic in X-mode (X2), and 1.5 MW from three gyrotrons at 118 GHz (X3). This overview highlights the progress accomplished on TCV during the 2004–2006 campaigns, focussed on five main themes: 1) particle, energy and momentum transport in shaped plasmas, investigated over a large range of normalized temperature gradients and including peaked density profiles measured even in the absence of a Ware pinch or a core particle source; 2) plasma edge physics, addressing the question of the origin of anomalous cross-field transport in the SOL; 3) H-mode physics under strong electron heating at reactor relevant beta, e.g. using third harmonic X3 heating (1.5 MW); 4) ECH and ECCD physics, including phase space fast electron transport and electron Bernstein wave heating, demonstrated in the O-X-B scheme; 5) physics of improved steady-state tokamak regimes with internal transport barriers, with or without inductive currents, and with large (over 70%) bootstrap current fractions, confirming the key role of the current profile in the transition to improved confinement and the necessity of a negative core magnetic shear for obtaining eITBs.

OV/3-4 · Overview of the FTU results

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Abstract: The FTU ($R = 0.93\,m$, $a = 0.3\,m$, $B_T \lesssim 8\,T$, $I_p \lesssim 1.6MA$) program aims to develop advanced scenarios relevant to ITER, not only for the magnetic field and density ($n_e \gtrsim 10^{20}$ m^{-3}) but also for the pure electron heating, LH (8 GHz) and EC (140 GHz) RF waves and for the absence of a momentum source. Steady internal transport barriers (ITB) in almost full current drive (LHCD), are obtained at peak density $> 1.3 \times 10^{20}$ m^{-3} with the energy confinement time about 1.6 times the ITER 97-L scaling. Ion collisional heating does not affect the barrier dynamics: turbulence is strongly suppressed and the ion transport tends to be neoclassical, lower than ohmic value. The barrier radius can be successfully controlled by LHCD: steady ITB radii up to $r/a = 0.67$ are obtained by peripheral LH absorption, favored primarily by low safety factors q. Central counter ECCD is also effective. Feedback control/suppression of MHD tearing modes (TM, $m = 2$) with EC heating has been demonstrated. A fast digital analysis of the EC emission detects in real-time the TM presence, determines its radial position, selects and automatically switches on which one of the 4 available EC beams, is absorbed closest to it. A liquid lithium limiter (LLL) has been tested successfully. The novelty of the design complements that of the LLL concept. A mesh of capillary tubes is filled with the liquid and the associated surface tension withstands the tearing-off $J \times B$ forces. The LLL surface showed no damage up to the maximum thermal load of 10 MW/m^2. With walls "lithized", the plasma is cleaner (all other impurities almost disappear) and the neutral gas recycling strongly drops. Progress on the issue of mitigating disruptions has shown the efficiency of the EC power in avoiding or softening them. Experimental tests on the collective Thomson scattering in ITER-relevant configuration have shown how it is very crucial that backscattered radiation does not excite spurious gyrotron modes. Theory has fully explained the evolution of fishbone-like instabilities driven by LH generated supra-thermal electrons in FTU. The relevance for burning plasmas is for the interaction of low-frequency MHD modes with trapped alpha particles that are characterized by small dimensionless orbits, as electrons; besides the trapped particle bounce averaged dynamics depends on energy and not mass.

OV/4-1 · Overview of HL-2A Experiment Results

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Abstract: The specific results of recent experiment campaigns on HL-2A are presented. The stable and reproducible discharges with divertor configuration have been obtained using the reliable feedback control and wall conditioning techniques. Up to now, the main plasma parameters are as follows: $I_p = 400$ kA, $B_T = 2.65$ T, $n_e \sim 6.0 \times 10^{19}$ m^{-3} and the discharge duration $\sim$3.0 s. The advanced scenario with weak positive/negative shear of current profiles, calculated by TRANSP and EFIT codes using experimental data, have been achieved by the central fuelling of the pellet injection. The improved confinement can keeps about 500 ms. The studied physics issues include transport, zonal flows, impurity, disruption, confinement and divertor physics, and so on. Three dimensional features of geodesic acoustic mode zonal flows are determined with 3-step Langmuir probes at the edge plasmas. The symmetries ($m \approx 0 \sim 1$, $n \approx 0$) of the directly measured low frequency ($7 \sim 9$ kHz) electric potential and field are simultaneously observed for the first time. A local deposition of silicon during plasma discharges by silane gas puffing is adapted. The results of the subsequent discharges show a similar effect on the plasma performance with normal siliconization. Titanium and Aluminium are injected into plasma by laser blow-off. The diffusion coefficient and the inward convection velocity are given by an impurity transport code. The major disruption features and the disruption database on HL-2A ohmic plasma are given. A new criterion can predict 95% of the disruptions, based on the MHD activity. And a primary off-line neural network is also developed. The disruption mitigation using noble gas puffing and MBI have been demonstrated. The huge radiation loss during noble gas injecting can obviously increase the quench time and decrease the runaways during major disruption. The molecular beam injection (MBI) with liquid nitrogen temperature is used. It is thought that the low temperature MBI can form the hydrogen cluster and penetrate into plasma more deeply and efficiently. The asymmetric cold pulse perturbation is observed by means of ECE and soft X-ray arrays during pulse-modulated MBI experiment. The experimental results indirectly provide evidence of the shielding mechanism of the MBI physics.

OV/4-2 · Overview of TJ-II experiments

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Abstract: This paper presents an overview of the experimental progress made in TJ-II. Confinement and electric fields: Energy confinement time follows the ISS04 scaling. Local electron heat diffusion decreases in regions close to low order rationals and increase with heating power. A direct link between electric fields, density and plasma confinement has been found. Positive radial electric fields are measured in ECRH plasmas; and negative plasma potentials with NBI. Transitions and magnetic topology: The threshold density needed to trigger e-ITBs depends on the order of the rational magnetic surface. Quasi-coherent modes, found close to the radial location of the internal barrier foot vanish once the internal barrier is fully developed. Electrode biasing experiments show that it is possible to modify the edge radial electric field and the particle confinement for both polarities Plasma rotation and momentum re-distribution: Impurity poloidal rotation measurements show a link between plasma density and rotation. The generation of spontaneous perpendicular edge sheared flows requires a minimum plasma density. It has been recently 2-D visualized by means of fast cameras. As sheared flows develop turbulence structures become stretched indicating a modification in the perpendicular degree of turbulence anisotropy. Experimental results show significant turbulent parallel forces at plasma density rises above the threshold value. Shear flow physics involves 3-D phenomena in which both perpendicular and parallel dynamics play a role. MHD and plasma stability: Although magnetic well is the main stabilizing mechanism, plasma profiles are not dramatically affected when magnetic well is removed in the plasma edge. This result, consistent with recent findings in LHD, calls into question stability calculations (Mercier, Ballooning) based on assuming smooth pressure profiles. Analysis of ELM like events based on the coupling of ion-temperature-gradient modes to Alfvén and acoustic modes will be reported. The impact of magnetic topology (ripple) on the appearance of ELMs is under investigation. Plasma-wall studies: Configurations having rational numbers resulting in island chains at the edge have been explored for possible application in "divertor-type" plasmas. Experiments aimed at characterizing the fuelling efficiency of plasma particles and injected impurities will be reported.

OV/4-3 · Overview of T-10 Results

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Abstract: Physics of plasma confinement and stability in the regimes with electron cyclotron heating (ECH) is under investigation in T-10 tokamak. High-density plasmas with the energy confinement time that attains the H-mode scaling predictions have been obtained both with gas puffing and with deuterium pellet injection. Formation of electron internal transport barrier has been observed during the current ramp-up phase with ECH. Essential progress has been made in understanding the links between energy and particle transport and plasma turbulence properties. An interplay between evolution of the plasma confinement with density and turbulence properties related to ion tenperature gradient (ITG) and dissipative trapped electron (DTE) modes has been studied. Degradation of the energy confinement with an increase of ECH power correlates with the growth of the turbulence level at the low magnetic field side while it has been found to be almost unchanged at the high magnetic field side. Sawtooth control by localized electron cyclotron current drive (ECCD) in the vicinity of q = 1 surface is under investigation. Dependence of the stabilizing/destabilizing effect on the profile of the driven current has been studied and the optimal profile of the driven current has been found.

OV/4-4 · Experimental Progress on Zonal Flow Physics in Toroidal Plasmas

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Abstract: Fundamental physical understanding of zonal flows (ZFs) is absolutely necessary to precisely predict confinement properties of forth-coming devices such as ITER. The experiments on zonal flows (ZF) – a crucial element to clarify the turbulence transport – have made a large progress recently regardless of difficulty in their detection. Besides the trials to identify stationary zonal flows and geodesic acoustic modes using modern diagnostics techniques, the couplings between turbulence and zonal flows are quantified with advanced analyzing techniques. The paper aims at integrating the experimental knowledge on ZFs distributed in worldwide devices to obtain a unified view of ZF structure and dynamics, dependencies of zonal flow characteristics on plasma parameters and magnetic configurations, their impacts on the

turbulence and transport. In this paper, the present status of the ZF experiments is overviewed and the integrated knowledge on the ZF is presented.

OV/5-1 · Recent Progress on FIREX Project and Related Fusion Researches at ILE, Osaka

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Abstract: In the April of 2003, the FIREX project has started. In the project, the heating laser of 10 kJ/10 ps/1.06 μm, target fabrication and irradiation system of foam cryogenic target, and integrated fast ignition simulation code are developed as the collaboration program between Osaka University, NIFS (National Institute for Fusion Science) and other universities. After the completion of the laser, we will irradiate a foam cryogenic cone shell target with LFEX in late 2007. Cone shell target implosion has been simulated by 2D PINOCO. The shape of imploded core plasma of experiments is well reproduced in the simulation. Namely, the top of the cone is damaged by the jet produced by the stagnation of imploding fuel shell. The PINOCO simulation results also show that the plasma density is as high as the maximum density of spherical implosion and the area density can be higher than the spherical implosion. From the present understanding on the heating physics and the cone shell implosion hydrodynamics, we believe that the area density will reach 0.45 g/cm^2 and 1.2 g/cm^2 respectively in the FIREX-I, and -II. It is found that implosion of cone target is relatively insensitive to the hydrodynamic instabilities in comparison with spherical target. Planer cryogenic foam layer targets were irradiated with GEKKO XII laser recently. As a result, we found that a deuterium layer is compressed and accelerated by ablation pressure as expected in simulations. We also irradiated a foam cryogenic deuterium layer with the peta watt laser to investigate the heat transport. DD neutron from a cryogenic foam layer target has been observed. The neutron yield of cryogenic target is compared with CD plastic target to find that heat transport is strongly inhibited on a surface of a cryogenic DD layer. As an alternative fast heating scheme, the impact of a highly accelerated foil has been proposed, where the ablative acceleration to the order of 10^8 cm/sec is the key issue. The recent experiments show that the planer target is accelerated to 8×10^7 cm/sec. Finally the future scope of the fast ignition laser fusion will be discussed through reactor design.

OV/5-2 · Overview of Inertial Fusion Research in the United States

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Abstract: Laboratory-based thermonuclear ignition may be achieved by imploding spherical cryogenic capsules of frozen deuterium and tritium using MJ-class laser facilities currently under construction in the United States (the National Ignition Facility, NIF) and France (the Laser MegaJoule). The approach to ignition conditions and the physics of cryogenic capsule implosions are being validated by imploding directly-driven, ignition-scaled (hydrodynamically equivalent) cryogenic D_2 and DT-filled targets on the 60-beam, 30-kJ OMEGA laser at the University of Rochester Laboratory for Laser Energetics (LLE). A coordinated effort among the major US national laboratories and industrial partners (the National Ignition Campaign) is also underway to establish the scientific basis for an indirect drive ignition demonstration campaign on the NIF beginning in 2010. Recent technical progress includes the demonstration of improved hohlraum efficiency through the use of high-Z material mixtures and improved hydrodynamic stability through the use of a radially varying Cu dopant in the Be ablator. The broader US inertial fusion program is also developing high-gain target designs, including the Fast Ignition concept, and the technology needed for inertial fusion energy production. A new high-energy petawatt capability is nearing completion at LLE where it will be used with the OMEGA laser to validate FI relevant physics using ignition-relevant targets. Taken together, these results indicate a robust IFE program with the goal of laboratory-based ignition validation by the end of the decade.

OV/5-3 · Theory of Alfvén waves and energetic particle physics in burning plasmas

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Abstract: We discuss two issues of practical interest for burning plasmas: (i) whether fast ions and charged fusion products are sufficiently well confined that they transfer their energy and/or momentum to the thermal plasma without appreciable degradation due to collective modes; (ii) whether mutual

interactions between collective modes and energetic ion dynamics on the one side and drift wave turbulence and turbulent transport on the other side, may decrease, on long time scales, the thermonuclear efficiency of the considered system. In the present work, we analyze both of these aspects starting from their first principle theoretical grounds, i.e. the identification of burning plasma stability boundaries as well as their nonlinear dynamics above threshold. We also discuss the investigations of such processes via computer simulations as well as the importance of benchmarking with existing or future experimental observations.
* This work is supported by U.S. D.o.E.

OV/5-4 · Status of R&D Activities on Materials for Fusion Power Reactors

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Abstract: Current R&D activities on materials for fusion power reactors are mainly focused on plasma facing, structural and tritium breeding materials for plasma facing (first wall, divertor) and breeding blanket components. Most of these activities are being performed in Europe, Japan, P.R. China, Russia and the USA. They relate to development of new high temperature, radiation resistant materials, development of coatings that shall act as erosion, corrosion, permeation, electrical or MHD barriers, characterization of the whole candidate materials in terms of mechanical and physical properties, assessment of irradiation effects, compatibility experiments, development of reliable joints, and development and/or validation of design rules. Priorities defined worldwide in the field of materials for fusion power reactors will be summarized, as well as the main achievements obtained during the last few years and the near-term perspectives in the different investigation areas.

OV/6-1 · Status of Inertial Fusion Energy Program in China

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Abstract: The goal of the first milestone of the inertial fusion energy (IFE) program in China is to reach fusion ignition and plasma burning in about 2020. Under the program, in the past years, the target physics research achieved great progress; SG-II has been operating with high quality since 2000: SG-III prototype began operating in 2005 and SG-III has been designed and will be completed in about 2012, and the support technologies for laser drivers are developed and improved; precise diagnostic techniques are developed and relatively integrated system is set up; precise target fabrications are coordinately developed. In addition, the architecture of SG-IV laser facility, National Ignition Driver, with laser energy output of about MJ serving for ignition and burning is being planned.

OV/P-1 · Recent experiments in the HT-7 superconducting tokamak

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Abstract: Since the last IAEA meeting, experiments in the HT-7 tokamak focused on long pulse discharges under different scenarios. Investigation shows that current drive efficient with the combination of LHW and IBW was 10 $\sim$ 20% higher than LHCD only. The discharges with optimized current drive efficient in this scenario has been extended longer than 10 s at Ip = 100 kA, $n_e(0) \sim 2\times10^{19}m^{-3}$ with the loop voltage close to zero. The long pulse discharges at the level of Ip $\sim$ 60 kA, $n_e(0) \sim 0.8 - 1 \times 10^{19}m^{-3}$ has been extended to longer than 6 minutes using about 150 kW of LHCD only, which are the new records of both the pulse length and injected energy in HT-7. A new steady-state AC operation mode has been demonstrated up to 35 s at the level of Ip = 125 kA, $n_e(0) \sim 1.5 - 2.0 \times 10^{19}m^{-3}$. Various issues around long pulse discharges including plasma and wall interaction, recycling, retention, new materials and etc were investigated. The relevant physics concerning on the lower hybrid current drive, synergy effect of lower hybrid wave and ion Bernstein wave on plasma confinement and current drive efficient and dynamics of runaway electrons were studied. Radial propagation of electrostatic turbulence and intermittently occurring large-scale coherent structures were measured using the Langmuir probe arrays to study the turbulent transport in the plasma peripheral region.

OV/P-2 · Overview of Recent Experimental Studies on TRIAM-1M

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Abstract: An overview of recent experimental studies on TRIAM-1M is presented. Several issues of plasma-wall interaction including particle retention in wall materials, co-deposition, and impurity behavior are investigated. It has turned out that the particle behavior in one super long discharge is not equivalent with that of the summation of successive medium long term discharges. Also, a clear increase of the number of dust particles has been observed as a function of discharge time in the case of long term discharges. Some issues of current drive with lower hybrid frequency and of the combination with electron cyclotron frequency (ECCD) are studied. In addition, the ECCD experiment relevant to ITER has been carried out with high toroidal magnetic field ($\sim$6 T).

OV/P-3 · Overview of the Globus-M Spherical Tokamak Results

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Abstract: In the Globus-M spherical tokamak a diminishing of impurities concentration in the discharge and the loop voltage decrease was achieved as a result of technological improvements. Plasma current plateau duration reached 2–3 current diffusion times and some plasma pulses were terminated artificially by the preprogrammed current ramp down. During experiments on density limit investigation in OH and NBI heating regimes the record absolute values of density in 0.4 T magnetic field was reached in excess of 1.1×10^{20} m^{-3} with gas puff supply only. The Greenwald limit was achieved and overcame. During NBI two distinct regimes, one – low density with overheated ions and another – high density with features of electrons heating were obtained and studied. The fundamental harmonics ICR heating of hydrogen “minority” in deuterium plasma showed that the heating efficiency is relatively high and is insensitive to hydrogen concentration in the range of 10–70%. The role of the second harmonic is clarified. Plasma jet injection into the Globus-M with the help of a double-stage plasma gun showed that jet penetrated into the plasma as a time-of-flight recombined dense flux of neutrals. First numerical simulations showed that after crossing the separatrix and ionization the jet particles decelerates due to emission of Alfvén wave. The deceleration time of the jet with initial velocity of $\sim$100 km/s should penetrate deep (10–20 cm) into the Globus-M plasma. This is consistent with interferometer measurements and local density measurements performed by Thomson scattering. MHD activity during all the experiments was monitored by different diagnostic tools. The role of core long wave fluctuations and their coupling with peripherial modes is outlined in low q_{95} regimes.

EX

Magnetic Confinement Experiments

EX/1-1 · The performance of improved H-modes at ASDEX Upgrade and projection to ITER

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Abstract: At ASDEX Upgrade, stationary discharges with improved confinement ($H_{98}(y,2) > 1$) and improved stability ($\beta > 2.5$) compared to standard H-mode have been developed since 1998. New results presented here concentrate on extending the operational range of these improved H-modes at ASDEX Upgrade and extrapolating these new results to ITER. The performance is optimised for q_{95} ranging from 3 to 5, for different values of collisionality, at high edge density, with significant central electron heating and with a first wall predominantly covered by tungsten coated carbon tiles. For the extrapolation to ITER, the fusion gain that could be obtained is evaluated using different confinement scalings, while the ASTRA transport code is used to provide an extrapolation of the fusion power in ITER. These ASDEX Upgrade results indicate that improved H-modes (candidate for a hybrid scenario) could extend ITER operation beyond what is currently foreseen using standard H-modes.

EX/1-2 · Progress Toward High Performance Steady-State Operation in DIII-D

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Abstract: Advanced Tokamak (AT) research in DIII-D works toward development of a scientific basis for steady-state high performance scenarios. These scenarios are a high level goal for ITER, in recognition of their potential for high duty cycle and reduced cyclic fatigue in a power plant. Bootstrap supplies most of the current ($f_{BS} \approx 60\%$–75%), with the remainder supplied by neutral beam (NBCD) and electron cyclotron (ECCD) current drive. In recent experiments, $\beta_N \approx 4 \approx 6l_i$ has been maintained for 2 s. These plasmas operate well above the no-wall stability limit ($\sim 4l_i$), enabled by active error field and resistive wall mode control. This has been achieved only during toroidal magnetic field ramps, which appear to broaden the current profile to improve coupling with the wall and control coils. These plasmas have internal transport barriers, contrasting previous experience where β_N is limited below $\sim$2. In other experiments, fully noninductive conditions $f_{NI} \approx 100\%$ have been sustained for several confinement times or about half a current relaxation time (τ_R). Similar discharges, with $f_{NI} \approx 90\%$–95%, are stationary for the entire 2 s ($\sim 1\tau_R$) ECCD pulse. These plasmas have $\beta_N \approx 3.5$, $q_{min} \geq 1.5$ and weakly reversed magnetic shear in the core. Recent efforts in this scenario focus on mapping the operational space. f_{NI} is found to increase with both q_{95} and β_N. Fusion performance decreases with q_{95}, so these experiments suggest continued emphasis on increasing beta at moderate q_{95}. At the same time, low density is observed to be advantageous for noninductive operation, primarily through its impact on the effectiveness of external current profile control tools. Improvements now underway on DIII-D include additional long-pulse ECCD and fast wave (FWCD) and density control for high triangularity double-null configurations. Theory-based simulations including these new capabilities predict in-principle steady-state conditions with high β_N maintained for several τ_R. The same models, applied to ITER, extrapolate current DIII-D results to steady-state scenarios with $Q \geq 5$.
* Work supported by US DOE under DE-FC02-04ER54698, DE-AC05-00OR22725, DE-FG03-98ER53297, DE-FG03-01ER54615, W-7405-ENG-48, DE-AC02-76CH03073, and DE-FG02-92ER54141.

EX/1-3 · Improved Performance in Long-pulse ELMy H-mode Plasmas with Internal Transport Barrier in JT-60U

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Abstract: The performance of long-pulse ELMy H-mode plasma was improved after the installation of ferritic steel tile on JT-60U in terms of the sustained duration for both high normalized beta and high thermal confinement factor (H_{98}). High normalized beta larger than 2.3 simultaneously with $H_{98} \sim 1$ was sustained for 23.1 s ($\sim$12 times resistive diffusion time) at $q_{95} \sim 3.2$, which provide high normalized beta times $H_{98} > 2.2$ exceeding ITER reference scenario of $\sim$1.8. A sustainable high normalized beta times H_{98} for maximum duration of 28.6 s was 2.0. Improved confinement is characterized by the larger thermal components at given density maintained by smaller heating power than previous experiments. A peaked pressure profile indicating better confinement was obtained at the same shape of the heating power deposition profile in the core region ($r/a < 0.6$) as that in the previous experiments before the installation of ferritic steel tile.

EX/1-4 · Evolution of Bootstrap-Sustained Discharge in JT-60U

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Abstract: A self-sustained state driven by the bootstrap current was achieved in JT-60U. Only perpendicular and counter tangential neutral beam injections were used so that the neutral beam driven current was negative. The current in the Ohmic heating coil was kept constant so it does not cause induction. The dynamics of such a completely self-driven system with no external current drive is studied. At a toroidal field of 3.7 T, beta collapses were often observed. The internal transport barrier (ITB) shrinks radially at such collapses, and the stored energy and the plasma current are reduced. Subsequent self-recovery of the ITB radius, as well as the stored energy and the plasma current, is observed, although the recovery is not complete. It is also demonstrated that beta collapses can be avoided by raising the toroidal field to 4 T. A plasma current of greater than 0.55 MA was maintained for 2.5 s. After an initial transient in which the plasma current increases due to stored energy increase in spite of the reduced heating power, the loop voltage becomes zero and the plasma current is maintained at a nearly constant level for 1.3 s. This plasma has a normalized beta of 1.2, poloidal beta of 3.0, a large ITB radius, and a current density profile peaked near the edge. The neutral beam driven current is calculated to be approximately −50 kA and the calculated bootstrap current is approximately equal to the total plasma current. The inductively driven current is nearly zero, but still slightly positive in the outer half radius.

EX/1-5 · Maintaining the Quasi-Steady State Central Current Density Profile in Hybrid Discharges

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Abstract: The Hybrid scenario, with confinement and beta improved above the baseline ELMy H-mode, is an attractive operation scenario for ITER. One intriguing feature of these discharges in DIII-D as well as in JET is that the plasma evolves into a quasi-steady state without sawteeth. The central safety factor (q_0) is kept close to 1 and correlates with a rotating 3/2 magnetic island. The central current density is found to be smaller than expected without taking into account the effect of the 3/2 island. In this work, we report on results from investigation on the causal relationship between the 3/2 island and the non-sawteething nature of the discharge. Two mechanisms were invoked; both rely on a large co-rotating 2/2 sideband excited by the rotating 3/2 island when the plasma central q drops towards 1. The first mechanism is the excitation of the kinetic Alfvén wave (KAW) by the 2/2 sideband in the central region, which (due to its short perpendicular wavelength) provides an efficient current driver in direction counter to the plasma current. In the second mechanism, the 3/2 and its 2/2 sideband are assumed to redistribute the energetic ions resulting from the injected neutral beam. Thus, reducing the beam-driven current in the plasma center. Equilibria reconstructed from discharges in DIII-D are analyzed using the MHD stability codes. The 3/2 island is found to develop a 2/2 sideband with increasing amplitude as q_0 approaches 1, as expected from standard MHD theory. The 2/2 sideband is shown to become resonant in the plasma center and converts into KAWs. The ONETWO transport code, which takes into account the neutral beam current drive from energetic ions, has been utilized to simulate the development of the discharge. When there is additional transport for the energetic ions, the profile of neutral beam driven current becomes broader, indicating equivalent counter current drive inside the 3/2 surface. These two mechanisms have been found sufficient for driving the negative current necessary for maintaining the steady current profile.
* Work supported by U.S. DOE under DE-FG03-95ER54309, DE-FC02-04ER54698, W-7405-ENG-48, DE-FC02-04ER54698, and DE-AC05-76OR00033.

EX/1-6 · Physics and operational integrated controls for steady state scenario

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Abstract: Assembling the relevant physics features using integrated controls is expected to be a major challenge in the operation of ITER steady state scenario. In recent experimental campaigns, Tore Supra has therefore focused its efforts on the physics optimisation and operation of steady state scenario with high input power close to 7 MW and vanishing loop voltage (more than 90% of non-inductive current) with duration of typically 60 s and above. To use its long pulse capabilities Tore Supra has also been equipped with a large number of new real time systems. With these tools, the control of the LH deposition profile width measured by the Hard X-ray camera has been achieved with the parallel index and power of the LH-wave using different type of control algorithms. Temperature gradient has also been controlled during internal transport barriers and the effect of Electron Cyclotron Heating (ECH) on control assessed. Then, the control of the current profile with the parallel index has been achieved in combination with loop voltage control (at 60 mV) using the central solenoid voltage as actuator. The RF antennae are receiving power fluxes from their private power but also from the plasma convective power and large orbit fast ions generated by ICRH. On the basis of the experimental power load and hot spot analyses, the seven infra-red cameras monitoring the five antennae (2 LH-launchers and 3 ICRH antennae) and the bottom limiter have been used for real time protection and combined successfully to LH deposition profile and loop voltage control. These experiments and modelling are pioneering the integration work that will be required on ITER when combining physics requirements for achieving the requested plasma performances and machine technologic constraints.

EX/2-1 · Study of Turbulence and Radial Electric Field Transitions in ASDEX Upgrade using Doppler Reflectometry

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Abstract: The radial electric field (E_r) is a crucial factor in the performance of magnetically confined plasmas and on density turbulence. On ASDEX Upgrade Doppler reflectometry has been developed for direct measurement of E_r profiles, its radial shear and its fluctuations. In the plasma edge the E_r radial profile shows the narrow negative well coinciding with the steep pedestal pressure gradient whose depth scales with the plasma confinement: from typically -50 V/cm in ohmic & L-mode conditions to -300 V/cm in H-modes, to over -500 V/cm in improved H-modes. The structure of the edge E_r profile is notably robust, but the core E_r of non-NBI heated discharges when increasing collisionality reverses from positive to negative with a transition in the dominant turbulence from TEM to ITG. Coherent E_r fluctuations with geodesic acoustic mode (GAM) behaviour are observed in the plasma edge, coinciding with region of high plasma vorticity and E_r shearing. GAMs are not detected in the core or in H-mode. The mode has the expected frequency scaling of sound speed over major radius but with additional dependency on plasma elongation and q.

EX/2-2 · Measurement and analysis of the fluctuations and poloidal flow on JFT-2M tokamak

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Abstract: The potential/density fluctuations and E $\times$ B poloidal flow inside of the separatrix were measured in the JFT-2M tokamak using heavy ion beam probe (HIBP),reciprocate probe and microwave reflectometer. The strength of the three-wave interaction between the background turbulence and the geodesic acoustic mode (GAM) was obtained by estimating the bispectrum function, and its convergence property was clarified for the first time. The GAM disappears in the H-mode, instead negative DC electric field dominates. The observed DC poloidal flow at the edge transport barrier(ETB) of ECH H-mode is found to be typically $20 \sim 30$ times larger than the oscillating ($\sim$15 kHz) poloidal flow by the GAM in the L-mode.

EX/2-3 · Characterization of Zonal Flows and Their Dynamics in the DIII-D Tokamak, Laboratory Plasmas, and Simulation

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Abstract: Zonal flows, including the zero-mean-frequency, spectrally broad flows and the higher frequency, oscillatory geodesic acoustic mode (GAM), have been characterized in the DIII-D tokamak core. These flows have long been predicted theoretically to be the crucial turbulence saturation mechanism in magnetically confined plasmas. The zero-mean-frequency zonal flow has been identified for the first time in the core of a tokamak plasma using multipoint, 2-D measurements of the density turbulence and its resulting velocity-field. These zonal flows are observed as a low frequency, spectrally broad feature in the derived poloidal turbulence velocity spectrum, peaking near zero frequency and exhibiting a width of $\Delta f \approx 10$ kHz. These flows exhibit a long poloidal wavelength and a short radial correlation length of a few cm. This velocity spectrum is dominated near the edge of the plasma ($0.9 < r/a < 0.95$) by the GAM, while the low-frequency zonal flow becomes dominant deeper in the plasma core. GAM amplitude is shown to be a strong function of the safety factor, q_{95}, consistent with theoretical predictions based on ion Landau damping. Experiments in a laboratory plasma device show the presence of an azimuthally sheared flow

consistent with a turbulent momentum balance that includes the measured turbulent Reynolds stress and flow damping. Zonal flow is shown to quench the turbulent particle flux. A nonlinear transfer of turbulent energy to higher frequency is measured in the tokamak fluctuation data and suggests that the GAM plays a role in saturating the turbulence in the near edge transition region. Application of an algorithm that calculates energy transfer between density fluctuations of different frequencies demonstrates a GAM-mediated transfer of energy between density fluctuations with frequency f_0 to poloidal density gradient fluctuations with frequency $f_0 \pm f_{\mathrm{GAM}}$, leading to a net transfer of energy from low to high frequencies. These experimental observations of zonal flows from confinement and laboratory devices and comparison with turbulence simulation validate the fundamental nonlinear dynamics of turbulence in plasmas and aid in the development of a fully predictive transport capability in future burning plasma experiments.
* Work supported by US DOE under DE-FG03-96ER54373, DE-FG02-04ER54758, DE-FG03-95ER54309, DE-FG02-92ER54141.

EX/3-1 · ELM transport in the JET scrape-off layer

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Abstract: Experiments and modelling at JET are advancing our understanding of scrape-off layer (SOL) edge localised mode (ELM) transport, providing many of the elements required for a quantitative treatment of the ELM energy fluxes and their consequences for plasma-wall interaction. Infra-red thermographic observations reveal that ELMs deposit energy preferentially at the outer divertor target at low pedestal collisionality, but that the in/out heat flux ratio approaches or even exceeds unity with rising collisionality. The IR data, in combination with fast triple Langmuir probe measurements, provide direct experimental evidence for strong variations in the ELM sheath heat transmission coefficients. The magnitude and temporal variation of these variations are in good agreement with new kinetic simulations of parallel transport in the JET Type I ELMing SOL using the BIT1+ particle-in-cell (PIC) code. Parallel heat flux limits and transmission coefficients increase by an order of magnitude during the ELM, with the amplitude of this rise scaling linearly with the pedestal temperature and with the distribution of ion energies at the targets strongly weighted to higher energies. To include these time dependent kinetic effects into the 2D fluid-Monte Carlo code packages in use for SOL modelling at JET (SOLPS5 and EDGE2D), a new transient model of ELM filament energy evolution due to parallel losses has been favourably benchmarked against the PIC simulations and is used to derive parametric expressions for the kinetic coefficients. This same transient model matches a wide range of ELM observations in the JET SOL, including target IR in/out heat flux ratios and pulse delays, characteristic ELM temperature widths and direct measurements of ion energies in the far SOL obtained with a retarding field analyser probe. This data clearly shows that ions, convected by the ELM to the main walls, can arrive there with energies up to 50% of those found in the pedestal plasma. It also demonstrates, as do dedicated fluctuation probe measurements, that the ELM is a turbulent phenomena, comprising a wavetrain of pulses consistent with plasma filaments released at multiple toroidal locations from a rotating pedestal. Radial ELM particle fluxes increase with increasing ELM size, as do the ELM filament cross-field velocities, observations supported by divertor and main chamber thermography.

EX/3-2 · Density Regime of Complete Detachment and Operational Density Limit in LHD

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Abstract: The highest central density in net current free helical plasmas of 5 times 10 to 20th per cubic meter has been demonstrated in the recent Large Helical Device (LHD) experiment. The volume-averaged electron density exceeds $3 \times 10^{20}\mathrm{m}^{-3}$. These are attained in the plasmas with strongly peaked density profile generated by pellet injection. These results indicate that helical plasmas are suitable for the high-density operation, which easily leads to the remarkable reduction of divertor heat load by detachment and therefore is favorable for realizing future fusion reactors. In LHD, sustainable complete detachment, named the Serpens mode, has been found in the density regime just below the operational density limit where plasmas are terminated by radiative collapse. So-called Sudo density limit scaling has been often used to discuss the operational density limit of helical plasmas. The Sudo scaling is based on the power balance between the heating power and the radiation loss that is proportional to the square of the electron density. However, radiative collapse is triggered even at a small radiation loss fraction of about 0.3, while it ranges from 0.3 to 1 at complete detachment. It is therefore difficult to determine a threshold radiation loss fraction that triggers radiative collapse. Other than the radiation loss fraction, the position of the

hot plasma boundary is important. Complete detachment takes place when the hot plasma boundary shrinks below the last-closed-flux-surface (LCFS). At detachment, the ionization front moves from the ergodic region to inside of the LCFS and the particle confinement effectively improves. The edge density that results in this shrinking corresponds to the maximum edge density achievable under the attached condition. Even though the Sudo fraction, which is the ratio of the volume-averaged density to the Sudo scaling, reaches as high as 3.5 in pellet fueled plasmas, the edge density in these cases are similar to that of gas-fueled plasmas at the threshold for complete detachment. Indeed, the Sudo fraction is proportional to the peaking factor defined by the ratio of the volume-averaged density to the edge density, which reaches 4 in pellet-fueled plasmas. In detached plasmas, the edge densities are roughly twice as large as those in the attached plasmas, reflecting the improved particle confinement.

EX/3-3Ra · Tungsten as First Wall Material in ASDEX Upgrade

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Abstract: ASDEX Upgrade has pursued a continuous increase of tungsten surfaces during the last seven years in order to investigate: tungsten erosion, radial impurity transport, control of divertor temperature by radiative cooling, and influences of tungsten on the gas balance. At present, 85% of the plasma facing components (PFCs) are tungsten coated. The last enhancements focused on the poloidal limiters at the low field side which recieve the highest power load in the main chamber. For the tungsten erosion at the these limiters, fast particles from NBI as well as impurity ions accelerated in the rectified sheath in front of the active ICRF antennas play an important role, while CX neutrals yield only a minor contribution. The fast ion load onto the limiters has been calculated with a detailed Monte-Carlo model, and the calculated W influx densities are in good quantitative agreement with spectroscopically measured spatial distributions. The increase of W-coated areas is also reflected in an increase of the W content in the plasma core. However, the techniques for impurity transport control, i.e. ELM pace making and sufficient central wave heating, allow keeping the W concentration below 10^{-5}. Changes in the total gas balance are not yet detectable, however, the release of short term retained noble gases in the tungsten coated tiles shows a pronounced increase with respect to the pure graphite tiles.

EX/3-3Rb · The Implications of High-Z First Wall Materials on Noble Gas Wall Recycling

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Abstract: With the transition from low-Z first wall materials like graphite (C) to high-Z materials like tungsten (W) the operating parameters of fusion experiments like ASDEX Upgrade have to be adjusted to be compatible with the new material capabilities. Recent experiments in ASDEX Upgrade have experienced surprisingly high He plasma impurity concentrations. Such high He concentrations have not been observed with C walls, they were only observed since the increase of the W first wall coverage of ASDEX Upgrade to 85%. The high He plasma concentration appears to be linked to the fraction of W surfaces open to plasma contact that are not covered by boronization layers and to the number of He glow discharges performed for wall conditioning prior to normal plasma operation. This pointed to the different retention and release properties of W and C for He. To elucidate these differences dedicated experiments have been performed. To study the retention, W and C samples identical to those used in ASDEX Upgrade were implanted with He at 200 eV and 600 eV and the amount of retained He was determined through TDS and IBA methods. They showed that W can retain up to 10 times more He than C depending on the energy of the implanted He. The differences in the release rate of He from W and C surfaces due to particle bombardment were investigated by exposing the He implanted W and C surfaces to a H plasma and measuring the loss of He. It was found that for 200 eV H energy three times more He is released from W than from C. The differences between W and C could be qualitatively explained by TRIDYN calculations: The higher retention of He in W at high implantation energies compared to C can be attributed to the stronger erosion of the He implantation zone in C and the resulting loss of He during the implantation. The stronger release of He from W due to particle bombardment could be explained by the higher retention and denser collision cascades close to the surface in W resulting in a high He sputter yield. From these experimental results one can conclude that the use of a high-Z first wall material results in different boundary conditions for the use of medium Z noble gases for wall conditioning or plasma seeding due to its potential for higher retention of noble gases compared to C.

EX/3-4 · Operation of Alcator C-Mod with High-Z Plasma Facing Components with and without Boronization

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Abstract: Present plans for reactor Plasma Facing Component (PFC) surface materials call for tungsten due its low tritium (T) retention, capability to handle high heat fluxes with low erosion, and robustness to nuclear damage and activation. ITER, to provide required operational experience for DEMO, will likely at some point in its lifetime operate with all tungsten high-Z PFCs. Recent C-Mod experiments, utilizing molybdenum PFCs, provide unique divertor tokamak operational experience with a high-Z PFC (molybdenum), comparing boronized and un-boronized surfaces. After boron was removed from vessel & PFC surfaces, RF-heated H-modes were readily achieved although the resultant enhancement in energy confinement over L-mode was small. Molybdenum fractional densities approach 0.1%, rapidly rising after the H-mode transition, cooling the plasma, reducing confinement and/or causing back H/L transitions. After boronization, the situation is dramatically changed: Mo density was reduced by a factor of more than 10, and energy confinement doubled. Under these conditions, a tokamak world record volume-average plasma pressure of 1.8 atmospheres at 5.4 T was achieved at the ITER beta-N. The positive effects of boronization wear off, correlated with integrated ICRF input energy; the boron erosion rate is significantly reduced in discharges without ICRF input. Inter-shot boronization techniques were also successfully applied. These experiments indicate that certain regions of the PFCs, away from the divertor strike points, may be primarily responsible for the Mo influx. With or without boronization, the H/D retention could be significant. This initial comparison indicates that high-Z operation without boronization carries the risk of degraded confinement; boronization, or other low-Z wall coating, might be required with high-Z PFCs.

EX/3-5 · Material erosion and redeposition during the JET MkIIGB-SRP divertor campaign

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Abstract: Wall erosion, material migration and the associated problems of long-term tritium retention in deposited layers and wall lifetime are among the most critical issues to be solved for ITER and all future steady state burning devices. This contribution provides an overview of experimental results of material erosion and migration during the JET campaigns with the MkIIGB-SRP divertor configuration (2001–2004) and summarises recent modelling activities aimed at understanding these data. New insights into material erosion and deposition in the JET divertor have been made possible through the use of $\sim$3 micron thick tungsten stripes deposited on a full poloidal set of divertor tiles at a single toroidal location. At all inner divertor tiles, metallographic cross-sections of the W-stripes confirm the post-mortem analysis from previous campaigns of strong net carbon deposition, with layer thicknesses ranging from 10 to 300 microns. Analysis of the W marker stripe at the outer divertor reveals heavy erosion at positions near the strike point and on the upper regions and apron of the upper tile. Modelling with the Monte-Carlo code ERO shows that the W erosion is dominated by carbon impurity sputtering. A dedicated material transport study has been performed at the end of the campaign by injecting ^{13}C marked methane into the outer divertor. Post mortem surface analysis of a poloidal set of divertor tiles finds $\sim$16% of the injected ^{13}C on the outer divertor tiles and $\sim$6% at the inner divertor, clearly indicating material transport from outer to inner divertor. A significant quantity of ^{13}C is also found on the horizontal outer tiles. Results will be presented from two dimensional modelling of the ^{13}C transport and deposition using the fluid Monte-Carlo code packages EDGE2D and SOLPS5. Shot resolved measurement of deposition in the inner divertor louver region using a quartz microbalance (QMB) show large deposition with the strike point on the inner base plate offering a direct line-of-sight to the QMB. In contrast, deposition on the QMB is small if the strike point is located on the lower part of the vertical target. It is at or below the detection limit once the strike point is moved further up. Simulations with ERO reproduce the importance of strike point location for the carbon migration toward the QMB.

EX/3-6 · Gas Balance and Fuel Retention in Fusion Devices

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Abstract: The evaluation of hydrogenic retention in present tokamaks is of a crucial importance to estimate the expected tritium (T) vessel inventory in ITER, limited for safety considerations to 350 g. In the frame of the European Task Force on Plasma Wall Interaction (EU TF PWI), efforts are underway to investigate the gas balance and fuel retention during discharges and compare with that from post-mortem tile analysis integrated over experimental campaigns. The aim is to assess the dominant processes that determine the fuel retention in order to extrapolate to ITER. This paper summarizes the principal findings from coordinated studies on gas balance and fuel retention from a number of European tokamaks e.g. ASDEX-Upgrade (AUG), JET, TEXTOR and Tore Supra. Gas balance is one of the few possibilities to evaluate the fuel retention in ITER in the non-activated phase. This contribution will also discuss the work that is required to improve the fuel inventory evaluation from gas balance and tile analysis. However, from the database available from tokamaks with carbon as main PFCs, the important conclusion is that T inventory limit could be reached in ITER within only ∼100 discharges and could therefore seriously impact the device operation unless efficient T removal processes are developed.

EX/4-1Ra · Active Control of Neoclassical Tearing Modes toward Stationary High-Beta Plasmas in JT-60U

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Abstract: Results from active control of neoclassical tearing modes (NTMs) with electron cyclotron current drive (ECCD) are described. In JT-60U, it was previously demonstrated that the amplitude and period of sawtooth oscillations can be controlled by optimizing the location and direction of the ECCD. In particular, ECCD inside the sawtooth inversion radius can increase the frequency and amplitude of the sawtooth oscillations. By utilizing the sawtooth destabilization, a new discharge scenario to control the evolution of an m/n = 3/2 NTM through sawtooth control has been developed (m and n are poloidal and toroidal mode numbers, respectively). For the cases of no EC wave injection and 1.3 MW of EC wave injection, a 3/2 NTM appears and grows steadily. On the other hand, for the case of 2.6 MW EC wave injection (EC-driven current is ∼10% of the total plasma current), the growth is suppressed and the amplitude of the NTM is kept low throughout the high-beta phase even after the turn-off of EC wave injection. Frequency spectrum of magnetic perturbations shows that the mode growth is interrupted by a sawtooth crash, which clearly shows the interaction between the sawtooth crash and the 3/2 NTM. The value of the normalized beta for the 2.6 MW EC case is higher than that for the other cases, suggesting the improvement of the beta value and confinement by the NTM control. An NTM with m/n = 2/1, which causes larger degradation of confinement than a 3/2 NTM, has been also successfully stabilized by injecting EC wave to the mode rational surface. Analysis with the ACCOME code and a Fokker-Planck code shows that EC-driven current density at the q = 2 surface, which is located at ∼0.5 in the normalized minor radius, is comparable to the local bootstrap current density. In addition, a transport code TOPICS has been improved to solve the modified Rutherford equation, which enables a self-consistent analysis of the evolution of island width and current profile during NTM stabilization. By using the TOPICS code, effects of ECCD width on NTM stabilization have been quantitatively evaluated. It has been found that the stabilization effect significantly increases with decreasing the ECCD width. The simulation also clarifies the importance of ECCD width: EC wave power required for complete stabilization can be reduced less than half by narrowing the ECCD with by about 30%.

EX/4-1Rb · Control of MHD Instabilities by ECCD: ASDEX Upgrade Results and Implications for ITER

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Abstract: MHD instabilities limit the operational space of tokamaks; their control is therefore of great interest for present day and future tokamaks, such as ITER. Based on recent success in control of sawteeth and Neoclassical Tearing Modes (NTMs), ECCD is foreseen as an MHD control tool in ITER. Hence, it is important to expand our physics base of MHD control by ECCD and to verify the control strategies proposed for ITER in present day experiments. In this paper, we report on recent experiments in ASDEX Upgrade in this area. Experiments on sawtooth tailoring of plasmas with dominant NBI heating have been performed with broad and narrow deposition of ECCD, indicating a better control with the narrower deposition profile. Using B_t-ramps, it was possible to map out precisely the different deposition regions suited for either stabilization or destabilization of sawteeth. The effects of co and ctr-ECCD as opposed to pure heating have clearly been documented. In addition, ICRH is used to study the fast particle stabilisation of sawteeth, exploring the possibility to approach the ITER situation where alpha-particles are supposed to have a large stabilising effect on sawteeth. In the area of NTM control, experiments with optimised deposition, maximising the figure of merit $\eta_{NTM} = j_{ECCD}/j_{bs}$, verify the relevance of

this figure of merit, which has been adopted by ITER for the optimisation of the NTM stabilization system and have increased significantly the β_N range in which (2,1) NTMs can be completely stabilized in ASDEX Upgrade. Another crucial point for ITER is the question if injection has to be phased with the O-point of the island or can be just DC. Experiments in ASDEX Upgrade with an artificially broadened deposition profile mimic the situation in ITER, where it is expected that the marginal island size will be significantly smaller than the deposition. These experiments, conducted at $W_{marg}/d = 0.5$, whereas previous experiments had $W_{marg}/d = 1$ and ITER is expected to have $W_{marg}/d = 0.1$–0.5, indicate an advantage of modulated injection, also in the FIR regime of NTMs at higher β_N. The experiments are accompanied by modelling using a fully nonlinear resistive MHD code as well as an analysis based on the modified Rutherford equation in order to enable the prediction for ITER. Finally, we will also discuss the next steps undertaken at ASDEX Upgrade in this area.

EX/4-2 · Prevention of the 2/1 Neoclassical Tearing Mode in DIII-D

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Abstract: Onset of the $m/n = 2/1$ neoclassical tearing mode (NTM) has been prevented in high-performance DIII-D discharges using localized electron cyclotron current drive (ECCD). Active tracking of the $q = 2$ surface location, using real-time equilibrium reconstructions with motional Stark effect data, allows the current drive to be maintained at the rational surface even in the absence of a detectable mode. With the application of this technique in DIII-D hybrid discharges, the 2/1 mode is avoided and good energy confinement is maintained for more than 1 second with beta at the estimated no-wall stability limit for ideal kink modes (β_T approximately equals 4.2% and normalized beta β_N approximately equals 3.2). A real-time correction of the ECCD location, taking account of refraction effects as the density varies, has recently been implemented and will be incorporated into future experiments.
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EX/4-3 · Gas Jet Disruption Mitigation Studies on Alcator C-Mod and DIII-D

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Abstract: Noble gas jet injection on Alcator C-Mod and DIII-D provides good mitigation of deleterious disruption effects, even though the jet does not penetrate deeply into the plasma as neutral gas. Disruption damage can come from overheating of divertor surfaces, electromagnetic loads on conducting structures, and localised impact of relativistic electrons. High-pressure noble gas jet injection is a mitigation technique which potentially satisfies the requirements of fast response time and reliability, without degrading subsequent discharges. Previous experiments on DIII-D showed good success at reducing all of the deleterious effects. More recently, gas jet experiments on Alcator C-Mod have tested the effectiveness of this approach on the higher pressure, higher energy density plasmas representative of C-Mod and ITER. Initial results confirm the ability of this technique to radiate away very high energy densities on timescales consistent with C-Mod's fast current quench. Higher-Z gas jets (Ar, Kr) result in a 50% reduction in halo current and also reduce the heating of divertor surfaces, similar to the DIII-D results, with no deleterious effects on subsequent discharges. High-speed imaging on both DIII-D and C-Mod shows only shallow penetration of the gas jets. However, the plasma core is affected on a timescale which is much faster than normal transport processes. An understanding of this paradox is obtained by modeling with the NIMROD MHD code, which shows that the initial cooling of the plasma periphery triggers a very rapid growth of low-order tearing modes, resulting in a stochastic region over much of the plasma. This allows rapid transport across the entire plasma, and could explain the effectiveness of gas jet mitigation in C-Mod and DIII-D, and presumably in ITER, in spite of the shallow penetration of the neutral gas jet. In DIII-D the onset time of the thermal quench is observed to increase monotonically with increased $q = 2$ surface depth, strongly suggesting that a 2/1 instability is involved, in agreement with the NIMROD modeling. Although runaway electrons are not observed in the gas jet experiments on C-Mod or DIII-D, the quantity of gas atoms injected to date is insufficient to avoid a runaway avalanche on ITER. A new, larger valve ($\sim 25\times$ more throughput) will be used soon on DIII-D to test collisional avalanche suppression.

EX/4-4 · Influence of plasma opacity on current decay after disruptions in tokamaks

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Abstract: Current decays after disruptions as well as noble gas injections in tokamak are examined. KPRAD model of [D.G. Whyte, T.C. Jernigan, D.A. Humphreys, A.W. Hyatt et al., Journal of Nuclear Materials 313–316 (2003) 1239] is used as a basic one. In addition, opacity effects are included in accordance. Zero dimensional problem is analyzed. The thermal balance determined by Ohmic heating and radiative losses is supposed. As it is shown, the cooled plasmas at the stage of current decay are opaque for radiation in lines giving the main impact into total thermal losses. Impurity distribution over ionization states is calculated from the time-dependent set of equations. Opacity effects are found to be important for all cases examined. The opacity effects are found to be most important for simulation of JET disruption experiments with beryllium seeded plasmas. Using the coronal model for radiation one can find jumps in temperature and extremely short decay times. If one take into account opacity effects, current calculated decays smoothly in agreement with JET experiments. The decay times are also close to the experimental values. Current decay in argon seeded and carbon seeded plasmas for ITER parameters are simulated. The temperature after thermal quench is shown to be twice higher in comparison with the coronal model.

EX/4-5Ra · Study of erosion products in experiments simulating ELMs and disruptions in ITER on plasma gun QSPA-facility

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Abstract: The experimental date on structure of deposited material from eroded CFC and W macrobrush targets under ITER ELMs and disruptions heat loads are presented. Experiments were realized in QSPA plasma gun facility (TRINITI, Troitsk). CFC and tungsten macrobrush divertor plates were manufactured in EU according to the ITER divertor target specifications. Analysis of the deposits of CFC and W eroded determine the transient load magnitudes dependence of the deposits characteristics and determine correlation of the deposit characteristics with the dominant mechanism for target damage. Macrobrush targets were exposed to a large number of repetitive pulse (up to 100) of QSPA plasma gun with heat loads 0.5–1.5 MJ/m^2 and 0.5 msec time duration. Scanning electron and tunnelling microscopes were used as basic diagnostics for films and dust analysis with spatial resolution up to 2 nm. Exposure of CFC target gives flake-like deposits on the sample surface, but the amount of dust particles was negligible small. Surface of deposits consist of the globules structured with the scale 150–300 nm. The same structure detected in nano scale range ($d \sim 30$ nm). Dates of correlation analysis are presented. Erosion of W target gives much more dust particles. Many of W dust have sphere-like, cauliflower shape. Dimensions of particles vary from 0.2 to 3 microns. Some zones of samples were cover by complex agglomerated clusters (2–3 micron in diameter and 8–15 micron longwise) with fractal structure of the surface. The distribution of particles sizes and parameters of their distribution, obtained in the QSPA experiments will be discussed in the paper. Fractal dimension of W dust are presented.

EX/4-5Rb · Material Damage Characterisation of Divertor Targets Exposed to ITER-Relevant Type I ELM and Disruption Transient Loads

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Abstract: To evaluate the response of carbon and tungsten based materials to repeated intense thermal wall loads in the MJ/m^2-range, a collaborative European-Russian research programme has been initiated. These activities include the modelling of the transient events using specific 2- and 3-dimensional numerical codes to predict the material erosion and damage under intense plasma impact. Beside sublimation, in particular macroscopic material erosion due to brittle destruction processes and melt motion have been implemented. To validate these codes, test specimens which represent the divertor targets in ITER have been exposed to repetitive ELM and disruption specific thermal loads by electron beams and by plasma impact. To converge the conditions in ITER, medium scale target plates with ITER-grade wall armour materials (3-directional carbon fibre composite NB31, pure and La_2O_3-doped deformed tungsten with a grain orientation perpendicular to the target surface) have been produced according the ITER standards. These target plates have been exposed to about 100 plasma discharges in the Russian plasma gun facility QSPA in the Troitsk Institute for Innovation and Fusion Research. To approximate the ELM conditions in ITER, tests have been performed at different power density levels from 0.5 to 1.5 MJ/m^2; the pulse

duration was 500 μs. During these experiments the samples were preheated to a temperature of 500°C to make sure that the metallic wall armour is clearly above the ductile-brittle-transition-temperature (DBTT); the target was tilted at an angle of 30° to guarantee a flat incidence angle of the plasma stream. Before and after plasma exposure the target plates have been subject to a thorough characterization using laser profilometry, optical and electron microscopy; in addition, a detailed metallographic examination has been performed to evaluate the melt layer thickness and the depth of thermally induced cracks. These analyses have revealed intense crack formation on all tungsten targets, even on those which have been exposed to thermal loads clearly below the melting threshold (i.e. at E = 0.2 MJ/m^2). On CFC targets the thermally induced material erosion showed a clear correlation with the microstructure: an enhanced erosion of the PAN fibres while the pitch fibres remain almost unaffected due to their high thermal conductivity.

EX/4-5Rc · Modelling of Material Damage of CFC and W Macro-Brush Divertor Targets under ELMs and Disruptions at Plasma Gun Facilities and Prediction for ITER

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Abstract: Operation of ITER at high fusion gain is assumed to be the H-mode. A characteristic feature of this regime is the transient release of energy from the confined plasma onto plasma facing components (PFCs) by multiple ELMs, which can play a determining role in lifetime of PFCs as well as the transient power fluxes during disruptions. The expected energy fluxes on the ITER divertor during transients are: Type I ELM of 0.5–4 MJ/m^2 during 300–600 μs, and disruptions of 2–13 MJ/m^2 during 1–3 ms. CFC and tungsten macrobrush armour are foreseen as PFC for ITER divertor. During the intense transient events in ITER the evaporation, surface melting and melt splashing (W) are seen as the main mechanisms of PFC erosion. Due to rather different heat conductivities of CFC fibers, a noticeable erosion of the PAN fibers may occur at a rather small heat loads at which the damage to the tungsten armour is not substantial. The expected erosion of the ITER plasma facing components under transient energy loads can be properly estimated by numerical simulations validated against target erosion experiments at the plasma gun facility QSPA-T, in which components manufactured according to the EU specifications for the ITER divertor targets have been tested, and at the plasma gun facilities MK-200UG and QSPA-Kh50.The measured material erosion data have been used to validate the codes, PEGASUS, MEMOS, PHERMOBRID and FOREV-2, which are then applied to model the erosion of the divertor and main chamber ITER PFCs under the expected loads in ITER. Numerical simulations performed for the expected ITER-like loads demonstrated: a significant erosion of the CFC target is expected for ITER-like loads in excess of ~0.5 MJ/m^2; the W macrobrush structure is effective in preventing gross melt layer displacement during ELM-like loads leading to the overall erosion of W. Optimisation of macrobrush geometry in order to minimize erosion under ITER-like transient loads has been carried out. The erosion of the dome armour and the gaps between the divertor cassettes under the radiation from the plasma shield is numerically investigated for ITER disruption conditions. Different mechanisms of melt splashing are analyzed and implemented in the code MEMOS. For estimation of the W crack formation an analytical model that links the crack depth with the characteristic size of the crack mesh is proposed.

EX/4-5Rd · Modelling of ITER Edge Plasma Dynamics Following Type I ELMs and Consequences for Tokamak Operation

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Abstract: The reference regime for ITER is the H-mode in which the stationary operation will be repetitively interrupted by outbreaks of the ELMs. At each ELM the lost DT plasma produces surface damages and following contamination of the confined plasma. The transient energy release concerns also the poloidal magnetic field that undergoes substantial changes drastically broadening the footprint of the power flux. Thus computer modelling for ITER implies enhanced plasma transport across both the magnetic surfaces and the field lines ending at the vessel walls, multi-species plasma and neutral inflow in real toroidal configuration, account of poloidal field coils and plasma currents, line radiation losses, and injections of noble gases mitigating the disruptions. In order to predict the expected fluxes and consequent contamination of the vessel, a collaboration research has been developed in frame of fusion programme of EU and RF. The FZK (Karlsruhe, Germany) performs numerical modelling for the high energy fluxes relevant to ITER transient events and the TRINITI (Troitsk, Russia) experiments with a powerful plasma gun aiming at validation the codes. The paper will describe the physics basis employed to model the transient loads in ITER and summarise the main results of the experimental and modelling studies. The carbon fibre

composite (CFC) and tungsten targets are investigated. The computer modelling revealed that significant impurity contamination of the edge plasma can occur, which can cause the collapse of the confinement at lesser ELM sizes than that determined by armour lifetime limitations. At the plasma gun facility MK–200 UG, experiments were carried out on the interaction of hot magnetized plasma streams with CFC and W targets. Plasma impact was investigated at the loads relevant to hard disruptions and Type I ELMs of ITER. First experiments aimed at impurity generation were performed in order to quantify the evaporation thresholds of W and CFC. Radiation properties of evaporated carbon were studied. First simulations showed that ELMs do not clean the plasma from impurities. The tolerable ELM size was estimated as 1 MJ/m^2 for ELM frequency of 0.5 Hz. Latest modelling results on the energy distribution over the vessel surface during ELM and post-ELM stages will be presented, accounting also for neutral influxes into the confinement regions.

EX/5-1 · Enhanced H-mode pedestal and energy confinement by reduction of toroidal field ripple in JT-60U

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Abstract: The pedestal pressure and energy confinement of H-mode plasmas were improved by the installation of ferritic steel tile (FST) in JT-60U. As the fast ion loss decreased, the H-mode pedestal temperature (and pressure) increased by ∼20–25% and the H-factor is increased by ∼ 20%. The ratio of ELM frequency to the power crossing the separatrix tends to become lower accompanied by larger type-I ELM spike. The experiments have shown that the energy confinement can be improved with the increase in the pedestal pressure accompanied by decreasing the fast ions loss ratio and/or enhancing the toroidal rotation in the co-direction. In JT-60U, the FST was installed in the vacuum vessel to reduce the toroidal field (TF) ripple ratio in 2005. In this study, comparing between the plasmas with and without FST, the effects of the fast ions loss and the toroidal rotation on the pedestal and core confinement properties are examined. At the identical plasma configuration between the discharges with and without FST, the pedestal temperature in case with FST can be raised by ∼20–25% compared to the case without FST at a given pedestal density. Larger type-I ELM spikes appear due to the increase in the pedestal pressure in case with FST. The ELM frequency normalized by the power crossing the sepratrix becomes relatively lower at a given line-averaged electron density. At a given absorbed power and pedestal density, the ion temperature profile becomes higher by ∼20% over the whole range of plasma core in case with FST. The stored energy also becomes larger by ∼20%. The ion temperature gradient in the ETB region becomes steeper by ∼5% and the pedestal width becomes wider by ∼10% with decreasing the fast ion loss fraction by ∼20%. After the FST installation, the toroidal rotation profile is shifted to the co-direction over the whole radial range including the plasma edge. The energy confinement can be improved as the fast ion loss fraction decreases. However, the toroidal rotation shifts to the co-direction monotonically together with the reduction of the fast ion loss. Since the radial electric field is influenced by the fast ion loss and toroidal rotation, the role of the toroidal rotation on the pedestal structure distinguished from that of fast ion loss should be analyzed in the next step study.

EX/5-2 · Reduction of Neoclassical Transport and Observation of a Fast Electron Driven Instability with Quasisymmetry in HSX

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Abstract: The Helically Symmetric Experiment (HSX) is the first operational quasisymmetric stellarator, with a helical direction of symmetry in the magnetic field strength. As a result of this symmetry, neoclassical transport at low collisionality is predicted to be greatly reduced compared to a conventional stellarator. Here we report experimental differences in the density and temperature profiles between the quasihelically symmetric (QHS) configuration and a configuration with the symmetry intentionally broken. The central electron temperature in the QHS configuration is significantly higher than in the non-symmetric configuration (∼450 vs. ∼250 eV). The density profile in the QHS configuration is always centrally peaked, regardless of heating location, whereas in the non-symmetric configuration the core density profile is flattened with central heating. Transport analysis shows that this flattening is due to neoclassical thermodiffusion, which is reduced in the QHS configuration. Measurements of the particle source rate and absorbed power confirm that the profile differences are due to the improved transport properties in plasmas with quasisymmetry. An unexpected consequence of the improved confinement in the QHS configuration is that a fast particle population drives Alfvénic instabilities. We report on the first experimental observations of fast electron

driven Alfvénic modes in QHS plasmas. These modes are coherent and global, peaking near the plasma core. The measured frequency range, propagation direction, and scaling with ion mass density are consistent with theoretical predictions for the Global Alfvén Eigenmode. When symmetry is broken, the modes are no longer observed.

EX/5-3 · Core Electron-Root Confinement (CERC) in Helical Plasmas

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Abstract: The core heat confinement improvement has been realized in helical devices with central ECH power deposition. Peaked electron temperature profiles and strongly positive radial electric fields ($E_{\rm r}$) in the core region have been commonly observed in CHS, LHD, TJ-II and W7-AS. The positive $E_{\rm r}$ is required within neoclassical (NC) theory to satisfy the ambipolarity condition. This "electron-root" feature exists only in low-collisionality helical plasmas. ECH power thresholds for this improvement have been recognized associated with this transition of $E_{\rm r}$. Core electron heat diffusivity is reduced significantly compared to its NC level with $E_{\rm r} = 0$. This improved core confinement, referred collectively as "Core Electron-Root Confinement (CERC)", has signatures quite different than those of tokamak ITBs. The contribution of the convective electron flux driven by strong ECH has also been revealed, illustrating the configuration dependence of the ECH power threshold. The important role of low-order rational surfaces/magnetic islands for CERC has been recognized in LHD and TJ-II. An external field is applied in LHD to examine the effect of the 2/1 island. The 2/1 island is found to promote CERC formation, in a sense, by reducing the ECH power threshold. The rotational transform profile control in TJ-II has found that the order of the rational surfaces influences the density threshold. Sheared poloidal rotation at the boundary of the magnetic island, as observed in LHD, may be responsible for this favorable effect. The accumulation of CERC discharges and related theoretical analyses has led to the initiation of "International Stellarator Profile DataBase". The comparison of collisionality regime for CERC in four devices has revealed that CERC is established at lower collisionality in CHS and LHD than those for TJ-II and W7-AS. It has been recognized that the larger the effective helicity (level of the ripple transport) the higher the maximum collisionality for CERC. This trend can be understood by the change of the ripple-transport level providing the bifurcation nature of $E_{\rm r}$. It has also been confirmed that larger convective electron flux and higher ECH power density are reasons for CERC in higher collisionality in W7-AS.

EX/5-4 · Gyrokinetic Theory and Simulation of Zonal Flows and Turbulence in Helical Systems

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Abstract: Gyrokinetic theory and simulation of plasma turbulence in magnetic confinement fusion have advanced physical understandings on the anomalous transport mechanism as well as the zonal-flow dynamics. Damping processes of zonal flows in ion temperature gradient (ITG) turbulence in helical systems are analytically investigated based on the gyrokinetic theory as a generalization of the previous work by Rosenbluth and Hinton for tokamaks. It is elucidated how the zonal-flow dynamics are influenced by magnetic geometry and vary between the ITG and ETG (electron temperature gradient) turbulence cases. A complete response of the zonal flow to given source terms is derived by taking account of the helical geometry and finite-orbit-width effects, and is used to construct a new closure model. Analytical predictions about geometrical effects on the geodesic acoustic mode (GAM) dispersion relation, the residual zonal flows, and the velocity-space structure of the distribution function are also verified by our gyrokinetic simulations with very-high resolution of the phase space. The recently developed gyrokinetic simulation code (GKV code) can properly reproduce fine velocity-space structures of the ion gyrocenter distribution function associated with the turbulence. The GKV simulation is extended to take account of helically-trapped particles, and is also applied to the ITG turbulence in helical systems in order to elucidate how the helical geometry can be optimized to enhance the residual zonal flow level and accordingly reduce the anomalous transport.

EX/5-5Ra · Configuration Control Studies of Heliotron J

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Abstract: Results of Heliotron J experimental progress during 2005–2006 are presented. Highlights are the configuration control studies of Heliotron J (a low shear helical-axis heliotron having 4 periods, $\iota(\mathrm{a})/2\pi = 0.3-0.8$, R $\sim$ 1.2 m, a = 0.1–0.2 m and $\mathrm{B_0} \leq 1.5$ T) with special reference to thermal confinement. The thermal confinement has been studied with an emphasis on the confinement improvement by the bumpy field which should play a key role in the neoclassical optimization of the helical-axis heliotron. In order to extract the key physics ingredient linked with drift optimization for the improvement of the core and/or edge transport in L- and H-modes, measurements of the enhancement factor (H_{ISS04}) of the global energy confinement time (τ_E^{exp}) with regard to the recent international stellarator scaling law (τ_E^{ISS04}) have been made for 0.3 MW, 70 GHz ECH plasmas by changing the bumpiness (ϵ_b) under almost the same average magnetic axis position (R_{ax}), minor plasma radius (a) and edge rotational transform ($\iota(a)/2\pi$) conditions. It has been found that the reduction of the effective helical ripple, ϵ_{eff}, produces a desired effect on the improvement of H_{ISS04} not only in L-mode but also in H-mode. The maximum H_{ISS04} has reached about 1.5 in the H-mode for the lowest ϵ_{eff} configuration (medium bumpy). As a consequence, these data indicate that the lower ϵ_{eff} helical-axis heliotron can be a promising solution to the design problem of developing an advanced heliotron line. Contrary to the thermal confinement, it should be commented here that the fast ion confinement in NBI/ICRF heating has been found to be enhanced with an increase in bumpiness. To determine what effect in the thermal confinement – ignored here, e.g. electric field, particle loss cone, turbulent flow shear, etc. – makes up this apparent contradiction would shed an interesting light on the different properties of thermal and fast ion confinement in the helical-axis heliotron.

EX/5-5Rb · Progress of Confinement Physics Study in Compact Helical System

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Abstract: Recent experimental results in CHS on the transport barrier formations and the related turbulence studies are reported. In H-mode experiments, the dependence of the power threshold on the magnetic configuration was studied making a full use of the configuration flexibility of CHS. Operational range of H-mode has been extended to the high magnetic field and high density ($1 \times 10^{20} m^{-3}$) plasmas where the combination of the H-mode and the reheat mode was observed producing the highest energy discharges in CHS. In the discharges with the internal transport barrier (ITB), improvement of the ion confinement was observed as well as the electrons. An increase of ion temperature gradient was measured by a new diagnostic technique of dynamical spatial scan of the charge exchange spectroscopy (CXS). It was confirmed that the radial locations of barriers for ions and electrons are clearly different. For ITB discharges, suppression of the local turbulence due to the barrier formation was measured by two sets of heavy ion beam probes (HIBP). Changes of fluctuation frequency spectra were obtained for before and after the transition at the electron barrier location. It was found that mainly high frequency components (>30 kHz) were reduced for ITB while the lower frequency components remained at a similar level. The reduction of the spatial coherence of fluctuations was also confirmed in the ITB formation.

EX/5-6 · Impact of Nonlocal Electron Heat Transport on the High Temperature Plasmas of LHD

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Abstract: Edge cooling experiments in the Large Helical Device show a significant rise of core electron temperature (the maximum rise is around 1 keV) as well as in many tokamaks. This indicates the possible presence of the nonlocality of electron heat transport in plasmas where turbulence as a cause of anomalous transport is dominated. The nonlocal electron temperature rise in the LHD takes place in almost the same parametric domain (e.g. in a low density) as in the tokamaks. Meanwhile, the LHD experiment shows some new aspects of nonlocal temperature rise, for example the delay of the nonlocal rise of core electron temperature increases with the increase in electron density.

EX/6-1 · Fast Particle Physics on ASDEX Upgrade

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Abstract: ASDEX Upgrade has broad capabilities to produce and to diagnose fast particles. The flexible heating system (up to 20 MW NBI at 60/100 keV; up to 6 MW ICRH, up to 2 MW ECRH) allows to partially decouple the effects of bulk plasma heating from the fast ion population. A fast-ion detector, mounted on a movable manipulator at the low-field side mid-plane, gives energy and pitch-angle resolved distributions of the lost particles with a time resolution approaching their toroidal transit time (1 MHz). Extensions to our ICRH system allow us now to operate it stably at two close-by frequencies. They can be swept during a discharge, making the system ideally suited for beat wave excitation of Alfvén-type waves.

On the modelling side we have developed a non-perturbative, fully gyrokinetic, linear stability code using exact particle orbits (LIGKA), and have extended a gyrokinetic turbulence code (GENE) to include fast ions as a third particle species.

EX/6-2 · Confinement Degradation of Energetic Ions due to Alfvén Eigenmodes in JT-60U Negative-Ion-Based Neutral Beam Injection Plasmas

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Abstract: Confinement degradation of energetic ions due to Alfvén Eigenmodes (AEs) induced by negative-ion-based neutral beam injection for the classical confinement is quantitatively evaluated for the first time. AEs, whose frequency rapidly sweeps and then saturates as the minimum value of the safety factor decrease, have been observed in JT-60U. These mode behavior can be explained by reversed-shear induced AE (RSAE) and the transition from RSAEs to TAEs. Measured total neutron emission rate (Sn) in the presence of these AEs is compared with that predicted by classical theory. As a result, confinement degradation of energetic ions is confirmed, especially, it is largest in the transition phase from RSAEs to TAEs, where the maximum reduction rate for the classical confinement is estimated as $(\Delta\mathrm{Sn}/\mathrm{Sn})_{\mathrm{max}} \sim 45\%$. Line-integrated neutron emission profile is also compared with that predicted when assuming that the confinement is classical. The result indicates energetic ions are transported from core region of the plasma due to these AEs.

EX/6-3 · Alfvén Instabilities in DIII-D: Fluctuation Profiles, Thermal-Ion Excitation, and Fast-Ion Transport

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Abstract: To understand and predict alpha-particle driven instabilities and the resultant alpha-particle transport in ITER, it is essential to measure fast-ion transport arising from Alfvén instabilities in present day experiments. In DIII-D, several diagnostic systems have improved sensitivity and bandwidth for the detection of Alfvén modes, including the interferometer, reflectometer, beam-emission spectroscopy, and electron cyclotron emission diagnostics. A new technique, fast-ion D_α (FIDA) spectroscopy, measures the spatial fast-ion profile. These diagnostics are applied to the toroidicity-induced Alfvén eigenmodes (TAE), reversed-shear Alfvén eigenmodes (RSAE), and compressional Alfvén eigenmodes (CAE) that are driven by deuterium neutral beam ions in DIII-D. FIDA measurements indicate large reductions in the fast-ion density at the center of the plasma during Alfvén activity. The measured fluctuations indicate spatial localization of RSAEs near the minimum q surface, as theoretically expected. Unstable toroidal mode numbers as large as $n = 40$ are inferred with measured poloidal wavenumbers on the thermal ion gyroradius scale. Calculations indicate the high n modes are excited by the thermal ion population.
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EX/6-4 · Off-axis Current Drive and Current Profile Control in JT-60U

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Abstract: For the first time, we have measured the current density profile for off-axis neutral beam current drive (NBCD), using motional Stark effect (MSE) diagnostic. A spatially localized NBCD profile was clearly observed at r/a = 0.6–0.8. The location was also confirmed by multi-chordal neutron emission profile measurement. The total amount of the driven current (0.15 MA) was consistent with the decrease in the surface loop voltage. The off-axis current drive can raise safety factor (q) in the center and help to avoid instability that limits performance of the plasma. We have developed a real-time control system of the minimum q ($\mathrm{q}_{\mathrm{min}}$), using the off-axis current drive. Injection power of lower hybrid (LH) waves, and hence, its off-axis driven current controls qmin. In a high b plasma ($\beta_\mathrm{N} = 1.7$, $\beta_\mathrm{p} = 1.5$), the system was adopted to control $\mathrm{q}_{\mathrm{min}}$. With the control, qmin was raised and MHD fluctuations were suppressed. The stored energy increased by 16% with the MHD fluctuations suppressed.

EX/7-1Ra · Active Control of Resistive Wall Modes in High Beta, Low Rotation DIII-D Plasmas

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Abstract: Active control of the $n = 1$ resistive wall mode (RWM) with rotation below the critical value for stabilization has opened access to new regimes of high performance in recent DIII-D experiments. Very high plasma pressure combined with elevated $q_{\min}$ for high bootstrap current fraction, and internal transport barriers (ITBs), for high energy confinement, are sustained for almost 2 s, or 10 energy confinement times. In these high beta discharges the rotation threshold for RWM stabilization is higher at $q_{\min} > 2$, and the plasma rotation obtained with optimal error field correction is only marginal for rotational RWM stabilization. (This may be the same BOUT regime in ITER steady-state scenario for which the predicted values of the plasma rotation and of the rotation threshold for RWM stabilization are close.) Therefore, the improvement in attained plasma pressure compared to previous plasmas with ITBs may be explained by stabilization of the RWM using simultaneous feedback control of the two sets of non-axisymmetric coils in DIII-D. High gain, slow feedback control of the external coils is used for dynamic error field correction so as to maintain high levels of plasma rotation for rotational stabilization of the RWM; low gain, fast feedback control of the internal non-axisymmetric coils, powered by high bandwidth audio amplifiers, provides direct RWM stabilization during transient periods of low rotation, e.g. following a large edge localized mode. Stability modeling shows that the ideal MHD, ideal-wall limits for low-n kink modes increase with $q_{\min}$ increasing above 2. These beta limits, which are consistent with the experimental results, indicate the possibility to operate at plasma beta of ~6% even in the presence of an ITB, as long as $q_{\min} > 2$. The new DIII-D capability of near-balanced beam injection in 2006 will allow more systematic tests of RWM feedback control in plasmas with near-zero rotation, for extrapolations to the more pessimistic predictions for the ITER steady-state scenario, where the plasma rotation is insufficient for RWM stabilization.
* Work supported by US DOE under DE-FG02-89ER53297, DE-FC02-04ER54698, DE-FG03-01ER54615, DE-FG03-99ER82791, W-7405-ENG-48, and DE-AC02-76CH03073.

EX/7-1Rb · Plasma Rotation and Wall effects on Resistive Wall Mode in JT-60U

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Abstract: This paper reports the exploration of the RWM onset using co-, near zero, counter plasma rotation profiles in JT-60U. This is the first experimental result which demonstrates the dependence of critical RWM onset of plasma rotation and beta on initial plasma rotation with the variation of the angular momentum input and without magnetic braking. In the JT-60U with newly ferretic wall installed, it is possible to produce high beta plasma tightly coupled with the wall ($b/a \approx 1.2$) above $\beta_{\rm N} > \beta_{\rm N.no-walllimit}$. In near-zero plasma rotation, the RWM started to grow at $\beta_{\rm N} \approx \beta_{\rm N.no-walllimit}$ and with finite plasma rotation, the plasma pressure survives up to much higher $\beta_{\rm N}$ level than that with small rotation. The observed critical beta onset $\beta_{\rm c}$ and the RWM growth rate $\gamma_{\rm RWM}$ are discussed along with theoretical predictions using experimentally observed q-, pressure-, and rotation profiles.

EX/7-2Ra · MHD Studies in MAST

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Abstract: The tight aspect ratio tokamak, with its high-beta capability, allows for an increased understanding of MHD stability, including of course aspect ratio scaling effects. The MAST tokamak is equipped with neutral beam heating, a range of excellent diagnostics and external error field corrections coils that allow non-axisymmetric (dominantly n = 1) fields to be applied. MAST has made considerable progress in studying sawtooth behaviour with co and counter-NBI, in observations of fast ion instabilities and on studies of the threshold for error field locked modes, as reported in this paper. Control of sawteeth, vital for achieving high performance, has been investigated by looking at the effects of various levels of co and counter-current NBI in otherwise matched conditions. Stability calculations have been extended to take into account the effects of plasma rotation and ion diamagnetic drifts. The minimum sawtooth period is found for weak counter-current NBI at the point where the sawtooth precursor frequency changes sign and the NBI induced rotation balances the intrinsic mode rotation. The radius of the q = 1 surface propagates between sawtooth crashes and modelling has indicated that the minimum marginally stable q = 1 radius occurs at the toroidal rotation rate when the experimental sawtooth period is also minimised. The

high current, low field environment of MAST provides ideal opportunities for looking at the effects of well confined, super-Alfvénic fast ions upon plasma stability. Nonlinear theory and simulations have been applied to frequency sweeping global Alfvén eigenmodes in MAST, validating the link between the rate of frequency sweep and the mode amplitude, and opening up diagnostic opportunities in environments where high frequency magnetic probes are precluded. Studies with the non-axisymmetric error field correction coils have confirmed conventional aspect ratio scalings, increasing confidence in ITER predictions. New modelling studies also show how MAST accesses the rotationally stabilised regime that governs the beta limit, enabling it to probe this physics and address critical questions for the advanced tokamak operation of ITER and an ST power plant.
* This work was partly supported by the UK EPSRC and by the European Communities under the contract of Association between EURATOM and UKAEA.

EX/7-2Rb · Investigation of Resistive Wall Mode Stabilization Physics in High Beta Plasmas Using Applied Non-axisymmetric Fields in NSTX

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Abstract: NSTX offers an operational space characterized by high-beta (beta toroidal = 39%, beta normal > 7) and low aspect ratio ($A > 1.27$) to leverage the plasma parameter dependences of RWM stabilization physics and plasma rotation damping. NSTX has added significant new capability to study and diagnose the RWM with the commissioning of a set of six non-axisymmetric magnetic field coils, allowing generation of fields with dominant toroidal mode number, n, of 1–3. First tests of this system for dynamic error field correction and mode control are planned for the 2006 run campaign. Recent NSTX research has focused on uncovering details of passive RWM stabilization. It was found previously that maintaining the entire toroidal rotation profile normalized to the Alfvén speed above $1/4\,q^2$ led to sustained wall-stabilized operation. Modification of stabilized rotation profiles using applied non-axisymmetric fields has since been performed, showing that rotation outside $q = 2.5$ is not required for passive RWM stability. NSTX has shown that rotation is strongly and globally damped by neoclassical toroidal viscous (NTV) forces. Plasma rotation damping is measured and allows quantitative comparison to NTV theory, applicable for all aspect ratios and for the first time including the full NTV formulation and non-axisymmetric field (δB) spectrum for both the plateau and low collisionality regimes, which scale as $\delta B^2 T_i^{0.5}$ and $\delta B^2 (T_i/\nu_i)(1/A)^{1.5}$ respectively. Quantitative agreement to theory within a factor of two requires the inclusion of trapped particle effects. The plasma response to rotating external perturbations show a peak in the resonant field amplification (RFA) magnitude (plasma response/applied field measured at the sensor location) when the perturbation rotates at approximately 30 Hz in the direction of plasma flow, consistent with drag on the mode by the rotating plasma. Observations also indicate that rotation braking near the plasma edge due to RFA is responsible for an increase in core ion pressure which results in a sudden rise in beta, increasing the RFA and the associated rotation damping. This process results in destabilization of the RWM, but often triggers an internal kink/ballooning mode which reduces beta, re-stabilizing the RWM.
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EX/7-3 · Overview of RFX-mod results with active MHD control

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Abstract: Plasma experiments resumed in December 2004 on RFX-mod. The machine now has a thin (3mm) Cu shell with one overlapped poloidal gap and one toroidal gap. Shell penetration time for Bv has been lowered from 450 to 50 ms and shell/plasma proximity from b/a = 1.24 to 1.1. Toroidal equilibrium is feedback-controlled and new power supplies provide a better control of the toroidal field. Newly designed graphite tiles protect the vessel from highly localized power deposition. The MHD Control System, MHD-CS, a set of 192 external saddle coils controlled by a digital feedback system, is used to control radial fields due to field errors, MHD modes and Resistive Wall Modes (RWMs). A dramatic improvement of plasma performance was obtained by using the MHD-CS to cancel all of the radial field components, an operational mode dubbed Virtual Shell (VS). The toroidal loop voltage was lowered by more than 40% and the plasma pulse duration tripled. In practice, steady state RFP pulses are now limited only by the applied volt-seconds. Hence RFX-mod initial operation demonstrated the possibility to operate a large RFP without a thick conducting shell, and opened enhanced RFP scenarios. Indeed the improved magnetic boundary in VS mode, which mimics an ideal closely fitting shell, has a stabilizing effect on the tearing modes underlying the sustainment of the RFP configuration, the so-called dynamo modes, which

are also responsible for field line stochastization in the plasma core and confinement limitation. With the VS the amplitude of such modes in the plasma centre was nearly halved. As expected, this led to improved particle and energy confinement. For instance, peak electron temperature in reference pulses at 600 kA was doubled (from 200 to 400 eV) with more peaked profiles, which corresponded to a reduction of the thermal conductivity by a factor >2 in the region r/a < 0.9. The MHD-CS is extremely flexible and can be used for a variety of mode control experiments. The most important result already obtained was the demonstration of the active control of RWMs. We found that full VS control completely inhibits the growth of RWMs, whereas such modes are indeed seen to grow in agreement with the theoretical prediction if the MHD-CS operated in Selective VS mode, i.e. leaving one or more mode helicity uncontrolled.

EX/7-4Ra · Integrated modelling of sawtooth oscillations in tokamak plasmas

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Abstract: The present understanding of sawtooth oscillations in tokamak plasmas is discussed. Predictive sawtooth modelling requires the integration of various aspects of tokamak physics, including transport, heating and current drive, MHD stability, kinetic effects and fast particle dynamics. In particular, fast particle distributions must be obtained self-consistently, taking into account auxiliary heating, fusion-produced alpha particles and the redistribution of fast ions due to sawtooth crashes and the relevant fast particle driven instabilities. Concerning the nonlinear evolution, a partial sawtototh reconnection model is proposed, assuming that full reconnection may be prevented by the onset of secondary instabilities. An exploration of secondary ballooning instabilities based on the XTOR and NIMROD codes will be presented. The model proposed here is shown to be in fair quantitative agreement with experimental data and to provide a useful tool for the definition of a possible strategy for sawtooth control in fusion burning plasmas.

EX/7-4Rb · Nonlinear Simulations of Fishbone Instability and Sawteeth in Tokamaks and Spherical Torus

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Abstract: Understanding of nonlinear behavior of m = 1 mode in tokamaks is of fundamental importance for burning plasmas. Here we report new results of self-consistent nonlinear simulations of fishbone instability and sawtooth oscillations obtained using the extended MHD code M3D. Hybrid simulations of energetic particle-driven fishbone instability in a circular tokamak show dynamic mode saturation as the particle distribution is flattened and mode frequency is reduced strongly. MHD nonlinearity reduces the mode saturation level. Resistive MHD simulations of the CDX-U spherical torus show repeated sawtooth cycles with calculated periodicity consistent with the experimental observation. The two-fluids simulations show that the induced sheared plasma rotation suppresses the island growth so that the magnetic field no longer passes through a stochastic state as observed in the MHD simulations. Resistive MHD simulations of sawtooth crashes in the TEXTOR tokamak show that the localization effect of reconnection is rather small. Even when the pressure in the core is flattened due to stochasticity in a nonlinear state, the complete reconnection process still tends to proceed with q(0) rises above one.

EX/7-5 · Stability in High-Beta Plasmas of LHD

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Abstract: In Large Helical Device (LHD), the volume averaged beta value of 4.5%, which is the highest value in all of heliotron/stellarators, was achieved by optimizing the magnetic configration. In this study, characteristics of MHD activities in extended beta range over 4% and the configuration dependence have been investigated. The dominant MHD modes moved from inner region to outer one when beta increases, and the mode excited in the outermost resonance near plasma edge are enhanced in the beta range over 4%. The dependence of saturated amplitude of the mode on magnetic Reynolds number is close to that of linear growth rate of resistive interhcnage mode. When the magnetic shear was decreased and the plasma is close to m/n = 1/1 ideal stability boundary, the m/n = 1/1 mode suddenly growed and led to a minor collapse in the core region. The results suggest the significance of magnetic shear and a validity of a linear thory on ideal interchange mode.

EX/8-1 · Super Dense Core Plasma due to Internal Diffusion Barrier in LHD

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Abstract: A Super Dense Core (SDC) plasma operational regime has been discovered in the LID (Local Island divertor) discharges in LHD. The observed new confinement improvement regime is very attractive to future helical device. The SDC plasma has been established naturally during the decay phase of a peaked high density profile, generated by multiple pellet injection. A core region with density as high as $\sim 3 \times 10^{20}\,\mathrm{m}^{-3}$ and temperature of ~ 0.8 keV is maintained by an Internal Diffusion Barrier (IDB), of which foot is located at the normalized minor radius of 0.6. The density gradient in the IDB is very high and the particle confinement in the core is about 200 ms. The temperature profile is flat in the core. The temperature gradient in the outer region beyond the IDB is similar to that in the outer region of the non-SDC discharge with similar plasma conditions. The SDC density profile appears only when the plasma particles are fueled by pellet injection. The maximum achieved stored energy is 980 kJ (at 14.7 MW of NBI), a factor of 2 higher than that of the gas puff LID discharge. The discharges with gas puffing, exhibit conventional density profiles, flat or slightly inverted density profile. In the LHD configuration, there exist two distinct regions, low shear core region and high shear outer region. The separation between two regions is fairly clear and its location varies with magnetic axis and beta. The IDB foot, at which density gradient suddenly increases appears to be located slightly outside (0–5 cm) of the separation. For the magnetic configuration with vacuum magnetic axis of 3.75 m and average beta of 1% (corresponding central value of 4.5%), the normalized radius of the IDB foot location is 0.6. It increases with increasing magnetic axis and increasing beta. For the outward shifted configuration with vacuum magnetic axis of 3.85 m with average beta of 1.2%, the IDB foot is located near the last closed surface. The density in the outer region, is kept low by pumping of the recycled particles by the LID. Even with Helical Divertor, however, similar SDC plasma is generated when the wall is well conditioned for the pumping. Low density in the outer region helps to raise the temperature gradient there and hence the core temperature.

EX/8-2 · Overview of Results in the MST Reversed-Field Pinch Experiment

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Abstract: In the general area of confinement improvement and concept advancement, recent results in the MST reversed-field pinch include 1) improved confinement at 550 kA plasma current, with copious energetic electrons and sustained elevated ion temperature greater than 1 keV; 2) raised density in improved-confinement plasmas from injection of frozen deuterium pellets; 3) sustainment of 10% of the plasma current by oscillating field current drive (ac helicity injection), in agreement with theory for the applied power; 4) wave injection from electron Bernstein wave and lower hybrid wave antennas, satisfying theoretical expectations of plasma loading at power levels of about 100 kW. Substantial progress has been made in understanding the causes and effects of magnetic reconnection in the RFP: 1) discovery that fast ions with large gyro-radii are well-confined in the standard RFP, despite the underlying stochasticity of the magnetic field; 2) determination that ion heating during reconnection events is global, as measured by charge-exchange recombination spectroscopy; 3) observation that global magnetic self-organization (characterized by sudden changes in magnetic energy, plasma momentum, and ion temperature) occurs only when spontaneous ($m = 1$) and driven ($m = 0$) reconnection are present simultaneously. Diagnostic system development continues, including charge-exchange recombination spectroscopy for fluctuation measurement, multi-point and multi-pulse Thomson scattering, spectral motional Stark effect measurement of low-field $|\,\mathbf{B}\,|$, and multi-color soft X-ray detector arrays for tomographic measurement of time-resolved electron temperature.

EX/8-3 · Progress in understanding anomalous impurity transport at JET

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Abstract: Progress in understanding of impurity transport at JET has been gained from a comparison of the experimentally determined transport with its predictions by theoretical models. It was observed in H-mode plasmas that Ni transport is dominated by turbulent transport and that the pinch velocity of Ni changes direction from strongly inward to slightly outwards when moving from a dominant ion heating to an increased electron heating. A mechanism of particle pinch connected with the parallel dynamics of impurities has been identified recently and the direction of such a pinch matches the pinch reversal observed experimentally. Progress in theoretical models allows theory to be tested at least quantitatively against experiments. The JET database of impurity injection provides results on the Z dependence of transport to validate the current theoretical understanding. Results on light (T), medium (Ne and Ar) and high Z in H-mode and hybrid discharges will be discussed in the light of theoretical predictions. Moreover, the transport of Ne, Ar and Ni is also compared in internal transport barrier discharges.

EX/8-4 · Peaked Density Profiles in Low Collisionality H-modes in JET, ASDEX Upgrade and TCV

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Abstract: Results from an extensive database analysis of JET and AUG density profiles in H-mode, show that the density peaking factor (central density divided by average density) increases to above 1.5 as the effective collisionality $\nu_{\mathrm{eff}} \sim R_0 n_e/T_e^2$ drops to near 0.1, as expected for ITER. On any single device density peaking is also strongly correlated with the Greenwald number N_G and the particle outward flux from the neutral beam source. The combination of the JET and AUG databases presented here has allowed to lift colinearities between ν_{eff}, $\mathrm{N_G}$ and the neutral beam particle source, identifying ν_{eff} as the statistically most relevant parameter for density peaking in H-modes. Correlations with $\mathrm{l_i}$, $\mathrm{q_{95}}$, β_{N}, $\rho^\star$, $\mathrm{L_{Te}}$, $\mathrm{L_{Ti}}$, the toroidal Mach number and its shear are weak. Scaling expressions using the combined database yield a prediction of a peaking factor near 1.5 in ITER, providing a boost of fusion power of 30% for fixed β and $\mathrm{N_G}$ with respect to the usual assumption of a flat density profile. Peaked density profiles may preempt the negative consequences on fusion power production in the case of a lower than expected density limit n ITER. The Ware pinch at low collisionality is too weak to sustain the observed density gradients. Neutral penetration code calculations and experiments in He plasmas show that the edge particle source also cannot account for density peaking, thereby experimentally establishing the anomalous nature of this phenomenon in H-modes. H-modes heated only by ICRH are on average slightly less peaked than H-modes dominated by NBI, demonstrating that neutral beam fuelling can explain a modest part (<20%) of the peaking. Density profiles tend to be slightly flatter at low values of $\mathrm{T_i/T_e}$, which is in qualitative agreement with drift wave turbulence predictions. A theoretical expectation for a reactor is that the strong core electron heating by slowing-down alpha particles may strongly destabilise TEM's, driving an outward flux, which may lead to flattening of the density profile. However purely electron heated H-modes with $\beta_{\mathrm{N}} \sim 2$ and $\mathrm{T_e/T_i} \sim 2$, obtained in TCV using 1.5 MW of ECH, show that flattening is only partial and significantly peaked density profiles (peaking factor ~1.6 in TCV) persist in electron heated regimes at reactor relevant values of β_N.

EX/8-5Ra · Microturbulence in Magnetic Fusion Devices: New Insights from Gyrokinetic Simulation and Theory

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Abstract: Gyrokinetic simulation of microturbulence in magnetic fusion devices is gradually reaching a level of maturity that allows for direct comparisons with experiments. Much of the work presented here is based on the nonlinear gyrokinetic turbulence code GENE which has been upgraded recently both in terms of physics and numerics. Using GENE simulations and complementing them by theoretical studies, we will address four critical issues in current plasma microturbulence research: 1) Electron thermal transport in plasmas with dominant electron heating; 2) Quenching of particle pinch effects by collisions; 3) Strong interaction of energetic particles with turbulent fluctuations; 4) Core turbulence in stellarators and its interplay with neoclassical transport.

EX/8-5Rb · Theoretical Understanding of Core Transport Phenomena in ASDEX Upgrade

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Abstract: The theoretical understanding of experimentally observed transport is a necessary step when assessing if regimes and performance achieved in present devices will also be obtained in burning plasmas. The present contribution reports the important progress obtained in this research field at ASDEX Upgrade over the last two years. The goal has been to enhance the physics understanding of observed core transport phenomena through the development and use of both fluid and gyrokinetic models and related numerical tools. The investigation of several transport channels simultaneously makes the comparison between theory predictions and experimental observations more stringent. Electron and ion heat transport, particle and impurity transport, as well as momentum transport and its relationship with heat transport are investigated. It is shown that low density electron heated plasmas in low confinement mode have a core electron heat and particle transport behaviour which agrees with the theoretical predictions for the trapped electron modes. Moreover, a physics mechanism for impurity pinch produced by compression of parallel velocity fluctuations, and reversing its direction depending on the direction of propagation of the turbulence, is identified for the first time. This pinch mechanism does not vanish for highly charged impurities and is the best candidate to account for the observed suppression of highly charged impurity accumulation in case of strong central electron heating. Gyrokinetic calculations and a quasi–linear model show that momentum transport can be written as the sum of a diffusive term, with a diffusion coefficient proportional to the ion heat conductivity, and a pinch term. Experimentally, a strong correlation is found between the radial gradient of the toroidal velocity and the logarithmic ion temperature gradient, the drive of the ion temperature gradient mode. This agrees with the theoretical expectation that the level of transport is reduced by shear flows. In contrast with theory, a very weak correlation is observed between the logarithmic temperature gradient and the temperature ratio. The implications of the acquired theoretical understanding in the extrapolation of present scenarios to burning plasmas are discussed.

EX/8-6 · Confinement and Local Transport in the National Spherical Torus Experiment (NSTX)

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Abstract: NSTX operates at low aspect ratio (R/a = 1.3) and high beta (up to 40%), allowing tests of global confinement and local transport properties that have been established from higher aspect ratio devices. NSTX plasmas are heated by up to 7 MW of deuterium neutral beams with preferential electron heating and with the electron channel dominating local transport losses as expected for ITER. New results on the role of the electrons in controlling global confinement, the characteristics of the electron transport, its relation to measured magnetic shear and high-k turbulence, and how it can be improved will be presented. Studies of energy confinement in H-modes found that these plasmas were not well described by parametric scalings developed for higher aspect ratios, showing a weaker current dependence and a significant dependence on BT, related to a reduction in the ratio of core electron to ion thermal diffusivity as BT increases. The parametric dependences transform into favorable dependences with decreasing normalized gyroradius and collisionality and a beta dependence that ranges from unfavorable to null. Local transport studies indicate that the electron channel usually dominates the loss of energy of NSTX plasmas. Improved electron confinement has been observed in plasmas with strong reversed magnetic shear, showing the existence of an electron internal transport barrier (eITB). Perturbative studies show that while L-mode plasmas with reversed magnetic shear and an eITB exhibit slow changes of LTe across the profile after the pellet injection, H-mode plasmas with a monotonic q-profile and no eITB show no change in this parameter after pellet injection, indicating the existence of a critical gradient that may be related to the q-profile. Further improvements in transport at the edge were observed during periods of an enhanced pedestal H-mode, where a second bifurcation led to a doubling of the plasma stored energy relative to that in the pre-enhanced H-phase and high plasma temperatures, with $T_i \sim T_e \sim 600$ eV, at the top of the pedestal. The ion thermal diffusivity in the core is typically 2 to 10 times greater than the neoclassical value, but is near neoclassical farther out in the plasma. The source and variations of both the electron and ion transport in NSTX plasmas will be studied with linear and non-linear gyrokinetic codes over a wide range of wavenumber.

EX/9-1 · Evolution of the pedestal on MAST and the implications for ELM power loadings

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Abstract: Studies of pedestal characteristics and quantities determining ELM energy losses in MAST are presented. New results from high temperature pedestal plasmas will be presented which have collisionalities one order of magnitude lower than previous results. The pedestal widths obtained in these low collisionality plasmas are found to be in better agreement with banana orbit scalings than previous high collisionality plasmas, suggesting that banana orbits can only play a role in determining the minimum width when the collisionality is sufficiently low. A stability analysis performed on these plasmas shows them to be near the ballooning limit and to have broad mode structures suggesting larger ELM energy losses, which are observed. The larger losses observed at low collisionality are consistent with measurements from various experiments which indicate that there is an ordering of the ELM energy loss with collisionality with increasing ELM size at low collisionality. The energy transported from the pedestal by an ELM can be examined in terms of the convective particle loss of electrons and ions and the conductive losses of electron and ion energy. During an ELM on MAST the fractional change in density pedestal is effectively independent of the pre-crash pedestal conditions with $\sim 4\%$ of the pre-ELM particles being expelled by the ELM, indicating that convective losses are not affected by the pedestal conditions. However, as in other devices, the ELM energy loss increases at low collisionality because the fraction of conducted energy increases. In this paper new results from MAST on the evolution of filamentary structures during ELMs will presented. Using the new insight gained from these measurements the following model for ELM energy losses is constructed: for the first 50–100 microseconds filaments remain near to the LCFS, rotate toroidally with the plasma and are aligned with the field lines. During this period 50–75% of the total ELM particle and energy loss occurs due to an increase in transport across the perturbed field lines associated with the filaments. After this time the filaments detach, accelerate radially away from the LCFS and their energy and particle content are subsequently lost by parallel transport along open field lines to the targets. It will be shown that this can explain both the convective and conductive losses observed.

EX/9-2 · ELM Propagation and Fluctuations Characteristics in H- and L-mode SOL Plasmas on JT-60U

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Abstract: Radial transport of the SOL plasma was recently studied in order to understand heat and particle loading to the first wall and the SOL formation mechanism. Fast propagation of the ELM plasma and fluctuation properties in H- and L-modes were determined both at high-field-side (HFS) and low-field-side (LFS) SOL using reciprocating Mach probes in JT-60U. Large and short (10–20 μs) peaks were found in ELM plasma flux (j_s) mostly at LFS SOL, which was enhanced after each ELM event (magnetic turbulence). Transport dynamics of the first large j_s peak was determined: it propagated to near the LFS first wall with the fast velocity of 1.3–2.5 km/s with large decay length of 9 cm. Statistical analysis of a probability distribution function (p.d.f.) was applied to describe intermittent (non-diffusion) transport in SOL plasma fluctuations. It was, for the first time, found that the positive bursty events appeared most frequently at LFS midplane distance from separatrix (Δr)$\sim$ 5 cm, and flat far SOL was formed in outer flux surfaces ($\Delta r > 5$ cm). Positive bursty events were seen in wide SOL radii ($\Delta r < 7$ cm) only at LFS midplane, where the "flow reversal" of the SOL plasma was observed. Influences of the radial transport of the convective blobby plasma on the SOL formation and the flow reversal were investigated.

EX/9-3 · Edge Localized Mode Control in DIII-D Using Magnetic Perturbation-Induced Pedestal Transport Changes

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Abstract: Edge localized mode (ELM) control is a critical issue for ITER because the impulsive heat load from ELMs is predicted to erode the divertor target plates and limit the divertor lifetime. In DIII-D, edge resonant magnetic perturbations (RMPs) with n = 3 toroidal symmetry have been used to eliminate large Type I ELMs without degrading confinement at pedestal collisionalities of 0.2, typical of those expected for ITER. ELMs are completely eliminated during the RMP pulse provided the edge safety factor is near the resonant condition $q_{95} \sim 3.7$. The RMPs enhance the radial transport across the H mode pedestal, reducing the pressure gradient enough to stabilize the MHD modes that trigger ELMs. The reduction is controlled by changing the RMP amplitude. Linear peeling-ballooning stability analysis with the ELITE code confirms that the ELMs are suppressed by reducing the pressure gradient below the peeling-ballooning limit in all cases analyzed to date. Surprisingly, the pressure gradient reduction results primarily from an increase in particle transport, not electron thermal transport, a result that is inconsistent with stochastic layer transport theory. There is no edge harmonic oscillation (seen in QH modes) or intermittent transport (seen in higher collisionality ELMing discharges) during the ELM-suppressed phase. Instead, pedestal density fluctuations increase, consistent with enhanced anomalous transport replacing the impulsive ELM transport and maintaining steady state H-mode. In ELM control experiments with collisionalities of 4, large ELMs are replaced by small recycling fluctuations, possibly Type II ELMs, that are correlated with increased intermittent transport. The increased intermittency yields a similar level of transport, leaves the pressure gradient unchanged near the peeling-ballooning boundary, and reduces the heat impulses to the divertor. Together, these results suggest that optimization of the ELM suppression might be possible.

These results will be extended to ITER-relevant pedestal rotation and triangularity using the new balanced neutral beam injection and high triangularity divertor pumping capabilities and will help assess the viability of using edge RMPs to control ELMs in ITER.

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EX/9-4 · Influence of the Dynamic Ergodic Divertor on TEXTOR Discharges

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Abstract: Recent results from the Dynamic Ergodic Divertor (DED) of TEXTOR are presented. At first, the structure of the ergodic and laminar zones are discussed for the different discharge conditions. A special subset in the open chaotic system is the structure of the so called manifolds. Then properties of the helical divertor formed by the near field of the DED are discussed including measurement data from upstream and downstream density and temperature. A new area of research is the limiter H-mode on TEXTOR which is achieved by operation of reduced Bt at full heating power. When the DED-field is applied to these discharges, the ELMs are suppressed; it is a topic of research in how far the barrier is destroyed when mitigating the ELMs.

EX/P1-1 · JET Hybrid Scenarios with Improved Core Confinement

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Abstract: The data from the Hybrid scenario experiments has been incorporated into a database that comprises ≈100 discharges and ≈280 time points, and contains more than 30 plasma and scenario parameters. Analysis of the database has revealed that there is a spectrum of temperature gradient scale lengths that exceed the critical level considered at JET to indicate the presence of an ITB. In the best example ($B_T = 2.5$ T, $I_p = 2.1$ MA, $P_{NBI} = 13.4$ MW, $P_{ICRH} = 2.5$ MW, $1 < q(0) < 1.5$) it was $R/L_{Ti} \sim 14$, $R/L_{Te} \sim 11$) on a wide region ($r \sim 0.2$ m) of the plasma core ($r/a \sim 0.4$). These steep gradients were steady for almost 5 seconds. The central ion and electron temperature were $T_i(0) \sim 17$ keV and $T_e(0) \sim 8$ keV. In the initial phase of the heating pulse the edge temperatures had been quite low ($T_i \approx T_e \sim 2$ keV), but suddenly they increased to ∼4 keV while, at the same time, the edge density decreased providing a more or less constant edge pressure. The discharge was essentially without ELMs, although the additional heating power was well above both the usual H-mode power threshold. The global performance was comparable to, or slightly better than, that of an equivalent standard Hybrid discharge with type I ELMs ($\beta_N \sim 2$, $H_{89} \sim 2.2$). The plasma energy associated with the pedestal was $W_{ped} \sim 37\%$ in this discharge, whilst the part within the steep gradient region was $W_{core} \sim 23\%$. Despite the relatively low density the toroidal rotation was only about half that of a typical JET ITB plasma. Nevertheless, the $E \times B$ shearing rate is calculated to be very large. Analysis using the gyrokinetic code, KINEZERO shows that, in the improved confinement region, ITG-TEM wavelength instabilities should be stable. An important factor in the plasmas with improved core confinement appears to be the density, which was higher in the case of similar Hybrid plasmas without core confinement improvement. Transport and stability analysis of discharges with and without core confinement improvement will be presented and compared. So far plasmas with improved core confinement in the Hybrid domain of JET represent a small minority of the database and occur only at low density and plasma collisionality. However, the possibility to obtain good global confinement in plasmas without significant ELM activity would be a useful extension of the Hybrid scenario operational domain and very attractive for ITER operations.

EX/P1-2 · New Dynamic-Model Approach for Simultaneous Control of Distributed Magnetic and Kinetic Parameters in the ITER-like JET Plasmas

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Abstract: Real-time control of radially distributed parameters was achieved for the first time on JET in 2004, in view of developing integrated control of steady state advanced tokamak scenarios in ITER. The controller was based on the static plasma response only. It was successful in achieving various targets but was found too sensitive to some rapid plasma events. A technique for the experimental identification of a dynamic plasma model has now been developed, taking into account the physical structure and couplings of the transport equations, but making no quantitative assumptions on the transport coefficients or on their dependences. The approach newly implemented aims to use the combination of heating, current drive and poloidal field systems in an optimal way to achieve a set of simultaneous tasks. Theoretical plasma transport analysis has led to the choice of the relevant state variables, and of a set of constraints to be imposed on the corresponding state-space model in order to best reproduce the dynamic response of the plasma profiles to heating power and loop voltage changes. To cope with the high dimensionality of the state space and the large ratio between the various time scales involved, our model identification procedure and controller design make use of a multiple-time-scale approximation and of the theory of singularly perturbed systems. Conventional optimal control is recovered in the limiting case where the ratio of the thermal confinement time to the resistive diffusion time vanishes. Closed-loop simulations of the new controller have been performed in preparation for experiments in 2006, using either the full-order dynamic plasma response or the two-time-scale approximate response. Comparisons show that the reduced-order controller can perform almost as well as the full-order optimal one. Feedforward compensation of some disturbances such as density changes is also included. Finally, the possibility of controlling the plasma boundary flux together with the plasma shape is a new feature to be exploited on JET together with the experimentally identified physics-based plasma model. Simultaneous control of the plasma shape, the magnetic and kinetic plasma profiles and the boundary flux can thus be attempted and will be needed to achieve non-inductively driven advanced tokamak discharges in ITER. The most recent experimental progress on JET is reported.

EX/P1-3 · Progress on NSTX towards a stable steady state at low aspect ratio

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Abstract: Modifications to the NSTX poloidal field coils have led to a significant enhancement in shaping capability and has led to has lead to the achievement of a record shape factor ($S = q_{95}(I_p/aB_t)$) of ∼37 [MA/m·Tesla]. Achieving high shape factor is an important result for future ST burning plasma experiments as exemplified by studies for future ST reactor concepts, such as ARIES-ST [S. Jardin, et al., J. Comput. Phys. 66 481 (1986)], as well as neutron producing devices such as the Component Test Facility (CTF) [F. Najmabadi, et al., Fusion Engineering and Design 65 143 (2003)], which rely on achieving even higher shape factors in order to achieve steady-state operation while maintaining MHD stability at high β_t. Plasmas with high shape factor have been sustained for pulse lengths which correspond to $\tau_{pulse} = 1.6\,\mathrm{s} \sim 50\tau_E \sim 5\tau_{CR}$, where τ_{CR} is the current relaxation time and τ_E is the energy confinement time. The non-inductive current fraction in the longest pulse discharges has reached ∼65%, with ∼55% pressure driven current and ∼10% neutral beam driven current. An interesting feature of these discharges is the observation that the central value of the safety factor q_0 remains elevated for several current diffusion times. Use of the "early H-mode" scenario has been further optimized during the 2005, exhibiting a substantial reduction in the frequency and size of ELMs. The reduction in ELM magnitude and frequency has improved energy confinement time. NSTX operates with peak divertor heat fluxes which are in the same range as those expected for the ITER device, i.e. with $P_{heat_{max}} \sim 10$ MW/m^2. High triangularity, high elongation plasmas on NSTX have been demonstrated to have reduced peak heat flux to the divertor plates to < 3 MW/m^2.
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EX/P1-4 · Feedback Control of the Safety Factor Profile in DIII-D Advanced Tokamak Discharges

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Abstract: Active feedback control for regulation of the safety factor (q) profile at the start of the high stored energy phase of an advanced tokamak discharge has been developed for the DIII-D tokamak. The time evolution of the on-axis and minimum values of q is controlled during and just following the period of ramp-up of the plasma current, using electron heating to modify the rate of relaxation of the inductive component of the plasma current profile. This concept of using changes in the σ profile to modify the current profile evolution is unique to DIII-D experiments and contrasts with maintenance of a constant current profile in steady-state, which utilizes sources of localized current drive and which has been the focus of experiments at other tokamaks. In L-mode and H-mode discharges, feedback control of q is effective with the appropriate choice of either off-axis ECH or neutral beam heating as the actuator. The q profile is calculated in real time from a complete equilibrium reconstruction fitted to external magnetic field and flux measurements and internal poloidal field measurements from the motional Stark effect diagnostic. Comparisons of experimental measurements and transport code predictions of the time evolution of the

tokamak equilibrium are used to validate transport codes for use in testing of real-time feedback control algorithms. Improved real-time controllers are being developed by including the feedback algorithm into the transport code simulation.
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EX/P1-5 · Studies on impact of electron cyclotron wave injection on the internal transport barriers on JT-60U

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Abstract: In ITER advanced operations and a steady-state tokamak reactor, an internal transport barrier (ITB) is expected to play an important role in confinement improvement and increasing the bootstrap current. In many tokamaks, formation of ITB has been observed and contributed to improving performance. A weak shear (WS) ITB plasma with the safety factor at the plasma center slightly above unity can be a good candidate for the ITER hybrid scenario and even for an attractive operational scenario in a reactor. However it has been observed that the ion temperature (T_i) ITB in a WS plasma can be degraded by a electron cyclotron range of frequency wave (ECRF). Since ECRF directly heats electrons, this phenomenon can be a critical issue in a burning plasma where electron heating is predominant. Even if this phenomenon is intrinsic to ECRF, this should be avoided. Because localized current drive by ECRF is one of the best ways to suppress the neo-classical tearing modes, which can be destabilized in an ITER hybrid discharge with an ITB for example. Therefore understanding of this phenomenon and identification of the operational space for this phenomenon to occur are important issues not only for plasma physics but also fusion development. Towards the understanding of this phenomenon, impact of ECRF injection on ITBs in a WS plasma has been investigated in JT-60U with ECRF of 110 GHz. It is observed that the T_i ITB in a WS plasma can be degraded by ECRF. It is clarified for the first time that the degradation depends increasingly on the EC power but decreasingly on the plasma current (I_p). Moreover it is confirmed that ECRF affects the toroidal rotation (V_t) indirectly and results in flattening of V_t(r) regardless to the direction of the target V_t(r), peaking co- (relative to the I_p direction) or counter. Furthermore, it is newly found that the central T_i and V_t are affected with almost no delay from the EC onset even with off-axis EC deposition. These results suggest that an EC injection acts on semi-global structure that characterizes T_i ITB in a WS plasma. In this paper, details of these dependences of the T_i ITB degradation by ECRF on plasma and operational parameters will be discussed.

EX/P1-6 · Steady-state AC Plasma Current Operation in the HT-7 Tokamak

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Abstract: A steady-state AC operation has been achieved on HT-7 superconducting tokamak with preset of 35-second duration. The plasma sustainment through the plasma current zero without lose the ionization was demonstrated under steady-state condition. Efforts are made for several important issues of the steady state AC operation, such as the equilibrium configuration, precisely feedback control of plasma position, current profile, plasma confinement property, plasma wall interaction, and fueling and recycling during the plasma current reversion. The modeling results showed that two equilibrium configurations might exit during the current across zero condition. Configuration one is that there are two oppositely flowing current components on the high field side and low field side. This is coincident with experimental observation in HT-7 by using plasma current ramping rate 1.5 MA/s. The second configuration is that one flowing current component in plasma center and another oppositely flowing current component in plasma edge which can be realized either by fast ramping the current or by lower hybrid current driven in the edge region. This is also partially demonstrated by experiment.

EX/P1-7 · Physics studies of the improved H-mode scenario in ASDEX Upgrade

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Abstract: The 'improved H-mode' regime of ASDEX Upgrade, a candidate for the ITER hybrid scenario, has been confirmed on several other devices. However, significant debate remains on which ingredients are key for improved H-mode operation and what the differences are compared to 'standard' H-modes. This contribution reports on recent studies at ASDEX Upgrade to further characterise and understand the physics of the improved H-mode. The main focus is on the influence of the ramp-up scenario for plasma current and heating on energy confinement and MHD-activity during the subsequent steady state phase. Additionally, new results for the transfer of the improved H-mode scenario to ITER are discussed, such as dependence on T_e/T_i, toroidal rotation and $\rho^\star$. Finally, the modifications of the H-mode pedestal profiles with the observed variations of stored energy and H-factor are reported and compared to the standard H-mode database.

EX/P1-8 · ITB-events and their Triggers in T-10 and JT-60U

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Abstract: Non-local transport bifurcations inside and around ITB (abrupt variations of transport in a ms timescale within 30–40% of minor radius) were found in various JT-60U reverse shear (RS) and NrS plasmas and called ITB-events. The abrupt reduction of transport in central part of plasma column often interrupts slow diffusive inward cold pulse propagation (CPP) in T-10. CPP is created by cut-off of the off-axis ECRH. This phenomenon may be called ITB-event as well. In the present paper, we report analysis of ITB-events, which have been observed at T-10 under various plasma parameters regularly. In many cases transport is halved within 40% of minor radius, which correlates with the appearance of q = 1 at r = 0 (calculated value). In some cases, ITB-events are also observed during CPP created by the reduction of ECRH power. In this case sawteeth do not appear in new steady-state, and q = 1 should not exist at the time of ITB-event. In any case, it is not clear why the appearance of q = 1 at r = 0 is able to create the abrupt and non-local reduction of heat transport. In JT-60U low-heated RS plasmas, ITB-events are connected with crossing of $\mathrm{q_{min}} = 3.5, 3, 2.5$ values. Internal MHD-activity n = 1 has been reported earlier as ITB-events trigger in JT-60U. The fishbone activity in ASDEX-U and coupling of edge and core MHD-activity (between q = 4 and q = 2) at JET assist ITB formation. In addition, the present paper shows the new MHD triggers of ITB-events in JT-60U. ITB-event is triggered by series of small internal disruptions probably associated with q = 2.5 surface in RS plasmas. ITB-event occurs in ms timescale correlation with the start of ELMs series that create enhanced level of H_α in high-βp NrS shot. The total heat flux reduces abruptly in the zone between r/a = 0.3 and 0.7 (by 3 times at the reduction maximum). The calculated radial electric field (with assumed neoclassical poloidal rotation) does not vary at ITB-events. The ms correlation between the start of ELMs series and the above-mentioned reduction of the heat flux gives possibility to control the ITB formation immediately and non-locally by inducing the ELM-like MHD activity. The new examples of ITB-triggers presented above highlight the importance of further systematical study of ITB triggering with internal and external MHD-activity. The success should bring new knobs for ITER scenario .

EX/P1-9 · Physics Advances in the ITER Hybrid Scenario on DIII-D

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Abstract: Experiments on the DIII-D tokamak have developed a long duration, high performance discharge that is an attractive operating scenario for ITER. This "hybrid scenario" regime is inductively driven with bootstrap current fractions of 35%–50% and a fully penetrated current profile with $\mathrm{q_0} \sim 1$. The remarkably good transport properties of the hybrid scenario are demonstrated by a DIII-D discharge with high normalized fusion performance, $\beta_\mathrm{N}\mathrm{H}_\mathrm{89P}/q_{95}^2 = 0.7$, that is sustained for 5 current relaxation times. The measured ion thermal diffusivity for this case is equal to the neoclassical value across the plasma cross-section. Electron heat conduction (although small in an absolute sense) dominates the energy loss process, which is consistent with nonlinear GYRO simulations that show the TEM and ETG mode cause the majority of transport. For discharges with $q_{95} > 4$, a $\rho^\star$ scan with the other dimensionless parameters held fixed showed that the effective thermal diffusivity has a scaling close to gyroBohm-like in the core (although more Bohm-like near the edge). A radiative divertor has been successfully applied to the hybrid scenario using argon injection, with good core confinement and high impurity enrichment in the divertor. Hybrid scenario discharges on DIII-D can have either a dominant 3/2 NTM or a dominant 4/3 NTM, depending upon initial conditions, with the latter having typically 15% higher H-factors (maximum 30% higher). One explanation for the lower confinement in 3/2 NTM hybrid plasmas is the flattening of the pressure profile near the $q = 1.5$ surface; the resulting "missing" bootstrap current has been explicitly observed for the first time using a direct analysis of the MSE signals. The 3/2 NTM has the beneficial effect in hybrid plasmas of broadening the current profile and maintaining $q_0 \geq 1$. There is evidence from the MSE signals that this may be due to poloidal flux pumping, associated with the ELM modulation of the 3/2 NTM. Another possible explanation is counter current drive near the axis by the 2/2 component

of the NTM, which can mode convert to a KAW and damp on electrons. Also being examined is the radial transport of fast ions, which may reduce the NBCD near the axis.

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EX/P1-10 · Controllability of Large Bootstrap Current Fraction Plasmas in JT-60U

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Abstract: A large fraction of bootstrap current driven by the high beta plasma itself is required for the steady-state tokamak concept. The profiles of pressure, current and flow should be optimized to keep stability limit and confinement as high as possible, because for example local reduction of diffusivity improves confinement while the produced steep pressure gradient sometimes causes MHD instability. The internal transport barrier (ITB) contributes to enhance the bootstrap current franction. Since the large bootstrap current fraction strengthens the linkage among these profiles, such a plasma is characterized as a self-regulating system. Active control of the ITB structure is required for reaching steady-state condition, because total pressure and current profiles are mostly determined by its structure such as the ITB radius, the ITB strength and its width. Towards the control of ITB structure, parameter linkage and controllability in the large bootstrap current fraction plasma have been investigated in JT-60U. In the case of a discharge with a large fraction of bootstrap current (∼75%) sustained as long as for 2.7 times longer than the current diffusion time, it is demonstrated that the evolution of the inductive field is greatly affected by the change in the structure of the ITB, indicating a strong linkage between pressure and current profiles. Real time control logic for rotation control based on real time calculation of the minimum value of safety factor using Motional Stark Effect diagnostic has been newly installed. Active control of the ITB radius is demonstrated by changing the plasma current profile, where the change in the loop voltage is found as one of the key factors for control of the radii of ITB and the minimum safety factor.

EX/P1-11 · The Physics of Electron Internal Transport Barriers in the TCV Tokamak

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Abstract: Electron internal transport barriers (eITBs) are generated in the TCV tokamak with strong electron cyclotron resonance heating (ECRH) in a variety of conditions, ranging from steady-state fully non-inductive scenarios to stationary discharges with a finite inductive component, and finally to transient current ramps without current drive. The confinement improvement over L-mode ranges from 3 to 6; the bootstrap current fraction is invariably large and is above 70% in the highest confinement cases, with good current-profile alignment permitting the attainment of steady state. Barriers are observed both in the electron temperature and density profiles, with a strong correlation both in location and in steepness; in particular, the density gradient being one-half as steep as the temperature gradient suggests a possible transition from an anomalous to a neoclassical regime. The dominant role of the current profile in the formation and properties of eITBs has been conclusively proven in a TCV experiment exploiting the large current-drive efficiency of the Ohmic transformer: small current perturbations accompanied by negligible energy transfer dramatically alter the confinement. The crucial element in the formation of the barrier is the appearance of a central region of negative magnetic shear, with the barrier strength improving with increasingly steep shear. This connection has also been corroborated by transport modeling assisted by gyro-fluid simulations. Rational safety-factor (q) values do not appear to play a role in the barrier formation, at least in the q range 1.3–2.3, as evidenced by the smooth dependence of the confinement enhancement on the loop voltage over a broad eITB database. MHD mode activity is however influenced by rational q values and results in a complex, sometimes cyclic, dynamical evolution.

EX/P1-12 · Prospects for Steady-State Scenarios on JET

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Abstract: In addition to the development of the inductive ELMy-H mode, much effort and progress have been made to achieve plasma regimes more suitable for steady state tokamak operations, i.e. where a large fraction of the plasma current is non-inductively driven. These challenging scenarios (high beta operation) could lead to more efficient electricity production in a reactor without the need for large plasma current. It will be argued in the paper based on recent modelling benchmarked on experimental results that the

power upgrade on JET will allow the development of high fusion performance non-inductive scenarios that could be made in conditions as close as possible to those required for ITER. The tools of the long term JET programme in preparation of ITER operation is the new ITER-like ICRH antenna (∼15 MW), an upgrade of the NB power (35 MW/20 s or 17.5 MW/40 s), a new ITER-like first wall, a new pellet injector for ELM control together with improved diagnostic and control capability. Operation with the new wall will set new constraints on non-inductive scenarios that already needed to be addressed. In this context, to estimate the fusion performances and the non-dimensional parameters that could be reached at high density and power, a simplified 0-D model has been developed based on global scaling for the energy confinement time and the current drive efficiencies. This model has been benchmarked on the existing JET database where the prediction are compared to the experimental results obtained in full current drive regimes. A power level of 45 MW gives access to JET regime where, as in a future steady state reactor, the bootstrap current is maximised together with the fusion yield and not at its expense. This approach that allows to scan the operating space has been complemented with more sophisticated 1-D modelling of the full time dependent scenario with the complete suite of heating and current drive models. Finally, it will report on the 2006 experimental results where with the available heating and current drive power levels these long term scientific issues are addressed but in a separate manner in view of preparing the full scenario integration when the JET power upgrades will be completed.

EX/P1-13 · Characteristic Features of Edge Transport Barrier Formed in Helical Divertor Configuration of the Large Helical Device

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Abstract: Edge transport barrier (ETB) was formed through low to high confinement (L-H) transition in a helical divertor configuration of the Large Helical Device (LHD) where the last closed magnetic surface is surrounded by ergodic layer defined in the vacuum field. Relationship between ETB formation and ergodic layer, and effects of sizable magnetic island in the edge region on ETB formation were investigated in this unique magnetic configuration of LHD. Electron density and its gradient in the edge region increase noticeably across the transition, while electron temperature profile has no obvious change. Thus formed ETB extends into ergodic layer. The improvement of energy confinement time is modest so far, compared to the ISS95 international stellarator scaling. The threshold power for the L-H transition is almost the same as the ITER power threshold scaling. The width of ETB is in the range of 8 cm to 16 cm for the averaged minor plasma radius of ∼60 cm, and has no clear dependence on the toroidal field strength over 0.5 T to 1.5 T, where the rotational transform at ETB is fixed. The ETB width is much larger than the poloidal ion gyro-radius which is ∼1 cm in these H-mode plasmas at the toroidal field 0.5 T. When resonant helical field perturbations were applied to expand a magnetic island size near the edge, the L-H transition was triggered at lower electron density compared with the case without application of the field perturbations, suppressing fluctuations of H-alpha emission like ELM like activities.

EX/P1-14 · Steady-State Operation of ICRF Heated Plasma in the Large Helical Device

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Abstract: Achieving steady-state plasma operation at high plasma temperatures is one of the important goals of worldwide magnetic fusion research. High temperatures of approximately 1–2 keV, and steady-state plasma-sustainment operations are reported. After the last IAEA conference, the steady state operation regime is largely extended in the Large Helical Device (LHD). A High-temperature plasma was created and maintained for more than 30 min with a world record injected heating energy of 1.3 GJ in 2004 and recently for 54 min with 1.6 GJ in 2006. The three-dimensional heat-deposition profile of the LHD helical divertor was modified and during long-pulse discharges it effectively dispersed the heat load using a magnetic-axis swing technique developed at the LHD. A sweep of only 3 cm of the major radius of the magnetic axis position (less than 1% of the major radius of the LHD) was enough to disperse the divertor heat load. The steady-state plasma was heated and sustained mainly by hydrogen minority ion heating using ion cyclotron range of frequencies (ICRF). The operations lasted until a sudden increase of radiation loss occurred, presumably because of wall metal flakes dropping into the plasma. The sustained line-averaged electron density was approximately 0.8×10^{19} m^{-3} at the 1.3 GJ discharge (#53776) and 0.4×10^{19} m^{-3} at the 1.6 GJ discharge (#66053). The average input power was 680 kW and 490 kW, and the plasma duration was 32 min and 54 min respectively. These successful long operations show that the heliotron configuration has a high potential as a steady-state fusion reactor.

EX/P1-15 · Internal Transport Barriers in FTU at ITER relevant plasma density with pure electron heating and current drive

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Abstract: The advanced tokamak scenario eligible for the steady state scenario in ITER foresees to built and sustain an internal transport barrier (ITB) at high plasma density (n_e up to 10^{20} m^{-3}), with largely dominant electron heating and negligible momentum input. Contrarily to most present tokamaks, FTU can simultaneously satisfy these two requests: high plasma densities are easily accessed, while electrons are heated by two radiofrequency systems: electron cyclotron (EC, up to 1.5 MW at 140 GHz) and lower hybrid (LH, up to 1.9 MW at 8 GHz). LH also forms and sustains the current profile j(r) suitable for the ITB, by driving off-axis a large fraction of the plasma current I_p: an almost flat or slightly reversed profile q(r) (safety factor) with $q_{min} \sim 1.3$ is built inside the ITB radius, r_{ITB}. Successful methods to vary r_{ITB} through j(r) have recently allowed obtaining steadily r_{ITB}/a up to 0.67. Peripheral LH absorption and current drive (CD) is favored primarily by low q, and, at a bit lower extent, by broader $T_e(r)$ that can in turn be affected by off-axis EC heating. Off-axis EC-CD and central counter ECCD proved tools for direct finer shaping of j(r). Full CD conditions and τ_E (energy confinement time) improved up to $1.6\tau_{E,ITER97-L}$ are maintained in plasmas with peak densities and electron temperatures up to $\bar{n}_{e0} \geq 1.3 \times 10^{20}$ m^{-3} and $T_{e0} \sim 5.5$ keV, for the whole duration of the heating pulse, longer than $35\tau_E$, and about one $\tau_{R/L}$ (ohmic current relaxation time). Good alignment of the bootstrap current is always obtained, with I_b/I_p up to 30%. At $n_{e0} \sim 0.8 \times 10^{20}$ m^{-3} T_{e0} can be larger than 11 keV. The ITB strength is instead controlled by the level of the EC heating inside the barrier. The significant collisional ion heating inside the ITB ($\Delta T_{i0}/T_{i0} \sim 35\%$) indicates that e^-–i^+ collisions do not affect the barrier dynamics. The ion heat diffusivity is lowered to the neoclassical value but the long e^-–i^+ equipartition time ($\sim 4-5\tau_E$) still prevents ion thermal equilibrium. Reflectometry shows a clear change in the turbulence close to the ITB radius, consistent with the reduced electron transport. An anomalous inward particle pinch persists also at high density, as the quite peaked profiles, $n_{e0} \sim 1.7n_e$(averaged), in full CD plasmas show, despite the absence of the neoclassical Ware pinch.

EX/P1-16 · Compatibility of the Radiating Divertor With High Performance Plasmas in DIII-D

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Abstract: We report on recent DIII-D experiments that successfully applied a radiating divertor scenario to high performance "hybrid" plasmas [T.C. Luce, et al., Nucl. Fusion 43 (2003) 321]. In the puff-and-pump approach [M.J. Schaffer, et al., Nucl. Mater. 241-243 (1997) 585] used here, argon was injected near the outer divertor target, plasma flows into both the inner and outer divertors were enhanced by a combination of particle pumping near both divertor targets and deuterium gas puffing upstream of the divertor targets, and a "dome" structure in the private flux region isolated the inner divertor from the outer divertor. Good hybrid conditions were maintained (e.g. energy confinement time normalized to ITER89p ≥ 2 and normalized plasma $\beta \cong 2.4$), and the argon accumulation in the main plasma was modest. The peak heat flux at the outer divertor target was reduced by a factor of $\cong 2.5$, while the peak heat flux at the inner target fell by only $\approx$20%. This was largely due to a much higher argon concentration near the outer divertor target than near the inner target ($\approx$7 times). Exhaust enrichment (ER) as high as 64 were obtained, and ER was insensitive to the argon injection rate. (ER is defined as the ratio of the neutral argon pressure in the baffle plenum to the atomic-equivalent pressure of deuterium in the baffle plenum, divided by the ratio of argon density to electron density in the main plasma.) The asymmetry in the argon distribution and the favorable enrichment values arose largely from the closed and partitioned divertor geometry and from the frictional forces due to the enhanced divertor flow, which impeded the escape of argon from the outer divertor. Although the argon density profiles were more peaked than the electron profiles at high argon injection rates, the emissivity profiles in the main plasma remained "hollow". Our results suggest that independent control of both the radiating properties at the inner and outer divertor targets can be controlled independently. UEDGE [T.D. Rognlien, et al., Plasma Phys. 34 (1994) 362] and MIST [R.A. Hulse, Nucl. Technol. Fusion 3 (1983) 259] modelings are used to assess the argon behavior in the divertor and main plasmas.
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EX/P1-17 · Control and Optimization of Current Profile under Dominant Electron Heating in HL-2A

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Abstract: Establishment of the current profile like in the hybrid scenario is studied under the condition of dominant electron heating in HL-2A. In the discharge of $q_a \approx 3.4$ a sawtooth-free period was produced following the pellet injection. The discharge is analysed with TRANSP. It is shown that a q-profile of weak negative shear is produced immediately after the pellet injection, and it then evolves to a broad flat profile with $q_0 > 0$. The measured MHD mode structures evidence consistencies of the evaluated q-profile with the locations of $q = 1$ and $q = 2$ surfaces in the sawtoothing period and $q = 2$ surface in the sawtooth-free period. Both the diamagnetic measurement and TRANSP analysis indicate that the energy confinement is enhanced substantially after pellet injection, which would be resulted from the q-profile optimization. The discharges with injecting LH and EC waves are simulated with TRANSP. Carefully adjusting the position of non-inductive current driven by EC, an optimized q-profile was obtained with $q_a = 3.78$ and low shear region extending to $x \sim 0.45$ in the low-density discharges (line averaged density $= 1.0 \times 10^{19}\text{m}^{-3}$). When 0.5 MW LH power in CD mode and 0.9 MW EC power mainly for plasma heating are used to control the current profile, a q-profile of low shear region extending to $x = 0.6$ and $q_a = 3.21$ is established through controlling the EC absorption position in the low-density plasma (1.0×10^{19} m^{-3}). Corresponding to the optimized current profile an electron-ITB is developed on the T_e profile of $T_e(0) = 3.5$ keV. With the similar control scheme the q-profile of low shear region extending to $x = 0.45$ and $q_a = 3.36$ can be produced in a higher density plasma of 2.32×10^{19} m^{-3}. As the constraint imposed by wave propagation condition in the HL-2A plasma limits $n_\parallel$-upshift, the LH wave absorption is bounded by the strong Landau-damping limit and the boundary of wave propagation domain. This mechanism of the LH wave absorption causes interplay of the distribution of the LH driven current with the modification of the plasma configuration, which constitutes non-linearity in the LH wave deposition. The LH wave deposition position changes spontaneously because of the non-linearity. Therefore, the feedback control of the plasma current profile through controlling the LH driven current is a challenge in the high performance operation.

EX/P3-1 · Transport Improvement Near Low Order Rational q Surfaces in DIII-D

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Abstract: In DIII-D and other tokamaks, changes in transport have been linked to low-order rational q surfaces, especially integer q surfaces. Recent experiments on DIII-D have probed the confinement changes near integer q in reverse shear discharges with detailed, time-resolved measurements of electron and ion temperature, impurity rotation, radial electric field, turbulent fluctuation levels, and q profile evolution. In discharges marginal for transport barrier formation, it is found that transport either transiently improves or a transition to an sustaining core barrier occurs in the vicinity of integer values of the minimum in q, $q_{\rm min}$, and that the change in confinement precedes the actual crossing of integer $q_{\rm min}$ by a short time. The timing of the transport changes plus measurements from magnetic probes and electron cyclotron emission indicate that magnetic reconnection is not acting as a trigger for the enhanced confinement state. The measured electron temperature gradient behavior near the radius of $q_{\rm min}$ matches predictions from nonlinear GYRO code simulations of time-averaged zonal flow structures near rational q values. These turbulence driven zonal flow structures occur due to the gap in rational surfaces near integer q surfaces and explain the observed temperature changes just before and just after $q_{\rm min}$ =integer is crossed. Measurements of turbulent density fluctuations near the integer $q_{\rm min}$ times show decreases in low and intermediate-k fluctuation amplitude coincident with the transport improvement. In addition, the BES diagnostic has detected localized increases in poloidal velocity at the $q_{\rm min}$ radius around the start of the barrier formation. The fluctuation level reductions and poloidal flow increases are consistent with the expected nature of the zonal flow structures. In cases of discharges with E $\times$ B shear close to a threshold for turbulence decorrelation the zonal-flow-induced transient can trigger changes in the equilibrium profiles which lead to formation of a sustained core transport barrier.

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EX/P3-2 · Experimental Tests of Paleoclassical Transport

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Abstract: A new model for an irreducible minimum level of radial electron heat transport, the paleoclassical model, was introduced at the 2004 IAEA Vilamoura meeting; its basic features and details have now been published. The key hypothesis of the paleoclassical transport model is that in resistive, current-carrying toroidal plasmas the electron guiding centers and hence electron heat diffuse radially with small bundles of poloidal magnetic flux on the magnetic ("skin") diffusion time scale. The initial paleoclassical model provided a number of encouraging preliminary interpretations of electron heat transport in toroidal plasmas. This paper carries the experimental comparisons to the next higher level via a number of more detailed comparisons of paleoclassical transport predictions with experimental data from a variety of toroidal plasma experiments: electron heat diffusivity in DIII-D, C-Mod and NSTX ohmic and near-ohmic plasmas; transport modeling of DIII-D ohmic discharges, of the RTP ECH "stair-step" experiments with eITBs at low order rational surfaces, and of strong eITBs in JT-60U; critical electron temperature gradient scale lengths; H-mode T_e pedestal profiles and magnitudes in DIII-D; and electron heat diffusivities in non-tokamak experiments (NSTX/ST, MST/RFP, SSPX/spheromak). The preliminary conclusion is that the paleoclassical model apparently sets the lower limit (within a factor of about two) on the radial electron heat transport in resistive, current-carrying toroidal plasmas – when it is not exceeded by fluctuation-induced transport, which typically occurs in the edge of L-mode plasmas and when the electron temperature is high (above about 1 keV in present experiments, 4 keV in ITER) because then paleoclassical transport becomes less than gyro-Bohm-level anomalous transport.

EX/P3-3 · Modulated ECCD experiments on TCV

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Abstract: Experiments were previously performed on TCV using Switched Electron Cyclotron Current Drive (SECCD) in which the total heating power and plasma current were kept nearly constant in time while the current density profile was modulated. It was observed that SECCD leads to a relative electron diffusivity modulation amplitude of ±65%. A direct measurement of the current density profile is not available on TCV; however, electrodynamic calculations show that ±55% shear modulation (from $s = 0.20$ to 0.70) is achieved during SECCD experiments, and that the shear modulation is localized at the CD layer position. These results provide strong experimental evidence that electron heat transport is shear dependent: heat transport is reduced when shear is low, confirming a general observation. New experiments have been performed to address some open issues raised after the previous campaign. In particular, the scaling of transport modulation with shear, the localization of transport modulation with respect to shear, and the impact of shear modulation on particle diffusion are studied. The work is also aimed at confirmation of the electrodynamic model of the SECCD technique. To these ends, the experiments have been performed at different deposition radii, different SECCD amplitudes and different toroidal injection angles. In addition, more effort has been dedicated to modelling using the ASTRA transport/diffusion code. This provides a more accurate prediction of current density profile evolution, together with the associated global (internal inductance, energy content, confinement time) and local (magnetic shear, electron thermal diffusivity, particle diffusivity) parameters, relevant to the issue of the shear/confinement relationship. Results from these new experiments as well as modelling will be presented.

EX/P3-4 · Interaction of Runaway Electrons with Magnetic Field Ripple in the HT-7 Tokamak

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Abstract: The runaway electrons have been measured in combination with hard x-ray detectors and thermographic camera in the HT-7 tokamak. The dynamics of runaways in the core and edge regions is monitored simultaneously. The synchrotron radiation signal diagnoses the runaways in the interior of the plasma, and the HXR signal served as a supplement to monitor the lost runaways and the dynamics of runaways in the edge region. The magnetic field ripple can play an important role on the energy of runaways located in the edge region. The maximum energy of runaways in the edge region could be blocked by the resonance of gyromotion with the nth harmonic of the magnetic field ripple. This resonance interaction creates a barrier to a further increase in the runaway energy. The mechanism can quantitatively account for the observed energy limit of the runaways in HT-7. Runaway electrons in the core have energy of about 26 MeV, while the energy limit of runaways in the edge is only several MeVs. The resonance interaction of runaways with magnetic ripple is experimentally investigated in the HT-7 tokamak. There are abnormal energy gap in the HXR spectra. The energy presents a cutoff which is resulted from the resonance interaction between the runaway electron gyromotion and the nth harmonic of magnetic ripple. The energy limit of HXR spectra increases with increasing plasma current (loop voltage). The value of energy limit is consistent with the resonance energy with harmonic number n. The energy gap increases with decreasing resonance harmonic number. It is shown that, the strength of the resonance increases with decreasing harmonic number. The energy gap under low harmonic number is larger due to the strong interaction. Interaction of runaways with magnetic ripple act an additional barrier to limit the energy of runaways to a few MeVs in the edge is favorable for reducing the effect of runaways on first wall during disruptions. By exploiting this virtue, safer operation can be achieved.

EX/P3-5 · Multi-machine Dimensionless Transport Experiments

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Abstract: Integrated dimensionless confinement studies have been performed on JET, C-Mod, JT-60U, DIII-D, and Tore Supra. ELMy H-mode identity experiments on JET and C-Mod have shown that, despite results indicating confinement falls as density approaches the Greenwald limit, Greenwald fraction is not a relevant parameter for confinement scaling, but that collisionality is. Studies on JT-60U show a fall in ELMy H-mode confinement with increasing β, contrasting with those on DIII-D and JET which showed a negligible effect. Analysis of a multi-machine database indicates that the differing results may be due to a change in β dependency with plasma shape. Tore Supra experiments have shown a negligible β effect on L-Mode confinement. Statistical studies of multi-machine core and edge confinement databases lead to an improved explanation of these results. The impact of these results on the understanding of plasma transport and its extrapolation to ITER will be discussed.

EX/P3-6 · Confinement of High Energy and High Temperature Ions in the MST Reversed Field Pinch

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Abstract: Confinement of both high energy and high temperature ions has been investigated in the Madison Symmetric Torus (MST) Reversed Field Pinch (RFP), both in standard plasmas with magnetic stochasticity, and improved confinement plasmas in which stochasticity is reduced. We find that (1) energetic ions (produced by neutral beam injection) are very well-confined in the standard RFP plasma despite the presence of significant stochasticity. This is understood from theory and simulation, and has positive implications for feasibility of neutral beam injection and alpha-particle confinement; (2) when magnetic stochasticity is reduced the thermal ion confinement is substantially increased as well. This is evidenced from direct observations of long lasting periods of sustained high ion temperature as well as from estimation of ion thermal diffusivity.

EX/P3-7 · Particle and Impurity Transport in Electron-Heated Discharges in TCV

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Abstract: The behaviour of particle and impurity transport in electron-heated TCV discharges in view of developing physics understanding for a predictive capability for alpha-heated, ignited reactor conditions. A key question with respect to ITER is whether peaked density profiles persist in electron heated plasmas at reactor relevant β. In TCV the addition of core ECH leads to partial flattening of the density profile (often dubbed 'pumpout'). This flattening effect saturates for a total power exceeding the Ohmic power in the target discharge by a factor of 3 or more, leaving a moderately peaked density profile, both in L-mode and H-mode plasmas independently of the ECH power. Although the core flattening by ECH is in qualitative agreement with drift wave turbulence theory, the saturation is not predicted by theory. Stationary H-modes with $n_e(0)/\langle n_e \rangle \sim 1.6$ heated with 1.5 MW of ECH to $\beta_N \approx 2$ have been obtained both in type I ELMy and in stationary ELM-free regimes with low particle confinement. These observations suggest that even in the presence of strong alpha heating in a reactor, density profiles may remain peaked enough for significantly improving the fusion power output. Steady-state electron ITB's are obtained in TCV by fully sustained off-axis ECCD, which leads to shear reversal. Unlike L-modes, density gradients

in the barrier do not experience flattening when central ECH is applied. Instead of a shear dependence, as observed in L-mode, the barrier region is characterized by a relation between electron temperature and density gradient lengths, which, for the strongest barriers, is expressed as $L_{ne} \sim 2.2\ L_{Te}$. Another issue is the behaviour of impurities. The presence of convective transport is clearly identified for intrinsic impurities and can differ significantly from convective electron transport. Experimental carbon density profiles from CXRS in stationary Ohmic and ECR heated L-mode discharges are always peaked. Both electron and carbon density peaking factors scale with the peaking of the current profile and with the minor radius of ECH deposition. For $q_{95} > 4$, the carbon profiles are significantly more peaked than the electron density profiles. Since these discharges are dominated by anomalous transport, the convective effects must be interpreted as being of anomalous origin.

EX/P3-8 · Improved Confinement regimes by the Control of Reversal of Toroidal Magnetic Field in the TPE-RX Reversed-Field Pinch Plasmas

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Abstract: Improved plasma performances have been achieved in the reversed field pinch (RFP) experiment on TPE-RX (major/minor radii = 1.72 m/0.45 m, plasma current < 0.5 MA) by the pulsed poloidal current drive (PPCD) and the quasi-single helicity (QSH) state, which were obtained by the control of the reversal of toroidal magnetic field. The steady high density state has been obtained during the PPCD period with a single ice pellet injection, which indicates the drastic (five-hold) increase of the particle confinement time in PPCD compared with the ordinary shots. It has been also found that the better plasma performance is observed in the QSH state compared with the multi-helicity (MH) state, both of which alternatively appear in a single shot under particular operating conditions. It has been demonstrated for the first time that the QSH state can be obtained with an almost 100% probability by applying a small positive pulse to the reversed negative toroidal magnetic field. The result of the high power (∼ up to 2 MW) NBI experiment will be presented, which has been connected to the torus and whose injection experiment has been just started.

EX/P3-9 · Edge Profile Stiffness and Insensitivity of the Density Pedestal to Neutral Fueling in Alcator C-Mod Edge Transport Barriers

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Abstract: Mechanisms determining the structure of edge "pedestals" in temperature and density, which are associated with edge transport barrier (ETB) formation in tokamaks, are investigated on Alcator C-Mod, using techniques that include empirical scaling studies and both experimental diagnosis and modeling of neutral fueling effects on the density pedestal. Experiments suggest a strong role for critical gradient behavior in setting profile characteristics of edge plasma. Maximum pressure gradient scales as the square of plasma current in both H-modes without edge-localized modes, and in the near scrape-off layer (SOL) in ohmic discharges, demonstrating a ballooning-like scaling. In either case, the obtained pressure gradient, normalized to the square of plasma current, is a function of local collisionality, hinting that common physics may contribute to setting profile gradients in both confinement regimes. The near SOL findings connect well with first-principles numerical results that suggest an underlying physical explanation based on electromagnetic fluid drift turbulence. The electron density pedestal in H-mode shows a linear dependence on plasma current, and inferred effective cross-field transport coefficients increase markedly as current is lowered. Varying the neutral fueling source alone has little effect on density gradient scale lengths in the ETB and a relatively weak impact on the height of the density pedestal, even during aggressive deuterium puffing. Altogether, these data indicate a substantial role for plasma transport in determining the density pedestal and gradient, with details of the neutral fueling source being less important. A modeled response of the density pedestal to perturbations to the edge neutral source couples a kinetic neutral treatment with a diffusive model for the plasma transport. The modeled response to small source perturbations is qualitatively consistent with experimental measurements, though the response to large perturbations does not reproduce the typically clamped density gradients seen in experiment during H-mode puffing. Together these results suggest that a simple diffusive model for plasma transport is deficient, and that a critical gradient assumption for transport may be essential for pedestal modeling.

EX/P3-10 · Transport and Confinement Studies in RFX-mod Reversed-Field Pinch Experiment

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Abstract: In reversed-field pinch (RFP) devices the magnetic configuration is characterized by a toroidal magnetic field whose direction in the plasma edge is opposite that in the core. With an edge safety factor $q_a \approx -0.02$ ($q = aB_t/RB_p$), the toroidal magnetic field in the RFP is much smaller than in the Tokamak and stability is ensured by the large magnetic shear (r/q dq/dr). Since in the RFP the safety factor is everywhere less than one, a large number of MHD modes are resonant and their amplitude largely affect the confinement. In RFX, as usual in RFP devices, MHD modes amplitude had been controlled by the use of a passive thick conductive shell that vanish the radial magnetic field at the surface. In RFX-mod a different approach has been followed: radial magnetic field control is obtained by external magnetic field coils and a close fitting thin conductive shell. In the so-called Virtual Shell operation (VS), radial field zeroing at the thin shell radius is stationary provided by the externally controlled coils. First experiments of RFX-mod proved the capability of the active scheme to control the radial magnetic field at any time. Furthermore it has been found that such edge magnetic field control extends its beneficial effects to the whole plasma, leading to a stationary 2 to 3-fold reduction of the core B_r field amplitude. The reduction of field fluctuations positively reflects on confinement. In fact, a strong reduction of the loop voltage is observed and correspondingly a 3-fold increase in pulse length is achieved by using the same poloidal flux swing. Temperature and particle measurements confirm the improved confinement properties of the virtual shell operation. With a lower ohmic input power, higher electron temperature in the plasma core and a steeper profile is measured. Particle and heat transport have been studied by means of a 1-d code. Local power balance was used to compute the heat conductivity profile: for the VS discharges a lower conductivity over a significant region of the plasma is found. The improved properties of RFX-mod VS operation provide a better confinement scaling both in terms of plasma current and density. The results show that compared to the thick shell solution, a significant confinement improvement can be obtained under stationary conditions by actively controlling the plasma magnetic boundary.

EX/P3-11 · Transport Barriers and H-mode in Regimes with Deuterium Pellets Injected into T-10 Plasma Heated by ECR

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Abstract: The confinement of energy and particles has been observed to improve in T-10 discharges when deuterium pellets are injected into the plasma heated by the ECR. The H factor found in the ITER scaling is equal to 1–1.4 in these regimes. Successive injections of deuterium pellets lead to corresponding increase in both the plasma energy and the energy confinement time. Moreover, the confinement time depends practically linearly on the averaged plasma density that increases after each pellet. In these regimes, the profile of the plasma density was observed to change abruptly. Namely, (i) a density pedestal, typical of an external transport barrier in the H mode, forms at the periphery of the plasma column; (ii) the power of auxiliary heating needed for transition to the H mode is several times less than that predicted by the ITER scaling; (iii) regions of steep density gradients, typical of internal transport barriers, form in the inner area of the plasma column. A change in the plasma temperature profile is also observed in the same regions. In addition, these regions are located near rational surfaces of low m and n values. The successive injection of deuterium pellets into the plasma has allowed the discharges with ELMs of type III to be obtained. In this case: (i) the plasma energy reduces when a packet of ELMs arises and restores when ELMs die out; (ii) a density pedestal at the plasma boundary is partially destroyed when an ELM train appears and forms again when ELMs decay; (iii) the rise of ELMs affects also the internal plasma area. The emergence of the ELM packet is accompanied by rapid decrease in ion temperature (about 15%) measured by a neutral particle analyzer, and a neutron flux decrease during an ELM train (in ~ 1.5 times) and restores after ELMs damp. Such a behavior of ion temperature can be associated with the destruction and reconstruction of an ion transport barrier resulted from the pellet injection. An analysis of huge experimental data allows us to conclude that the injection of deuterium pellets forms the transport barriers (both external and internal ones) in the T-10 plasma heated by the ECR.

EX/P3-12 · Inter-Machine Comparison of Spontaneous Toroidal Rotation

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Abstract: Plasma rotation plays an important role in the L-H transition, ITB formation and in the stabilization of resistive wall modes. In the current generation of tokamaks, this rotation is usually provided by the external momentum input from neutral beams. In future reactor-grade devices, this may not be available. The spontaneous rotation observed in many tokamaks without external momentum input may provide the solution. Since the mechanism driving spontaneous rotation is not well understood, and in order to anticipate the level of rotation expected in ITER and other reactor devices, a database of observations on several current devices has been constructed. In H-mode or in other enhanced confinement regimes, spontaneous toroidal rotation is generally observed to be in the co-current direction. Co-current spontaneous rotation has been seen on many devices and produced with a wide variety of techniques, demonstrating its fundamental nature. Substantial rotation velocities have been generated with ICRF heating on JET, Alcator C-Mod and Tore Supra. Co-current rotation has been seen in Ohmic H-mode plasmas in COMPASS-D, C-Mod and DIII-D. Co-current rotation has been observed at the edge of ECH plasmas on DIII-D and on JT-60U with a combination of LH waves and ECH. Velocities up to 150 km/s have been measured, without external momentum input. A common feature of all of these observations is a correlation between the toroidal rotation velocity and the plasma pressure or stored energy. On devices which can operate with a large range of plasma current, the rotation velocity is found to be inversely proportional to I. The coefficient of this scaling is different on different devices, and probably includes some machine size scaling information. In order to unify the rotation measurements on all of these devices, the approach of utilizing dimensionless variables has been followed. For the rotation velocity, a Mach number using the background ion sound speed has been chosen. The plasma stored energy or pressure normalized to the plasma current is contained in the normalized plasma β. All of these devices follow a scaling of M increasing with β_N. Based on this simple scaling, a Mach number of 0.3 may be expected for an ITER discharge with $\beta_N = 1.8$, and for T = 15 keV, this corresponds to a spontaneous rotation velocity of 250 km/s or 40 kRad/s.

EX/P3-13 · The Influence of Beam Injection Geometry upon Transport and Current Drive in the MAST Spherical Tokamak

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Abstract: Future Spherical Tokamaks (STs) require a large pressure driven component of plasma current, combined with off-axis auxiliary current for sustaining central safety factors $q_0 > 1$ (thereby avoiding sawteeth). To achieve long pulse ($\sim 5\,$s), discharges with $q_0 > 1$ on MAST, TRANSP simulations indicate that neutral beam current drive (NBCD) from ~10 MW of NBI, comprising a combination of off-axis and co- and counter-directed mid-plane beams is required. A number of experiments are being performed in order to confirm modelling predictions. Changes to the beam geometry can be effected by either a) reversing Ip (resulting in counter-NBI) and/or b) moving the plasma vertically either to a Lower (L) or Upper (U) single null diverted (SND) configuration (resulting in off-axis injection). Using mid-plane co-NBI, ELMing H-mode discharges are routinely studied, as are Internal Transport Barriers (ITBs), with ion energy transport reduced almost to the underlying neoclassical level. For sawtooth free operation, however, 2.5 MW of counter-NBI is needed to balance the mid-plane tangential co-NBI. 40–60 keV counter injected deuterium orbits are, unfortunately, only weakly confined by the MAST field (this having been confirmed using I.R. imaging). Nevertheless, counter-NBI results in temperatures rivalling those of co-NBI, highlighting the profound influence the E × B shearing rate has upon plasma performance. Wide ITBs can be routinely produced, with both ion and electron thermal diffusivity reduced to the ion-neoclassical level. Due to the large B_{pol}/B_{tor} ratio in the ST, NBCD efficiency for horizontally oriented off-axis NBI is predicted to be significantly different if the beam is injected above, rather than below the magnetic axis. Experiments confirm predictions, with sawtooth onset time later and l_i lower for LSND. In order to carry out current diffusion analysis, an MSE system is being commissioned and new 2D imaging diagnostics are being deployed in order to map high resolution data measured at the "tokamak" mid-plane (TS, high resolution CXRS, Z_{eff} etc.) to the "plasma" mid-plane for the SND regime. In this paper we discuss all three injection scenarios and the resulting impact upon energy, particle and momentum transport, together with the interpreted NBCD and changes to the q-profile evolution, comparing and contrasting with data from ASDEX-U and JET where possible.

EX/P3-14 · Interaction of T-10 Plasma with Impurity Pellets and Supersonic Gas Jet

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Abstract: Discharge control and diagnostics of high temperature plasma are important tasks of the controlled fusion program [ITER Physics Basis. Chapter 1 Nucl. Fusion 39 (1999) 2137]. Pellet and supersonic gas jet injection are tools for fuelling, diagnostics, wall conditioning and mitigation of major disruption [Kuteev B.V. Journal of Technical Physics 69 (1999) 63]. Recent results of T-10 experiment with impurity pellet injection and supersonic gas jet for studying and developing these applications are presented. Study of the effects during the pellet ablation phase (∼1 millisecond) with main objective to clarify the potential role of both MHD events in plasma initiated by pellet injection and pellet ablation bursts by supra-thermal electrons. It was observed that local bursts of pellet ablation appeared with increasing the pellet size while the plasma parameters were fixed. In the Li pellet experiments, fairly large cylindrical pellets were accelerated into T-10 ECRH heated plasmas. Effects of T-10 plasma response on Li pellet injection such as modifications of density and temperature profiles, snake formation are presented and described. Only weak effects of Li-wall conditioning were detected using impurity NIII, OII monitors and neutron rate signal. Reasons of this observation are discussed. The supersonic gas jet injection was arranged through a fast valve with a Laval nozzle. A forming of compact helium jets with supersonic 1.5 km/s velocity was expected. Injections were carried out in Ohmically heated plasma with $\langle n_e \rangle = 3 \times 10^{13}$ cm^{-3}, I = 270 kA and the different 1.3×10^{19} (case I) and 3×10^{19} (case II) numbers of injected helium atoms. It was observed that for case I the jet penetrated up to 18 cm of the 30 cm plasma minor radius and the injected atom content was completely absorbed by plasma. Total electron content was doubled without deterioration of plasma conditions. It implies that the supersonic jet can be used for plasma fuelling. For case II, the jet injection caused the major disruption occurring at the density limit scenario. Experimental observation of jet penetration and absorption are compared with results of modeling.

EX/P3-15 · Plasma Heating by Neutral Beam Injection in the TUMAN-3M Tokamak

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Abstract: Paper describes experiments on Neutral Beam Injection (NBI) heating (input power 330 kW) in the TUMAN-3M tokamak. A moderate ion heating was observed in the NBI experiments. The central ion temperature $T_i(0)$, measured by the Neutral Particle Analyzer, was increased from 180 to 350 eV in the shots in the nonboronized vessel. Although up to 70% of beam power was absorbed by electrons, the $T_e(0)$ increase was even smaller than ion one. An essential increase was observed in the electron density. Thus, the major effect of NBI power absorption is the increase in the energy content due to the density increase rather than temperature rise. According to the diamagnetic measurements the increase in the perpendicular stored energy during NBI was up to factor of 2.5 as compared to the initial level. Measured energy confinement times are in good agreement with ITER98(y,2) scaling in the all regimes under consideration (ohmic H-mode, NBI H-mode: boronized/nonboronized). The SXR measurement has shown some peaking of j(r) in the co-injection NBI scenario. Seemingly, the peaking indicates appearance of noninductively driven current in the vicinity of magnetic axis. The fraction of driven current could be estimated as ∼10% of total plasma current, that agrees with the ASTRA transport simulation predictions. Clear peaking of the electron density profile n(r) was observed as well. Since central particle source from the NB is negligible the n(r) peaking could be attributed to $Z_{eff}(0)$ increase due to accumulation of impurities. Application of NB heating was found to effect MHD stability of the plasma. Practically all high density shots are terminated by disruptions, but shots with NBI were systematically longer than the ohmic ones in the similar vessel conditions. Although global stability was better in auxiliary heated shots, the level of MHD activity registered by high frequency magnetic probes was higher. Bursts of wide spectrum MHD oscillations were observed during NBI. The bursts strongly correlated with sawtooth crashes. In the experiments planned for the spring 2006 the effect of current drive by NBI will be further explored in counter-injection scenario. The increase in the input NBI power up to design value of 600 kW is planned as well. Effect of toroidal plasma rotation on the edge transport barrier formation will be investigated at the highest available NBI power.

EX/P3-16 · Overview of Poloidal and Toroidal Momentum Transport Studies in JET

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Abstract: It is well known that the ExB shear flow is one of the major players in quenching turbulence. As $E \times B$ shear flow is closely related to poloidal and toroidal rotation, understanding of momentum transport is one of the key elements to achieve high fusion performance, good confinement and access to regimes with transport barriers. This paper reports on the recent studies of poloidal and toroidal momentum transport in JET. The experimental results show that the carbon poloidal rotation of an order of magnitude above the neo-classical estimate within the ITB. The main candidate to explain the large measured poloidal rotation is thought to be the turbulence driven poloidal velocity through the Reynolds stress. Simulation results with both TRB and CUTIE turbulence codes indicate that the turbulent driven poloidal velocity can well be of the order a few 10 km/s and thus, could explain the difference between meas-ured and neo-classical values. Concerning the toroidal momentum transport, the analysis of high density ELMy H-mode discharges with ITG dominated turbulence shows that the ratio of the effective momentum and ion heat diffusivity is in the range 0.1–0.4. This ratio seems to be lower on JET than on other tokamaks, and lower than the commonly assumed ratio of 1. While the ion temperature profiles are stiff, the rotation profiles are not, as increasing torque does not increase the momentum diffusion coefficient. A deeper theoretical understanding of this is in progress. The torque profiles in these high density plasmas are hollow while rotation profiles are peaked. Thus, an inward momentum pinch might be needed, and this is also predicted by the new version of the Weiland model with self-consistent calculation of toroidal rotation. The existence of the pinch combined with low momentum diffusion coefficient gives rise to the possibility to have non-flat and possibly larger than expected toroidal rotation profiles in ITER.

EX/P3-17 · Edge transport properties of RFX-mod approaching the Greenwald density limit

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Abstract: The turbulent transport at the edge of the RFX-mod Reversed Field Pinch is described with regard to its evolution as the Greenwald density limit is approached. RFX features in fact an upper density limit remarkably well defined by the Greenwald's law found in Tokamaks despite the many differences between the two magnetic configurations. The outer regions of RFP's and Tokamaks share several analogies: edge particle transport is mainly driven by electrostatic turbulence and is characterized by a highly sheared $\nu_{E\times B}$ flow. The Reynolds' stress, which is known to drive the $E \times B$ flow against anomalous viscosity, exhibits a strong gradient in the layer of high velocity shear, almost entirely due to the electrostatic component. Coherent structures emerge from the turbulent background as intermittent bursts, co-rotating with the local mean velocity shear. In RFX such structures contributed up to 50% of the total radial diffusivity. The novelty is represented by new measurements of the RFX-mod edge plasma from a Gas Puffing Imaging diagnostic (GPI) and a sophisticated array of Langmuir and electromagnetic probes. As in Tokamaks and Stellarators the probability distribution function of the edge fluctuations data is typically non Gaussian, where the deviation decreases moving from the far scrape off layer into the plasma. The paper concentrates on the results obtained in the high density regimes close to the Greenwald limit. As the density is increased – at least up to $n/n_G = 0.8$, where n_G is the Greenwald density – the frequency of the highly non Gaussian intermittent events increases. The toroidal velocity of such intermittent events at the edge decreases with the density but saturates beyond $n/n_G > 0.35$. Further experiments are planned to explore the parameters space beyond $n/n_G = 0.8$ in order to verify whether the Greenwald limit is defined by a degradation of the edge transport. The analysis of the edge data is complemented by a one-dimensional modeling of the RFP transport based on the RITM code which helps understanding the relative role of transport and neutral gas influx in the determination of the density limit. The discussion includes the comparison with helium discharges, which have been seen to slightly overcome the Greenwald limit.

EX/P3-18 · High Kinetic Energy Jet Injection into Globus-M Spherical Tokamak

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Abstract: Progress in theoretical and experimental development of the plasma jet source and injection of hydrogen plasma and neutral gas jets into the Globus-M spherical tokamak is discussed. There are: (i) injection of plasma jets with higher densities than reached in earlier study; (ii) injection of high-density

neutral gas jets with a directed energy of the particles of few tens of eV; (iii) plasma startup in tokamak with the help of the plasma gun in conditions of double swing CS operation regime. Injection of pure, highly ionized hydrogen plasma jet with a density up to 2×10^{22} m^{-3}, total number of accelerated particles (1–5) $\times 10^{19}$ and a flow velocity of $\sim$110 km/s was used as instrument for the density control. First numerical simulations showed that due to the large initial velocity such jet should penetrate deep (10–20 cm) in the Globus-M plasma. The density raise recorded by interferometer and Thomson scattering is consistent with simulations. The fuelling by high velocity plasma jet might be preferable than Laval nozzle technique, especially for injection from high field side. The gas and plasma jets were injected into Globus-M and penetrated efficiently into the magnetic field. The density rise time of $\sim$2.5 ms (for gas jet) observed is shorter than that achieved customarily with conventional gas puffing (4–5 ms) while being much longer than that of characteristic density raise time by plasma jet injection (<0.5 ms). It increased plasma particle inventory in Globus-M by $\sim$50% (from 0.65×10^{19} to 1×10^{19}) in a single shot without plasma parameter degradation. Fast density increase in all spatial points of the plasma column including the plasma central region confirmed density rise during the time less than 1 ms, which is the signature of deep jet penetration. The plasma discharge, initiated by means of plasma jet demonstrated faster heating of the discharge as compared to RF pre-ionisation.

EX/P3-19 · Tearing Mode Driven Charge Transport and Zonal Flow in the MST Reversed Field Pinch

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Abstract: Magnetic reconnection is characterized by discrete sawtooth-like bursts in many high temperature toroidal plasmas such as tokamaks and the Reversed Field Pinch (RFP). These magnetic reconnection events, which are associated with magnetic field and current density fluctuations, enhance radial transport and degrade energy confinement. In this paper, we report the first measurement of magnetic fluctuation-induced charge transport. This transport is driven by resistive tearing modes and occurs near a resonant surface in the core of a hot plasma. Charge transport results from an imbalance between the ion and electron radial flux due to particles streaming along stochastic field lines. It bursts during the sawtooth crash. Transport related charge flux has two important consequences. First, it generates a potential well along with locally strong electric field and electric field shear at the resonant surface. Second, this electric field results in a spontaneous driven zonal flow which impacts plasma momentum and energy transport. Magnetic and current density fluctuation measurements are achieved using a high-speed laser-based Faraday rotation diagnostic on MST RFP plasmas providing an opportunity to explore magnetic flutter induced transport in the core of a plasma.

EX/P3-20 · Impact of Plasma Shaping on Electron Heat Tranport in TCV L-mode Plasmas at Various Collisionalities

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Abstract: The observed increase of energy confinement towards negative plasma triangularities, found in EC heated TCV L-mode plasmas and not explained by a change of the temperature gradient due solely to geometrical effects, has motivated a dedicated study of the impact of plasma shaping on local electron heat transport. In the present study, the plasma triangularity is varied over a wide range to investigate the effect of plasma shaping, and various plasmas conditions are explored by changing the total EC power and plasma density. When decreasing the plasma triangularity from +0.4 to −0.4, the mid-radius electron heat diffusivity is observed to decrease by 30%, and the central temperature to increase by the same amount, while the plasma density, electron heat flux, safety factor profiles and average effective charge are kept constant. Alternatively, only half of the EC power is required at this negative triangularity compared to the positive one to obtain the same temperature profile. In addition, the observed dependence of the electron heat diffusivity on the electron temperature, electron density and effective charge can be cast in a unique dependence on the plasma effective collisionality. In summary, the electron heat transport level exhibits a continuous decrease towards negative triangularities and high collisionalities, in EC heated plasmas of effective collisionality ranging from 0.2 to 1. Local gyro-fluid and global gyro-kinetic simulations predict the trapped electron modes to be the most unstable modes in these rather low collisionality plasmas. For higher effective collisionality, ranging from 1 to 2 and achieved in ohmic plasmas, the electron heat transport is no more observed to decrease towards negative triangularities.

EX/P3-21 · Plasma Behavior with Hydrogen Cluster Jet Injection in the Hl-2A Tokamak

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Abstract: A cluster jet, which is like micro-pellets, will be beneficial to deeper injection and higher fuelling efficiency than that for both of supersonic molecular beam injection and gas puffing. The experiment on cryogenically cooled high-pressure hydrogen (CJI) into the HL-2A plasma was carried out and the fuelling effects were distinctly better than that of the room temperature (RT) one. For example, under the same gas pressure 1.8 MPa and similar Ohm discharge parameters, the density increment for shot 4413 with 80 K hydrogen CJI is twice of that for shot 4512 with RT gas. The injected particle front observed by the tangential optical detection array arrives at r = 8 cm and r = 26 cm for shot 4413 and shot 4512, respectively. The electron temperature gradient measured by ECE shows that the location of maximum gradient of Te is at r = 15 cm and r = 21 cm for shot 4413 and shot 4512, respectively.

EX/P3-22 · Driving Mechanism of Toroidal Rotation and Momentum Transport in JT-60U

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Abstract: It is widely recognized that plasma rotation profiles play one of the most critical roles for plasma transport and MHD stability. In JT-60U, the toroidal rotation in the direction antiparallel to the plasma current, i.e. counter (CTR) direction has been observed with no or small momentum input from near-perpendicular neutral beam injection (PERP-NBI). An inward electric field induced by a ripple loss of fast ions was considered as a candidate for the CTR rotation in the peripheral region [Koide Y., et al., Plasma Phys. Control. Nucl. Fusion Research 1, 777 (1992)]. In the core region, momentum diffusivity and inward convective velocity were separately estimated from the transient momentum transport analysis [Nagashima K., et al., Nucl. Fusion 34, 449 (1994)]. In this paper, the driving mechanism and momentum transport were systematically investigated both in the peripheral and core regions by using the various combinations of NBIs (CO, CTR, and PERP) and wide dynamic range of toroidal field ripple with and without ferritic steel tile (FST) in JT-60U. Main results are as follows: (i) fast ion losses due to the toroidal field ripple induce CTR rotation in the peripheral region, (ii) the magnitude of CTR rotation increases with increasing the ripple loss power in the peripheral region. (iii) We have also found that toroidal rotation velocity profiles in the core region can be almost explained by momentum transport considering diffusivity and convective velocity estimated from transient momentum transport analysis.

EX/P4-1 · Transport and Deposition of ^{13}C From Methane Injection Into L- and H-Mode Plasmas in DIII-D

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Abstract: The study of impurity transport on DIII-D is primarily focused on carbon as nearly all of the plasma facing surfaces are graphite. In turn, deposition regions of carbon may be particularly important for the International Thermonuclear Experimental Reactor (ITER), as tritium can be co-deposited with the carbon layers, resulting in a large in-vessel inventory. We have examined carbon transport in both L- and H-mode plasmas in DIII D by injecting $^{13}CH_4$ from a toroidally symmetric source into the top of lower single null plasmas. The divertor plasma was well characterized by divertor diagnostics, including a 2-D tangential view of the upper half of the plasma. At the conclusion of the each run campaign, carbon tiles were removed and analyzed by two different Nuclear Reaction Analysis (NRA) surface techniques. In both H- and L-mode plasmas, high ^{13}C coverage ($\sim$ 2–2.5 $\times 10^{17}$ atoms cm^{-2}) was found just inboard of the inner divertor. In the H-mode case, the deposition extends into the private flux region between the divertor strike points. The divertor deposition accounts for about one-third of the injected ^{13}C. A more sensitive NRA shows a low concentration of ^{13}C, but if uniformly distributed over the large area of the main chamber walls, this can account for another one-third of the injected ^{13}C. These experimental results are being compared with UEDGE, DIVIMP, and OEDGE models. These data, along with the modeling results, will contribute to the database required for estimating the ITER tritium inventory.
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EX/P4-2 · Particle and Energy Transport in the SOL of DIII-D and NSTX

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Abstract: We find that that SOL radial transport and plasma-wall contact is mediated by two main SOL transport vehicles, intermittent transport and ELMs. While intermittent transport is the only vehicle in L-mode, the relative importance of inter-ELM and ELM particle flux at the wall is an important question for ITER. We find that both intermittent transport and ELMs are comprised of filaments of hot, dense plasma ($n_e \sim 1 \times 10^{13}$ cm^{-3}, $T_e \sim 100$ eV) originating at the pedestal and convective in nature and thus capable of transporting both particles and heat into the SOL, increasing wall interaction and potential sputtering and impurity release. Both intermittent transport and ELMs leave the pedestal region at speeds of ∼0.5–1 km/s, losing heat and particles by parallel transport as they travel through the SOL. We find that whereas the intermittency and ELM heat is rapidly lost, resulting in short temperature radial decay lengths (∼1–2 cm), the particles are not, resulting in radial density decay lengths that can be quite long (∼13 cm) and depend on SOL collisionality. In the low collisionality case, the ELMs impact the walls quite directly. It is found that type I ELM transport can be momentarily comparable or higher than L-mode transport in most conditions, but since both the ELMs and inter-ELM (intermittency mediated) transport depend sharply on edge collisionality (density) a survey was performed using density scans. It is found that the ELMs account for ∼90% of the wall particle flux at low Greenwald fraction ($f_g \sim 0.4$), decreasing to ∼30% at $f_g \sim 1.0$. Whereas in DIII-D the intermittency decays in both intensity and frequency in H-mode (as compared to L-mode), it only decays in frequency in NSTX.
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EX/P4-3 · Erosion of Fusion Materials under High-Power Steady-State Plasma Stream on the LENTA Facility

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Abstract: The LENTA linear plasma facility has been used to study erosion of plasma-facing materials (tungsten, graphites, CFCs) under impact of deuterium plasma. The plasma stream propagated along the axial magnetic field through a gas target in steady state and simulated the conditions relevant to the ITER divertor channel. These conditions are characterized by combination of a few eV plasma temperatures with a high divertor target surface temperature. This combination is obtained on the LENTA due to the inherent plasma production method when an electron beam generates the plasma with a few eV temperatures and the traversing beam portion provides the needed high surface temperature (∼1000–1500 K). The materials have been subjected to plasma exposure, the plasma parameters were $\mathrm{N_e} \sim 10^{12} - 10^{13}$ cm^{-3}, $\mathrm{T_e} \sim 0.5 - 10$ eV, the accumulated ion dose about 10^{26} m^{-2}. Tungsten has shown the erosion effect to occur at ion energies essentially lower than the energy threshold for the physical sputtering. The highest erosion yield was detected for W–1%La_2O_3 containing the lanthanum oxide, it achieved Y ∼ (4–5)$\times 10^{-4}$ atom/ion. The subthreshold sputtering effect is explained by the sputtering of tungsten adatoms from the surface at high temperatures. Chemical erosion of carbon material was observed at room temperature. An experimental investigation of neutron irradiation effect on material erosion in plasma is considered on the basis of radiative damage modeling with a MeV-level ion accelerator (to 1–3 dpa) followed by plasma exposure of damaged materials.

EX/P4-4 · Correlations of Dust Particles at DIII-D With Plasma Parameters

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Abstract: The first quantitative measurements of dust size and spatially localized number density at DIII-D during plasma discharges have been made. The particles are observed by Rayleigh/Mie scattering of ND:YAG lasers during plasma operations. Previous dust measurements at DIII-D have included dust sample collection after operation periods and fast camera measurements of particle trajectories. Cameras have observed large dust particles moving rapidly during plasma operations but are not sensitive to very small particles. The particles observed during discharges are significantly smaller (80 nm mean radius) than those injected for trajectory studies, observed by cameras and collected with wipes of the tiles after run campaigns. The small volume (0.5 cm^3) of each observation location and short laser pulse length provide a

good localization of the dust in the tokamak. Dust produced in the vessel is comprised primarily of carbon from the plasma facing components. The observed dust particles do not penetrate into the plasma core and event rates inside the plasma edge are consistent with the neutron background rate. A typical shot has 0.7 observed particles in the SOL, producing a sample set of 2500 dust particles from the 3630 discharges in the DIII-D 2004/2005 run campaign which corresponds to a mean number density of 4000 m^{-3} in the SOL. Studies of these particles show significant asymmetries in the dust densities for different plasma configurations. There is a significant increase in dust density with H-mode discharges relative to L-mode discharges. Plasma configuration is also very important for dust production and upper single null plasmas have double the dust density in the SOL compared to lower single null plasmas with similar confinement.
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EX/P4-5 · ITER Divertor Relevant Plasma Achieved in the Magnum-PSI Programme

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Abstract: This paper presents a breakthrough in magnetized hydrogen plasma beam production,achieved in the Pilot-PSI device at FOM. Using a cascaded arc source in an axial B-field of up to 1.6 T, a 1 cm diameter H^+ beam was generated with $n_e = 7.5 \times 10^{21}$ m^{-3} at $T_e = T_i = 1.5$ eV (Thomson scattering measurements) and a forward velocity of 3.5 km/s, resulting in an ion flux density = 2.6×10^{24} ions/m^2 s. This places the Pilot-PSI beam in an ITER divertor relevant parameter range. Carbon targets have been subjected to this plasma beam and erosion measurements will be reported. Pilot-PSI is a forerunner of the larger, super conducting Magnum-PSI, presently under construction. Magnum-PSI will produce a CW 10 cm beam with $B < 3$ T, and will be equipped with extensive in situ PSI diagnostics. The design and physics issues of Magnum-PSI, to become operational in 2009, will be presented. Both linear plasma generators are part of a comprehensive Plasma Surface Interaction laboratory (PS-Lab) at FOM-Rijnhuizen, which further includes a surface analysis laboratory. The experiments go hand-in-hand with numerical modeling, for which a suite of established codes are used. These include the B2-Eirene, the DS2V Direct Simulation Monte Carlo Code, the VAC-code (MHD), a PIC/MC approach (plasma expansion and dust behaviour) and the PLASIMO code (for the plasma production in the arc). For near-surface interactions, EIRENE and the MD5 molecular dynamics code are used. Together, these provide an integrated approach from the edge plasma to the material surfaces. Modeling results of the linear plasma generator experiments are presented.

EX/P4-6 · Analysis of L-mode turbulence in the MAST boundary plasma

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Abstract: Regimes with an L-mode edge remain an important candidate operating scenario for fusion power plants due to the absence of ELM power loads. Edge fluid codes have had success in modelling many features of phenomena observed in the SOL and boundary of these plasmas but recent observations on a number of devices have been difficult to reproduce, including significant recycling at surfaces far from the plasma boundary. One possible explanation is that radially propagating plasma filaments play a key role in cross-field transport, modifying or even dominating the diffusive-like processes usually represented in the codes by poloidally and radially constant perpendicular diffusivities. Images from a wide-angle camera operating at >20,000 fps on MAST clearly show the evolution of field-aligned, filamentary structures at the edge of L-mode plasmas and density and temperature profiles from an edge Thomson scattering system with sub-microsecond time resolution and 1 cm spatial resolution show spatially isolated peaks. These observations are complemented by data from a reciprocating Gundestrup probe with integral triple probe, giving 1 microsecond time resolution of particle flux; density; temperature and plasma flow across the LFS mid-plane SOL. These data reveal a highly intermittent characteristic, with the fluctuating component dominating the time averaged value even close to the separatrix. Initial analysis confirms that the intermittent bursts are correlated with the filamentary structures. Experiments have been conducted to estimate the total radial particle flux due to the filaments from an analytic particle balance model that utilises reciprocating probe and mid-plane D_α measurements as input. These analyses are allowing a picture to be constructed of intermittent L-mode transport in MAST in three dimensions. The size of the filaments; their mode number; frequency; temperature and density evolution; radial and toroidal velocity have all been estimated. Simulations of these plasmas using the BOUT turbulence code show characteristics of intermittent transport similar to those observed experimentally. Quantitative, model independent

information on the extent of agreement between the experimental data set and BOUT simulations has been obtained using an array of powerful statistical techniques, including differencing and rescaling.

EX/P4-7 · Radial Propagation of Eruptive Turbulent Transport Events in the Scrape-Off Layer of Alcator C-Mod

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Abstract: This paper reports on investigations of propagating turbulent spatiotemporal structures in the scrape-off layer (SOL) of Alcator C-Mod. Structures are diagnosed using turbulence imaging techniques. A radial array of views coupled to diodes filtered for D_α and a fast framing camera system viewing D_α emission intensity in a poloidal-radial cross section utilize "gas-puff-imaging" to visualize the turbulence. The radial evolution of the cross-correlation function in the SOL reveals that fluctuation structures propagate radially outwards across the entire SOL. For quantification of the propagation speeds the camera images are decomposed into large-amplitude fluctuation structures, which are tracked across the camera's field of view. In poloidal direction the propagation is mainly in the direction of the time-averaged plasma drifts. The number distribution of radial velocities reveal that more than 80% of structures propagate radially outwards over distances larger than the typical structure size. The mean velocity is 1% of the ion sound speed, which is much smaller than predicted by basic structure propagation models.

EX/P4-8 · Mechanisms for Carbon Migration and Deuterium Retention in Tore Supra CIEL Long Discharges

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Abstract: A model has been created to examine, in greater detail, the co-deposition mechanism in long discharges with high extracted power. The model couples 1D core, 3D scrape-off layer and detailed 3D wall impurity generation models to follow the C balance in steady state. The model describes the complex 3D erosion/deposition zones of the Tore Supra CIEL surface, and includes a description of impurity generation from intra-tile gaps and from the poorly adhered layers of re-deposited material in shadowed areas. Results have been compared with CII/D_α spectroscopy for a power scan in a database of 50 discharges. The model reproduces the observed increase in CII emission with power. However sources due to D^+ physical sputtering are found to saturate at higher net power levels, while self-sputtering contributions continue to increase. Significantly different scaling trends are predicted for sources due to physical/self-sputtering, and for chemical erosion using flux-independent and empirical flux suppression models. Contributions due to intra-tile-gap emission are found to be important in reconciling predicted chemical erosion rates with observation in regimes where chemical erosion can be important. The model incorporates IR measurements of local temperature to evaluate the enhanced chemical erosion rates. A new mechanism for transport of deposited C to remote areas is suggested from the model: radiant heating of such layers, combined with CX bombardment, can cause layer decomposition and further transport. This contributes to the resolution of questions concerning the observation of largescale deposits (flakes) in areas on JET and in other devices which are very remote from plasma fluxes. A partial solution has previously been advanced (scrape-off layers flows move the C in the right direction) but the proposed radiation/CX mechanism can solve the puzzling aspect of transport to these remote surfaces.

EX/P4-9 · Links Between Wide Scrape-Off Layers, Large Parallel Flows, and Bursty Transport in Tokamaks

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Abstract: We will present original measurements that demonstrate the universality of many phenomena concerning SOL flows that have been observed in X-point divertor tokamaks such as JET. Surprisingly large values of parallel Mach number are measured midway between the two strike zones on top of the torus (Fig. 1), where one would expect to find nearly stagnant plasma if the particle source were poloidally uniform. Two-dimensional fluid modelling including classical parallel transport and anomalous ballooning-type transport fails to produce solutions with large flow. Tore Supra is a large tokamak with a bottom toroidal limiter. The scrape-off layer (SOL) is usually in the high temperature, low recycling regime. We choose to suppose that our Mach probe measurements are correct, and use a simple 1D model to deduce

what must be happening in the SOL to explain them. Our conclusion is that a significant fraction of the effective particle source must be poloidally localized near the outboard midplane of the torus. In order to verify this hypothesis one would need to measure the poloidal distribution of parallel flow. This cannot be done in Tore Supra because the probe is fixed, but it is possible to change the probe location relative to the strike zones by moving the plasma contact point around the poloidal section. The resulting variation of the flow is in excellent agreement with the prediction of the 1D model, if we assume that the source is always concentrated near the outboard midplane. For example, asymmetric flows are observed in cases that should be symmetric from simple geometrical considerations. We have obtained striking evidence that this localized source is due to enhanced radial transport. Plasma appears to be ejected into the SOL near the outboard midplane and can travel significant radial distances. Our measurements imply that the enhanced radial transport is localized to a small region of around 30 degrees in poloidal extent near the outboard midplane.

EX/P4-10 · Studies of hydrogen isotope accumulation in plasma facing materials performed in Kurchatov Institute

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Abstract: The report presents an overview of recent studies of hydrogen isotope accumulation in ITER plasma facing materials performed in Kurchatov Institute that aimed at search of approaches to decrease tritium retention inside the vacuum vessel of a fusion reactor. These studies address the modeling of plasma interaction with carbon, beryllium, tungsten and their mixtures at both normal fusion reactor operation and at high-intensity deuteron fluxes. Big attention was paid to the H isotope inventory in the carbon flakes forming inside the tokamak vacuum chamber during the facility operation. Deuterocarbon flakes with atomic ratio D/C from 0.002 to values exceeding 1.2 were found inside the T-10 tokamak vacuum chamber. Their color, that is an indicator of the H concentration varied from black at D/C = ∼0.4 to reddish-brown at D/C = ∼0.8 and to yellow at D/C above 1/0. Dependence of D/C and flake relative mass on temperature of annealing in air and steam was analyzed. Using the Fourier-transform infra-red spectroscopy, we found that more than 90% of D atoms in the flakes are in two states with different binding energies. The first one is characterized by the C-D sp^3 stretching modes at 2100–2220 1/cm which appear at D concentration above D/C = 0.2–0.3. The second one is characterized by the weak C-D sp^3 bending modes at 600 and 1100 1/cm. All the C-D sp^3 modes include 2–3 D atoms. The analysis of spectroscopic characteristics of C-D films provides qualitative insight into the physical reasons behind the isotope effect contributing to the retention of the heavier H isotopes in tokamak erosion products and hampering their desorption. D/C in the films redeposited under D plasma interaction with Be decreased from 0.15 at 375 K to 0.05 at 575 K. Carbon admixture in Be-C-D films increases D retention. The formation of mixed Be + C layers on carbon tiles decreases erosion of underlying carbon and hydrogen accumulation. The D concentration in the near-surface layers of W specimens after their exposure to the stationary D plasma, decreased almost by an order of magnitude under exposure to high-energy deuteron fluxes. The D content in W specimens resulting from their exposure to stationary D plasma after preceding exposure to high-energy deuteron fluxes was about two times lower than in specimens that previously were not exposed to such high-energy fluxes simulating.

EX/P4-11 · Particle Control under Wall Saturation in Long-pulse High-density H-mode Plasmas of JT-60U

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Abstract: Particle control is one of the key issues for steady-state operation. In short discharges, the first wall absorbs hydrogen particles and it works as a pump (wall pumping). The wall pumping is effective to control the plasma density. However, in future tokamak devices, wall retention increases in a long discharge. Then, it is expected that the wall retention is saturated and the wall pumping does not work. Therefore, for steady-state operation, particle control by active divertor pumping without the wall pumping should be established. In JT-60U, long-pulse operation up to 65 s with NBI of ∼12 MW for 30 s has become available since 2003. Global saturation of the wall retention has been observed in a latter phase of an ELMy H-mode plasma after several long-pulse discharges. In this paper, the particle control has been studied under the wall saturation in long-pulse high-density H-mode plasmas of JT-60U. The electron density has been successfully controlled by active divertor pumping in long-pulse high-density ELMy H-mode plasmas where wall pumping does not work and even outgas appears. The energy confinement and

ELM activity was sustained, while the outgas rate increased and the gas puff rate decreased. Particle balance has been investigated to clarify mechanisms of the wall saturation. We have observed immediate outgas with appearance of divertor plasma detachment. The observation shows that dynamic equilibrium between particle injection and desorption is established under high flux in the attached divertor plasma and the wall retention decreases with particle flux reduction by the detachment. The wall saturation can be attributed to three mechanisms: increase in the wall retention by repeating the long-pulse high-density discharges, outgas from the divertor tiles due to a rise in the divertor tile temperature, and dynamic equilibrium between particle injection and desorption.

EX/P4-12 · Surface Analysis for the TFTR Armor Tile exposed to D-T Plasmas using Nuclear Technique

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Abstract: Fuel and impurity particles show complicated behavior on the surface of plasma facing components (PFC) in fusion devices. The study is important for the design of the fuel recycling, plasma control, safety management of the tritium inventory, etc. In this study, quantitative measurements of hydrogen isotope, lithium isotope and other impurities were performed on the PFC surface exposed to D-T plasmas in the Tokamak Fusion Test Reactor (TFTR) to understand the fuel and impurity particle behaviors. The analyzed tile made of carbon-fiber composite was used as the inner bumper limiter at the position KC-16 in D-T experiments from 1993 to 1997. Lithium conditionings on the PFC surface were performed by the lithium pellet injection. After the D-T experiments, air ventilation and bake out were performed to clean the PFC surfaces. The sample tile was analyzed with the deuteron-induced nuclear reaction analysis, imaging plate method, full combustion method and activation analysis. The tritium retention of the side surface of the tile was 2.3 times larger than that of the plasma facing surface and its areal distribution increased toward the plasma facing surface. So carbon co-deposition with hydrogen isotopes occurred on tile surfaces without direct plasma contact. The tritium depth profile was different from deuterium one on the side surface. The tritium depth profile had a peak at 0.5 μm, while the deuterium depth profile was broadened up to the depth of 1.5 μm. The retained T/D ratio of 1% fairly agreed with the total injected T/D ratio of 3%. Of course the difference of both absolute densities in the profile reflects the history of injected deuterium and tritium fuels in the D-T experiments. The averaged tritium density in the near surface region was estimated to be 50% of the whole tritium density in the bulk. On the other hand, the retained isotope ratio of lithium-6 to whole lithium was 0.107 near the surface, which is 1.4 times larger than the natural isotope ratio 0.075. The ratio change relates to the injection of 95.6% enriched lithium-6 pellets to avoid resonant absorption of RF heating power in some campaigns. The lithium retention was the same order of magnitude as the deuterium retention, suggesting that lithium was deposited as lithium-deuteride. No other impurities such as carbon-14 or radio active intermediate mass nuclei were detected.

EX/P4-13 · Studies of Dust Dynamics in High Temperature Plasmas

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Abstract: Dust generation and transport in apparatuses with high temperature plasmas create substantial problems on the way to fusion reactor with magnetic confinement [G. Federici, C. H. Skinner, J. N. Brooks, et al., Nucl. Fusion 41 (2001) 1967]. The dust influences likely upon operation regimes of the divertor and core plasmas. Accounting for strong requirements to tritium losses in fuel cycle, it is desirable to clarify in next future the role of dust in accumulation and transport of tritium in the reactor. In this paper the results of RosAtom Program "Dust in high temperature plasmas" are summarized. The program has begun in Russia in 2004 and it has been supported by Kazakhstan activity in 2005. Experiments aimed at studying the dust generation in conditions corresponding to ELM and disruptions have been started using plasma guns and electron beam-plasma discharges. A new ablation regime characterized by generation of 10 μm dust particles was discovered in graphite pellet experiments on W7-AS stellarator. Studies of the Z-pinch stability with dust doped plasma are started in Kurchatov Institute. Dynamics of dust flows in gas discharges is investigated by IHED. Laser blow-off diagnostics of the films deposited on tokamak walls, including their thickness and atomic composition, is developed in Ioffe Institute. Surface and dust structure analysis is intensively developed on the basis of tunnel and atomic-force microscopy. Opportunities of Synchrotron Source of Kurchatov institute are used in analysis of films and dust composition and structure. A review of film/dust studies on tokamak T-10 is presented in [B.V. Kuteev, in Dust in Fusion Plasmas, Napa, California, April 5, 2005. ⟨http://maemail.ucsd.edu/∼dust/⟩].First experiments with carbon dust injection into tokamak plasma showed that 2–10 μm dust at 300 m/s velocity penetrates 3–5 cm beyond the last closed magnetic surface. Thus it reaches plasma with the electron temperature up to 100 eV and the density about 10^{13} cm^{-3}. The data obtained points out significant influence of radiative cooling on dust ablation. Dust penetration beyond the separatrix in reactor conditions seems possible.

EX/P4-14 · Hydrogen Retention and Carbon Deposition in Plasma Facing Wall and Shadowed Area of JT-60U

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Abstract: Evaluation of fuel inventory and its retention process are critical issues for a next-step fusion device, especially with carbon-based wall. In order to resolve the issues, the hydrogen retention and carbon deposition for the plasma facing surfaces and plasma shadowed areas of JT-60U have been performed. In JT-60U, erosion/deposition analyses for the plasma facing wall and carbon thirteen methane gas puffing experiment have shown that carbon impurity produced by erosion in the outer divertor can be transported to the inner divertor through the private region as well as the SOL plasma, and deposited in the inner divertor. For the plasma shadowed area, local carbon transport to the inboard direction was appreciable in addition to long-path transports. The amount of hydrogen isotopes in the inner divertor (deposition dominant) was larger than that of the outer dvertor (erosion dominant). The highest hydrogen isotope retention was observed in the redeposition layers at the bottom of the outer dome wing tile. This is probably because the bottom of the outer dome wing tile was facing to the outer divertor and its surface temperature was kept rather low, and neutral pressure at the pumping slot was high. This indicates that carbon eroded at the outer divertor was directly transported to the outer dome wing and deposited with large amount of hydrogen retention. This local carbon transport is also very important in carbon deposition and hydrogen retention. Nevertheless, the amount of the hydrogen isotope retention in such area (0.13 in a ratio of hydrogen isotopes over carbon) is still smaller than those observed in JET, because of high baking temperature (600 K) and high surface temperature during NB heated discharges. The local carbon redeposition and the hydrogen retention seem strongly dependent on the divertor geometry and the position of the pumping slots as well as the surface temperature. In other words, a divertor structure with pumping slots in the private flux region and high base temperature could be effective in suppressing carbon deposition in the plasma shadowed area of the inner pumping slot and in reducing the hydrogen retention.

EX/P4-15 · Particle Balance Study in the Large Helical Device

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Abstract: Achieving an effective particle control using a divertor system is a crucial issue in fusion experiment. Particle balance studies are necessary for understanding neutral particle behavior, and are also important from the viewpoint of tritium inventory in vacuum vessel for next step fusion devices. In the Large Helical Device (LHD), plasma experiments with an intrinsic helical divertor (HD) and a local island divertor (LID) have been performed, respectively. The HD is an open divertor at this stage, and the LID is a closed divertor equipping baffle structure and pump-system. In the LID configuration, most particles from the core are well guided by the outer-separatrix of $m/n = 1/1$ island to the closed divertor module located locally in a toroidal section, and the substantial part of fuelled particles are evacuated. On the other hand, in the HD configuration, the most part of fuelled particles are retained in the vacuum vessel. In a typical ice-pellet fueled discharge, over 80% of fuelled particles are evacuated in the LID configuration, and about 75% of fuelled particles remain in the vacuum vessel in the HD configuration, respectively. Therefore the discharge history strongly affects the density control in the HD configuration. The difference in the neutral particle behavior between the HD and the LID configurations is considered to be explained by following conditions in the LID configuration: (1) The low charge exchange particle flux to the first wall due to the relatively low edge neutral density reduces the amount of implantation of the neutral particles in the first wall. (2) The small carbon amount released by physical and chemical sputtering from the helical divertor plates due to the small ion flux to the plates in the LID configuration leads the co-deposited particles to be small. (3) For the high operational temperature (over 1000 K) of the LID divertor plates in the LID configuration, the amount of desorbed particle from the plates is large. On the contrary, in the HD configuration, the operational temperature of the helical divertor plates is typically less than 700 K, and it is too low for the particle desroption from the carbon divertor plates.

EX/P4-16 · Lithium as a Liquid Limiter in FTU

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Abstract: At the end of 2005, an experimental program on FTU has started, to test for the first time a Liquid Lithium Limiter (LLL) with Capillary Porous System (CPS) configuration. This program aims to extend the physical knowledge, acquired on FTU, on plasma characteristics with different dominant impurities in the discharge (B, Si, Ti, Ni, Mo, W). The main goal of the first experiments on FTU has been to test the CPS LLL behaviour in a high field machine and to use it like a conditioning system to deposit a lithium film on the chamber walls ("litization"). The lithium limiter is made of three separated and electrically insulated modules which consist of a surface layer of wire meshes of stainless steel similar to that of T-11 lithium limiter. The surface faced to the plasma is refilled through capillary forces by a liquid lithium reservoir placed on the bottom of this structure. LLL has been equipped with different diagnostics to monitor lithium surface and plasma parameters close to the LLL position. In this first experimental campaign the LLL has been tested in ohmic plasma discharges with $B_t = 6$ T, $I_p = 0.5$–0.9 MA and average electron density n_e from 0.15 up to 2.6×10^{20} m^{-3}. Litization is performed by combining the physical sputtering by plasma ions with the evaporation due to the thermal load that can be controlled acting on the plasma position through the equilibrium control coils. The experimental data clearly show that the litization of FTU vacuum vessel permits to obtain better plasma performances also in comparison with boronization. The behaviour of the lithium limiter as first wall material has been tested successfully for thermal loads in the range 1–10 MW/m^2. No anomalous phenomena like "lithium bloom" occur during plasma discharges and no surface damage has been observed on the LLL after operations.

EX/P4-17 · Lithium experiment in tokamak T-11M and concept of limiter tokamak-reactor

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Abstract: The paper suggests the replacement of magnetic divertor concept by concept of lithium limiter, using the unique properties of lithium as material of plasma facing components (PFC). Main fundamental property of lithium injected in tokamak plasma is its poor penetration to plasma center and localization close to plasma boundary. As was showed experimentally in tokamaks with Li (TFTR, T-11M, CDXU, FTU), the Z_{eff} in plasma center drops after Li-injection up to 1–1.5. This property of lithium behavior can be used for creature the lithium irradiative blanket around the plasma core. It should prevent tokamak first wall and divertor plates from local high power local loads. But if the problems of tokamak power load and impurity contamination will be solved by PFC material choose the concept of magnetic divertor might be replaced by limiter concept. This limiter can be, for example, mushroom kind limiter. Main advantage of such limiters is ability to catch the impurity flux from wall by "mushroom leg" area. Its disadvantage is too high power flux to "hat" area. In condition of intensive plasma boundary cooling by lithium radiation we can hope to decrease the energy flux to "hat" area up to low magnitude, which can be avoided by water cooling. The "hat" area in such conditions can work as lithium emitter and colder "leg" area can work as collector of back lithium ions flux to limiter shadow. If the limiter will be coated by capillary porous system (CPS) its surface tension forces can return the arrived lithium from "legs" to "hat'. As results we can close the lithium circuit: "hat"-plasma-"leg"-CPS-"hat". The non coronal radiation of lithium ions during its plasma wandering should work as virtual limiter. Such scheme was established partly in T-11M experiments with CPS lithium rail limiter. In paper the main results of lithium behavior in T-11M are discussed. The total lithium radiation power in T-11M experiment increased up to 80% of ohmic heating power. The main radiation power was localized in thin (5 cm) layer close plasma boundary. The lithium spectral line radiation (LiI, LiII, LiIII) and SXR from plasma center showed the quasi steady state character plasma processes with $Z_{\mathrm{eff}}(0) = 1.2 \pm 0.2$. The T-11M experiment showed, that temperature of mushroom limiter "hat" can be in interval 500–700°C and "legs" temperature can be from 250–350°C.

EX/P4-18 · Spontaneous Shift of Divertor Plasma Footprints during a Discharge in a Helical-Axis Heliotron Device

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Abstract: A spontaneous shift of divertor plasma footprints during a discharge is investigated in Heliotron J. Heliotron J is a low-magnetic-shear helical-axis heliotron device with an $L = 1/\mathrm{M} = 4$ helical coil ($R_0 = 1.2$ m, $B_0 \leq 1.5$ T), aiming at experimental optimization of the helical-axis heliotron concept. The

shift of δR $\sim$ 4 cm was observed for a discharge with a non-inductive small plasma current of $I_p < 2.4$ kA. The most plausible candidate for the observed shift is the change of the field topology caused by I_p. This experiment points out not only the importance of current control to fix the divertor plasma position in a low shear helical device but also the possibility of "divertor swing" for reduction of the divertor particle/heat load by a small plasma current drive without bad influence on plasma performance.

EX/P4-19 · Radiation processes of impurities and hydrogen in detached divertor plasmas of JT-60U

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Abstract: Volume recombination of C and e into C is observed for the first time in detached plasmas with MARFE. It is found that the recombination flux of C^{4+} to C^{3+} is comparable to the ionization flux of C^{3+} to C^{4+}, and that the recombination zone is above an X-point and beneath the ionization zone. This result suggests that this volume recombination predominantly produces C^{3+} ions, which contribute 60–80% to the total radiation power in the divertor plasma.

EX/P4-20 · Bursty Fluctuation Characteristics in SOL/Divertor Plasmas of Large Helical Device

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Abstract: Fluctuation properties in the SOL plasmas were intensively studied to understand the cross-field plasma transport, which determines the SOL structure and heat/particle deposition onto the first wall. Recent studies in tokamaks showed that the SOL density fluctuation is highly intermittent. Convective cross-field transport associated with the intermittent events would have strong influence on recycling processes and impurity generation from the first wall. On the other hand, in helical devices, there are few systematic studies on the SOL fluctuation property focusing on the intermittent bursty fluctuations related to plasma blob transport. Recent theory predicts that the blobs propagate toward a low field side in tokamaks. On the other hand, in the Large Helical Device (LHD), the direction of the gradient in B is not uniform because the high-field and the low-field sides rotates poloidally along the torus in the helical system. Comparison between the intermittent bursty fluctuations in the edge plasma of tokamaks and helical devices makes it possible to understand the essential physics of the blob transport. Recently, fast camera observation showed the radial motion of filaments in the edge of the LHD, suggesting the convective cross-field transport. In this paper, bursty fluctuation properties in the edge of the LHD have been investigated by analyzing the ion saturation currents measured with a probe array embedded in an outboard divertor plate. Statistical analysis based on probability distribution function was employed to determine the intermittent evens in the density fluctuation. Large positive bursty events were often observed in the ion saturation current measured with a divertor probe near a divertor leg at which the magnetic line of force connected to the area of a low-field side with a short connection length. Condition averaging result of the positive bursty events indicates the intermittent feature with a rapid increase and a slow decay is similar to that of plasma blobs observed in tokamaks. On the other hand, at a striking point with a long connection length, negative spikes were observed. Complex Morlet wavelet decomposition indicates the positive and negatives spikes have a cross-correlation, which could suggest that the origin of the intermittent events is same.

EX/P4-21 · On the origin of anomalous radial transport in the tokamak SOL

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Abstract: The high rate of cross-field transport in the tokamak scrape-off layer (SOL) is rightly attributed to turbulence, but typically only in a qualitative sense. By comparing statistical analysis of density fluctuation and E $\times$ B driven cross-field turbulent flux measurements in the TCV SOL with that of time series from 2D fluid turbulence simulations, this paper demonstrates, quantitatively, that convective SOL radial transport may be ascribed to turbulent interchange motions. Analysis of radial profiles of particle flux and electric potential from the outboard midplane region of a wide range of TCV ohmic discharges show that density and turbulent flux probability distribution functions (PDF) exhibit a high degree of statistical similarity. Across the entire SOL width, the strength of turbulence, quantified in terms of relative fluctuation level, is close to that expected from Lognormal and Gamma PDFs. Extreme value distributions, such as the Gumbel or BHP, do not vary with the relative fluctuation level and so cannot

adequately describe the data. At the SOL-main chamber interface, the density fluctuations are self-similar over a wide frequency range, with both density and flux PDFs found to be universal in shape. This implies, and is observed, that both the absolute flux and the fluctuation amplitude must scale with the local mean density near the wall. The density is further found to scale with the square of the line averaged density, providing a link between a main operating parameter and the turbulence driven wall flux. Results of 2D fluid turbulence simulations of the outboard midplane TCV SOL using the ESEL code are in remarkably good agreement with experiment. Quantitative matches are found for radial profiles of mean values, fluctuation levels, PDF shapes, timescales and power spectra of both density and flux. The fluctuation time series can be used to estimate the radial variation of an effective radial particle diffusivity and convective velocity. Excellent agreement between ESEL and experiment for TCV is again found in magnitude and radial variation. Both $D_\perp$ and $v_\perp$ increase with increasing radial distance in the SOL, consistent with the picture of radially advecting plasma filaments propagating into the far SOL at substantial fractions of the local acoustic speed. The physical mechanism driving these filaments is the interchange motion.

EX/P4-22 · Plasma Turbulence studies in ADITYA tokamak

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Abstract: The exchange of energy between large-scale sheared $E\times B$ mean flows and small-scale turbulent fluctuations plays a crucial role in controlling anomalous heat and particle losses in fusion devices. For a better understanding of this process, we have carried out measurements of the Reynolds stress and the poloidal flow velocity gradient in the edge plasma region of ADITYA tokamak with poloidal and radial arrays of Langmuir probes. Two different discharge conditions are considered: a normal Ohmic plasma and a short (5 ms) pulse of gas-feed during plasma current flat-top. It is observed that during the gas-feed pulse, turbulence levels get significantly reduced together with a reduction in the radial and poloidal correlation lengths. During the normal plasma current flat-top, energy transfer takes place from the poloidal flows to the turbulence whereas during the gas-feed pulse, the transfer process is reversed. This may be a possible reason for the suppression of turbulence during the gas-feed pulse (GP). This type of turbulence suppression has also been observed in earlier molecular beam injection (MBI) experiments on ADITYA. Thus, short pulses of GP and MBI are not only effective means of increasing plasma density in a tokamak, they also alter cross-field flow conditions and dampen edge turbulence. For additional insight into turbulence characteristics we have analyzed the data using a novel technique based on the Hilbert transform and found that turbulence consists of a finite (about 10) number of intrinsic mode functions (IMF). The instantaneous amplitudes and instantaneous frequencies obtained from the Hilbert transforms of these mode functions are used to determine some useful averaged quantities like the instantaneous energy (IE) and the degree of non-stationarity (DNS). The IE shows intermittent bursts that exceed three times the mean turbulence energy. The characteristic time-scale of IE is about 20 times larger than that of fluctuation data. From the DNS, which is a measure of the deviation of time-frequency spectrum from the mean marginal spectrum, we find that frequencies larger than 20 kHz are non-stationary. We have also studied triplet interaction among mode functions by measuring the IMF bi-coherency and found it to be statistically significant among a few high frequency modes.

EX/P4-23 · Results from the CDX-U Lithium Wall and NSTX Lithium Pellet Experiments

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Abstract: CDX-U has been operated with the vacuum vessel wall and limiter surfaces nearly completely coated with lithium, with dramatic improvements to plasma performance. Discharges achieved global energy confinement times of up to 5–6 ms, exceeding previous CDX-U results by 5x, and ITER98P(y1) scaling by 2–3x. Lithium wall coatings were applied by electron beam-induced evaporation of a lithium-filled limiter and vapor deposition from a resistively heated lithium-filled oven, up to 1000 Ångstrom thick, between discharges. The e-beam experiments also produced up to 60 MW/m^2 power density in a 0.3 cm^2 spot. Total beam power was modest (1.6 kW), but duration was up to 300 seconds. Convective transport of heat away from the beam spot was so effective that the entire lithium inventory (140 g), rather than just the area under the beam spot, was heated to evaporation (400–500°C). There was no observable hot spot within the beam footprint to the limit of resolution (10°C) of an infrared-red camera, which imaged the surface of the lithium during beam heating. These results are promising for the future implementation of lithium plasma-facing components in reactor scale devices. Lithium coating of carbon PFCs has been

initiated also on NSTX, to control the density rise during long-duration H-modes, and to optimize density profiles for non-inductive current sustainment. Lithium pellets were injected into ohmically heated helium plasmas to deposit lithium on plasma contact areas. In the first experiment, pellets were injected into repeated ohmic helium discharges to deposit about 30 mg of lithium on the Center Stack Limiter. A following deuterium reference discharge with NBI exhibited a reduction in the volume-average density by a factor of about four and a peaked density profile. In a similar subsequent experiment, diverted, Lower Single-Null helium discharges about 25 mg of lithium was deposited on the lower divertor. The density exhibited a factor of about two reduction compared to a similar reference discharge run prior to the lithium deposition, and the density profile was again peaked. These results demonstrated that lithium coating could produce the required edge pumping of diverted plasmas and motivated the recent installation of a lithium evaporator for performing routine lithium coatings over a significant fraction of the plasma facing surfaces.

EX/P4-24 · Co-deposition and fuel inventory in castellated plasma-facing components at JET

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Abstract: Castellated structure of plasma-facing components will be employed in ITER to ensure their durability and integrity under high heat loads. Material eroded from the wall may be transported and co-deposited with fuel species into the grooves of castellation leading to a high tritium inventory. Until recently the largest castellated structures were used at JET: beryllium limiters and the Mk-I divertor operated first with carbon and later with Be tiles. Detailed studies were performed to assess and compare the influence both of materials (C and Be) and PFC structure on fuel inventory. The most important results may be summarised by the following: (i) significant co-deposition of carbon and deuterium has been observed up to a few cm deep in the gaps between the tiles, in both the CFC and Be divertors; (ii) in the gaps between inner divertor CFC tiles, the fuel inventory exceeds that on plasma-facing surfaces by up to a factor of 2; (iii) in the gaps between the inner divertor Be tiles the fuel inventory reaches 30% of that on plasma-facing surfaces; (iv) in the narrow castellated grooves the co-deposited fuel is only around 2% of that found on top surfaces; (v) the deposition profile of deuterium found in the grooves has a short e-folding length of $\sim$1.5 mm; (vi) the presence of fuel is always associated with the co-deposition of carbon. This last point constitutes an important observation, particularly with regard to the current ITER first wall material mix which still envisages carbon at the divertor targets. The small inventory in the castellated grooves of Be tiles points to the influence of the gap width on the overall in-vessel fuel retention. This is also an important consideration when choosing the optimum width particularly in regions of glancing magnetic field line impact where Debye sheath effects must be accounted for. The modeling efforts will be reported both for ion and neutral transport into the gaps. The results presented here imply that the material transport and resulting fuel inventory would be strongly reduced in a machine with all-metal walls in the main chamber. These issues will be addressed in the ITER-like Wall Project being in preparation at JET.

EX/P4-25 · Multiscale Phenomena of Plasma-Wall Interaction in Long Duration Discharges on TRIAM-1M

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Abstract: In TRIAM-1M, which has capability of the steady state operation, various phenomena of the plasma-wall interaction of long duration discharges have been investigated focusing on their length scales, which are classified according to the length from the order of a diameter of the torus to that of the microstructure of deposits on the wall. The transport of neutral hydrogen in the torus has been studied by using a toroidal profile of $I_{H\alpha}$ which is measured at 6 sections in the toroidal direction and the DEGAS simulation. The hydrogen retention rate in the codeposited layer which was obtained using a material probe during the long duration discharges is found to be consistent with the global wall pumping rate estimated from the particle balance in the vacuum vessel. The global wall pumping rate in the initial phase of the discharge seems to correlate with the oxygen impurity flux, which dominates erosion and deposition on the wall and the hydrogen retention property of the codeposited layer. An in situ measurement system of erosion and deposition has been developed which is based on the interference of light on a thin semi-transparent layer of deposited material. It is found that OII line intensity is higher during the erosion phase although H_α line intensity keeps constant. The codeposition and oxygen impurity play an important role in the bridge between the scales of the PWI phenomena.

EX/P4-26 · New techniques for of tritium inventory control in carbon-based scenarios by reactive gas injection

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Abstract: The issue of tritium retention in carbon-based plasma facing materials has triggered the development of several techniques of different applicability to the in-situ, non-perturbative control of the tritium inventory in ITER. These techniques are aimed at any of the following aspects of the problem (1) Inhibition of co-deposit formation during plasma shots; (2) Release of T from films at plasma facing areas and gaps between shots; (3) Release of T from the films formed in remote areas between shots. As a type 1 technique, the scavenger concept offers the possibility of inhibiting the formation of co-deposits during plasma operation. Recently, its application to Asdex Upgrade has led to a reduction of the carbon re-deposition up to 80%. This has been achieved by injecting nitrogen into the subdivertor region, while keeping the plasma parameters unperturbed. On the other hand, experiments aimed at understanding the underlying mechanism, of critical impact on the extrapolation of the technique to ITER-relevant divertor conditions, have been carried out at the Ciemat. For the removal of carbon deposits in plasma facing areas, glow discharges in He/O_2 mixture, seems to be an efficient approach. However, the effectiveness of this technique for cleaning into the gaps, or in the presence of strong oxygen getters as Be or B, is strongly reduced. Experiments at the Ciemat have shown a surprisingly high efficiency for the removal of carbon coatings in narrow (1 mm), 4 mm deep gaps in He/O_2 discharges, with a removal rate for the films a factor >50 greater than previous reports. This is due to the survival of atomic species, which seems to be an important factor in the effectiveness of the process. If strong getters are present during film removal, use of oxygen can be precluded. The use of nitrogen containing glow discharges as one of the alternatives will be also described in the conference. Finally, baking of the full vessel in pure oxygen at sub-atmospheric pressure represents today one possible way to remove re-deposited carbon films with high T content. However, low efficiencies are typically obtained at $T < 523\,K$ allowed for ITER in its present design. Alternatives such as the activated isotope interchange induced by the use of catalysers, as hydrogen peroxide, have been investigated in laboratory experiments.

EX/P4-27 · Quantitative analysis of plasma particles in materials exposed to LHD divertor plasmas

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Abstract: The divertor armor tiles in the Large Helical Device (LHD) are suffered high flux heat load and plasma particles load. Thus, the selection of the divertor materials is one of the key issues to get the high performance plasmas. In 1999, isotropic graphite divertor tiles were installed by replacing the stainless steel tiles. In the present study, hydrogen atoms retained in a LHD divertor tile used for 6 years was analyzed by using ion beam analytical techniques (RBS&ERD). Areal density of the retained hydrogen is low at the left-hand side area ($\sim 1\times10^{21}$ H/m^2), where thick Fe deposits covered the surface. In the central area, erosion dominant area, retained hydrogen is $\sim 3\times10^{21}$ H/m^2. On the other hand, in the right-hand side (private side) area, which was covered by a thick deposition of Ti, Fe, C and O, hydrogen retention is highest ($\sim 5\times10^{21}$ H/m^2). Judging from the connection length (Lc) profile, influx of hydrogen on the right-hand side area is lower than that of the left-hand side area. This fact implies that the hydrogen retention does not simply depend on its influx. Present results indicate that changes of structure and chemical composition of the surface are very important factors to determine the hydrogen retention.

EX/P4-28 · Divertor Heat Flux Reduction and Detachment in NSTX

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Abstract: We report the first successful experiments at achieving outer divertor leg partial detachment with high auxiliary heating in a spherical torus. Two approaches to high steady-state heat flux mitigation on the outer divertor plate have been investigated in NSTX in 1–6 MW NBI-heated L- and H-mode plasmas with elongation 1.8–2.4 and triangularity 0.45–0.75. In higher triangularity and elongation plasma shapes used for the extended pulse small ELM H-mode scenario a natural reduction of the peak heat flux to 2–4 MW/m^2 due to high poloidal flux expansion (up to 20) at the outer strike point (OSP) was attained.

Another approach – a dissipative divertor scenario with deuterium or impurity (methane, neon) puffing – was employed for plasma shapes with lower delta and kappa where typical OSP steady-state peak heat flux was measured to be 4–6 MW/m^2. Steady-state midplane deuterium puffing at $R = 3-7 \times 10^{21}$ particles/s apparently led to a radiative divertor operation with a two to four-fold reduction of the peak heat flux. However, no signs of volume recombination were observed at the OSP. Midplane neon puffing led to a radiative layer power exhaust with the total radiated power being 30% of the input power and a 50–75% reduction of the OSP peak heat flux. The outer scrape-off layer (SOL), however, remained in the high-recycling regime. Deuterium injection into the lower divertor region resulted in an increase of the divertor neutral density and midplane SOL collisionality to 60–100, and led to the OSP partial detachment while the core plasma remained in the H-mode. The extent of detachment was localized to a small radial region nearby the OSP where the D_γ/D_α brightness ratio – a spectroscopic signature of volume recombination onset – increased two-fold and approached that of the detached inner divertor. An 80% reduction in the peak heat flux along with a shift of the peak location were also observed. These results are explained by a power balance calculation based on the two point model with power and momentum losses: the short parallel connection length at the OSP (4–6 m) is a key factor limiting the radiative exhaust channel while a local steady-state neutral source provides the required momentum sink.
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EX/P4-29 · Hydrogenic fuel recovery and retention with metallic plasma-facing walls in the Alcator C-Mod tokamak

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Abstract: We report on new high-Z PFC hydrogenic (H/D) retention measurements as well as the first successful demonstration of controlling H/D wall inventory in a high-Z divertor tokamak. We exploit an in-situ tritium recovery method that has been proposed for ITER that uses planned radiative terminations. All C-Mod PFC surfaces are solid molybdenum. Low-Z, oxygen-gettering boron coatings are added at times. These Mo and B surfaces serve as a proxy for tungsten and beryllium, the primary wall materials in ITER. Initial measurements of the D_2 retention in C-Mod discharges shows the surprising result that a large fraction of injected gas is retained in PFC surfaces implying D/Mo $\sim 10^{-3} - 10^{-2}$, much higher than expected (10^{-5}). While retention rates may evolve and depend somewhat on plasma parameters, the wall D inventory does not saturate; large retention fractions are found for repeated discharges. The retention rate is also dependent on divertor strike point position and plasma duration indicating that the retention is likely localized in regions of high plasma flux and/or fluence. Planned disruptions have been used to provide localized surface heating to recover the D/H from surfaces. We find a threshold energy above which the H removed becomes significant, consistent with expectations that a critical surface temperature must be obtained to promote H diffusion and release on a ms timescale. Changing the disruption type and pre-disruption plasma shape varied the location of heating and hence removal efficiency. The amount of H removed during a single C-Mod run day is ~1/3 of the H inventory measured to be absorbed into PFC surfaces. The initially large amount of H in tiles is due to a prior vacuum vent with concomitant exposure of PFCs to H_2O laden air. Subsequently, all discharges and discharge cleaning is performed with D_2. This demonstrates that planned disruptions can result in net global fuel depletion (i.e. the opposite of retention) of the wall, an important demonstration of H isotope control in a confinement experiment. The rate of H/D removal demonstrated has been determined to give a 5–10x higher global H/D recovery rate than either EC discharge cleaning or non-disruptive discharges.

EX/P4-30 · Control of the Edge Turbulent Transport by Emissive Electrode Biasing on the Tokamak ISTTOK

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Abstract: In this work results are presented on the changes induced by emissive electrode biasing in the ISTTOK edge transport. The boundary plasma is characterized with focus on the relation between $E \times B$ sheared flows and particle transport. We suggest that the distinct behaviour of the particle confinement for positive and negative bias observed in ISTTOK is related with the low $E \times B$ shear induced by positive bias in the core periphery region associated with the appearance of large amplitude fluctuation. In addition, the effect of electrode bias on the edge turbulent transport has been investigated identifying the changes induced on the fluctuations frequency spectrum and PDF. We have shown that negative electrode bias reduces the propagation of large-scale events, making the fluctuations distribution more Gaussian and resulting in low amplitude fluctuations across most of the edge plasma region. For positive bias, large amplitude, broad spectrum fluctuations appear in the core periphery, which increase the cross-field diffusion and contribute to the observed asymmetry in particle transport with the bias polarity.

EX/P4-31 · Short Wavelength Density and Low Frequency MHD Fluctuation Measurements in the STOR-M Tokamak

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Abstract: A microwave small angle forward scattering system (13 W, 140 GHz) has been used in the STOR-M tokamak to measure density fluctuations of the order of the skin depth and modified ion Larmor radius (k = 5–10 cm) during ohmic discharges. The measured relative density fluctuation level (10%) and the spectral density scaling with respect to the temperature and magnetic field suggest that the spectrum is dominated by low frequency MHD fluctuations at about 30 kHz. Geodesic acoustic mode and its asymmetry in the poloidal direction have been identified from the potential and density fluctuations measured by Langmuir probes in the edge region. Low frequency magnetic fluctuations around 30 kHz have also been measured by Mirnov coils and are suppressed by tangential compact torus (CT) injection into the STOR-M discharges. CT injection has also dramatically reduced the fluctuation correlation time.

EX/P4-32 · Investigation of ETG mode micro turbulence in FT-2 tokamak

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Abstract: Two small-scale modes are found at the FT-2 tokamak as a result of correlative analysis of the upper hybrid resonance back scattering (UHR BS) signals under conditions when the threshold for the ETG mode instability is overcome. The first possessing frequency less than 1 MHz is maximal at the plasma periphery and associated with the edge drift wave turbulence or ITG mode. The second possessing frequency higher than 2 MHz and radial wave number higher than 125 cm^{-1} is associated with the ETG mode. Its phase velocity is twice as high and amplitude is growing towards the centre. In the region of observations its level is comparable to that of the low frequency mode, been however much smaller than the later one at the edge. Behavior of the small-scale turbulence frequency and wave number spectra in dynamic lower hybrid heating and current ramp up experiments leading to confinement improvement is also studied. It is shown in particular that suppression of electron transport after the current ramp up is accompanied by decrease of the high frequency component in the UHR BS signal. The possibility to determine the poloidal plasma velocity from the correlative UHR BS data is demonstrated. The rotation profiles obtained with UHR BS, Doppler reflectometry and impurity spectroscopy diagnostics are compared.

EX/P4-33 · Radial Propagation of Electrostatic Turbulence in the HT-7 Tokamak

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Abstract: Radial propagation of electrostatic turbulence was systematically measured using a triple Langmuir probe array and a pyramid Langmuir probe array in the plasma peripheral region ($r/a = 0.6 \sim 1.1$) of the HT-7 superconducting tokamak ($R_0 = 1.22$ m, a = 0.27 m, $B_t = 2$ T). In order to improve the accessibility of Langmuir probe, the experiments were conducted in low T_e ohmic discharges. The standard two point correlation technique was used to analyze the radial propagation characteristics of the electrostatic potential fluctuations. It was found that the electrostatic turbulence propagated in the radial direction not only in the confinement region but also in the scrap-off layer. In the confinement region turbulence propagates outward with a relative small phase velocity $V_r \sim 300$ m/s compared with poloidal propagating velocity $V_p \sim -1$ km/s, negative indicates electron diamagnetic direction. Not only the poloidal dispersion relation but also the radial one shows typical electron drift-wave-like mode characteristics. Radial average wavenumber is $k_r \sim 2.8$ rad/cm, which is much higher than the poloidal average wavenumber $k_p \sim -0.6$. This means that turbulence is dominated by long wavelength components in the poloidal direction, but their radial wavelength is relative short. Radial correlation length is about 0.5 cm, which is about a half of the poloidal correlation length. This implies turbulence eddy structures are elongated in the poloidal direction and correlation is stronger within magnetic surface than across the surface. The possible reason is due to magnetic shear. Only in the edge shearing layer, the poloidal correlation length is reduced and become close to the radial correlation length, which means large-scale turbulence eddy are suppressed there. These radial-propagation characteristics from the HT-7 tokamak are a little different from those observed

in small tokamaks and purely toroidal devices, but seem to be consistent with the early experimental observations in the TEXT tokamak. Our experimental results support the turbulence theory proposed by Mattor and Diamond. They suggested that the radial propagation is a candidate mechanism responsible for driving the edge turbulence by fluctuations in the plasma core region. This scenario explains the higher fluctuation levels observed in the plasma edge as compared to the core.

EX/P4-34 · Joint Experiments on Small Tokamaks

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Abstract: The first Joint Experiment (JE) in the framework of the International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP) on "Joint Research Using Small Tokamaks" (JRUST) has been carried out on the tokamak CASTOR of the IPP Prague (Academy of Sciences of the Czech Republic). The main experimental programme of this first JE was aimed to diagnose and characterize the edge plasma in a tokamak. This JE has clearly demonstrated that small tokamaks are suitable and important for broad international cooperation, if the necessary environment and manpower to conduct dedicated joint research programmes are available. The contribution of small tokamaks to the mainstream fusion research such as edge turbulence and improved confinement in the present case can be enhanced through coordinated planning.

EX/P4-35 · Three Dimensional Identification of Zonal Flows in the HL-2A Tokamak

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Abstract: Zonal Flows (ZFs) have been studied in a few fusion devices. For example, the poloidal structures of geodesic acoustic mode (GAM) are observed in the DIII-D (Jakubowski, PRL, 2002), JFT-2M (Nagashima, PRL, 2005) and ASDEX Upgrade (Conway, PPCF, 2005) tokamaks. The toroidal structure of zonal flow is only measured on the Compact Helical System (Fujisawa, PRL, 2004). It has not been reported from a tokamak. Especially, simultaneous determination of poloidal and toroidal features of ZFs has never been conducted in a fusion device. Recently, a novel design of three-step Langmuir probe (TSLP) array has been developed for ZF measurements in the HL-2A tokamak. Three movable TSLP arrays are used to identify the properties of zonal flows. They have the poloidal distance of 6.5 cm and toroidal span of 80 cm. The fourth TSLP array driven by pneumatically reciprocating system can measure edge plasma profiles in 8 cm. Three dimensional GAM features are analyzed for the first time. The poloidal mode (m $\sim 0-1$) and toroidal mode (n ~ 0) of electric potential and field perturbations are simultaneously determined. Corresponding frequencies are estimated as 7–9 kHz, which are in good agreement with theoretical prediction. The radial scale lengths of ZFs are 2.4–4.2 cm. The formation mechanism of the flows is identified to be nonlinear three wave coupling of ambient turbulence. The modulation effect of ZFs on ambient turbulence is also observed. Poloidal dependence of density perturbations and detailed radial structures of electric field are under consideration for next campaign while the existent evidence of the low frequency ZFs will be identified.

EX/P4-36 · Major progress in high spatial and spectral resolution of reflectometry in Tore Supra: density peaking and fluctuation measurements

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Abstract: Fluctuations measurements provide key insight for understanding the turbulent transport. Localised measurements over the whole plasma core reveals detailed properties of turbulent transport, especially the changes in its spatial distribution or the localised zone where the transport falls to neo-classical values. Complementary spectral sensitivity, from low to medium or high k permits to address the question of ion or electron transport channel and dominant modes. Tore Supra is equipped with a set of four reflectometers: fast swept reflectometers provide accurate density measurements over the whole profile; a fixed frequency reflectometer detects large scale fluctuations or coherent modes with a high sensitivity; a Doppler reflectometer measures the poloidal rotation and fluctuations amplitude at selectable poloidal wave numbers ($3 < k_\theta < 20$ cm^{-1}). High resolution profile reflectometry has shown the existence of a localised density peaking inside the $q = 1$ surface. The density profile is flat over a region larger than the $q = 1$ surface, except in the very centre. Particle transport coefficient and pinch velocity derived from the dynamics of this peaked profile are close to neo-classical values. The pinch

effect disappears in non inductive discharges. When present, the density fluctuations profile is hollow inside the $q = 1$ surface, over the peak region supporting the assumption of a very low level of turbulent transport inside the $q = 1$ surface. L-mode discharges in Tore-Supra are particularly suitable for studying the β dependence of turbulent transport without edge MHD (ELMs) and pedestal physics to interfere. The fluctuation level profile measured in the β scan experiment, with similar $\rho^\star, \nu^\star$, q profile, shows no dependence in the gradient zone, which is consistent with the weak β-dependence of the global confinement time. This supports the low dependence observed in JET and DIIID, in contradiction to the ITER H-mode scaling law, which predicts a degradation of the energy confinement with β. Finally, Doppler reflectometry shows a fast decrease of density fluctuation level at high k: the spectral index changing from the usual -3 to -7 for $k_\perp \rho_s \sim 1.5$. This suggests a minor role of Electron Temperature Gradient driven mode. The k spectrum deviation from usual power law.

EX/P4-37 · ETG Scale Turbulence and Transport in the DIII-D Tokamak

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Abstract: On the DIII-D tokamak, high wavenumber, sub ion gyroradius scale ($k_\perp \sim$ 35–40 cm^{-1}, $k_\perp \rho_i$ = 4–10 or ETG scale, where ρ_i is the ion gyroradius) plasma density turbulence has been found to vary independently of low wavenumber turbulence and to correlate with changes in electron thermal flux. These results indicate that the high k fluctuations ($k_\perp \sim$ 35–40 cm^{-1}) are not remnants of the low k fluctuations and are consistent with the high k turbulence driving at least part of the electron heat transport in the outer half of the plasma. Recent theoretical work points to high k turbulence as a potential source of anomalous electron heat transport making such measurements and comparisons highly relevant to fusion energy research. The level of high and low k density fluctuations are reported and compared as well as the spatial distribution of the high k fluctuations Finally, changes in the measured turbulence levels are consistent with relative changes in linear gyrokinetic growth rates only if damping due to electric field shear is taken into account. From these measurements an improved understanding of high k turbulence, its relation to low k turbulence and transport, and its spatial distribution and level has emerged. These and other similar comparisons of transport properties and broad wavenumber turbulence measurements to theoretical predictions are essential in developing confidence in the predictive capability of simulations and transport codes.
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EX/P4-38 · Investigation of Different Plasma Components Confinement and Turbulence Characteristics in T-10 Tokamak

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Abstract: The turbulence plasma state was characterized by means the comparison of impurity and energy confinement times. The transport of high Z ions was studied using KCl and Ti pellet injection and fast argon gas puffing. The impurities transport was analyzed by means of X-ray crystal monochromator RM-2. Main plasma confinement was investigated, using deuterium gas puff and energy time was estimated from the total heating power and total energy in plasma. In the neoclassical case, the different plasma species should have different confinement times, in contrast to the conditions of strong drift-like long wave turbulence, where the transport could be the same for all plasma components. The confinement times were equal in ohmic plasma. They greatly decreased and became different in ECRH discharges, suggesting preferential deterioration of the electron energy transport due to possible appearance of the short-wave turbulence in ECRH in addition to the long-wave drift one, which is dominant in the OH discharges. The experiments with Ar fast gas puff were made in ECRH discharge. The confinement of the argon significantly decreased in the ECRH discharge with respect to the Ohmic one. The comparison of the argon Ar^{+16} and Ar^{+17} lines and the recombination continuum intensities at the stationary phase of discharge after injection enables us to determine the deviation of the ions populations from coronal equilibrium. This deviation could be explained by such physical processes as finite confinement time, charge exchange with neutrals, sawtooth activity and the deviation of the electron distribution function from the Maxwell one. The reflectometry experiments with the newly installed HFS antenna reveal the lower in a factor of two turbulence level, with respect to the LFS. This level at HFS do not increase in ECR heated plasmas, suggesting the ballooning turbulence nature. The relative contrast of the quasi-coherent maxima were much less at HFS.

EX/P4-39 · Dynamic Transport Study of the Plasmas with Transport Improvement in LHD and JT-60U

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Abstract: A transport analysis during the transient phase of heating (a dynamic transport study) applied to the plasma with internal transport barriers (ITBs) in Large Helical Device (LHD) Heliotron and JT-60U tokamak is described. The dependence of electron thermal diffusivity on T_e during the density decay phase after the pellet injection is studied in LHD. There are clear 4 phases characterized by the sign of the time derivative of the T_e gradients. ($d(Q_e/n_e)/d(-dT_e/dr) < 0$ in phase I and III and >0 in phase II and IV). There are two branches in the transport, one is characterized by a weak T_e dependence (phase II) and the other by a strong T_e dependence (phase IV). Phase III is a back-transition phase between the strong and the weak T_e dependence branches. A slow transition between two transport branches, a weak T_e dependence and a strong T_e dependence transport, is observed. The time of the transition from the L-mode plasma to the ITB plasma is clearly determined by the start of flattening of temperature profile in the core region by investigating the time evolutions of the electron temperature gradients in the high gradient ITB region and at the shoulder during the formation of the ITBs in LHD and JT-60U with positive magnetic shear. An abrupt change in the sign of the time derivative of the temperature gradient at "shoulder" shows there is a clear transition from the L-mode plasma to the ITB plasma. It should be noted that the flattening of the temperature profile in the core region is observed both in positive and negative magnetic shear configurations in JT-60U and in LHD plasmas, where the magnetic shear is always negative and unchanged. A spontaneous transition from a weak ITB to a strong, narrow ITB is observed during the phase of constant heating power. The electron thermal diffusivity values are almost identical both at $r/a = 0.45$ and 0.65 in the wide weak ITB phase. However, the electron thermal diffusivity at $r/a = 0.45$ in the core region increases (the T_e gradient decreases) associated with the decrease of electron thermal diffusivity at $r/a = 0.65$ at the highest gradient region. This behaviour also indicates that there are multi meta-stable states in the transport and the spontaneous transition between these meta-stable states occurs on a transport time scale.

EX/P4-40 · Transport and fluctuations during electrode biasing experiments on the TJ-II stellarator

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Abstract: Results of edge electrode biasing experiments carried out in TJ-II are reported. Electrode biasing experiments in TJ-II show that it is possible to modify the edge radial electric field and the particle confinement for both positive and negative bias. The existence of a threshold density value for the development of the spontaneous shear layer has lead to the study of the effect of bias at different plasma densities, particularly in what concerns their effects on the edge plasma parameters and particle confinement. Biasing experiments have been performed in different TJ-II configurations having different volumes (and consequently distinct ripple) and results indicate that the fast decay time of the edge plasma potential depends both on plasma configuration and plasma density. The role of neoclassical/anomalous viscosities to explain those results is under investigation. Externally applied radial electric fields have a crucial impact in non-exponential SOL decays of plasma parameters; once edge biasing is on the level of fluctuation and the radial velocity of turbulent events are reduced and non-exponential tails disappear. The obtained results show a direct link between the level of turbulence and the development of non-exponential tails in the SOL region.

EX/P4-41 · The Radial Structure of Zonal Flows and Spectral Transfer in Turbulence of the H-1 Heliac

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Abstract: Spectral transfer of turbulent energy is an important aspect of turbulence. The nature and direction of the spectral energy transfer have direct impact on the generation of large turbulent structures, on the magnitude and shape of the spectrum, and ultimately on the particle and energy transport produced by the turbulence. For plasma turbulence described using a single field approximation, a method of inferring the spectral power transfer function (PTF) from experimental data has been applied to the H-1 plasma turbulence. Based on the analysis of the spectral energy transfer in the broadband turbulence in the H-1 heliac, the first experimental evidence of the inverse energy cascade in toroidal plasma has been presented The inverse energy cascade from the unstable spectral range (region of the maximum growth rate) has been identified as the main mechanism of generation of the broadband turbulence in H-1. Linear growth

rate determined in the process of the PTF analysis can be used as a tool for the identification of the instability. This unstable spectral range cannot be easily seen in the broad turbulence spectra. In this paper, we study spatial localization the instability, zonal flows and the potential fluctuations in order to understand the interplay between these components of the plasma turbulence. Zonal flows (f close to 0, m = 0 and n = 0) are identified as the spectral broadening of the potential fluctuations at f = 0. It is found that the maximum of the low frequency zonal flows (<1 kHz) in L-mode discharges does not coincide with the maximum in the fluctuations of the electrostatic potential. These results are discussed in the context of the recently suggested mechanism of the nonlocal relationship between the microscopic fluctuations and the resulting turbulent transport, referred to as, the turbulence spreading. Results on the localization of the linear growth rate determined using the PTF technique are also presented.

EX/P4-42 · Confinement improvement in high-ion temperature plasmas heated with high-energy negative-NBI in LHD

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Abstract: The increase in the ion temperature (T_i) due to transport improvement has been observed in plasmas heated with high-energy negative-NBI (>150 keV, H), in which electrons are dominantly heated, in Large Helical Device (LHD). When the centrally focused ECH is superposed on the NBI plasma, the central electron temperature (T_e) is increased with a steep gradient in a core region of around r/a = 0.4, indicating the formation of the electron ITB. Simultaneously, the ion temperature is also raised. The electron ITBs in helical systems are characterized by improvement of the core electron transport due to the neoclassical electron root. The neoclassical calculation shows the formation of positive radial electric field (E_r) in the core region in the electron ITB plasma, in which the transport improvement of both ions and electrons is theoretically predicted. In the CXRS measurement a positive increase in the E_r is observed in a core region with the superposition of the ECH, and an increase in the T_i is also observed. Therefore, the T_i rise is ascribed to the ion transport improvement with the transition to the neoclassical electron root. Although the electron heating is dominant in the high-energy hydrogen NB heating, the ion heating is effectively enhanced in high-Z discharges with Ar or Ne seeding. In the high-Z plasmas, the ion temperature is increased with an increase in the ion heating power, and reaches 13.5 keV. The central T_i increases with an increase in a gradient of the T_e in an outer region of r/a = 0.8. Since the helical ripple is increased toward the outer region, the transition condition to the electron root would be mitigated in the outer region. In the neoclassical ambipolar calculation considering multi-ion species, generation of strong positive Er is indicated in the outer region of the high-T_i plasma, and the positive E_r is observed in the outer region in the CXRS measurement. These suggest the ion transport improvement in the outer region induced by the neoclassical electron root, and that should lead to the central T_i rise. The results presented here show the effectiveness of the electron-root scenario for obtaining high-ion temperature plasmas in helical systems.

EX/P6-1 · Dependence of the Confinement of Fast Ions Generated by ICRF Heating on the Field Configuration in Heliotron J

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Abstract: The formation and confinement experiment for fast ions is performed using the ICRF minority heating scheme with a proton minority and a deuteron majority in Heliotron J, a low-shear helical-axis heliotron. The effect of the magnetic configuration on the fast ion confinement is one of the most important issues in helical devices. In this paper, the effect of the bumpiness on the trapped fast ion confinement is clarified by using ICRF minority-heating. The role of one of the Fourier components, the bumpiness, is a key issue for the design principle of the magnetic filed of Heliotron J, where the particle confinement is controlled by the bumpiness. The proper bumpiness causes deeply trapped particles to be confined in the small grad-B region. Two loop antennas are installed on the low-field side of the corner section of the Helitoron J. The high energy ions are produced up to 10 keV by injecting an ICRF pulse into an ECH target plasma where $T_i(0) = 0.2$ keV, $T_e(0) = 0.8$ keV and $\bar{n}(e) = 0.4 \times 10^{19}$ m^{-3}. For the study of the configuration dependence on the fast particle confinement, three configurations are selected; the bumpy ripples (B_{04}/B_{00}, where B_{04} is the bumpy component and B_{00} is the averaged magnetic field strength) are 0.01, 0.06 and 0.15 at r/a = 0.67. The measured tail temperatures by using a charge-exchange neutral energy analyze are 1.04, 0.87 and 0.47 keV for the ripples of 0.15, 0.06 and 0.01, respectively. The heating efficiency of the bulk ion is also better in the high bumpy case.

EX/P6-2 · Electron Bernstein Wave Heating of High Density H-modes in the TCV Tokamak

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Abstract: Unlike conventional electron-cyclotron (EC) waves, the propagation of electrostatic Electron Bernstein Wave (EBW) is not limited by a density cut-off. In a toroidal machine, EBW can be generated through a double mode conversion scheme O-X-B, with O-mode injection from the plasma low field side. The O-X conversion requires a density above the O-mode cut-off, a steep edge density gradient and a specific injection angle. The X-B conversion occurs naturally in a hot plasma and the B wave is absorbed at a harmonic of the EC resonance. EBW absorption was demonstrated for the first time on a stellarator. In the conventional aspect ratio tokamak TCV ($B = 1.5$ T), high edge density gradients are produced in diverted high-density Ohmic H-modes at high triangularity, high elongation and low safety factor. EBW heating experiments were performed with up to four of the six available 0.5 MW 2nd harmonic gyrotrons, injecting power through independent steerable launchers. The optimum O-mode injection angle was determined experimentally in a 2D angle scan, by injecting EC power at low duty cycle, such as not to perturb the plasma absorption conditions. The quality of the power absorption is measured by several EC stray field detectors. The experimental optimum angles differ from the ray tracing prediction by less than 2 degrees. The required high edge density gradients were obtained during ELM-free phases. High average power EBH experiments were subsequently performed with O-mode injection at the optimum angle. The fraction of absorbed power is measured with a diamagnetic loop to be above 60%. The power deposition location is measured from a high spatial resolution soft X-ray camera. The spatial distribution of the modulated component of the line-integrated soft X-ray emission clearly shows two radial peaks corresponding to the deposition layer on the high- and low-field sides. Upon tomographic inversion of these signals, the power deposition is determined to be centred at a normalised radius of 0.7, at a density above the O-mode cut-off, a clear signature of EBH. A satisfactory match is found with the ray tracing prediction. Significant central electron temperature and beta increase were achieved in recent high power EBH experiments (CW, 2MW) at a reduced magnetic field to favour a more central deposition.

EX/P6-3 · Observations of Alfvénic MHD Activity in the H-1 Heliac

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Abstract: The H-1 heliac is a medium sized helical axis stellarator ($\mathrm{R} = 1\,\mathrm{m}$, $\mathrm{a} = 0.15$–0.20 m). Its flexible heliac coil set permits access to a wide range of magnetic configurations, achieved by precise control of the helical winding current, providing rotational transform (ι) in the range $0.9 < \iota < 1.5$ for $\mathrm{B}_0 < 1\,\mathrm{T}$. Poloidal Mirnov arrays of 20 coils are located in two of the 3 toroidal periods. Observations of MHD activity during RF plasma production are presented for magnetic configurations of both positive (stellarator-like) and negative (tokamak-like) shear, including configurations where the sign of the shear reverses. Data mining techniques, SVD, wavelet and Fourier analysis are applied. Signals range from highly coherent, often multi- frequency, to near broad band ($\delta\mathrm{f}/\mathrm{f} \sim 0.02$–$0.5$), and in many cases, observed frequencies exhibit Alfvénic scaling with electron density variation. Clear structure is found near resonant transforms, and evidence is given for a relationship of the observed frequencies to the lowest stationary point of the Alfvén resonant frequency for low order resonances such as $\iota \sim 4/3$ and $5/4$. Density fluctuation profiles, measured by a fast sweeping interferometer (2 ms, 200 GHz) indicate that these modes are large amplitude ($\delta\mathrm{n_e}/\mathrm{n_e} < 0.05$) and may extend beyond radii at which the Alfvén resonance condition is met. Mode structure and possible localisation are investigated, and mode numbers up to $\mathrm{m} \sim 6$ are found, including values consistent with the above resonant transforms ($\mathrm{m} = 3, 4$). Fast particle driving sources are under investigation, and include both fast electrons and minority heated H ions. The complex plasma shape creates problems in analysis, possibly broadening the poloidal mode spectrum beyond expected toroidal coupling effects, and the variable plasma – probe distance makes mode localisation difficult to interpret. This work demonstrates that the well diagnosed and finely controlled H-1 plasmas, when coupled with selective ion and electron rf heating, provides a productive environment in which to develop integrated models of Alfvén eigenmodes, to contribute to assessment of Alfvénic activity in modern-day and planned extremely energetic fusion plasmas.

EX/P6-4 · Evidence for Anomalous Effects on the Current Evolution in Tokamak Operating Scenarios

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Abstract: In advanced tokamak (AT) modes of operation [T.S. Taylor, Plasma Phys. Control. Fusion 39, B47 (1997)], the desired current density profile is fully sustained using a combination of external, noninductive current drive sources along with the internally driven bootstrap current density (J_{BS}). These AT modes impose significant demands on the control of the pressure profile to properly align J_{BS} along with external sources to achieve the desired current density profile for optimal stability. Alternatives to the usual picture of AT discharges are those that form when anomalous effects alter the plasma current and pressure profiles, and/or those discharges that achieve stationarity through self-organizing mechanisms, so that the desired AT characteristics are maintained without external current-profile control. Two regimes that exhibit this behavior are those with evolution of the safety factor (q) to a stationary profile with $q_0 > 1$, and those with a deeply hollow current channel. Operating scenarios with high fusion performance at low current, and where the inductively-driven current density achieves a stationary configuration without sawteeth, may enhance the neutron fluence per pulse on ITER and future burning plasmas. Hollow current profile discharges exhibit high confinement and a strong “box-like” internal transport barrier (ITB). In assessing the effects on the q profiles due to anomalous conductivity, we use a combination of time-dependent analysis of experimental data to compare and contrast with modeling of the discharge evolution. Determination of the underlying physical processes leading to these anomalous effects is important for scaling of current experiments to potential application in future burning plasma devices.
* Work supported by US DOE under W-7405-ENG-48, DE-FC02-04ER54698, DE-AC05-00OR22725, and DE-AC02-76CH03073.

EX/P6-5 · Power dependence of density and current drive efficiency in full LHCD plasmas on TRIAM-1M

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Abstract: Full lower hybrid current drive (LHCD) plasmas on TRIAM-1M have been investigated in the wide ranges of line-averaged electron density, $\overline{n}_e = 0.1$–$4.7 \times 10^{19}\mathrm{m}^{-3}$, plasma current, $I_p = 15$–100 kA, and injected power, $P_{LH} = 0.003$–0.2 MW. Strong gas-puff makes a termination of the discharge in high density region, where the density is still much lower than the limit predicted by wave propagation characteristics. The achieved and current drive efficiency have a significant relation to P_{LH}, and their power dependences are close to the one predicted by a model derived from the balance between the energy confinement and current drive efficiency of the LHCD scalings. The termination in high density region can be explained by the proposed model.

EX/P6-6 · Spherical Tokamak Startup and Formation by ECH without Central Solenoid on LATE

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Abstract: Removal of central solenoids (CS) from the fusion reactor based on Spherical Tokamak (ST) concept is essential and expected to reduce the construction cost greatly. Among start-up scenarios without CS, use of electron cyclotron heating (ECH) is promising because breakdown and current initiation can be fulfilled simultaneously and the required equipment for microwave power injection is only a small launcher remote from the plasma. The purpose of the LATE device is to demonstrate the startup of ST plasmas by ECH without CS and to establish the physical bases. The initial closed field equilibrium is formed spontaneously under a steady weak vertical field through the current jump which bridges the gap between the open field equilibrium maintained by a pressure-driven current and the closed field equilibrium at a larger current. It may be due to the efficient current generation by the asymmetric confinement of electrons along the field line appearing upon the transition of field topology. After the formation of the initial closed field equilibrium, plasma current Ip is increased by increasing both the microwave power and external vertical field for equilibrium at larger plasma current, up to 8.1 kA by 2.45 GHz, 35 kW, 2 sec pulse and 12.1 kA by 5 GHz, 130 kW, 60 ms pulse. In both cases, the final value of Ip amounts to 13.5 % of the total toroidal coil current flowing through the center post and the magnetic field line on LCFS has a large pitch angle and shows the characteristics of the ST configuration. The bulk electron density is more than the plasma cutoff density. The profiles of plasma current, visible light emission and soft X-ray emission encompass the 2nd and 3rd EC resonance layers. These facts suggest that the injected microwave is mode-converted to electron Bernstein wave (EBW) and then absorbed at EC harmonic resonances. Hard X-ray pulse height analysis suggests that a directional high energy electron tail is developed as plasma current increases, which may be generated by the EBW heating with high refractive indices along the magnetic field.

EX/P6-7 · Observation of spontaneously excited waves near the ion cyclotron range of frequency on JT-60U

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Abstract: In magnetically confined plasmas, fluctuations in the ion cyclotron range of frequency (ICRF) may be driven by the presence of non-thermal ion distribution. On the GAMMA 10 tandem mirror, plasmas with a strong temperature anisotropy have been formed. In a typical discharge, Alfvén-ion-cyclotron (AIC) modes are spontaneously excited due to the strong temperature anisotropy. On the while, in fusion-oriented devices with a toroidal configuration, neutral beam (NB) injections are commonly used to create high performance plasmas. Resultant high-energy ions are trapped in the local mirror configuration and will form the velocity distribution with the strong anisotropy. In burning plasma experiments on JET and TFTR, ion cyclotron emissions (ICEs) have been observed in the ion cyclotron frequency and its higher harmonic regions. To study the relation among the AIC modes, ICEs and beam-driven electrostatic instabilities in ICRF, and the basic physics in the magnetically confined plasmas with non-thermal energy distribution is the main motivation of this work. In this paper, experimental observations of spontaneously excited waves in ICRF on JT-60U are described. Two types of magnetic fluctuations are detected: one is due to high energy D ions from NB injections and the other is due to fusion products (FPs) of ^{3}He and T ions. This paper reports the first measurement of the toroidal wave number, k-para, of the excited waves. The fluctuations due to D ions have small k-para and will behave as electrostatic waves. On the while, the measurement of finite k-para supports the excitation of the magnetoacoustic cyclotron instability is the possible origin of FP-ICEs. It is also confirmed that frequency peaks due to FPs are sometimes split into doublet shape as observed in JET experiments. The phase differences of both peaks are measured and indicate that two waves are traveling in both toroidal directions. Both beam-driven ICEs and FP-ICEs are observed and the wave structures are clearly measured on JT-60U.

EX/P6-8 · Fast Ion-Driven MHD Instabilities and Consequent Fast Ion Losses in the Compact Helical System

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Abstract: Interaction between fast ions and kinetically driven MHD modes is one of the key physics issues in current fusion experiments. In particular, the energetic particle modes (EPM) and toroidicity-induced Aflvén eigenmodes (TAE) may lead to anomalous loss of energetic alpha particles produced by the D-T reaction in a future burning plasma. In this work, impacts of fast ion-driven MHD instabilities on fast ion transport and consequent losses are studied in the Compact Helical System (CHS). The fast ion experiment in CHS is characterized by the combination of a variety of fast particle diagnostics, i.e. scintillator-based lost fast ion probes (LIP) at outboard and inboard sides, a charge exchange neutral particle analyzer (NPA) whose viewing angle is horizontally variable and a directional Langmuir probe. Correlated with EPM bursts ($m = 3/n = 2$), pulsed increases are observed in LIP, NPA and H_α signals. LIP shows that enhanced fast ion losses due to EPMs begin after the mode amplitude reaches the maximum, suggesting that the EPM is stabilized after an expulsion of fast ions. NPA indicates that co-circulating fast ions having energy close to injection energy of NB drive EPMs and are expelled by EPMs. Fast ions are expelled to the outboard side due to EPMs and the loss rate is steeply augmented with the increase of the fluctuation amplitude. The effect of TAE instabilities on fast ion confinement is also of a concern as well as that of EPM. One of the new results of recent experiments is an observation of large repetitive anomalous fast ion losses due to $n = 1$ TAEs ; the gap for these modes is formed by a coupling of $m = 2$ and 3 poloidal modes. Assuming an internal mode structure of EPM activities motivated by heavy ion beam probe data, efforts to simulate the fast particle loss have been made using the improved DELTA5D code. The result indicates that a perturbed field whose amplitude is consistent with the CHS experiment can rapidly increase the fast ion losses as compared to losses in the stationary magnetic field. It was also shown as seen in the experiments that fast ion losses were strongly enhanced as the mode amplitude increases as seen in the experiments.

EX/P6-9 · Recent progress in JET on Heating and Current Drive studies in view of ITER

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Abstract: This paper summarizes recent results obtained on JET for optimizing heating and current drive systems, in view of ITER. Fast T ions, injected during off-axis beam heating show anomalous transport to the plasma centre for discharges with ITER relevant $q_{95} \sim 3$. This is not the case for discharges with high $q_{95} \sim 8$. Several possible reasons for this anomaly have been investigated, but so far the results are not explained. Similar effects have been seen on other tokamaks. The results of this study could show to be of importance to improve predictions for the off-axis NBI current drive for ITER. Fast fusion born alpha particles in ITER could induce sawteeth with long periods, which in turn could create seed islands large enough to trigger NTMs. Recent JET experiments demonstrated that fast ion induced long sawteeth can be destabilised with ICCD applied close to the $q = 1$ surface. This driven current significantly increases the shear at the $q = 1$ surface (as confirmed by simulations with the SELFO code) and leads to sawtooth destabilisation. This study shows also that any method capable of driving a current of sufficient magnitude around the $q = 1$ surface would be useful for ITER. Polychromatic ICRF heating should allow better ion heating on ITER compared to monochromatic heating. This has been tested in JET using ^{3}He and H minority heating in D plasmas with different frequencies simultaneously excited on the 4 JET ICRF antennas. Fast ion tail temperature and energy content are found to be lower with polychromatic heating; smaller-amplitude and shorter-period sawteeth, and higher ion to electron temperature ratios have been observed. Inverted heating scenarios are one of the few options for ICRH heating during the hydrogen phase of ITER. Exploration of such scenarios at JET showed the importance of small amounts (1–2%) of C impurity ions, as their presence prevents any D minority heating. Results are confirmed with TOMCAT and CYRANO simulations. Increasing the SOL electron density is one of the possibilities to improve LH wave coupling and this has been done in JET with gas puffing close to the LH launcher. To extrapolate this method to ITER, the detailed mechanisms have been studies with a modified version of EDGE-2D. Results show that the measured SOL electron density can be reproduced, assuming an increase of the far SOL electron temperature due to LH heating.

EX/P6-10 · Oscillating Field Current Drive in the MST Reversed Field Pinch

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Abstract: Oscillating Field Current Drive (OFCD) is an inductive current drive method which could sustain a DC plasma current without magnetizing flux accumulation. Its application in the reversed field pinch (or possibly other magnetic configurations) could provide the means to sustain a steady-state reactor plasma with Ohmic current drive efficiency. To create OFCD, oscillatory (audio-range) toroidal and poloidal inductive loop voltages are applied with a relative phase that is predicted optimum 90 degrees by a simple magnetic helicity balance. We report MST experiments in which 10% of the plasma current is driven by OFCD. The magnitude of the driven current agrees with theoretical expectations, but interestingly the maximum current does not occur for a relative phasing of the oscillators which produces maximum helicity injection. This might be explained by a combination of confinement or current profile changes. The OFCD current drive efficiency is 0.1 A/W. This is about the same as that for conventional Ohmic induction, which provides the balance of current drive in these experiments. Magnetic fluctuation amplitudes from MHD tearing modes are found to depend on the relative phase, with the minimum amplitudes occurring with maximum current drive. For poloidal mode $m = 0$ fluctuations, the time-average fluctuation amplitude can be smaller with OFCD than without. These results establish the capability of OFCD to drive a portion of the plasma current, motivating experiments at higher power to produce a larger fraction of current drive by OFCD.
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EX/P6-12 · Interactions of RF Antennas with the Edge Plasma in Tore Supra Steady-State Discharges

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Abstract: High frequency antennas operating in the Ion Cyclotron Range of Frequency (ICRF) or the Lower Hybrid Range of Frequency (LHRF) are exposed to high heat flux deposition from the plasma. The gap between the plasma and the last closed flux surface has to be as large as possible while still ensuring

good wave coupling. In ITER it is foreseen to have the antennas embedded in the first wall. This should guarantee low enough heat flux from the convected/conducted power. However specific phenomena arising from the intense oscillating electric field in front of the antennas, leading to the acceleration of ions and/or electrons, are up to now not fully taken into account. The involved mechanisms have to be investigated in order to ensure safe and controlled operation in future fusion devices. On Tore Supra, the total area of the three ICRF antennas and two LHRF launchers are monitored by an infrared (IR) system. The IR data is complemented by calorimetric measurements of the energy extracted by the cooling loops on the antennas and their side limiters. The long pulse capability of Tore Supra allows to assess the plasma-antenna interaction in steady-state plasmas. Both fixed and reciprocating Langmuir probes are used for studying the local density perturbations. In particular, a detailed 2D mapping of the scrape-off layer perturbation in the vicinity of an ICRF antenna has been performed for the first time, using a reciprocating Langmuir probe. The detailed analysis of the IR images of the ICRF antennas yield that at least four different phenomena can arise during combined ICRF-LHRF heating: i) The formation and rectification of an RF sheath. ii) Fast electrons ($\sim$1 keV) accelerated by Landau damping in the near-field of the LHRF antenna. iii) The conducted/convected power from the plasma. iv) Interaction by fast ions ($\sim$500 keV) created by ICRF in the hydrogen minority heating scheme. This identification of heat flux deposition sources has been integrated in the real-time control system of Tore Supra in order to optimize the RF power output without deleterious effects. The different scalings can be used for the design of the ITER RF antennas.

EX/P6-13 · Investigation of Collective Fast Ion Instability-induced Redistribution or Loss in NSTX

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Abstract: The National Spherical Torus Experiment (NSTX) is particularly well suited to investigate fast-ion driven instabilities because large values of the dimensionless parameters v_{fast}/v_{Alfven} and $\beta_{fast}(0)/\beta_{tot}(0)$ required to drive such instabilities occur routinely in neutral beam heated plasmas. The instabilities can be divided into three categories; chirping Energetic Particle Modes (EPM) in the frequency range 0–120 kHz, the Toroidal Alfvén Eigenmodes (TAE) with a frequency range of 50–200 kHz and the Global and Compressional Alfvén Eigenmodes (GAE and CAE, respectively) between 300 kHz and the ion cyclotron frequency. These modes are of particular interest because of their potential to cause substantial fast ion redistribution or loss. Both the volume-integrated neutron and the line-integrated charge exchange neutral particle diagnostics show signal depletion due to fast-ion driven instabilities, but cannot distinguish between fast-ion redistribution or loss. Two recently implemented diagnostics on NSTX, the Motional Stark Effect (MSE) and scintillator Fast Lost Ion Probe (sFLIP), facilitate separation of redistribution and loss effects. Outward redistribution of the core-peaked energetic beam ions modifies the beam-driven current profile and hence the core q-profile. MSE-constrained q-profiles are being used to assess this effect. sFLIP measures the pitch and energy of fast ions that are ejected from the plasma and intercept the wall-mounted probe thus identifying fast-ion loss. For certain H-mode discharges where NPA measurements of the NB energetic ion spectra exhibit MHD-induced fast-ion depletion, the sFLIP data confirm the existence of an ion loss that occurs primarily for passing particles near the NB full injection energy. Observations and TRANSP simulations of a range of fast ion instability-induced redistribution/loss phenomena in NSTX will be presented.
⋆ This research was supported by U.S. DOE contract DE-AC02-76-CH03073.

EX/P6-14 · Control of Non-inductive Current in Heliotron J

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Abstract: Non-inductive current including bootstrap current and EC driven current has been studied in Heliotron J ECH plasmas. The Heliotron J device has a high flexibility to control the magnetic field spectrum, making it possible to investigate the field configuration effect on the non-inductive current. The non-inductive current is accurately measured in the absence of Ohmic current without affecting plasma confinement. The measured toroidal current at medium density ($n_e = 1.0 \times 10^{19}$ m^{-3}) agrees with a theoretical prediction of bootstrap current considering the configuration effect. The ECCD determined by the balance between the Fisch-Boozer and Ohkawa effects is controlled by the EC power deposition and magnetic field configuration. Net free current state has been experimentally demonstrated by compensating the bootstrap current with the EC current.

EX/P6-15 · Behaviour of a Low Frequency Wave in a FRC Plasma

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Abstract: The FRC plasma has a toroidal geometry even if it is produced in a linear solenoidal field. In addition, it has extremely high beta value of 1 at its magnetic axis. These features are desirable for a reactor because higher pressure plasma can be contained in a simple geometry with smaller confining field than in low beta systems. But, different from low beta plasmas, a diversity of waves are not available for the heating; waves with the frequency between the ion gyro frequency and that of the electron are not accessible to the FRC plasma and a low frequency wave is adopted. The wave is applied to the plasma by an external antenna, the main component of which is the compressional mode, and a torsional wave is induced. The induced wave has an apparent peak at a certain radial location just inside the separatrix. The polarization of the wave is left handed outside (low density, high magnetic field side) of this location and inside, the polarization is right handed. These facts signify that this location is the Alfvén resonance layer. It is also observed that the wave penetrates deep into the region where the local ion gyro frequency becomes smaller than the frequency of the induced wave; deeper in the plasma, ion gyro frequency decreases gradually as the magnetic field becomes smaller, and eventually, it becomes smaller than the wave frequency. To this region, usual torsional Alfvén wave is not accessible. This fact, together with the already known fact that the ion heating takes place, may be explained by the kinetic effect.

EX/P6-16 · Experimental Studies and Analysis of Alfvén Eigenmodes in Alcator C-Mod

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Abstract: Reverse shear Alfvén eigen-modes (RSAE) or equivalently, Alfvén cascades (ACs) have been observed in Alcator C-Mod during current ramp experiments in the presence of intense ICRF driven energetic minority tails. These experiments were carried out at ITER relevant densities and magnetic field. Phase contrast imaging (PCI) has been used for detecting the Alfvén eigen-modes, although magnetic pick-up coils have also been used to measure mode numbers. The chirping evolution (increasing in time) of the frequency spectrum has allowed us to deduce the temporal evolution of the reversed shear q-profile. Numerical results from the latest version of the NOVA-K MHD code, including finite pressure (which allows to include the geodesic deformation of the Alfvén continuum) and appropriate filtering, reproduces the experimental measurements, including the minimum frequency. Our experimental data indicates a temperature dependence of the minimum cascade frequency that is in accord with theory. The PCI diagnostic was also used to measure the spatial structure of the Alfvén mode perturbed density. Future development of a PCI simulator using the outputs from NOVA-K will allow quantitative comparisons to be made. The harmonic of the cascade frequency was also observed by the PCI diagnostic. According to theory, the ratio of the measured fundamental and the nonlinearly generated harmonic signals may allow one to obtain independent information about the amplitudes of the internal modes. In a different set of C-Mod experiments a pair of active MHD antennas were used to excite ITER relevant intermediate toroidal mode number ($3 < n < 14$) toroidal Alven eigen-modes (TAE), and their measured weak damping rates indicate that such modes may be excited by alpha particles in ITER. Future studies include measuring the damping rates of these modes in the presence of ICRF generated minority hydrogen tails. We have also examined the possibility of adding PCI detection optics outside the ITER vessel that shares the CO_2 laser interferometer port hardware. Implementing PCI on ITER would greatly extend our capabilities to search for RSAE and TAE in the presence of alpha particles or energetic beam ions.

EX/P6-17 · Study of High-energy Ion Tail Formation with Second Harmonic ICRF Heating and NBI in LHD

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Abstract: Ion cyclotron range of frequencies (ICRF) heating is a good tool for the alpha-particle simulation of a future fusion reactor. In the large helical device (LHD), the second harmonic ICRF heating had been conducted with the target plasma heated by tangential neutral beam injection (NBI). High-energy particle detectors such as the silicon-diode-based fast neutral analyzer (SiFNA), the natural diamond detector (NDD), and the compact neutral particle analyzer (CNPA) were installed in LHD. In addition to the passive measurements, the CNPA is used for the pellet charge exchange (PCX) to measure the distribution of high-energy particles with the combination of a polystyrene pellet injection. ICRF antennas are located

in the vertically elongated plasma section in LHD. In addition to the tangential NBI, a perpendicular NBI was newly installed in the horizontally elongated plasma section in 2005. Population of 'seed' particles with the large Larmor radius was small in the case of second harmonic ICRF heating with tangential NBI. However, enhancement of high-energy ion tail formation was observed in the presence of the 'seed' particles having the large Larmor radius supplied by the perpendicular NBI. Moreover, it was clarified that the high-energy deeply trapped particles exist near ion cyclotron resonance layer by the PCX measurement. GNET code is a drift kinetic equation solver in 3D space and 2D velocity space based on the Monte Carlo technique. The spectrum of high-energy ion tail was consistent with the calculation by GNET code in the region of Larmor radius of alpha-particles normalized by the plasma minor radius in the heliotron reactor. This experimental result supports the GNET simulation that alpha-particles will be well confined in the heliotron reactor.

EX/P6-18 · Effective Heating and Improved Confinement Transition in Lower Hybrid Experiment on FT-2 Tokamak

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Abstract: The paper presents the new data of the effective Lower Hybrid Heating (LHH) which has been studied on FT-2 tokamak. The main goal of these experiments is to analyze the plasma core parameters, where the Internal Transport Barrier (ITB) during auxiliary LHH and than L – H transition after RF pulse has been observed. The task of this paper is both (1) the presentation of the characteristic features of the edge parameters changes, which was observed at $P_{LHH} = 90 \div 100$ kW when the edge transport barrier is formed. (2) the presentation of the first data with enhanced RF power $P_{LHH} \approx 2P_{OH} = 180$ kW. The considerable attention in the recent experiment researches is given to statistical analysis of fluctuation properties of the edge plasma's characteristics. The paper presents such data and illustrates how change of Probably Distribution Functions (PDFs) of fluctuation induces flux oscillations measured near of the LCFS on HFS and LFS of the plasma core. The new experimental data connected with fluctuation induces flux parameters changes are presented and discussed. In development of the LHH experiment the enhanced additional LH heating $P_{LHH} \approx 2P_{OH} = 180$ kW is applied. Useful increase electron and ion temperatures the same as plasma density are observed. Fast drop down of the H_β spectral line intensity as well as periphery radiation losses decrease indicates at L – H transition. The experiment demonstrates the L – H transition with suppression of the radial fluctuation induced flux near LCFS which observed just with LHH switch on. The new experimental data are presented and discussed.

EX/P6-19 · Development of Alfvén Spectroscopy in Advanced Scenarios on JET

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Abstract: Alfvén Eigenmodes (AEs) driven by fast ions are routinely seen in advanced tokamak plasmas heated by ICRF waves and/or by NBI. Measurements of low-amplitude AEs represent an attractive form of MHD spectroscopy which can identify the existence of reversed magnetic shear and provide data on the evolution of q_{min} from the experimentally observed AEs. A significant improvement in detecting Alfvén Cascades (ACs) and Toroidal Alfvén Eigenmodes (TAEs) in JET discharges achieved by using O-mode reflectometry with a probing frequency above the cut-off frequency (interferometry regime) is described. The possibility of using this technique for the detection of AEs on ITER is discussed. ACs driven by sub-Alfvénic NBI were detected on JET using the O-mode interferometry. The AC excitation by NBI was explained by the geodesic acoustic effect, which allows super-sonic, but sub-Alfvénic, NBI ions to resonate with the ACs. The theory of ACs has been extended to include thermal plasma effects and now promises to provide kinetic information on plasma pressure and electron-to-ion temperature ratio. Transitions from TAEs to ACs have been observed on JET when an Internal Transport Barrier (ITB) forms. A temporal evolution of the current density profile consistent with the observed TAE-to-AC transition has been identified using Alfvén spectroscopy. It was found that this transition is caused by a reversal of the magnetic shear, mainly associated with the bootstrap current. One of the most intriguing problems in advanced tokamak scenarios is the correlation between the ITB triggering events and low rational values of q_{min}. Using O-mode interferometry for ACs, it was possible to identify the exact time of q_{min} = integer events via grand ACs, which mark the appearance of integer q_{min} surfaces. It was found that in majority of JET discharges the ITB triggering event is observed before grand AC indicating strongly that the ITB triggering event is associated with the depletion of rational magnetic surfaces just before q_{min} reaches an

integer value, rather than with the presence of an integer q_{min} value itself. The correlation between ITB triggering events and grand ACs has been found to exist in JET plasmas with high densities, showing that the timing of ITB triggering from AC diagnosis may facilitate scenario development in machines with high densities (C-MOD, ITER).

EX/P6-20 · Plasma Dynamics with Second and Third Harmonic ECRH on TCV Tokamak

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Abstract: Intense electron cyclotron resonance heating (ECRH) is employed on the TCV tokamak both in second- and third-harmonic X-mode (X2 and X3). The plasma behaviour under such conditions is driven largely by the electron dynamics, motivating extensive studies on TCV of the heating and relaxation phenomena governing both the thermal and suprathermal electron populations. In particular, the dynamics of suprathermal electrons is intimately tied to the physics of X2 current drive (ECCD). The absorption of X3 radiation has been found to be enhanced by the presence of these high-energy electrons, which can also be generated by the X3 wave itself. Suprathermal electrons are also generated by the strong electric fields created by the magnetic reconnection events accompanying sawtooth crashes. Electrons play a crucial role in high-density plasmas where indirect ion heating can be achieved through ion-electron collisions. Such plasmas can be heated by the X3 system, thanks to its high density cut-off, enabling in particular H-mode studies with strong electron heating. Experiments have been performed by applying X3 heating to an Ohmic H-mode target, resulting in an ELM-free regime with constant density and stored energy, and elevated D_alpha and energy confinement, as well as increased ion temperature and rotation velocity. The toroidal β reached 2.5% ($\beta_N \sim 2$) in these plasmas, 70% of the ideal limit. ECRH is also an optimal tool for manipulating the electron distribution function in both physical and velocity space: examples in TCV are the internal transport barriers controlled by current-profile inversion and Fisch-Boozer ECCD, and the energy selectivity of vertically launched X3 waves. Fundamental studies of the energetic electron dynamics have been performed in TCV through periodic, low-duty-cycle bursts of ECRH, with negligible average power injection, and with electron-cyclotron-emission measurements coherently averaged over a long stationary period to improve the statistics. The characteristic times of the dynamical evolution, affected by rf diffusion and collisional scattering and slowing-down in velocity space as well as by cross-field transport in physical space, are clearly revealed by this technique.

EX/P6-21 · Fast Wave Current Drive and Direct Electron Heating in JET ITB Plasmas

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Abstract: Experiments with Fast Wave Current Drive, FWCD, and heating have been carried out in JET Internal Transport Barrier (ITB) discharges with strongly reversed magnetic shear. In order to maximize the current drive efficiency and increase the electron damping, and at the same time modifying the current profile in the transport barrier, hot low density ITB plasmas with strongly reversed magnetic shear, close to current hole, were created with Lower Hybrid Current Drive. It was difficult to strongly modify the central plasma current, even though the calculated current drive efficiency in terms of ampere per watts absorbed by the electrons was fairly high, 0.07 A/W, because of: the strongly inductive nature of the plasma current due to the high electric conductivity; the interplay between the fast wave driven current and the bootstrap current, which, due to the dependence of the bootstrap current on the poloidal magnetic field, decreases the bootstrap current as the driven current increases; and parasitic absorption of the waves that decreased the power absorbed by the electrons. The power absorbed by the electrons was measured with a power modulation technique and the associated fast wave current drive calculated. Current diffusion simulations using the JETTO transport code, assuming neoclassical resistivity, were then carried out to calculate what changes to the plasma current profile could be expected from the current drive. The simulations showed a much slower response to the current drive compared to the measured central current densities suggesting a faster current penetration in the experiments than expected from neoclassical theory. Whereas the direct electron heating by fast magnetosonic waves using dipole spectra has proven to be an effective method to heat electrons in high-temperature ITB plasmas, even for a single pass damping of only a few percent, the heating in FWCD experiments with $+90^\circ$ and -90° antenna phasings were, for similar single pass damping as for the dipole, strongly degraded by parasitic losses, and with a heating efficiency of about half that of the dipole. Observations supporting that the losses are primarily caused by the presence of rectified RF-sheath potentials come from the large differences in performance and in Beryllium-II and Carbon-IV line radiation intensities between the dipole and $\pm 90^\circ$ phasings.

EX/P6-22 · Electron Bernstein Wave Heating Experiments on MAST

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Abstract: Burning plasma spherical tokamaks (ST) rely on off-axis current drive and non-solenoid start-up techniques. Electron Bernstein waves (EBW) may provide efficient off-axis heating and current drive (CD) in high density ST plasmas. EBW may also be used in the plasma start-up phase due to the fact that EBW absorption and CD efficiency remain high even in relatively cold plasmas. EBW studies on the Mega Amp Spherical Tokamak (MAST) can be subdivided into four separate subjects: thermal EC emission observation from overdense plasmas; EBW modeling; proof-of-principle EBW heating experiments with the existing 60 GHz gyrotrons; EBW assisted plasma start-up at 28 GHz. These studies are also aimed at determining the potential for a high power EBW for heating and CD on MAST. The optimum choice of frequency and launch configuration for EBW heating and CD is a key issue for future applications in MAST. A major advance in the integrated modelling of EBW, involving a suite of codes, has been achieved. The EBW excitation in the plasma is first considered as a full wave 1D mode-coupling problem in slab geometry. Then propagation of the EBW is computed using an EBW ray-tracing code, which implements the fully electromagnetic, hot plasma dispersion function. The ray tracing data are then used in the BANDIT code in a self-consistent, relativistic 3D Fokker-Planck treatment to calculate the heating and driven current profiles. The modelling suggests the preferable operating frequency for efficient EBW heating and CD must be in the range of the fundamental EC resonance or its lower harmonics. This requires a high power RF source in the range of 16–28 GHz. Proof-of-principle EBW heating experiments have been conducted on MAST using the existing 60 GHz, 1 MW complex. EBW heating effects have clearly been observed for the first time in an ST. A 28 GHz start-up system (200 kW, 40 ms) is being commissioned. According to our modelling EBW can generate plasma current ∼100 kA during the plasma start-up phase giving the prospect of a fully non-inductive plasma start-up scenario.

EX/P6-23 · Bi-directional Lower Hybrid Current Drive and Electron Cyclotron Counter Current Drive Experiments in Full Current Drive Plasma in TRIAM-1M

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Abstract: In order to investigate the spectrum gap problem in LHCD and controllability for the current profile by means of counter current drive in full current drive plasma, experiments by combination of backward (BW) propagating LHW and/or ECW have been performed in a plasma sustained by forward (FW) propagating LHW. Three ctr-CD aspects have been investigated for relativistic resonance with respect to the power ratio of backward (P_{BW}) to forward (P_{FW}) LHWs; 1) a clear reduction of co-I_{CD} for $P_{BW}/P_{FW} < 0.8$; 2) a rapid positive change in co-I_{CD} and broadening in j(r) for $P_{BW}/P_{FW} > 0.8$ [H. Zushi et al., Nucl. Fusion 41 (2001) 1483]; and 3) a large positive change in co-I_{CD} by ctr ECW into FW LHCD plasma and further changes to negative value in co-ΔI_{CD} depending on P_{BW}/P_{FW} of LHW.

EX/P7-1 · Physical model assessment for the energy confinement time scaling in stellarators

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Abstract: Global scaling laws for the energy confinement of stellarators have the tendency to hide both configuration effects and physics dependencies if derived from engineering parameters directly. For the International Stellarator Scaling ISS04, these physical dependencies were considered by introducing configuration dependent normalizations. The resultant subgroups of data in the International Stellarator Confinement Database, however, are assumed to exhibit similar physics and can be tested against physical models. These tests are addressed here. They represent significant extensions of the scaling laws since a comparative revision also indicates validity ranges of scaling properties. The ranges of trustability are relevant for reactor extrapolations of 3-d confinement concepts and intermachine assessments are employed for validation. Examples of available data sets are configuration scans in LHD and data from high β discharges in W7-AS. For an assessment of high density operation and high confinement modes result from HDH and H-mode discharges, respectively. The strategy for model assessment in this study is twofold: First, systematic predictive transport modeling of Wendelstein 7-X is performed. As a stellarator specific example, density scans in electron cyclotron heated electron root plasmas are investigated and compared to results from W7-AS. A predictive configuration factor is then explored. Second, Bayesian model comparison techniques are employed for a systematic model assessment of scaling laws obeying invariance principles. Since high β operation data are available in single devices, a comparative study reveals the validity of the physical scaling laws which is to be compared with a global $\tau_E \sim \langle\beta\rangle^{-0.17}$ scaling from ISS04.

EX/P7-2 · On the link between edge momentum redistribution and turbulence in the TJ-II stellarator

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Abstract: This paper presents a view of the experimental results and progress in the investigation of the physics of sheared flow generation in the edge region of the TJ-II stellarator. Experiments in the TJ-II stellarator have shown that the generation of spontaneous perpendicular sheared flows (which organize itself to be close to marginal stability with fluctuations) requires a minimum plasma density. Near this critical density, the level of edge turbulent transport and the turbulent kinetic energy significantly increases in the plasma edge. The development of edge sheared flows, first reported by means of Langmuir probes measurements, has been recently 2-D visualized by means of Ultra Fast Speed cameras, reflectometry and HIBP measurements. Bright, long-living structures are frequently seen with a spatial extent of few centimetres. Those structures show predominant poloidal movements with typical speed in the range of 10^3-10^4 m s^{-1} in agreement with the expected E×B drift rotation direction. As sheared flows are developed turbulence structures became stretched which can be interpreted as a modification in the perpendicular degree of turbulence anisotropy. Turbulence suppression during biasing-induced improved confinement regime is clearly reflected on the number of turbulent structures, which turns out to depend on wave-number and polarity. Experimental results also show significant turbulent parallel forces at plasma densities above the threshold value to trigger perpendicular E × B sheared flows. Radial profiles of the parallel-radial Reynolds stress component, proportional to the cross-correlation between parallel and radial fluctuating velocities, have been measured in the plasma boundary region of the TJ-II stellarator. Experimental results show the existence of significant parallel turbulent forces at plasma densities above the threshold value to trigger perpendicular E×B sheared flows. This finding provides the first experimental evidence of the role of parallel turbulence forces on edge momentum redistribution in fusion devices.

EX/P7-3 · Study of plasma potential evolution in ECRH and NBI plasmas in the TJ-II stellarator

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Abstract: The Heavy Ion Beam Probe (HIBP) diagnostic is used in TJ-II stellarator to study directly plasma electric potential profiles with spatial and temporal resolution. Low density ECRH (n = 0.5–1.1×10^{13} cm^{-3}) plasmas in TJ-II are characterised by core positive plasma potential of order of 500–1000 V and positive electric fields. Edge radial electric fields remain positive at low densities and became negative at the threshold density that depends of plasma configuration. NBI plasmas are characterized by negative electric potential in the full plasma column and negative radial electric fields. The density rise during the NBI phase is accompanied by the decay of core plasma potential. When density is getting the level of n = 2.0–2.5 × 10^{19} cm^{-3}, the potential stops its evolution and remains constant. These observations show the clear link between plasma potential and plasma density. Edge quasi-coherent fluctuations have been observed in some configuration windows when plasma density/heating power are above a threshold in ECRH heated plasmas. The appearance of those modes in specific configuration windows suggests the role of low order rationals in the plasma edge. The existence of threshold densities and heating power points out the role of threshold gradients to trigger quasi-coherence modes. When rationals move towards the plasma core, the modes are clearly seen in ECE emission and in HIBP secondary current and potential signals. These quasi-coherent have been connected with the development of electron internal transport barriers (e-ITB).

EX/P7-4 · Recent Results from L-2M Stellarator

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Abstract: This report presents results of recent experiments on ECR plasma heating and confinement in the L-2M stellarator. The experiments were carried out with the use of boronization of the stellarator

vacuum chamber. This made it possible to change the conditions of the plasma experiment substantially, reducing the radiation loss power to 10–15% of the power input. Under these conditions, at a heating power above 120 kW, a jump (pedestal) by 100–150 eV appears in the electron temperature at the separatrix. When operating with a high heating power and plasma density above n > 10^{19} m^{-3}, we observed a spontaneous, sudden and very fast (about 0.2 ms) drop in the plasma energy by 5–10% of its value, measured by the diamagnetic diagnostics. After this fast drop, the plasma energy slowly restores or even increases somewhat. The probability of such events increases with increasing density and heating power. The fast change in the plasma parameters occurs only at the edge while not influencing the core plasma parameters. The fast drop is followed by a slow increase in the plasma energy, which correlates with an increase in the plasma density, which may be considered as a demonstration of somewhat better confinement of particles. The report presents the results of ion temperature measurements from a Doppler broadening of emission lines of impuity ions and their comparison with calculations using a TRANSZ transport code. Results of measurements of the microwave scattering by plasma fluctuations are discussed in the context of the relation between anomalous and neoclassical transport. The electron energy distribution function was reconstructed from the measured SXR spectra and was found to be essentially non-Maxwellian at low plasma densities.

EX/P7-5 · Study of Ion Viscosity by Spontaneous L-H Transitions under Marginal Hot Cathode Biasing in the Tohoku University Heliac

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Abstract: Neo-classical theories explain that the nonlinearity of the ion viscosity plays the important role in the bifurcation phenomena of the L-H transition observed in large tokamaks and stellarators. In Tohoku University Heliac (TU-Heliac) the effects of the ion viscosity maxima on the transition to an improved confinement mode have been experimentally investigated by the externally controlled J × B driving force for a poloidal rotation using the hot cathode [S. Kitajima et al., Nucl. Fusion 46, 200 (2006)]. One of further extended works is to clarify the effect of magnetic Fourier components on the neo-classical viscosity. In TU-Heliac the viscosities at the H-L transition point in various magnetic configurations have been evaluated experimentally by sweeping the J × B driving force and the experimental results showed that the viscosity maxima qualitatively agreed with neo-classical predictions [H. Takahashi et al., J. of Plasma and Fusion Research SERIES 6, 366 (2003)] and the hysteresis feature in a driving force were also observed [H. Takahashi et al., Plasma Phys. Control. Fusion 46, 39 (2006)]. In these current-sweep-biasing experiments it was difficult to explore precisely the time response of plasma parameters for the transition, because these parameters were actively changed by the electrode current. In order to find a direct trigger on the spontaneous L-H transition observed in large devices, it is appropriate to research plasma parameters at the transition point. In TU-Heliac the spontaneous L-H transitions appeared with delay times under the marginal biasing condition, which was lower condition than that required for the transition and was precisely tuned by the biasing voltage and heating power for a hot cathode. The ion viscosities were estimated in various magnetic configurations at the spontaneous transition point under the marginal hot cathode biasing. The critical viscosity experimentally estimated in different magnetic configurations agreed with the neoclassical predictions within a factor 2. Although the transition points were spread over a wide time range, poloidal Mach numbers at the transition point concentrated near the viscosity maxima predicted by the theory.

EX/P7-6 · Transitions to improved core electron heat confinement triggered by low order rational magnetic surfaces in the stellarator TJ-II

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Abstract: In stellarator devices, transitions to improved core electron heat confinement are established in conditions of high ECH power density and are characterized by peaked electron temperature profiles and large radial electric field and shear in the inner plasma region [1–7]. These transitions have been often referred as Neoclassical or electron Internal Transport Barriers (N-ITB or e-ITB) [1, 2, 4, 6] or as “electron root” feature [3]. TJ-II experiments show that e-ITBs can be easily triggered by positioning a low order rational surface at the plasma core region [6]. The rational surface contributes to the outward electron flux that creates a locally strong positive radial electric field [7]. Quasi-coherent modes are observed where the E_r × B shear flows develop at the e-ITB formation [8]. These quasi-coherent modes, characterized using HIBP and ECE diagnostics, are localized close to the radial location of the barrier foot and vanish

as the barrier is fully developed. A possible cause of the modes may be attributed to MHD instabilities. In that case, the sheared radial electric field developed at the e-ITB formation may act as the stabilizing mechanism. Experiments with different low order rationals (3/2, 4/2, 5/3...) show a dependence of the threshold density (and also of the barrier quality) on the order of the rational. The island width may be the relevant parameter in the modification of the radial electric field induced by the rational surface. Experiments are in progress to study the influence of the island width keeping the order of the rational surface and changing the magnetic shear by the induction of Ohmic current. Preliminary results indicate that the rational surface 3/2 triggers the transition within the accessible range of OH current, that is, within a certain range of magnetic shear. Modulation experiments have been performed to explore transport properties of plasmas with improved electron heat confinement.

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EX/P7-7 · Formation of Radial Electric Field Shear at Boundary of Magnetic Island in LHD

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Abstract: Active control of a radial electric field shear by a combination of pellet injection and a non-rotating magnetic island (n/m = 1/1) induced by external perturbation coils is demonstrated in the Large Helical Device (LHD). Six pellets are injected to produce negative radial electric field of ion root in an NBI sustained plasma. When there is a magnetic island, the large negative radial electric field is observed except for inside of the magnetic island, where the radial electric field vanishes because of damping of the poloidal flow, and large radial electric field shear is observed at the boundaries of the magnetic island. The radial electric field shear reaches 0.3 MV/m^2 and is sustained by the repetitive pellet injection while the no significant radial electric field shear is observed in the plasma without the magnetic island. The gradient of an ion pressure in the plasma with the magnetic island is almost twice that in the plasma without the magnetic island associated with the radial electric field shear formation near the magnetic island within a few centimeters. The improvement of an ion transport is observed near the boundary of the magnetic island. The combination of a magnetic island and pellet injection is considered to be a useful tool to control radial electric field shear and an ion energy transport near the magnetic island.

EX/P7-8 · Studies of plasma confinement in GOL-3 Multiple Mirror Trap

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Abstract: An idea of a multi-mirror fusion reactor originates from the early 70s [G. I. Budker, V. V. Mirnov, and D. D. Ryutov, JETP Letters, 14, 212 (1971) 1]. In such a system, if plasma density is high enough, its expansion along the magnetic field becomes diffusion-like due to effective "friction force" between the magnetic field and plasma particles. The main advantages of this approach are technical simplicity, absence of density and b limits. The experiments on multi-mirror plasma confinement are carried out at GOL-3 device in Novosibirsk. In present configuration of the device, plasma with a density of $10^{20} - 10^{22}$ m^{-3} is confined in a 12-meter-long solenoid, which comprises 55 corrugation cells with mirror ratio $B_{max}/B_{min} = 4.8/3.2$ T. The plasma in the solenoid is heated up to 1–2 keV temperature by a high power relativistic electron beam (∼1 MeV, ∼30 kA, ∼8 μs, ∼120 kJ) injected through one of the ends [A.V.Burdakov, et al., 32nd EPS Conf. on Plasma Physics, paper P5.061 (2005) 2]. In the paper, we discuss the results obtained in studies of plasma heating and confinement in the corrugated magnetic field. In particular, a mechanism of experimentally observed fast ion heating in GOL-3 is discussed. Additionally, we discuss possible application of energetic plasma exhaust from the GOL-3 device to study plasma-wall interaction in the conditions relevant to ITER divertor under major disruption event and operation with ELMs.

EX/P7-9 · Modern Magnetic Mirror Systems. Status and Perspectives

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Abstract: Principles of multi-mirror plasma confinement and gas-dynamic one were proposed in the Budker INP in early and late 70 s, correspondingly. The first experiments have demonstrated a correctness of these approaches. In the paper the status of the experiments on two modern magnetic mirror devices is discussed. At present, the multi-mirror system GOL-3 has 12 m long magnetic system which consists of 55 mirror cells. The magnetic field strength is 4.8 T in the mirrors and 3.2 T in middle planes of each cell. Dense ($n_e > 10^{21}$ m^{-3}) plasma in the solenoid is heated by high current relativistic electron beam (E = 1 MeV, I_b = 30 kA, $\tau_b = 8 \times 10^{-6}$ s) and is confined in the multi-mirror system. Note a simplicity of this method of plasma confinement (axial symmetry of the magnetic configuration), practicability of high β (order of 1 or even more). The present day parameters are: $T_e \approx T_i \approx 2$ keV. Such plasma is sustained during $\tau_E \sim 1$ ms and there are no physical limitations to increase them. But even with present day parameters a lot of experiments can be made on plasma-wall interaction (study of evaporation, erosion and ionization of wall materials, propagation of ions of these materials along magnetic field lines at long distances, disruption and ELM simulations, etc) using the streams (up to 50 MJ/m^2) flowing out along the axis. The second system is Gas Dynamic Trap (GDT). It is an axisymmetric mirror system with very high mirror ratio (R $\sim 10^2$). The distance between mirrors, L = 7 meters and magnetic field in them is 15 T. Recently it was demonstrated that the GDT concept based on use of "warm" plasma and oblique injection of D-T neutral beams can lead to creation of a simplest plasma neutron source with high density 14 MeV neutron flux reaching up to 2 MW/m^2 on the limited part (order of 1 m^2) of the device. The principles of the GDT operation have been already demonstrated at moderate parameters of plasma and NB injection. In particular, stable confinement of plasma with β as high as $\sim$0.7 was demonstrated in the GDT device. The goal of nearest experiments is to increase T_e of "warm" target plasma up to T_e = 300 eV, and the density of fast (up to 25 keV) ions up to 5×10^{19} m^{-3}. As a result, a practicability of the source with intermediate neutron flux density of 0.5 MW/m^2 will be demonstrated.

EX/P7-10 · Mirror Stabilization Experiments in the Hanbit Mirror Device

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Abstract: The Hanbit device is a magnetic mirror machine which has a central cell, one anchor cell and one plug cell plus associated vacuum chambers. It is about half of the original TARA mirror device from MIT. The Hanbit device has been involved in a series of experiments on stabilization of the MHD flute type mode. Earlier work showed that it was possible to stabilize the m = −1 flute type MHD instability with RF power near the cyclotron resonance. This stability has been attributed to the sideband coupling process. We have now undertaken investigations to see if a divertor and the Kinetic Stabilizer (KS) of R. F. Post can stabilize the MHD instability. Divertors were used previously in experiments on the TARA mirror device and the HIEI mirror device. The Hanbit divertor configuration uses one of the central cell coils with reversed current as the divertor coil and two adjacent coils with increased current to compensate for the field droop and to prevent the field lines from intercepting the bare ICRH antenna. The divertor strongly reduces the m = −1 instability when the null point (x-point) is sufficiently inside the vacuum tank. However, the diverted plasma is directed into a wall and the divertor cannot be used to eliminate impurities. The KS uses microwave produced plasmas on field lines in the cusp tank region. According to the theory, by locating a stabilizing plasma pressure on the field lines at a region with a strong second derivative and large radius in the expanding field region outside the mirrors, the main plasma in the mirror central cell in regions with unfavorable field line curvature can be stabilized. Two coils on the cusp tank are configured to produce expanding field lines with a large positive radius of curvature. A 5-kW 2.45 GHz magnetron is used to produce the stabilizing ECRH plasma pressure in this region. A reduction in the instability duration has been observed. More microwave power is expected this year. Details of both experiments will be given.

EX/P7-11 · Confinement Studies in High Temperature Spheromak Plasmas

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Abstract: Recent results from the SSPX spheromak experiment demonstrate the potential for obtaining good energy confinement ($T_e > 350$ eV and radial electron thermal diffusivity comparable to tokamak L-mode values) in a self-organized toroidal plasma. In this paper we discuss energy confinement and transport in ohmically-heated SSPX discharges and compare data against several transport models applicable to self-organized systems maintained by DC helicity injection. A strong decrease in thermal conductivity with temperature is observed and at the highest temperatures, transport is well below that expected from the Rechester-Rosenbluth model. Recent improvements to performance (raising T_e from 200 eV) result from increasing both gun flux and current to increase the magnetic field while keeping a relatively flat current profile to minimize magnetic fluctuations. In the near term, a new capacitor bank is expected to produce higher magnetic fields and longer pulses, allowing operation with temperatures as high as 0.5 keV. At temperatures above 300 eV, it becomes possible to use modest (1.8 MW) amounts of neutral beam injection (NBI) auxiliary heating to significantly change the power balance in the core plasma, making it an effective tool for improving transport analysis. We are now developing detailed designs for adding NBI to SSPX and have developed a new module for the CORSICA transport code to compute the correct fast-ion orbits in SSPX. This module, coupled to a deposition code (NFREYA), is used to calculate the particle, current and power deposition from Neutral Beam injection. Initial CORSICA results show that a substantial fraction of the injected bean, of order 70%, is confined as fast ions, which is sufficient to raise the electron temperature and total plasma pressure in the core by a factor of two.
* Work performed under the auspices of the US DOE by University of California Lawrence Livermore National Laboratory under contract W–7405–ENG–48.

EX/P7-12 · Transient and Intermittent Magnetic Reconnections in TS-3/UTST Merging Startup Experiments

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Abstract: The spherical tokamak (ST) merging has been studied in TS-3 device for high-power reconnection heating/startup without center-solenoid (CS) coil. Two STs with major radius ~0.2 m were merged together in the axial direction using magnetic compression by two acceleration coils. The magnetic reconnection transformed the magnetic energy of reconnecting magnetic field through the outflow energy finally to the ion thermal energy, increasing the plasma beta of ST up to 0.5. A new finding is that ejection of current sheet (or plasmoid) causes high-speed merging/reconnection as well as high-power heating. In the high-q ST merging, the sheet resistivity was almost classical due to the sheet thickness much larger than ion gyroradius. In the high inflow (compression) case, the large inflow flux and low current-sheet dissipation resulted in the flux pileup followed by the rapid growth of the current sheet. When the flux pileup exceeded a critical limit, the sheet was ejected mechanically from the squeezed X-point area. The reconnection (outflow) speed was slow during the flux pileup and was fast during the ejection, indicating intermittent reconnection similar to the solar flare. As the inflow flux was increased, the quasi-steady reconnection was transformed to the transient one and finally to the intermittent one. In the low inflow case, the inflow flux balanced with the outflow flux, indicating the quasi-steady reconnection like the conventional Sweet-Parker reconnection. In the intermediate inflow case, the inflow flux was larger than the outflow flux in the early reconnection phase and became smaller in the late phase, indicating that the flux pileup increased early reconnection speed. The transient effects such as the flux pileup and ejection enable us to have the high reconnection speed as well as the high-power reconnection heating, even if the merging high-q tokamaks have low current-sheet resistivity. We are now up-scaling those merging techniques of TS-3 and also the RF heating/current drive techniques of TST-2 to a new ST device: UTST ($R \sim 0.4$ m). In this device, all PF coils are located outside of the vacuum vessel to demonstrate (1) double-null startup of STs without CS coil, (2) their reactor-relevant reconnection heating for high-beta ST formation and (3) their sustainment by advanced RF and NBI techniques.

EX/P7-13 · Long Sustainment of FRC-Equilibrium by Use of Center Solenoid in TS-4

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Abstract: This paper addresses a basic equilibrium/confinement question whether a Field Reversed Configuration (FRC) with balanced amount of magnetic and thermal energies, is sustained solely by inductive magnetic flux (magnetic energy) injection of center solenoid (CS). A new finding is that the preferential injection of magnetic energy causes significant increase in resistivity of the FRC. The magnetic energy injected by Center Solenoid Current Drive (CSCD) was converted into its thermal energy through the large anomalous resistivity to maintain the FRC-equilibrium. A quasi-steady state of the high-beta (volume averaged beta ∼0.6–0.7) FRC for 0.15 ms (∼ energy confinement time of conventional FRCs) was achieved through the use of the present low power CSCD. This successful result leads us to the next stage steady sustainment experiment of an FRC by increased capacitor bank energy.

EX/P7-14 · Progress in Potential Formation and Radial-Transport-Barrier Production for Turbulence Suppression and Improved Confinement in GAMMA 10

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Abstract: (1) Four-time progress in ion-confining potential height, ϕ_c, for central-cell plasmas through recent a few years is achieved in the hot-ion mode with T_i = several keV having a favorably increasing scaling of ϕ_c with plug electron-cyclotron heating (ECH) powers. (2) The advance in the potential formation leads to a finding of remarkable effects of radially sheared electric fields dE_r/dr on turbulence suppression and transverse-loss reduction. (3) A preferable dependence of a weak decrease in ϕ_c with increasing the central-cell densities n_c ranging to $\sim 10^{19}$ m^{-3} along with the recovery of ϕ_c with increasing plug ECH powers is obtained. (4) A transverse energy-transport barrier is produced by off-axis ECH in a barrier mirror for the first time, with the formation of a cylindrical layer having energetic electrons flowing through the whole device. This leads to suppress intermittent turbulent vortex-like structures near the layer in the central cell, and results in T_e and T_i rises surrounded by the cylindrical layer having a localized bumped ambipolar potential Φ_c in the central cell. The radial transport barrier is explained in terms of the formation of a strong E_r shear or peaked vorticity W with the direction reversal of $E_r \times B_z$ sheared flow near the Φ_c peak for the turbulence suppression in the central cell. (5) Preliminary central-cell ECH (170 kW, 20 ms alone) in a standard tandem-mirror operation raises T_{e0} from 70 to 300 eV together with $T_{i\perp 0}$ from 4.5 to 6.1 keV, and $T_{i\parallel 0}$ from 0.5 to 1.2 keV with τ_{p0} = 95 ms for ϕ_c = 1.4 kV) trapped ions. The on-axis particle to energy confinement ratio of $\tau_{p0}/\tau_{\mathrm{E}0}$ is observed to be 1.7 for ϕ_c trapped ions (consistent with the Pastukhov theory) and 2.4 for central mirror-trapped ions with 240 kW plug ECH and 90 kW ICH ($\eta_{ICH} \sim 0.3$).

EX/P8-1 · Dynamics of the Pedestal Structure in the Edge Transport Barrier in CHS

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Abstract: Properties of the edge pedestal in tokamaks and/or helical devices play important roles not only in the core plasma confinement but also in the particle and heat fluxes onto the plasma facing components. Therefore, we investigate the dynamic behavior of the edge pedestal in the edge transport barrier (ETB) formation discharge (H-mode) in the compact helical system (CHS). ETB can be achieved with two high power neutral beam injection (NBI) systems, both of which are installed tangentially to the toroidal field in the co-direction. For a local measurement of both the density gradient and fluctuations, we have developed a beam emission spectroscopy (BES) system in CHS. In the case that the heating power exceeds a certain threshold, a transition phenomenon characterized by sudden drop in the temporal evolution of the H-alpha intensity can be observed. The density inside the last closed flux surface (LCFS) increases while that outside the LCFS decreases at the transition, indicating the formation of the edge pedestal. We categorize the waveform into three phases: (1) L-phase before the transition, (2) density build-up phase, and (3) the ETB-saturation phase. After the transition, the density gradient around the rotational transform $\iota = 1$ surface (r/a ∼ 0.95) becomes steeper, forming a density pedestal. Edge harmonic oscillations (EHOs) having the fundamental frequency of around 4.5 kHz and the 2nd harmonic frequency of around 9 kHz can be observed simultaneously with the saturation of the increase in the edge density gradient. An EHO can be observed only in the ETB saturation phase and only for the chord at r/a = 0.95, which correspond to the place of maximum density gradient and $\iota = 1$ rational surface. EHO

is enhanced when the gradient achieves a certain threshold. The gradient stays almost constant after the enhancement of the EHO. The phase analysis suggests that there is a radially decreasing phase shift which is consistent with a radially outward propagation with an apparent velocity of several hundred m/s.

EX/P8-2 · Structure and Dynamics of Spontaneous and Induced ELMs on ASDEX Upgrade

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Abstract: In order to assess the contribution of edge localized modes (ELMs) to plasma-wall interaction in future fusion experiments like ITER, a sound experimental data basis for model validation and extrapolation, and, to be prepared for the unfavorable case, the development of tools for ELM mitigation are required. On ASDEX Upgrade, exploiting the fast edge/divertor diagnostics as well as active ELM control tools, especially ELM pacing by cryogenic pellet injection, a large amount of experimental information has been accumulated on the structure and dynamics of natural and induced ELMs, as well as on related wall effects, e.g. on erosion of tungsten-coated first wall elements. In general, the results for spontaneous and triggered ELMs are consistent with weakly correlated, magnetic field-aligned structures, with rather low effective toroidal mode numbers prevailing in the initial phase, shifting to the intermediate range later on. For the smallest pellets applied, the basic signatures of induced ELMs are rather similar to those of spontaneous ones, while larger pellets with deep penetration tend to produce a longer tail with larger integral particle loss, though the ELM onset phase is again similar. The combined experimental evidence will be presented and discussed in the light of gradually improving theoretical models for the different ELM phases, from an initial seed perturbation, through non-linear formation of electromagnetic, attached helical structures, to fully detached, approximately electrostatic, outward drifting filaments in the scrape-off layer wing.

EX/P8-3 · Pedestal Performance Dependence Upon Plasma Shape in DIII-D

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Abstract: Higher moments of the plasma shape than triangularity, i.e. squareness, are found to significantly affect the pedestal pressure and edge localized mode (ELM) characteristics in DIII-D. The pedestal pressure was controlled over a range of 40% by minor changes to the plasma shape while maintaining a fixed divertor configuration with constant pumping and fueling. The pedestal pressure change is consistent with that expected from magnetohydrodynamic (MHD) stability analysis of the experimental profiles and magnetic equilibrium. The shape dependence of the pedestal pressure was experimentally examined by varying the squareness in the proposed ITER configuration while holding the triangularity fixed. Over this scan the pedestal pressure increased by ∼20% over the ITER target shape for low squareness while the pressure decreased by ∼20% at higher squareness. The ELM energy also varied with the shape to maintain a nearly constant fraction of the pedestal energy. The variation in pedestal energy with squareness was also used to optimize the advanced "Hybrid" discharges in DIII-D. In the "Hybrid" regime low squareness resulted in a high pedestal pressure with large infrequent ELMs that eventually triggered an internal 2/1 tearing mode that locked, resulting in a disruption. At higher squareness the pedestal pressure was reduced with smaller and more rapid ELMs, resulting in control of the total stored energy and maintenance of a steady beneficial internal 3/2 tearing mode. The resulting global beta was higher than could be maintained for the lower squareness shape, all the while maintaining the divertor configuration for optimal pumping and density control. The variation in pedestal pressure with squareness is well described by MHD stability analysis of the edge plasma. The magnetic equilibrium was reconstructed by constraining the pressure with measurements of all relevant profiles and the current profile with a collisional bootstrap model. Equilibrium reconstructions with variations of the pressure and current about the operating point produced a stability map consistent with the experimental measurements of the pedestal pressure. The pedestal stability limit dependence upon global beta and shape is also examined.
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EX/P8-4 · Characteristic Features of Edge Localized Mode under the Presence of Edge Ergodic Magnetic Field Layer in LHD

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Abstract: Magnetic configurations of LHD are characterized by the presence of stochastic magnetic field, so called ergodic layer, surrounding the core plasma. H-mode-like discharges have been newly obtained at an outwardly shifted configuration of $R_{ax} = 4.00$ m with a thick ergodic layer ($\lambda_n > 30$ cm), where the $\iota = 1$ position is located in the middle of the ergodic layer, although those discharges in helical devices have been obtained until now by forming an edge sharp boundary, varying the edge rotational transform at LCFS. ELM-like bursts (ELM) appeared near $\iota = 1$ position. This H-mode-like transition and the ELM can be mainly triggered by changing P_{NBI} (< 12 MW) from 3 beams to 2 beams in a density range of $4 - 8 \times 10^{19}\mathrm{m}^{-3}$, although the spontaneous transition occurs in few discharges without such an external help. The adjustment to a suitable P_{NBI}/n_e ratio ($0.5 < P_{NBI}/n_e < 2\mathrm{MW}/10^{19}\ \mathrm{m}^{-3}$) was a key for the transition, which could determine the edge plasma parameters and their gradients. The ELM vanished by a small change of the edge rotational transform. A precise profile was measured using a newly installed multichannel CO_2 interferometer with a spatial resolution of 1 cm at inboard side of the LHD where the ELM is strongly excited, and we confirmed that the ELM is triggered at $\iota = 1$ location. Density bursts was also observed during the ELM activities with propagation to both sides having opposite signs like the tokamak sawtooth oscillation as an inversion point of $\iota = 1$ position. A variety of the ELM is observed as a function of P_{NBI}/n_e, which indicates a difference of edge pressure gradient near $\iota = 1$. Reduction of the magnetic fluctuation is seen after the transition. However, no clear MHD mode numbers are observed on the present ELM. Increasing the pressure gradient at $\iota = 1$, the ELM frequencies increase. The ELM could be modified by application of m/n = 1/1 resonant field. Characteristics features of the ELM occurred in the ergodic layer are presented.

EX/P8-5 · Studies of the edge pedestal and behaviour of Edge Localised Modes in improved H-mode and small-ELM regimes in ASDEX Upgrade

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Abstract: In the ASDEX Upgrade tokamak, the behaviour of the High-confinement mode (H-mode) pedestal parameters is studied for two cases: 1. H-mode discharges, which gain confinement improvement from flat central magnetic shear profiles and a modification of pedestal parameters compared to conventional H-mode, and 2. Regimes with small Edge Localised Modes (ELMs), such as grassy ELMs at high poloidal beta, similar to the grassy ELM regime in JT-60U and obtained in ASDEX Upgrade in configurations near Double Null. The grassy ELM regime, which occurs at high edge safety factor and medium edge collisionality, is compared to the type II regime obtained with strong shaping (high plasma triangularity) near Double Null.

EX/P8-6 · Automatic Detection and Control of MHD Activity in FTU Tokamak by ECE and ECH/ECCD

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Abstract: Active control of MHD instabilities is important in order to improve tokamak performance. This paper describes experimental work done on FTU tokamak in this field, in particular for the automatic suppression of Tearing Modes (TM) via simultaneous detection and tracking of island position, sawteeth inversion radius and Electron Cyclotron Heating (ECH) power deposition. Effective control in real time is achieved through the implementation of a system for automatic detection of TM onset, and stabilizing reaction with ECH/ECCD. The system is composed by a particular arrangement of the 140 GHz, 4 $\times$ 0.5 MW ECH set-up, and a DSP-based (Digital Signal Processor) unit for the analysis of ECE and Mirnov data, and for the control of gyrotron power supplies. Main emphasis is given to the intrinsic capability of the arrangement for a fast reaction to an early warning of the TM appearance. In order to improve ECH power deposition identification by synchronous detection of ECE oscillations and ECH power modulation, special techniques based on non-periodic pulse sequences have been implemented.

EX/P8-7 · Mitigated Plasma Shut-Down with Fast Impurity Puff on ASDEX Upgrade

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Abstract: Experiments on disruption mitigation are carried out nowadays on all large tokamaks to investigate the influence of the injected impurities on the development of the disruption. The optimal disruption mitigation requires that all three deleterious effects of disruptions, namely the localized thermal load, the energy carried by runaway electrons and the mechanical forces are minimized. The kind of gas, the rate of injection and the injected quantities differ considerably according to the type of gas injection system available, the volume of the vessel and the specific problems affecting the machine after disruption. In this framework, ASDEX Upgrade has been conducting experiments for years and is routinely employing the fast injection of neon for the plasma shut-down of disrupting plasmas and machine protection. The injection is triggered by the locked mode (LM) signal and leads to the onset of a mitigated disruption within 5 ms. The impurity gas is injected into the plasma with two electromagnetic valves. The valves have an opening and closing time of 2 ms and remain open for 4–5 ms; they have been mostly operating with neon gas at a reservoir pressure of 5 bar and have been typically injecting 180 mbar (4.5×10^{21} atoms) of gas. Dedicated experiments have been carried out with 10 and 15 bar reservoir pressure. This paper will (1) describe the phenomenology of mitigation, (2) the experimental condition in ASDEX Upgrade, (3) the plasma response to the injection of impurities, (4) discuss the understanding of the observed phenomena, and (5) outline the future plans for the development of the mitigation system on ASDEX Upgrade.

EX/P8-8 · Density Limit in Discharges with High Internal Inductance on JT-60U

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Abstract: High densities exceeding the Greenwald limit by a factor of 1.7 have been obtained in discharges with high internal inductances of li as high as 2.8 in JT-60U. The internal inductance is controlled by ramping down the plasma current. While the density is beyond the Greenwald limit, confinement performance is kept as good as H_{89PL} and HH_{y2} factors of 1.5 and 1, respectively. In NBI heated discharges in JT-60U, the operational density limit has been surveyed by changing the current ramp down rate (0.175–0.75 MA/s) from a flat top (plasma current of 1 MA). Internal inductance ℓ_i goes up in this phase, which suggests a peaked current profile with enhanced magnetic shear in the edge region. In a scheme of monotonic current ramp down, ℓ_i (magnetic shear) and q have co-linearity. In order to separate these two factors, the current ramp down has been paused at the plasma current of 0.65 MA and the phase with decreasing ℓ_i (magnetic shear) at the constant q also has been investigated. In the case of no disruption, the plasma has good confinement performance even beyond the Greenwald limit and the density has reached 1.73 times the Greenwald limit. The confinement performance is better than the 89PL scaling and is consistent with the earlier work on high ℓ_i plasmas. A slight increase of the density has resulted in a disruption in the earlier phase of the current ramp down. The density normalized by the Greenwald limit of the case with no disruption is even higher than the case with a disruption in particular in the periphery. The temperature in the periphery is higher in the discharge without disruption than that of the discharge terminated by a disruption. The normalized density when the detachment characterized by the decrease in a D_α signal at the divertor occurs is even higher in the case with no disruption compared with the case with a disruption. This comparison suggests that the improvement in thermal and particle transport does exist in the periphery and the edge, and mitigation of density limit is observed coincidently. Although the high ℓ_i discharge studied here lies outside of the usual parameter space for tokamak operation, demonstration of a stable discharge with good confinement beyond the Greenwald limit uncovers its physical element.

EX/P8-9 · Suppression and Excitation of MHD Activity with Electrically Polarized Electrode at the TCABR Plasma Edge

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Abstract: The purpose of this experimental work performed in the TCABR tokamak, using electrically polarized electrode, was to clarify why in some discharges excitation of strong MHD activity was observed while in others this instability was suppressed. To do this, the plasma current was varied to cover a range of safety factor from 2.8 up to 3.6, allowing the H-mode barrier to partially scan the spectrum of MHD modes and interact with the magnetic islands. A set of 22 Mirnov coils was used to detect the magnetic oscillations and a triple Langmuir probe to measure plasma fluctuating potentials and ion saturation currents. The

investigation of MHD excitation with biased electrode was performed with discharges with low amplitude MHD activity. After applying the bias the amplitude of the MHD activity strongly increased. The analysis of the data indicated that in this case the dominant mode was m = 2, n = 1 and the m = 3, n = 1, the passive mode. The positive electric field showed strong and fast decrease near the onset of the MHD activity. On the other hand, to identify the mechanism of MHD suppression, the tokamak parameters were adjusted to obtain reproducible discharges with strong MHD activity without electrode biasing. For these discharges the dominant mode was m = 3, n = 1. After applying the bias to the electrode the MHD activity was strongly decreased. The proposed explanation for the suppression is that the m = 3 mode, located near the biased electrode at the plasma boundary, is suppressed together with the passive mode m = 2. Strong quenching of the turbulence was also detected.

EX/P8-10 · Investigation of the Resonant Perturbations Amplification at T-10 during ECR Heating

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Abstract: The dynamics of large-scale MHD instabilities is studied at T-10 during high power Electron Cyclotron Resonance Heating (ECRH). It is well known that, in a tokamak plasma, large scale MHD instabilities can develop near so-called resonant, or rational, magnetic surfaces , (here, is the safety factor; is the radius of the rational surface; and m, n = 1, 2, 3, 4, ... are, respectively, the numbers of poloidal and toroidal revolutions of the field line around the torus). In toroidal geometry, plasma perturbations are coupled in such a way that the development of an active (master) primary perturbation mode at one resonant magnetic surface can give rise to a cascade of passive (slave) secondary modes at other resonant surfaces. It is possible to observe the similar excitation of a cascade of resonant modes by external perturbations created by imperfections in the toroidal and poloidal field coils, (Error field), or by coils producing the fields for feedback suppression of instabilities. This process is actively studied today. On the other hand it will be interesting to observe the cascade excitation of resonant surfaces by internal perturbation. High power ECRH can create such internal stationary perturbation due to non symmetric current excitation. Resonant components of this perturbation can be amplified by resonant magnetic surfaces. Joint analysis of ECE, SXR, and Mirnov signals were used to detect modes structure. It was shown that the interactions of the resonant components excited (injected) by high power ECRH inside plasma with free running MHD eigen modes has the following features: (a) Amplification of internal perturbations; (b) Beat frequency behavior far from locking point; (c) Noncoherent phase rotation near the locking point; (d) Locking by internal perturbation (dominant mode). The results of this analysis were used for verification of the empirical model of the coupled MHD perturbation excitation based on coupled nonlinear the Van der Pole oscillators.

EX/P8-11 · Experiments on feedback control of multiple resistive wall modes comparing different active coil arrays and sensor types

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Abstract: Experiments have been carried out on the EXTRAP T2R reversed-field pinch device to study several important issues related to feedback control of resistive wall modes (RWMs). In the first series of experiments, the effect of side band coupling due to the limited number of coils in the array was investigated. Different feedback schemes have been used in order to overcome the coupling effect such as the mode control scheme, which includes real time spatial FFT to obtain action on individual modes. The unstable RWM spectrum consists of about 16 modes with m = 1 and different toroidal mode number n. In recent experiments using the intelligent shell scheme with a full PID controller action and higher feedback gains, complete stabilisation of the modes is achieved. The active array consists of 128 coils at 4 poloidal and 32 toroidal positions. The pulse length is equivalent to 10 wall times, limited by the power supply. Without feedback the discharge pulse ends prematurely after 3–4 wall times due effects associated with the RWM mode growth. With feedback stabilization, plasma rotation and tearing mode rotation is maintained throughout the pulse, thereby avoiding the locked mode phenomenon often observed in RFPs and manifested in an increased local plasma wall interaction. With feedback control the influx from the wall is maintained at a low level throughout the pulse. The first feedback experiments using a sensor array measuring the toroidal field component have been carried out. The critical gain required for suppression has been compared for the radial and toroidal field sensor cases, and found in qualitative agreement with theory. The phase shift of the control field has been varied. Optimal suppression is achieved at the predicted complex feedback gain phase. Mode rotation is induced at other complex gain phases,

in agreement with modelling. Previously developed linear models have guided the feedback experiments. Open-loop experiments have been used for validation of the cylindrical linear model. The model is based on a number of parameters: 1) RWM growth and damping rates, 2) wall diffusion time of different helical harmonics, 3) mutual inductances between coils and sensors. The results are satisfactory, showing that the cylindrical model with a continuous thin wall can describe the plasma-wall response.

EX/P8-12 · Evaluating Electron Cyclotron Current Drive Stabilization of Neoclassical Tearing Modes in ITER: Implications of Experiments in ASDEX-U, DIII-D, JET, and JT-60U

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Abstract: Resistive neoclassical tearing mode (NTM) islands will be the principal limit on stability and performance in ITER as beta is well below the ideal kink limit. NTM island control in ITER is predicted to be challenging both because of the relatively narrower marginal island widths and the relatively broader electron cyclotron current drive (ECCD). Measurements from ASDEX Upgrade, DIII-D, and JET in beta rampdown experiments are used to determine the marginal island size for m/n = 3/2 NTM removal. This is compared to data from ASDEX Upgrade, DIII-D and JT-60U with elimination of the m/n = 3/2 island by continuous ECCD at near constant beta. The empirical marginal island size is consistent in both sets of removal experiments and found to be about twice the ion banana width. A common methodology is developed for fitting the saturated m/n = 3/2 island before (or without) ECCD in all four experimental devices. To this is added (and model tested to experiments) the effect of unmodulated co-ECCD on island stabilization including both replacing the missing bootstrap current and making the classical tearing stability index more negative. The experimentally benchmarked model is then used to evaluate ITER. The ITER ECCD upper launcher with up to 20 MW of injected power is appraised with or without modulation for both the m/n = 3/2 mode and the m/n = 2/1 NTM (which can lock to the resistive wall and induce disruption). An m/n = 2/1 rotating island model with drag from eddy current induced in the resistive wall is used to predict the necessary ECCD to keep the island from locking as a function of the rotation in ITER. The planned relatively wide ECCD should be capable of regulating the island width to avoid mode locking with the anticipated rotation in ITER but there is little margin available for inevitable misalignment. Narrower ECCD of more power and/or more rotation in ITER would increase confidence in island control and successful operation.
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EX/P8-13 · Observation of m/n = 1/1 Mode Behaviors during Molecular Beam Fuelling and ECRH Discharges in HL-2A

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Abstract: On the HL-2A tokamaks, series of experiments have been conducted to shed light on the persistent m/n = 1/1 mode topology and its influence on transport properties in the plasma centre. Several new characteristics of central MHD activities during auxiliary fualling and heating are discussed, and some important information about the changes in local plasma parameters and how the plasma responds to the perturbation caused by molecular beam injection (MBI) or laser blow-off are provided. Several important central MHD activities, for example, sawtooth suppression, monster and compound sawtooth, and persistent m/n = 1/1 oscillations, have been observed. A large, long persistent m/n = 1/1 perturbation has newly been observed in the core region after molecular beam injection, a detailed study on the influence of MBI on central pressure gradient and the stability of central plasma has been made. Especially, possible mechanisms for the formation of the continuous mode or snake-like perturbation during MBI are discussed. In laser blow-off experiments, persistence of the m = 1 oscillation after the internal disruption of inverted sawtooth is found. With the illumination effect of the impurity radiation, the island structure after sawtooth crash can be studied. Such a delicate study provides much information about the evolution of the central q profile and the reconnection process. During electron cyclotron heating (ECRH), a strong m = 1/n = 1 mode is excited when the heating power is high enough and the resonance position is located just around the core of plasma. With the strong m = 1 oscillation driven by ECRH, a sawtooth tends to saturate or decrease in its ramp phase and the shape of sawtooth is usually changed, leading to formation of a saturated sawtooth, a hill, or a compound sawtooth. Experimental results indicate that there is evidence for an internal steep electron temperature profile associated with the q = 1 rational surface before

the compound crash. The effect of ECRH on plasma transport properties in the vicinity of the $q = 1$ surface are investigated.

EX/P8-14 · Integrated View of Disruption Dynamics on Internal Electromagnetic and Plasma Structures in the Small Tokamak HYBTOK-II

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Abstract: Integrated view of disruption dynamics on internal electromagnetic and plasma structures with a high time resolution is presented for the first time. Main results are as follows: (i) Observation of thermal quench just before the current quench. The electron pressure in the core region decreases rapidly in about 10 μs; (ii) finding of current profile broadening just before the current quench and then its rapid pump-out ($\sim$10 μs) in the core region, (iii) appearance of high-energy electron produced by the induced strong toroidal electric field associated with a rapid change in current profile. In addition, we have proposed the new definition of current quench time, taking into account both electromagnetic force induced by current quench and the mechanical impulse on the vacuum vessel. In the conference, the electron temperature dependence in the current quench time would be examined by using the proposed new definition.

EX/P8-15 · Two-Dimensional Structure of MHD Instabilities and their Non-Linear Evolution in the Large Helical Device

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Abstract: It is important to understand the role MHD instabilities play in plasma confinement in helical systems. Since mode amplitudes depend nonlinearly on the pressure gradient, it is not straightforward to make theoretical predictions of mode structures and amplitudes, and thus determine their effect on confinement. We experimentally study the evolution of pressure driven modes in the Large Helical Device (LHD) using images measured by a fast, tangentially-viewing soft X-ray camera . When ice-pellets are injected in a NB-heated plasma, the core pressure gradient is $2 \sim 3$ times larger than found in normal operation with gas fueling. As the pressure gradient increases, sawtooth-like events begin. These events are often followed by large amplitude oscillations which persist for 0.1–0.3 s. It is found that an $m = 3$ mode evolves within 500 μs and deforms the magnetic surface just before the sawtooth crash. When the mode saturates, an enhanced heat flux from the core to the edge is observed, causing flattening of the pressure profile. The enhanced flux could be caused by reconnection where the magnetic surfaces are heavily compressed due to an interchange mode. The magnetic Reynolds number S is used as a measure of the resistivity. For similar pressure gradients, sawtooth activity occurs for lower values of S and saturated oscillations occur at higher values of S.

EX/P8-16 · Solenoid-free Plasma Start-up in NSTX using Transient CHI

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Abstract: For the first time, 60 kA of closed flux current is produced in the National Spherical Torus Experiment (NSTX), without using a solenoid. A process called Transient Coaxial Helicity Injection (CHI) is used to generate the self-contained equilibrium. This is an important step in the production of a starting equilibrium for solenoid free operation. Until now, almost all tokamaks and spherical torus plasma confinement devices have relied on a solenoid, through the center of the device, to produce the plasma current needed to confine the plasma. An alternate method for plasma startup is essential for developing a fusion reactor based on the spherical torus concept and could also reduce the cost of a future tokamak reactor as well. In the CHI method, a plasma current is rapidly produced by forming a discharge between coaxial electrodes connected to an external power supply in the presence of toroidal and poloidal magnetic fields. The initial poloidal field configuration is chosen such that the plasma rapidly expands into the chamber. When the injected current is rapidly decreased, magnetic reconnection occurs near the injection electrodes, with the toroidal plasma current forming closed flux surfaces. The CHI technique has previously been studied in smaller experiments, such as the HIT-II device at the University of Washington. The significance of these results are (a) demonstration of the process in a vessel volume thirty times larger than the HIT-II concept exploration device, on a size scale more comparable to a reactor, (b) a remarkable multiplication factor of 60 between the injected current and the achieved toroidal current, compared to six in previous experiments, and (c) significantly more detailed experimental measurements, including,

for the first time, fast time-scale visible imaging of the entire process that shows discharge formation, disconnection from the injector and the reconnection of magnetic field lines to form closed flux. Results from these and other new experiments in NSTX will be presented.

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EX/P8-17 · Imaging and Manipulation of Sawteeth and Tearing Modes in TEXTOR

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Abstract: The real-time detection and control of instabilities in a thermonuclear plasma presently is and will continue to be one of the exciting challenges in fusion research on the way to a fusion reactor, as will be the understanding of these mechanisms. Thanks to a combination of an innovative 2D imaging technique for temperature fluctuations, a versatile ECRH/ECCD system and a unique possibility to externally induce tearing modes in the plasma, TEXTOR is able to make pioneering contributions in this field. This paper focuses on two different aspects: the sawtooth oscillation and the $m = 2$ tearing modes. In both cases the 2D-electron cyclotron emission imaging diagnostic (ECE-i) can resolve features not attainable before, allowing a direct comparison with theory and in both occurrences the ECRH system is able to control or suppress these instabilities.

EX/P8-18 · Prediction of Rotational Stabilisation of Resistive Wall Modes in ITER

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Abstract: The RWM is a kink mode whose stability is related to damping arising from relative rotation between the fast rotating plasma and the slowly rotating wall mode. Plasma rotation, and the ensuing RWM damping, is a passive stabilising mechanism making it an attractive route for RWM control. It is thus important to understand how plasma rotational stabilisation of the RWM will scale to ITER and future power plants. Cross machine scaling studies represent a good way of testing the various theories on RWM damping – in this paper such cross machine RWM studies between JET and DIII-D are presented. An excellent way of probing RWM stability has been to examine the plasma response to externally applied resonant fields – as the beta-limit without the stabilising influences of surrounding walls is exceeded, strong Resonant Field Amplification (RFA) occurs. Cross machine studies of RFA are discussed in this presentation and when geometric differences are taken into account show good agreement, indicating equivalent damping in the 2 tokamaks. A second way of probing RWM stability is to compare the critical rotation velocity below which an $n = 1$ RWM is destabilised, and studies comparing JET and DIII-D results on this issue are also discussed in this presentation. These indicate that the plasma rotation in the vicinity of $q \sim 2$ normalised by the Alfvén time is likely the controlling parameter in RWM stability. The magnitude of rotational stabilisation, indicated by these studies, that is required to stabilise RWMs, is predicted to be at best marginally achieved in ITER, and so active feedback schemes should be developed.

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EX/P8-19 · Control of resistive wall modes near the ideal limit using modular internal coils

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Abstract: We report on the first demonstration of active feedback stabilization of RWM near the ideal limit using a mode control algorithm implemented with a digital control system. Stabilization of resistive wall modes (RWMs) is a key issue for future tokamak reactors. An economic fusion reactor based on advanced tokamak scenarios relies on the ability to stabilize these modes up to beta values near the ideal wall limit. The 3D electromagnetic code VALEN has been extended to include important effects such as plasma mode rotation, multiple plasma modes, and realistic feedback systems that have noise and latency. Recent work on applying the concepts of observability and controllability from modern control theory to the RWM stabilization problem is described. The effectiveness of a RWM feedback system can be quantified in terms of a few coupling parameters that can be calculated using the VALEN code. Operation near the ideal wall limit implies the use of control coils internal to the main conducting structures, typically

the vacuum vessel. For ITER and any future fusion tokamak power plants, it is beneficial to make these internal coils as few and small as possible. Side band excitation, lack of mode rigidity and other effects will limit how sparse the coverage can be made without compromising the ability to feedback stabilize RWMs. We present plans to address these issues in the HBT-EP device in the near future.

EX/P8-20 · Potential Safe plasma termination Using Laser Ablation of High-Z Impurity in Tokamak

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Abstract: A preliminary experiment triggering a plasma current quench by laser ablation of high-z impurities has been performed in the HL-1M and HL-2A tokamak. Using injection of impurity with higher electric charge allows us to increase the radiation cooling. Resistive, highly radiating plasma formed prior to the thermal quench, can dissipate both the thermal and magnetic energy. It can be possibly a simple and potential approach to decrease significantly the plasma thermal energy and magnetic energy before a disruption then obtain a safe plasma termination.

EX/P8-21 · MHD issues in Tore Supra steady-state fully non-inductive scenario

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Abstract: Impact of MHD modes on non-inductive discharges: a crucial issue Fully non-inductive tokamak plasma discharges are attractive for a fusion reactor, and the understanding of their MHD limitations beyond the linear stability analysis is therefore essential. Specific features are due to the enhanced impact of non-linear effects: one is due to the bootstrap current perturbation, and the other is linked to the intrinsic physics of plasma current generation by the external system used. In the Tore Supra tokamak, a Lower Hybrid wave launcher provides the external source, which has non-linear dependences on the plasma current and electron temperature profiles. Such complex interplay is causing spontaneous temperature oscillations, but also a regime with permanent MHD activity, after the triggering of a double-tearing mode. We report observations and interpretations of MHD modes in such non-inductive discharges in Tore Supra. The double-tearing mode and its non-linear impact: experiment and simulations Due to the slightly hollow deposition profile of LH waves in Tore Supra, the magnetic configuration exhibits a flat or reversed q-profile. The plasma is therefore prone to the triggering of double-tearing modes. In its non-linear development, this mode can either produce a regime of periodic crashes (the $q = 2$ sawteeth), or degenerate into the MHD regime where islands are produced on $q = 3$ and $q = 4$. With its large radial extension, the double-tearing mode could have dramatic effect on a scenario with important bootstrap fraction, as confirmed by numerical simulations. MHD modes: a powerful diagnostic for investigating oscillating regimes In cases where oscillating regimes are present, MHD modes give valuable information about the dynamics of the current profile, as will be shown. Safe operation diagram for non-inductive discharges: preforming and steady-state phases Only the full reconnection of a double tearing mode has an important impact on a Tore Supra non-inductive discharge. On this basis, a diagram for safe operation can be drawn for fully non-inductive discharges. In the ohmic (preforming) phase, MHD modes can also undermine the development of a successful non-inductive phase when the edge safety factor is too high. This aspect will also be addressed.

TH

Magnetic Confinement Theory and Modelling

TH/1-1 · Studies of the Tokamak Edge with Self Consistent Turbulence, Equilibrium, and Flows

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Abstract: We report on the theory and computation of gyrofluid and gyrokinetic turbulence in the tokamak edge, with time-dependent self consistency not only in the large scale $E \times B$ flows but also the toroidal equilibrium. The physical mechanisms of flow generation and saturation are emphasised. Edge zonal flows are maintained by equilibrium processes, forced upon but not directly determined by the turbulence. Effects of flux surface shaping, ergodic magnetic field perturbations, and scrape-off layer boundary structure are subjected to control tests. Theoretical study within a Lagrangian framework yields rigorously consistent nonlocal gyrofluid equations. Computations at this level using both the gyrofluid and gyrokinetic field theory models will be presented. The emphasis throughout is on physical process and theoretical understanding. Nevertheless, some of the findings such as self consistent, long-term burst effects resulting from breakdown of the flow equilibrium have direct application to long standing problems.

TH/1-2 · A Comprehensive Theory-Based Transport Model

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Abstract: A new theory based transport model with comprehensive physics (trapping, general toroidal geometry, finite beta, collisions) has been developed. The core of the model is the new trapped-gyro-Landau-fluid (TGLF) equations which provide a fast and accurate approximation to the linear eigenmodes for gyrokinetic drift-wave instabilities (trapped ion and electron modes, ion and electron temperature gradient modes and kinetic ballooning modes). This new TGLF transport model removes the limitation of its predecessor GLF23 and is valid for the same conditions as the gyrokinetic equations. A theory-based philosophy is used in the model construction. The closure coefficients of the TGLF equations are fit to local kinetic theory to give accurate linear eigenmodes. The saturation model is fit to non-linear turbulence simulations. No fitting to experiment is done so applying the model to experiments is a true test of the theory it is approximating. The TGLF model unifies trapped and passing particles in a single set of gyrofluid equations. A model for the averaging of the Landau resonance by the trapped particles makes the equations work seamlessly over the whole drift-wave wavenumber range from trapped ion modes to electron temperature gradient modes. A fast eigenmode solution method enables unrestricted magnetic geometry. Electron-ion collisions and full electromagnetic fluctuations round out the physics. The linear eigenmodes have been benchmarked against comprehensive physics gyrokinetic calculations over a large range of plasma parameters. The deviation between the gyrokinetic and TGLF linear growth rates averages 11.4% in shifted circle geometry. The transport model uses the TGLF eigenmodes to compute quasilinear fluxes of energy and particles. A model for the saturated amplitude of the turbulence completes the calculation. The saturation model is constructed to fit a large set of nonlinear gyrokinetic turbulence simulations. The TGLF model is valid in new physical regimes that GLF23 was not. For example, the low aspect ratio spherical torus which has both a high trapped fraction and strong shaping of magnetic flux surfaces. The TGLF model is also valid close to the magnetic separatrix so the transport physics of the H-mode pedestal region can be explored.
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TH/1-3 · Identification of TEM Turbulence through Direct Comparison of Nonlinear Gyrokinetic Simulations with Phase Contrast Imaging Density Fluctuation Measurements

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Abstract: We have implemented a new synthetic diagnostic in the GS2 gyrokinetic code to simulate density fluctuations measured by 32 Phase Contrast Imaging (PCI) chords. The synthetic diagnostic enables a direct comparison between gyrokinetic simulations and fluctuation measurements. This comparison is more fundamental than the usual comparisons of transport fluxes, which were first used to provide a quantitative understanding of internal transport barrier formation and control in Alcator C-Mod. Excellent agreement between simulated and measured wavelength spectra over the range 1–8 cm^{-1} can be seen. The relative increase in density fluctuation level with on-axis heating is reproduced by the simulations. Including the PCI instrument function, accounting for finite aperture, reduced sensitivity at long wavelength, and Gaussian beam effects, further improves the already excellent agreement between simulations and measurements. This comparison amounts to a direct observation of TEM turbulence, underscoring

our previous results, but with more fundamental implications. In addition, the nonlinear upshift of the TEM critical density gradient is shown to increase strongly with collisionality, in marked contrast with the Dimits shift for ITG turbulence. Trapped electron mode turbulence is particularly relevant to particle and electron thermal energy transport. Achieving understanding of the mechanisms underlying particle and electron thermal energy transport crucial to to future devices such as ITER, where core fueling is greatly reduced, and electrons are heated directly by alpha-particles.

TH/2-1 · Coupled ITG/TEM-ETG Gyrokinetic Simulations

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Abstract: Very expensive high effective Reynolds number gyrokinetic simulations are performed with the GYRO code nonlinearly coupling low wave number (low-k) ion temperature gradient/trapped electron mode (ITG/TEM) turbulence to high-k electron temperature gradient (ETG) turbulence at reduced ion/electron mass ratios. GYRO contains all the physics required for simulation of core tokamak plasmas outside transport barriers. Previous simulations off DIII-D Bohm scaled L-modes and gyroBohm scaled H-modes with low-k ITG/TEM turbulence match transport flows within experimental error. Neoclassical ($k \sim 0$) transport has been shown to be simply additive to the low-k transport. The GYRO gyrokinetic code has now been extended to cover the high-k ETG electron transport (requiring electron gyro-averaging). All previous gyrokinetic simulations of ETG have assumed adiabatic ions valid only at very high-k: ETG-ai. GYRO simulations of ETG-ai turbulence have been benchmarked using three codes (GS2, PG3EQ, GENE); good agreement was found for the Cyclone base case at low shear ($s = 0.1$). However GYRO has shown that the ETG-ai model does not in fact nonlinearly saturate at moderate shear (e.g., the Cyclone base case at $s = 0.8$ with electron trapping). However GYRO simulations with fully kinetic ions (ETG-ki) reaching into the low-k non-adiabatic ion zonal flow regime do saturate at reasonable transport levels. We find that for very large ion gyroradius scale box simulations with resolution reaching down to the small electron gyroradius scales, the high-k ETG-ki driven electron energy transport is likely not very significant (<20%) compared to low-k ITG/TEM transport *except* when $E \times B$ shear stabilizes most of the low-k transport. Inside $E \times B$ stabilized inernal transport barriers, ETG driven transport likely dominates ITG/TEM driven transport. Comparing simulations with and without coupling, we tentatively conclude that the ITG/TEM-ETG coupling is weak.
* Work supported by US DOE under DE-FG03-95ER54309.

TH/2-2 · Beyond scale separation in gyrokinetic turbulence

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Abstract: Predicting turbulent transport in nearly collisionless fusion plasmas requires solving gyrokinetic equations coupled to Maxwell equations. This paper presents results obtained with a new method based on a semi-Lagrangian scheme applied to gyrokinetics. This technique allows to compute the full distribution function with little numerical noise. Several issues have been addressed. First, the kinetic and fluid descriptions of an interchange turbulence have been compared. It is found, as expected, that kinetic fluxes are well below the values calculated in the fluid approach. This difference is only partially due to different behaviours of Zonal Flows. In fact it appears that wave/particle resonant processes play a central role, so that the distribution function is far from a Maxwellian, and cannot be described by a small number of moments. Second, a slab Ion Temperature Gradient (ITG) driven turbulence has been studied. When the initial density profile is flat, the temperature profile evolves towards a state where its gradient is weak, i.e. with low confinement. For of a peaked initial density profile, the temperature profile steepens, i.e. a transport barrier develops. This difference of behaviour is due to turbulent flow generation driven by the density gradient. Finally the GYSELA code has been upgraded to run 5D simulations of toroidal ITG turbulence. A scan of thermal flux with temperature gradient has been produced and compared to the CYCLONE test case. It appears that choosing the equilibrium distribution function as function of the motion invariants is crucial for these non perturbative full torus simulations. Breaking this rule leads to the development of large scale steady flows, which prevent the onset of turbulence.

TH/2-3 · Simulations on the nonlinear mode coupling in multiple-scale drift-type turbulence with coherent flow structures

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Abstract: The dynamics of secondary anisotropic coherent structures behaving as stationary waves, including zonal/mean flows, streamers and secondary low-frequency long wavelength fluctuations, in multiple-scale turbulence in tokamak plasmas is studied based on 3D gyrofluid and gyrokinetic particle simulations as well as modeling analyses. We specifically discuss the nonlinear mode coupling as a ubiquitous principal interaction mechanism in the dual processes of the generation and back-action of secondary structures on ITG and ETG turbulence. Here we report two new results: (1) The nonlinear mode coupling resulting from the strong dynamics of secondary structures changes the spectral structure in inertia region from power-law into exponential-law dependence. The turbulence may be reduced due to the local and/or nonlocal free energy transfer to stable region. (2) Streamers, and also poloidally long wavelength fluctuations, can limit the saturation amplitude of ETG turbulence even though they are expected to enhance turbulent heat transport. Furthermore, the mean shear flows can reduce the generation of zonal flows in micro-scale turbulence.

TH/2-4 · Progress in Understanding Multi-Scale Dynamics of Drift Wave Turbulence

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Abstract: We report on recent progress on several closely related theoretical problems with a common focus on the multi-scale dynamics of drift wave turbulence. These topics are: (a) low q resonances and the fate of turbulence energy in an inverse cascade, (b) turbulence propagation and transport barrier dynamics. Topic (a) deals with narrow, m not equal zero secondary flows, which are generated by turbulence and resemble zonal flows. These vortex modes can produce local flat spots and corrugations on profiles, and sink energy from the inverse cascade. Topic (b) deals with the theory of turbulence spreading. In particular, we validate simple turbulence propagation models via a rigorous theory, which yields a flux-gradient relation for turbulence intensity. We also show which classes of wave interactions facilitate spreading and that zonal flows do not. We also solve exactly for the phase co-existence region in a simple two-field model of the L to H transition. We show that profile curvature must be retained in order to obtain a regularizable solution, and that hysteresis is weaker than conventionally assumed.

TH/2-5 · Linear and nonlinear aspects of edge turbulence and transport in tokamaks

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Abstract: Linear, nonideal, curvature driven instabilities in the weakly collisional edge of hot reactor like plasmas, and nonlinear physics of edge localized modes and pedestal transport in tokamaks are studied. Multiple fluid descriptions for the collisionless edge tokamak plasma in the presence of trapped electrons have been used for instability model. If electron inertia effects are included, the edge plasma is unstable to ballooning instability when the plasma beta is less than its critical value. We conclude that a robust non-ideal curvature driven instability persists in the tokamak edge plasma even when it is hot and weakly collisional. For nonlinear studies, a simple self-consistent theoretical model of multi-scale interaction of edge localized modes (ELMs) such as the ideal ballooning mode interacting with zonal magnetic fields and zonal flows has been used. The influence of zonal fields and zonal flows on short scale ballooning turbulence is calculated from standard wave kinetic equation. If beta exceeds beta critical and the edge pedestal is unstable to ideal mode, the magnetic Reynolds stress completely suppresses the zonal flow growth. Thus, the dynamics of self-consistent zonal flows in relaxation of ELMs is likely to be unimportant. As we are interested in studying saturation of ballooning mode by back reaction of zonal fields, we first investigate the stability of long-scale zonal magnetic field with non-zero radial and zero poloidal and parallel wave vectors to a finite amplitude spectrum of background ballooning mode turbulence. An instability is observed which can be viewed as a fast dynamo action by ideal ballooning instability. Secondly, we have studied the secondary instabilities of large-scale magnetic field with non-zero radial, poloidal and parallel wave vectors; this can be viewed as an interaction of ideal ballooning mode with tearing instability. Again the interactions show possible excitation of secondary instabilities. A simple zero-dimensional model to estimate the saturated amplitude of ballooning modes interacting with large-scale dynamo and tearing instabilities, and an estimation of edge pedestal transport are presented.

TH/2-6Ra · Gyrokinetic Studies of Nonlocal Properties of Turbulence-driven and Neoclassical Transport

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Abstract: The nonlocal physics associated with turbulent and neoclassical transport is investigated using newly developed simulation capabilities and analytic models. 1) For turbulence transport, we focus our studies on the turbulence spreading through a transport barrier characterized by an $E \times B$ shear layer. We perform a series of numerical experiments by placing a radial electric field well, with varying strength, next to the region where the ITG instability is linearly unstable. It is found that an $E \times B$ shear layer with an experimentally relevant level of the shearing rate can significantly reduce, and sometimes even block, turbulence spreading by reducing the spreading extent and speed. From the spatio-temporal evolution of the turbulence propagation front, we find that the spreading slows down significantly in the region of higher shearing rate, rather than at the bottom of the E_r well. Studies with more realistic E_r profiles are in progress alongside development of an analytic model. 2) Our global neoclassical particle simulation using GTC-Neo, which includes nonlocal physics due to large orbit effects, studies neoclassical physics of NSTX plasmas. Typically, near the magnetic axis, the ion heat flux is decoupled from the local temperature gradient, breaking the Fick's law type gradient-flux relation. Our simulation predicts an outward ion heat flux, even for a reversed local gradient of T_i near the magnetic axis, which is in the same direction as the experimental measurement.

TH/2-6Rb · Long Time Simulations of Microturbulence in Fusion Plasmas

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Abstract: Remarkable success has been achieved in the area of kinetic simulations of turbulence transport in tokamaks using the modern massively parallel computers. Here, we will report the use of the δf global toroidal gyrokinetic particle simulation code (GTC) for studying the long time behavior of microturbulence for fusion plasmas. Establishing the capability of this type is extremely important for making meaningful comparisons between the simulation observations and actual experimental measurements. One of our findings is that the often-neglected velocity space nonlinearity associated with parallel acceleration has a profound influence on the production of zonal flow and the temporal evolution of the ITG turbulence. This parallel acceleration term, which must be taken into account to ensure the energy conservation, is neglected in most of the fusion turbulence codes. Secondly, to address the questions about the intrinsic particle noise in PIC codes, we will present our recent findings by extending the Fluctuation-Dissipation Theorem to a nonlinearly saturated system arising from drift instabilities. Based on this "first principles" approach, it is demonstrated that the discrete particle noise, by using an insufficient number of particles, will always enhance the steady state particle (and thermal) flux in accordance with the entropy conservation property. This is in disagreement with the conclusion made by a recent study, which claims that discrete particle noise can suppress steady state thermal flux. In addressing the issues of special importance for burning plasmas such as ITER, we have also carried out studies of the influence of electromagnetic effects on turbulent transport using GEM, a δf particle code for turbulence studies with kinetic electron dynamics and electromagnetic perturbations. With the recent extension to handle general toroidal equilibrium magnetic field configurations to enable realistic applications to actual experimental scenarios, we have used GEM for high-β NSTX plasmas for studying kinetic ballooning modes (KBM) and toroidicity-induced-Alfvén-eigenmodes (TAE). Details will be reported.

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TH/3-1 · Interpretation of Mode Frequency Sweeping in JET and NSTX

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Abstract: Fast frequency chirping, on the millisecond time scale, has been observed in many tokamak experiments observing Alfvénic activity. The cause is generally attributed to energetic particles that produce a kinetic drive for instability that then form holes and clump phase space structures, that must sweep in frequency to balance the intrinsic dissipation present in the background plasma. A general theory, independent of a specific system, has been developed to describe this dynamics. Within the past few years fast chirping has been observed in the excitation of the geodesic acoustic mode in JET and compressional Alfvén wave (CAE) in NSTX. In addition an experimental effort on NSTX is attempting to alter the

nature of the observed chirping by introducing additional rf heating and thereby test the hypothesis that the observed chirping phenomena is associated with the formation of phase space structures. The GAM oscillations on JET manifest itself as repeated and prolonged $n = 0$ frequency sweeping behavior which can persist during the entire discharge. This particular excitation demonstrates that the problem of zonal flow can strongly correlate with the formation of kinetic phase space structures. On NSTX fast chirping of the (CAE) is due to the excitation of an ion cyclotron resonance and this signal may be explained by the application of the aforementioned chirping theory. In previous experiments on the Columbia dipole experiment, Terella, chirping signals were quenched by applying rf heating, with the explanation that the heating destroyed the phase space structures. A similar experiment has been attempted to destroy chirping signals observed on NSTX, with negative results at the large rf heating did not change the chirping characteristics. Analysis shows that there may not have been enough power supplied to alter the phase space structures.

TH/3-2 · Electron fishbones: theory and experimental evidence

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Abstract: We discuss the processes underlying the excitation of fishbone-like internal kink instabilities driven by supra-thermal electrons generated experimentally by different means: Electron Cyclotron Resonance Heating (ECRH) and by Lower Hybrid (LH) power injection. The peculiarity and interest of exciting these electron fishbones by ECRH only or by LH only is also analyzed. Not only the mode stability is explained, but also the transition between steady state nonlinear oscillations to bursting (almost regular) pulsations, as observed in FTU, is interpreted in terms of the LH power input. These results are directly relevant to the investigation of trapped alpha particle interactions with low-frequency MHD modes in burning plasmas: in fact, alpha particles in reactor relevant conditions are characterized by small dimensionless orbits, similarly to electrons; the trapped particle bounce averaged dynamics, meanwhile, depends on energy and not mass.

TH/4-1Ra · Stability and Dynamics of the Edge Pedestal in the Low Collisionality Regime: Physics Mechanisms for Steady-State ELM-Free Operation

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Abstract: The peeling-ballooning model proposes that intermediate wavelength MHD instabilities are responsible for edge localized modes (ELMs), and impose constraints on the pedestal height. In typical discharges with ELMs, the pedestal goes unstable to coupled peeling-ballooning modes shortly before an ELM is observed [P.B. Snyder et al., Phys. Plasmas 9 (2002) 2037]. [P.B. Snyder et al., Nucl Fusion 44 (2004) 320]. In nonlinear simulations these modes are found to evolve explosively, leading to the ELM crash [P.B. Snyder et al., Phys. Plasmas 12 (2005) 056115]. However, at low collisionality, the bootstrap current in the pedestal region can be large, even very near the separatrix, and the discharge can be most unstable to current-driven kink/peeling modes, typically at relatively low mode number ($n \sim 1$–10). Recently, interesting ELM-free regimes, including both quiescent (QH) [W.P. West et al., Plasma Phys. Control. Fusion 46 (2004) A179], and resonant magnetic perturbation (RMP) H-mode [T.E. Evans et al., Phys. Rev. Lett. 92 (2004) 235003] have been observed to occur in this low collisionality regime. We systematically explore MHD stability in this regime, including the effects of a conducting wall and sheared toroidal flow. Both theoretical studies focusing on the effects of low collisionality and strong toroidal flow shear, and detailed analysis of several QH and RMP discharges will be presented. We propose rotationally-destabilized edge localized kink/peeling modes as the mechanism for the edge harmonic oscillation (EHO), which allows steady-state ELM-free operation in QH-mode, and quantitatively compare these predictions with observations. We find that the RMP plays the role of the EHO in RMP ELM-free discharges, allowing steady state operation at a controllable pedestal height, below the peeling-ballooning threshold for ELM onset. We present quantitative predictions of the density and shape parameters required for ELM-free operation and successful comparisons to DIII-D data, notably including access to QH mode at much higher density in the presence of strong shaping. Detailed calculations for ITER are presented.

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TH/4-1Rb · ELM crash theory: Relaxation, filamentation, explosions and implosions

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Abstract: Edge Localised Modes, or ELMs, are plasma instabilities that can lead to large transient heat loads on the divertor target plates of tokamaks. Extrapolations based on data from present tokamaks give concern that ELMs would cause unacceptable erosion of the target plates on ITER, but the uncertainties are large. An improved theoretical understanding would be beneficial in two ways: (i) constraining extrapolations to reduce the uncertainties and, (ii) identifying techniques to reduce the size of ELMs. It is widely believed that ideal MHD "peeling-ballooning" modes provide the trigger for ELMs, but little is known about their non-linear evolution and resulting transport processes. As a consequence, a predictive ELM theory remains elusive. This paper presents an improved theoretical understanding of the non-linear evolution of ideal MHD modes associated with the edge pedestal region, addressing both peeling and ballooning modes. To study the peeling mode, an extension of Taylor relaxation theory is employed to describe the post-crash state of the ELM. This provides, amongst other things, a prediction for the energy lost, and is expected to be appropriate for small, Type III ELMs. Predictions are broadly in line with measurements of energy loss due to small ELMs on MAST. To understand larger ELMs requires consideration of the non-linear evolution of the ballooning mode. A rigorous expansion of the ideal MHD equations is employed to predict the formation of plasma filaments. A new result is that these filaments can either explode into the scrape-off layer, or implode into the core, depending on the pedestal current density. Explosion, which may be associated with the larger Type I ELMs, occurs for sufficiently high current density. At lower current density, the filaments implode, possibly giving rise to smaller ELMs (eg Type II).

TH/4-2 · Integrated Simulation of ELM Energy Loss Determined by Pedestal MHD and SOL Transport

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Abstract: An integrated transport simulation code with a stability code for the peeling-ballooning modes and the scrape-off-layer (SOL) model has been developed for the first time in the world to clarify self-consistent effects of edge localized modes (ELMs) and SOL on the plasma performance. Experimentally observed collisionality dependence of the ELM energy loss is found to be caused by both the edge bootstrap current and the SOL transport. The bootstrap current decreases with increasing the collisionality and intensifies the magnetic shear at the pedestal region. The increase of the magnetic shear reduces the width of eigenfunctions of unstable modes and changes the mode number from medium-n to high-n. These effects reduce the area and the edge value near the separatrix of the ELM enhanced transport. On the other hand, when an ELM crash occurs, the energy flows into the SOL and the SOL temperature rapidly increases. The increase of the SOL temperature lowers the ELM energy loss due to the flattening of the edge gradient. The parallel heat conduction determines how the SOL temperature increases. For higher collisionality, the conduction becomes lower and the SOL temperature increases more. We found that, by the above two mechanisms, the ELM energy loss decreases with increasing the collisionality.

TH/P2-1 · Nonlinear equilibria, stability and generation of zonal structures in toroidal plasmas

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Abstract: The long-lived saturated zonal flow (ZF) structures, spontaneously generated by drift-wave (DW) turbulence, are viewed as generators of nonlinear equilibria. Here, we derive a nonlinear evolution equation for the zonal response, valid for arbitrary wavelengths, which gives the temporal evolution of the zonal structures, whose time-asymptotic behavior corresponds to the nonlinear equilibrium. Its corresponding stability, meanwhile, determines the nature of the ZF instability and the nonlinear up-shift of turbulent transport thresholds. On a shorter-time scale, the temporal evolution of the zonal response describes the DW-ZF generation and the regulation of the DW intensity by the ZFs. As applications, we present a kinetic stability analysis of zonal structures in the ITG case and estimate the nonlinear up-shift of the threshold for ITG driven turbulent transport; we also derive the dispersion relation for the spontaneous excitation of ZFs by CTEM.
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TH/P2-2 · Turbulent Transport in Spherical Tokamaks with Transport Barriers

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Abstract: Spherical tokamaks (STs) such as MAST, particularly at high beta, provide an important test of theoretical models for turbulent transport, which can then be used in a predictive mode for tokamaks in general. A combination of computational studies using the turbulence simulation code GS2 and analytic models has been used to understand both the 'numerical experiments' and the real experimental results. Linear stability studies of L and H mode MAST discharges exhibit a range of dominant instabilities, depending on the wave-number: ion and electron temperature gradient (ITG and ETG, respectively) and trapped electron (TEM) modes, with electromagnetic effects being important even at modest beta values and giving rise to kinetic ballooning modes and micro-tearing modes. However the correspondence between these results for micro-tearing modes and large aspect ratio analytic theories is limited, pointing to the importance of geometry, as treated fully in GS2. While rotation shear in MAST is expected to stabilise longer wavelength modes, non-linear GS2 calculations of the electron thermal diffusivity from ETG modes yield values typical of MAST, $\sim 5\mathrm{m}^2\mathrm{s}^{-1}$. Analytic studies of the collisionality dependence of the TEM reveal a broad, stable spectrum at low collisionality. The value of the relevant critical collisionality parameter is consistent with the appearance of electron ITBs in MAST. The stabilising effect of rotation shear, characteristic of ITBs, has been demonstrated for a generic class of drift wave models. Finally, while this rotation can be generated initially by the Reynolds' stress from turbulence, we find off-diagonal elements of neoclassical toroidal momentum transport in the presence of impurities can sustain it by transmission from edge to core.
⋆ This work was jointly funded by the United Kingdom Engineering and Physical Sciences Research Council and by EURATOM.

TH/P2-3 · Gyrokinetic Simulations of ETG and ITG Turbulence

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Abstract: We have carried out an investigation [1], using simulations and analytical theory, of the discrepancy between gyrokinetic continuum-code [2] and particle-in-cell (PIC) simulations [3] of electron-temperature-gradient (ETG) turbulence. Ref. [2] indicated a sufficiently large value of the electron thermal conductivity to account for anomalous electron thermal transport in tokamaks, while [3] gave significantly lower values. We have reproduced the key features of the results reported in [3] using the flux-tube gyrokinetic PIC code, PG3EQ [4], thereby eliminating global effects and as the cause of the discrepancy [1]. We show [1] that the late-time low-transport state in both these PG3EQ simulations and those reported in [3] is a result of discrete particle noise. Since discrete particle noise is a numerical artifact, the low value reported in [3(b)], along with conclusions about anomalous transport drawn from it, are unjustified. A detailed theory of the spectrum of noise fluctuations in a gyrokinetic particle simulation has been developed which greatly facilitates this demonstration [1]. This theory has no free parameters, and gives excellent agreement both with the fluctuation spectrum and the transport levels observed at late times in gyrokinetic particle simulations when the noise dominates [1]. In our attempts to benchmark PIC [4] and continuum [2] codes for ETG turbulence at the plasma parameters used in [3], both produce very large intermittent transport. We have therefore undertaken benchmarks at an alternate reference point, magnetic shear $s = 0.1$ instead of $s = 0.796$, and have found that PIC and continuum codes reproduce the same transport levels. Scans about this new reference point have been used to investigate the parameter dependence of ETG transport and to elucidate previously proposed saturation mechanisms. New results on the balances of zonal-flow driving and damping terms in late time quasi-steady ITG turbulence and on real-geometry gyrokinetic simulations of shaped DIII-D discharges will also be reported.
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TH/P2-4 · Particle pinches in fluid and kinetic descriptions

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Abstract: It has been found that gyro-fluid resonances strongly reduce particle pinches. This has been seen both in a parameter scan and for the JET L-mode 51034 where only a non-dissipative model is able to support the peaked experimental density profile. Extrapolations to quasilinear kinetic models are made.

TH/P2-5 · Eigenmode Analysis of Geodesic Acoustic Mode and Zonal Flow

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Abstract: Both the low/zero frequency zonal flow (ZF) and higher frequency oscillating ZF, so called the geodesic acoustic mode (GAM), first and foremost, are plasma eigenmodes. The progress of the experimental research on multiple GAM oscillations stimulates the research on various types of branches in the family of ZFs. In the work, we employ a linear gyrokinetic model in collisionless toroidal plasmas with an electrostatic potential nearly constant around a magnetic surface, and then the plasma response is analytically solved. Besides the trivial zero frequency solution and the standard GAM solution, a branch of low frequency mode and an infinite series of ISW-like modes are found. The ISW-like modes has a frequency spectrum roughly with a progression of $\sqrt{n\pi}$ times the transit frequency and strongly damped. The low frequency eigenmode has a rigid zero frequency for low q but oscillates with a finite frequency for larger q. The critical value of q increases and the finite frequency decreases with an increasing electron/ion temperature ratio. This low frequency eigenmode relaxes on time scaling with the order of transit frequency. Moreover, the finite wavelength effect on the GAM can be studied by the model, and then the eigenmode structure is expected.

TH/P2-6 · News from the Geodesic Acoustic Mode: Magnetic Shear-, q-, and Geometry Effect

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Abstract: The generation of Geodesic Acoustic Modes (GAM), oscillating poloidal shear flows, has been studied in greater depth by three-dimensional turbulence simulations. A change of the magnetic shear, in particular, a switch to negative shear profoundly affects the amplification mechanism of the GAMs. Negative shear reverses the symmetry of the turbulence modes with respect to the shear flows, altering the sign of the Stringer-Winsor forces. The phenomenon readily suggests an experimental test, which would quantify the role of the Stringer-Winsor effect in comparison to the Reynolds stress in exciting the GAMs. The safety factor q controls the coupling of the GAMs to the parallel velocity, i.e., sound waves. Lowering q increases this coupling. Since the parallel sound waves in turn are heavily damped by the turbulence they act as a loss channel. Thus sufficiently low q cause a quench of the GAM activity, as has been found in recent experiments, too. Finally, the shape of the flux surfaces has great influence on the frequency of the GAMs and the relative strength of the Stringer-Winsor force. Elongation of the plasma column reduces the GAM frequency, and simultaneously increases the energy transfer term due to the up-down asymmetry of the anomalous flux, which can have a dramatic impact on the shear flow level and transport. Again, the results suggest a relatively straightforward comparison with experiments.

TH/P2-7 · Transport enhancing features of electron temperature gradient driven turbulence

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Abstract: Electron Temperature Gradient driven (ETG) modes have been proposed as a source of anomalous electron thermal loses in tokamaks. It is widely acknowledged that the electrostatic potential in ETG turbulence can develop into radially elongated structures known as streamers. Understanding the conditions that permit streamers to produce experimentally significant transport is a topic of great interest. Using the gyrokinetic fluxtube code, GS2, the effects of trapped electrons and charge non-neutrality (Debye length larger than the gyro-radius) on the thermal transport resulting from ETG turbulence is studied, in large aspect ratio, low beta equilibria (s-alpha equilibria), with adiabatic ions. It is found that trapped electrons are essential, in the cases studied here, for producing large thermal diffusivities. ETG mixing

length estimates of the thermal transport, derived by retaining the Debye term in Poissons equation, suggest an enhancement in the diffusivity. The simulation results presented here show an increase in the turbulent heat flux with the ratio of the Debye length to the gyro-radius.

TH/P2-8 · Electron Transport Driven by Short Wavelength Trapped Electron Mode Turbulence

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Abstract: An outstanding issue in tokamak confinement studies is the origin of the anomalous electron thermal transport in internal transport barriers (ITB), where the ion transport is reduced to the neoclassical level. In global gyrokinetic particle simulations, we found that the short wavelength fluctuations of trapped electron mode (TEM) turbulence drive a large electron heat flux, while the longer wavelength TEM fluctuations drive large ion heat and particle fluxes. Since the formation of the ITB is often accompanied by the generation of equilibrium sheared flows, these longer wavelength fluctuations can be easily suppressed or broken up into smaller eddies by the strong flow shear. On the other hand, the short wavelength fluctuations can survive the shearing effects. Therefore, the small scale TEM turbulence is a viable candidate for driving the electron thermal transport in the ITB regions. We also found that trapped electrons enhance the ion temperature gradient (ITG) fluctuation and transport level. However, these ITG fluctuations drive a smaller electron thermal transport because trapped electrons do not resonate with the ITG modes. The short wavelength TEM turbulence presents interesting nonlinear physics and challenging theoretical issues. The time scale separation between the EXB nonlinearity and the polarization nonlinearity no longer exists in TEM turbulence. Zonal flows and electron temperature perturbations with a short radial wavelength are strongly driven by the polarization between ions and electrons. Our simulations utilized a global GTC code, which has been upgraded to treat the electron dynamics and electromagnetic fluctuations. To ensure the physics fidelity of the simulation, we have studied extensively the convergence of the most challenging simulation of electron temperature gradient (ETG) mode turbulence, and have demonstrated convergence with respect to the number of particles in global ETG simulations using a total of 30 billion particles and 4800 processors with a computing power of 24 Tera-Flop.

TH/P2-9 · Statistical characteristics of turbulent transport dominated by zonal flow dynamics

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Abstract: Large scale structures (LSS) such as zonal flows (ZF), streamers, generalized Kelvin-Helmholtz (KH) mode, geodesic acoustic mode (GAM), etc, are considered to regulate the turbulent transport in magnetically confined fusion plasmas. However, the precise roles and underlying physical mechanisms of LSS have not been clearly understand. In order to control the turbulent transport, better understandings of the LSS are necessary. Thus, we have investigated the characteristics of electron/ion temperature gradient (ETG/ITG) turbulences based on our gyro-fluid turbulence simulations. Here, we compared the ZF-dominated plasmas with the turbulent plasmas, and studied the role of the LSS on turbulent transport, in particular, for the relation between transport suppression and the LSS in plasmas. Characteristics of turbulent transport dominated by zonal flows and nonlinearly excited large scale structures are investigated by ETG/ITG gyro-fluid simulations. Main results found in this research are as follows. (i) The zonal flows change the characteristics of turbulence from "homogeneous" to "inhomogeneous", in which disintegrated micro-scale vortices and nonlinearly excited macro-scale vortices appears at different radial zones, exhibiting a two-scale nature in turbulence. (ii) The fractal dimension is considerably reduced with the increase of the partition of zonal flows to the total fluctuation energy, accompanied by the disappearance of the exponential PDF tail of heat flux. This suggests that fractal dimension and PDF can be used for experimental identification of turbulent structures, such as the dominancy of zonal flows. (iii) The reduction of the heat flux by zonal flows results from the synergetic interplay of two exclusive mechanisms, i.e. the reduction of coherence and the phase synchronization between the poloidal electric field E_y and the pressure perturbation p.

TH/P2-10 · Evolution of Anomalous Transport in Shear Flow of Toroidal Devices

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Abstract: The experimentally determined velocity shear in Uragan-3M (U-3M) torsatron in (L-H) – like transition is up to ten times greater than the drift wave frequency. The temporal evolution of the drift and Alfvén turbulence and the resulted anomalous transport are considered under the conditions of modest (flow shear parameter ν_0' is of the order of the instabilities growth rate) and strong flow shear (flow shear parameter ν_0' is greater than the drift frequency) on the grounds of the non-modal approach. The studies of the dynamics of packets of nonmodal drift waves, eta-i modes, Alfvén waves have shown that the wave packets are stagnated or reflected by the shear flow. The components of the group velocity along the flow shear rapidly vanish with time. The calculated anomalous ion transport displays rapid decreasing with time. The comparative analysis of the linear non-modal effects and nonlinear turbulent effects, such as the effect of the enhanced by flow shear nonlinear decorrelation, on the instabilities evolution and saturation is performed. The renormalized kinetic theory of drift and drift-cyclotron instabilities of a plasma with a transverse inhomogeneous shear flow, which accounts for the turbulent scattering of ions across the shear flow and the enhanced scattering of ions along the shear flow, is developed. The saturation level of the shear-flow-modified drift turbulence and the reduced by the shear flow anomalous fluxes of ions and electrons are determined.

TH/P2-11 · Interplay between zonal flows/GAMs and ITG turbulence in tokamak plasmas

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Abstract: Zonal flow (ZF) behavior and its effect on turbulent transport in tokamak plasmas are investigated by global fluid simulations of electrostatic ion temperature gradient (ITG) driven turbulence. In the simulations, the frequency of oscillatory ZFs called geodesic acoustic modes (GAMs) appearing in a high q (safety factor) region does not vary continuously with a radius, but its radial variation is step-like and the GAMs have the same frequency in a certain radial region. It is found that the radial width of the region in which the GAMs have the same frequency is almost proportional to the geometric mean of an ion Larmor radius and a minor radius of a plasma in positive shear tokamaks, while the radial wavelength of the GAMs is proportional to the ion Larmor radius. Since the GAM frequencies are close to those of the ITG turbulence, the GAM dominant region is a high transport region and the statinoary ZF region is a low transport region. The turbulent transport is affected by the nonlocal behavior of the GAMs. It seems that the radial width of the region connecting the low tranport region where the stationary ZFs are dominant with the high transport region where the GAMs are dominant is related with the nonlocal width of the GAMs.

TH/P2-12 · Thermal Diffusion by Stochastic Electromagnetic Fluctuations

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Abstract: A new simple method has been developed to evaluate the thermal diffusion brought by coexisting homogeneously stochastic electrostatic and electromagnetic fluctuations. This method is significantly useful, because the synergetic treatments of electrons and ions in the electromagnetic gyrokinetic simulations are considered to be quite difficult in the standard experimental situations of LHD and tokamaks. As a most simple case, thermal conductivities for electrons and ions are considered in a large aspect straight tokamak with a small gyro-radius and negligible magnetic shear and negligible equilibrium $E \times B$ flow shear. Those analytical formulae applicable to the range beyond so-called quasi-linear limit show that the thermal diffusion of electrons (ions) is mainly dominated by magnetic (electrostatic) fluctuations in the experimentally relevant situations, even if both magnetic and electrostatic fluctuations coexist.

TH/P2-13 · Gyrokinetic full f modelling of plasma turbulence in tokamaks

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Abstract: Understanding turbulence in fusion plasmas has become a crucial task in order to achieve sufficient confinement level in large-scale devices like ITER. The ELMFIRE full f nonlinear gyrokinetic simulation code has been developed for calculation of plasma evolution and dynamics of turbulence in tokamak geometry. ELMFIRE is based on a particle-in-cell algorithm with direct implicit treatment of ion polarization and consideration of either kinetic or adiabatic electrons, and also impurities. The gyrokinetic model performs averages over the gyro-orbits while keeping accuracy over the most interesting time and length-scales regarding turbulence development. Opposite to delta f algorithms, the full f method is suitable for calculating also strongly perturbed plasmas including wide orbit effects, steep gradients and rapid dynamic changes. The Monte-Carlo method easily allows for changes in geometry like limiter consideration. ELMFIRE has been successfully tested against other relevant codes at calculating linear growth rate of unstable modes and nonlinear saturation of instabilities. It also produces a radial electric field consistent with the neoclassical theory even in the presence of turbulence. The code is also being tested against experimental data from FT-2 tokamak about density fluctuations from Doppler reflectometry measurements. ELMFIRE has been specifically prepared for large-scale runs on massively parallel computers, aiming for the global simulation of actual devices like ASDEX-U or JET. Use of multicluster grid computing is under consideration as parallel scalability to very high number of processors has been taken care of while keeping high efficiency.

TH/P2-14 · Simultaneous Enhancement of Core Electron Density and Temperature by Synergistic Effect of Molecular Beam Injection and Shock due to Toroidal Flow

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Abstract: High pressure supersonic molecular beam injecting (SMBI) into tokamaks can increase the core electron density. In the meantime, however, the core electron temperature decreases due to the density peaking. In the present paper, it is shown that the synergistic effect of the SMBI and the shock wave due to zonal flow can enhance both the core electron density and temperature.

TH/P2-15 · Global simulation of the GAM oscillation and damping in a drift kinetic model

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Abstract: The collisionless damping of the geodesic acoustic mode (GAM) is investigated by a drift kinetic simulation. The main subjects of the study are to analyze how the magnetic configuration and the finite-orbit-width (FOW) effect of the ion drift motion affect the collisionless damping rate of the GAM. We utilize the neoclassical transport code "FORTEC-3D", which solves the drift kinetic equation based on the delta-f method including the FOW effect, to study these issues. In FORTEC-3D, the ion distribution function in five-dimensional phase space is solved. The evolution of radial electric field is solved simultaneously in the simulation. The outstanding features of our simulation are that it can solve the time evolution of plasma in a complicated three-dimensional magnetic field, and that the whole confined region can be treated at once. It enables us to simulate the non-local, global behavior of the GAM oscillation including the FOW effect in realistic configurations like Large Helical Device (LHD) in NIFS, Japan. In recent analyses on the GAM and zonal flow it is found that the FOW effect and the helical components of magnetic field strength change the damping rate of the GAM oscillation. We confirmed the change of the damping rate by these effects in our simulation. For example, the dependence of the damping rate on the FOW effect is investigated in a simple tokamak geometry by changing the magnetic field strength. As the drift orbit width swells in weaker magnetic field, the GAM damping rate becomes larger. The collisionless damping is about 10 times as rapid as the expectation neglecting the FOW effect. On the other hand, from the simulations in LHD model helical configurations, it is found that the damping rate is mainly determined by the effect of helical ripples. In LHD, the strength of each helical component of magnetic field spectrum can be controlled by shifting the magnetic axis. Sideband components appear as the axis is shifted inward, and it is found that these components make the GAM damping rapider. This result suggests the possibility of controlling both the neoclassical transport level and the GAM oscillation, or zonal flow, in LHD plasma. In the conference, we will present more detailed, quantitative discussion on the simulation result and comparison with analytic estimation of the GAM damping rate.

TH/P2-16 · Nonlinear Inward Particle Flux in Trapped Electron Mode Turbulence

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Abstract: Weakly collisional trapped electron mode (TEM) turbulence has a robust inward particle flux component associated with a linearly stable eigenmode that is excited nonlinearly by spectral energy transfer from the unstable TEM mode. The nonlinear mixture of the two eigenmodes achieved in saturation cannot be described by the quasilinear approximation, hence the inward flux component, which combines with the outward quasilinear flux to produce the net flux, is fundamentally nonlinear. The net flux, which remains outward but is significantly reduced by the inward component, depends on the gradients of density and temperature. This dependence, which establishes whether the flux is diffusive, convective, or something else, is sensitive to the details of the saturation. Saturation is calculated asymptotically in an ordered expansion in collisionality and the ratio of density to temperature gradient scale length. Spectral transfer is highly anisotropic and saturation must account for the energy transfer to zonal modes with zero poloidal wavenumber. Even though zonal modes do not contribute directly to the particle flux they change the fluctuation level and gradient scaling of both the unstable and stable eigenmodes. The result is a flux that is neither diffusive nor convective, but is driven by temperature gradient and enhanced by density gradients that are flat or nearly so. Near the instability threshold the inward component is particularly strong.

TH/P2-17 · Pulse Propagation in a Simple Probabilistic Transport Model

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Abstract: Experimental observations of the characteristics of transport in fusion plasmas, such as the Bohm scaling of the energy confinement time, the existence of canonical profiles in the presence of on-axis and off-axis heating, or the anomalous propagation of cold and hot pulses, indicate that the description of transport based on a local relation between fluxes and thermodynamic forces is not completely satisfactory. Recently, we proposed an alternative (non-local) approach, based on the continuous time random walk, characterized by a step distribution that depends on the local gradient, which has had some success in modeling such behavior, at least in a qualitative sense. Here, we investigate the propagation of localized perturbations using a simplified model, with the purpose of improving the understanding of these complex systems, and showing how such analyses may provide a method for extracting the super-critical behavior of transport from experimental observation.

TH/P2-18 · Test Particle Statistics and Turbulence in Magnetically Confined Plasmas

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Abstract: A component of test particle motion in turbulent magnetized plasmas is the stochastic $E \times B$ drift. This drift determines a trapping effect or eddy motion in turbulence with slow time variation, which generates non-standard statistical behavior of the trajectories: memory effects, non-Gaussian probability and coherence. Two consequences of the trapping are discussed. The first is related to the memory effect and consists in a strong modification of the diffusion coefficients. We have developed a model that includes, besides the $E \times B$ stochastic drift, the parallel motion (uniform and produced by the parallel electric field of the turbulence), average flows, collisions and large Larmor radius effects. Trapping influences not only the values of the diffusion coefficients but also their scaling laws. We present systematic analyzes of the diffusion regimes and of their specific conditions with the aim of providing a tool for the experimental studies of transport coefficients. The second part of the paper deals with the quasicoherent behavior and with the trajectory structures that are shown to appear due to trapping. We present here the first results on the effects of trajectory trapping on the dynamics of the drift turbulence. We show that the presence of trajectory trapping determines an important influence on the turbulence, which essentially consists of the tendency of structure formation in the turbulent potential.

TH/P2-19 · Interaction of Drift-Tearing (Mesoscopic) Modes with Coherent and Turbulent Microscopic Structures

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Abstract: The excitation and evolution of the mesoscopic modes involving magnetic reconnection remains a challenging theoretical issue as experiments on high temperature plasmas provide evidence for their presence while the original and well established theories indicate that they should be hampered by the effect of electron Landau damping or longitudinal electron thermal conductivity. Considering the interaction of these modes with a relevant kind of background micro-structure (coherent or turbulent) then becomes a necessary development of the theory. Within this general framework, electromagnetic electron temperature gradient driven modes that produce localized magnetic reconnection at the scale of the collisionless electron inertial skin depth are considered the most appropriate source of relevant microstructures and are shown to sustain a significant electron temperature variation along the field lines defeating the effect of the longitudinal thermal conductivity. As an alternative and as a minimally working self-consistent model, we analyze the coupling between a tearing mode and electrostatic drift wave turbulence. In this case the principal effect of the drift waves is to pump the tearing mode via negative viscosity, consistent with the classical notion of the inverse cascade in quasi-2D turbulence.

TH/P2-20 · Effects of Magnetic Island Induced Symmetry Breaking on Plasma Confinement and Island Evolution in Tokamaks

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Abstract: Magnetic islands exist in most tokamak discharges. The toroidal symmetry in $|B|$ is broken when an island is embedded in the equilibrium magnetic field B in tokamaks. Plasma confinement properties in the vicinity of an island are different from those in the region away from the island. Physically, this is a result of the modifications on the plasma viscosity in the presence of the island. Interesting plasma confinement properties including, besides the usual particle and energy fluxes, the momentum transport and the bootstrap current, are derived from the island induced plasma viscosity. The consequence of the momentum transport process modifies plasma flow and the radial electric field in the vicinity of an $m = 1$ island, and provides an explanation for plasma confinement improvement in snakes. Here, m is the poloidal mode number. The additional bootstrap current density induced by the presence of an island modifies the island evolution. It is found that island-induced bootstrap current has stabilizing influence on the island dynamics for $m > 1$ islands in plasmas with high poloidal beta. Here, poloidal beta is the ratio of plasma pressure to the poloidal magnetic field pressure.

TH/P2-21 · Multi-Scale-Nonlinear Interactions among Micro-Turbulence, Magnetic Islands, and Zonal Flows

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Abstract: We investigate multi-scale-nonlinear interactions among micro-instabilities, macro-scale tearing instabilities and zonal flows, by solving reduced two-fluid equations numerically. We find that the nonlinear interactions of these instabilities lead to fast breaking of magnetic surfaces then this breaking spreads the micro-turbulence over the plasma. These multi-scale-nonlinear interactions can explain complicated evolution of fluctuation observed in torus plasma experiments because micro-turbulence and MHD instabilities usually appear in the plasma at the same time, in spite of the fact that effects of micro-turbulence and MHD instabilities on plasma confinement have been investigated separately. For instance, MHD activities are observed in reversed shear tokamak plasmas with a transport barrier related to zonal flows and micro-turbulence, and micro-turbulence is observed in Large Helical Device plasmas that usually exhibit MHD activities. Our goal is to understand the mechanism of the macro-scale MHD activities and their effects on the disruption in the reversed shear plasmas based on the analysis of multi-scale-nonlinear interactions among the micro-turbulence, macro-scale MHD and zonal flows.

TH/P2-22 · Integrated Tokamak Modelling: the way towards Fusion SimuIators

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Abstract: An European task force on integrated tokamak modelling has been activated with the long-term aim of providing the EU with a set of codes necessary for preparing and analysing future ITER discharges, with the highest degree of flexibility and reliability. The task force is structured under seven projects establishing the foundations of the fusion simulators.

TH/P2-23 · From extensive micro-stability analysis of experiments to integrated modelling

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Abstract: So far, interpretative or predictive simulations of fusion plasmas are performed using either empirical transport models or gyro-fluid, mixed gyro-fluid/gyro-kinetic models. The most complete turbulence simulations require full non-linear gyro-kinetic codes. However, such codes cannot be coupled to an integrated simulation since they are still too demanding in terms of computing time. Nevertheless, fast linear gyro-kinetic codes can provide quasi-linear heat, particle and momentum fluxes that can be then coupled to transport codes. This approach allows improving significantly turbulent transport interpretation and prediction. Indeed, linear gyro-kinetic codes predict precisely the turbulence threshold and its parametric behaviour. They also allow clarifying basic transport scaling. For this purpose, we used the code Kinezero [C. Bourdelle et al., Nucl. Fusion 42, 892 (2002)] coupled to the integrated transport code CRONOS [V. Basiuk et al., Nucl. Fusion 43, 822-830 (2003)]. In the first part of the paper, an extensive analysis of experiments is reported, mainly for advanced scenarios performed in existing tokamak devices. The impacts of the key parameters on the plasma performance are analysed, namely the magnetic shear s and the MHD a parameters [C. Bourdelle et al., Nucl. Fusion 45, 110-130 (2005)]. The a stabilization reinforced by high pressure gradient, |∇P|, is particularly discussed. In the second part of the paper, first results of CRONOS interpretative/predictive simulations are presented. In these simulations, we use a transport model derived from quasi-linear heat, particle (including impurity) and momentum fluxes provided by Kinezero. A possibility to navigate the plasma into the positive feedback loop thanks to the a stabilization is also discussed. Finally, we report the consequences of such an approach on transport prediction for ITER.

TH/P2-24 · Impurity Transport in ITER-like Plasmas

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Abstract: The presence of impurities in the edge region of tokamaks can be beneficial because a strongly radiating boundary distributes plasma power loss. But many tokamak discharges suffer from unwanted impurity accumulation in the plasma core. Impurity transport is critical for ITER and therefore it should be given careful consideration. In this work we compute the impurity transport from neoclassical theory and compare it with transport calculated from the reactive drift wave model of turbulent transport for a specific ITER-like scenario. Neoclassical transport is driven by parallel friction dynamics, and is not affected significantly by the fact that the ion cross-field transport is dominated by fluctuations. Thus it is likely that both neoclassical and anomalous transport co-exist. The reactive model used here (usually called Weiland model) has been used extensively in describing the present database and also for making ITER predictions It uses an "advanced" reactive fluid model where "advanced" here refers to the rule for closure which allows us to use the model close to the fluid resonance. In the version used here it has two independent ion species with the same physics included for both. The particle transport for the main species was tested successfully on JET discharges. The particle pinch depends particularly strongly on the magnetic drift frequency. Because of this, species with larger Z have a weaker particle pinch thus giving a favourable net effect on the effective Z. The particle pinch is particularly relevant for ITER since central fuelling will not be possible there. In fact the particle pinch has been found to improve ITER performance significantly. We consider a specific high-Q ITER-like scenario and keep Z_{eff} constant. Neoclassical and turbulent impurity transport are of the same order of magnitude and have opposite signs. The neoclassical transport is outwards due to temperature screening and it dominates for large Z. For low Z, neoclassical transport is weaker than turbulent transport except for the central part of the plasma. The turbulent transport leads to an inward pinch of the impurities. However, the main ion inward pinch

is much stronger, so the impurity accumulation should not be a problem in ITER-type plasmas, since the net flow is either very weak or outwards.

TH/P3-1 · MHD Flow Layer Formation in Vicinities of Rational Flux Surfaces of Tokamak Plasmas

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Abstract: The crucial role of sheared flows in formation of external and internal transport barriers (ITBs) in advanced tokamak discharges has been demonstrated in recent decades. Two sources driving such flows have been identified in the studies. The first is the mean sheared flows created by external sources and/or gradient of plasma pressure. The second is the zonal flows formed in nonlinear interaction of small scale turbulent fluctuations through wave coupling or inverse cascade. A third kind flow, MHD flows, is proposed and studied in this work. The driving mechanism for the MHD flows is the magnetic energy released in reconnection process of tearing mode nonlinear development. The flows have the same helical structure as the magnetic field at the resonant surfaces in the toroidal and poloidal directions. They are confined in the vicinities of the resonant surfaces and have strong shear at the boundaries of the reconnection layer. The radial structure and the amplitudes of the flows are estimated to be compatible with the requirements for suppression of transport producing turbulences. The possible correlation of the flows with ITB formation is discussed.

TH/P3-2 · Runaway Electron Generation during Plasma Shutdown by Killer Pellet Injection

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Abstract: Disruptions in high-current tokamaks can be accompanied by runaway electron generation. As the plasma cools down quickly in the thermal quench of a disruption, a large toroidal electric field is induced which accelerates some electrons to relativistic energies which severely damage the first wall on impact. To mitigate disruption problems it has been proposed that “killer” pellets could be injected into the plasma in order to safely terminate the discharge. Killer pellets enhance radiative energy loss and thereby lead to rapid cooling and shutdown of the discharge. However, pellets may also cause increased runaway production, as has been observed in several tokamaks. During pellet injection there are two competing effects that may affect runaway generation: the pellet increases the electron density and therefore suppresses acceleration of runaway electrons because of higher collisional friction, but the pellet also increases the plasma resistivity due to cooling and higher Z, leading to an increased toroidal electric field and more runaway acceleration. There are three runaway generation mechanisms: the “Dreicer” mechanism, caused by Fokker-Planck diffusion into the runaway region of velocity space; a ”burst” of runaway production caused by the rapid cooling of the bulk plasma; and the “avalanche” mechanism, where existing runaways produce new ones through knock-on collisions with thermal electrons. In high-current tokamaks, the role of the two first mechanisms is to provide a “seed” for the runaway avalanche. In the present work, two aspects of the runaway dynamics in connection with killer pellet induced fast plasma shutdown are considered. First we give a criterion for whether runaway bursting is more important than Dreicer production, by solving the kinetic equation in a cooling plasma and estimating the number of electrons in the runaway region. Second, we determine the post-disruption runaway current profile by solving the equation for runaway production coupled to an equation for the evolution of the toroidal electric field. To provide the evolution of the background plasma density and temperature we rely upon a pellet ablation code. In this way we can investigate the effect of varying pellet size and composition.

TH/P3-3 · High-m Multiple Tearing Modes in Tokamaks: MHD Turbulence Generation, Interaction with the Internal Kink and Sheared Flows

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Abstract: During the dynamical evolution of a tokamak plasma with a non-monotonic current profile, multiple resonant surfaces with the same safety factor (q) value may be formed. If the inter-resonance distances are sufficiently small, double or triple tearing modes (DTMs, TTMs) with high poloidal mode numbers may be strongly destabilized under such conditions. The growth rates of such high poloidal-number modes (i.e., high-m modes) are typically much higher than those of the lowest poloidal-number modes, especially that of the $m = 1$ mode, which typically has a large mode structure. In both DTM and TTM cases, it is shown that nonlinear evolution of low-m modes are significantly enhanced by the unstable high-m modes and often leads to the generation of MHD turbulence in the region bounded by the original rational surfaces. In this work, nonlinear evolutions of tokamak plasmas driven by high-m DTMs or TTMs instabilities at q = 1 or 2 resonant surfaces are examined. Their possible role for sawtooth crashes and effect on sheared flows associated with internal transport barriers (ITBs) are discussed.

TH/P3-4 · Equilibria and Stability in Partially Relaxed Plasma-Vacuum Systems

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Abstract: Since the early working of Grad (Phys. Fluids 10(1), 1967), the existence and stability of 3D magnetic configurations in ideal MHD has bedeviled magnetic containment theory. The 3D problem is related to a lack of integrability of the Hamiltonian poloidal flux function. In non-integrable systems, this Hamiltonian can be written as the sum of an integrable part plus a non-integrable perturbation. If the non-integrable perturbation is vanishingly small (for instance, a tokamak), flux surfaces exist everywhere. As the perturbation is increased, flux surfaces with rational rotational transform are destroyed. The last surviving flux surfaces are KAM surfaces with strongly irrational rotational transform. In regions between these KAM surfaces, the magnetic field ergodically spans the volume, and so the pressure gradient is zero. In this work, we develop a stepped-pressure profile model, in which the pressure across the plasma is piece-wise constant, and the field obeys the Beltrami equations in regions between ideal MHD barriers, at which pressure jumps occur. As a first step en-route to fully 3D ideal MHD solutions, we solve for a multiple-interface cylindrical plasma-vacuum system, where analytic solutions are available. In turn, these permit a detailed exploration of the equilibrium constraints, magnetic configuration and stability. For a given number of interfaces, we explore optimization of the field configuration (via variation of constraints) to achieve peak beta. Also, we compare the convergence of the field structure of a smooth pressure profile equilibrium to a stepped pressure profile as the number of interfaces increases. The existence of advanced tokamak-like magnetic configurations (weak reverse shear in core) also prompts an investigation into an energy based study of the existence of Internal Transport Barriers (ITB's), particularly around the q = 2 and q = 3 surfaces, where ITB's are often observed. Finally, in separate working, an algorithm to construct stepped pressure profile equilibria in arbitrary 3D geometry is developed, and an example in 3D computed.

TH/P3-5 · Explosive growth and nonlinear dynamics of the forced magnetic island

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Abstract: Novel features of the explosive growth and the nonlinear dynamics of the forced magnetic island subject of the suppression by plasma rotation are studied by nonlinear MHD simulations. The island formation is a critical issue severely affecting the performance of the tokamak plasmas. There are two main mechanisms for the origin of the magnetic island. One mechanism is related to the unstable tearing mode and the other one is due to the forced magnetic reconnection from the external perturbation. The latter process can also be an important source of the seed island for the neoclassical tearing modes (NTM), where the MHD event like as the sawtooth oscillation also acts as the external perturbation for the target mode through the toroidal mode coupling. So far the theoretical work has been done to understand the forced magnetic island suppression by the plasma rotation and to evaluate its threshold value, while less attention has been paid to the subsequent break up process and the long term behavior of the forced island. These latter processes are important for understanding the island effect on tokamak confinement. In this paper, we study the whole process of nonlinear dynamics of magnetic island due to the growing external perturbation in rotating tokamak plasmas. It was found that the magnetic island grows explosively with changing its structure to cause a new energy source. This new energy source appears to be localized plasma current around the X-point. Contrary to the standard magnetic reconnection theory, this localized plasma current causes the enhanced magnetic reconnection in the low resistivity regime. As the result, the long term evolution of the forced magnetic island, i.e. the seed island for NTM, is dominated by the secondary reconnection as the resistivity becomes small. The secondary island formation will affect the bootstrap current contributions to the NTM evolution.

TH/P3-6 · MHD simulation on ablation cloud in tokamak and heliotron

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Abstract: It is well known that an ablation cloud drifts to the lower field side in tokamak plasmas, which leads to a good performance on fueling in tokamak. Such a good performance, however, has not been obtained yet in the planar axis heliotron; Large Helical Device (LHD) experiments, even if a pellet has been injected from the high field side. The purpose of the study is to clarify the difference on the cloud motion between tokamak and LHD plasmas by using the MHD simulation including ablation processes. It is found in tokamaks that the drifting motion is induced by a tire tube force, and that the pressure and density of the plasmoid have oscillation due to fast compressional Alfvén wave. The first trial simulations on the motion of the plasmoid with helical symmetry in a straight helical plasma also show that the plasmoid drifts to the lower field side similarly to tokamaks. However, an actual plasmoid drifts inward or outward of the torus depending on the location, since the plasmoid expands along the magnetic field and the rotational transform is greater than that in tokamaks in the plasma periphery. Thus, it is suggested that the difference of the pellet injections between the high field and the low field sides is reduced in helical plasmas.

TH/P3-7 · Plasma geometry and current profile identification on ASDEX Upgrade using an integrated equilibrium generation and interpretation system

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Abstract: The identification of ideal MHD equilibrium states at ASDEX Upgrade is the starting point for interpreting any diagnostic data dependent on knowledge of the flux surface geometry. The method of Function Parameterization (FP) starts with the Monte Carlo generation of a simulated equilibrium database, regression analysis of which yields simple functional representations of plasma geometry whose arguments are information-rich, uncorrelated linear combinations of simulated diagnostic signals. Once calculated, these FP expressions can be rapidly evaluated using experimental data. FP using magnetic data is in routine realtime use on ASDEX Upgrade for plasma position and shape control. An extension to FP using MSE data has recently been developed for realtime identification and control of the current profile on ASDEX Upgrade. Post-discharge interpretive equilibrium solutions are generated by the CLISTE code, which best fits a set of specified diagnostic data. CLISTE can include kinetic data and poloidal halo currents in the scrape-off layer as constraints on the equilibrium solution, a valuable feature which has been applied to ELM analysis. The code has recently been extended to interpret dB/dt data from magnetics and $\mathrm{d}\gamma/\mathrm{dt}$ data from MSE to yield a best fit solution to the time derivative of the Grad-Shafranov equation $-\Delta^\star \partial\psi/\partial \mathrm{t} = 2\pi\mu_0 \mathrm{R}\ \partial/\partial \mathrm{t}\ \mathrm{j}_\phi$. The $\partial\psi/\partial \mathrm{t}$ solution is used to calculate the flux surface averaged profile $\langle \mathbf{E}\cdot\mathbf{B}\rangle$ which can be used to calculate current drive from auxiliary heating methods via the equation $\langle \mathbf{j}\cdot\mathbf{B}\rangle_{\mathrm{aux.heating}} = \langle \mathbf{j}\cdot\mathbf{B}\rangle_{\mathrm{equil}} - \sigma\langle \mathbf{E}\cdot\mathbf{B}\rangle - \langle \mathbf{j}\cdot\mathbf{B}\rangle_{\mathrm{boot}}$ where $\langle \mathbf{j}\cdot\mathbf{B}\rangle_{\mathrm{boot}}$ is calculated from kinetic profiles and neoclassical theory and $\langle \mathbf{j}\cdot\mathbf{B}\rangle_{\mathrm{equil}}$ is an equilibrium output. This technique is being applied to analyse current profile modification by off-axis NBI on ASDEX Upgrade.

TH/P3-8 · Feedback Stabilization of Resistive Wall Modes in the Presence of Multiply-connected Wall Structures

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Abstract: The feedback stabilization of resistive wall modes (RWM) for realistic wall structures has been studied splitting the problem in two parts. In the open-loop part the complete set of eigenvalues and eigenfunctions of the plasma-resistive wall system without feedback currents has been determined. Then, in the closed-loop part an initial value problem has been formulated for the time evolution of the RWM and the currents of the feedback coils. The interaction of the feedback currents and of the RWM's are given by prescribing the feedback logic. The effectiveness of the feedback can be studied by solving the characteristic equation of the closed-loop system. The procedure has been implemented numerically: STARWALL code. The equilibrium data are provided by the VMEC code, the potential energy of the plasma perturbation is provided by the CAS3D stability code. The magnetic field in the vacuum region with the resistive wall has been calculated with a finite element procedure, an appropriate method for treating cases with multiply-connected wall configurations. It is assumed that the resistive wall modes are slow so that the kinetic energy of the plasma perturbation can be neglected. Applications for ASDEX Upgrade are presented.

TH/P3-9 · Computation of Toroidal-Current Reversed Equilibria for the JT60-U Tokamak

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Abstract: A toroidal-current reversed Grad-Shafranov (GS) equilibrium is computed, using poloidal-magnetic-field and plasma-pressure profiles retrieved from experimental JT60-U data [T. Fujita et al., Phys. Rev. Lett. 87, 245001 (2001)], which is representative of a typical current-hole discharge. Since first observed, stable plasma configurations for which the measured poloidal field is nearly zero throughout a significant region around the magnetic axis (the so-called current-hole) raised a number of questions about possible GS solutions displaying toroidal-current reversal. Indeed, the development in such magnetic configurations of a poloidal-field reversal layer, for which the enclosing toroidal current does vanish, poses several problems to conventional GS equilibrium solvers, percluding their use in toroidal-current reversal scenarios. However, recent developments enabled GS codes to cope with the poloidal-field reversal layer and to handle a large variety of internal plasma profiles [P. Rodrigues, J.P.S. Bizarro, Phys. Rev. Lett. 95, 015001 (2005)], allowing in this way a suitable modeling of experimental data. In current-hole regimes, the poloidal-field profiles obtained from motional-stark-effect (MSE) measurements display significant relative errors inside the core region, with the error bar spanning from small positive values into small negative ones. Therefore, such uncertainty does not exclude (at least by itself) a toroidal-current reversal GS equilibrium, which is herein computed using poloidal-field and plasma-pressure profiles that best fit available experimental data for current-hole discharges in the JT60-U tokamak. Computing GS equilibria in these scenarios may aid to understand the physics behind the reported resilience of tokamak magnetic configurations with a current-hole [T. Fujita et al., Phys. Rev. Lett. 95, 075001 (2005)].

TH/P3-10 · Partial Stabilization and Control of Neoclassical Tearing Modes in Burning Plasmas

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Abstract: Neoclassical tearing modes (NTMs) are magnetic islands which increase locally the radial transport and therefore degrade the plasma performance. They are self-sustained by the bootstrap current perturbed by the enhanced radial transport. The confinement degradation is proportional to the island width and to the position of the resonant surface. The $q = 2$ NTMs are much more detrimental to the confinement than the 3/2 modes due to their larger radii. NTMs are metastable in typical scenarios with $\beta_N > 1$ and in the region where the safety factor is increasing with radius. This is due to the fact that the local pressure gradient is sufficient to self-sustain an existing magnetic island. The main questions for burning plasmas are whether there is a trigger mechanism which will destabilize NTMs, and what is the best strategy to control/avoid the modes. The latter has to take into account the main aim which is to maximize the Q factor, but also the controllability of the scenario. In this paper we present different aspects of the above questions, in particular the role of partial stabilization of NTMs, the possibility to control NTMs at small size with little electron cyclotron heating (ECH) power and the differences between controlling NTMs at the resonant surface or controlling the main trigger source, that is the sawteeth.

TH/P3-11 · Dynamical origin of shear flow induced modifications of nonlinear magnetic islands

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Abstract: The nonlinear evolution of magnetic islands due to unstable classical or neoclassical tearing modes is a topic of much current interest particularly in the context of confinement limits for long pulse experiments in superconducting tokamaks. The stability characteristics of these resistive modes can be strongly affected by the presence of equilibrium plasma flows and recent numerical investigations employing a fully toroidal code based on generalized reduced MHD equations have revealed a number of interesting results. It has been found that differential flow provides a strong stabilizing influence leading to lower saturated island widths for the classical tearing mode and reduced growth rates for the neoclassical tearing mode. The effect of velocity shear is found to depend on the sign of the shear at the mode resonant surface with negative shear providing a stabilizing effect and positive shear acting in a destabilizing fashion. In this paper we present a detailed analytic understanding of these results through model calculations that trace the dynamical origin of the various flow induced effects. To assess the changes in the outer layer dynamics we calculate the modifications in the parameter delta prime from a general set of ideal MHD equations that include inertial contributions of flow as well as flow induced pressure profile changes. Next a generalized Rutherford model equation incorporating such a modified delta prime and shear flow contributions in the inner layer dynamics is derived and used to estimate the nonlinear saturated island widths as well as threshold conditions. Our analytic results are found to compare very favourably with the numerical findings of the toroidal reduced MHD code and should prove useful for interpreting similar numerical investigations carried out on more complex codes like NIMROD.

TH/P3-12 · Helicity fluctuation, generation of linking number and effect on resistivity

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Abstract: Magnetic configurations subject to fast local variation of helicity (like in the current sheet of a reconnection event) must respond by increasing the topological linking number. We estimate the increase of topological complexity using the linking number and the crossing number. This latter quantity is further expressed in terms of the average curvature of the magnetic field lines. The curvature drifts of electrons and ions combined with non-vanishing collisionality lead to an enhanced resistivity.

TH/P3-13 · Theoretical Studies on the Physics of Magnetic Islands

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Abstract: Three aspects of the physics associated with magnetic islands are investigated. (a) The threshold for the onset of magnetic island in tokamak is studied using the two fluids equations. For a sufficiently high beta plasma like that of ASDEX-Upgrade, the stability of a small island is found to be mainly determined by the electron diamagnetic drift frequency and the perpendicular heat diffusivity, and it can be driven unstable by the electron temperature gradient for a certain range of the diamagnetic drift frequency. With experimental data as the input, the spontaneous growing tearing mode observed on ASDEX-Upgrade is obtained from the numerical simulation. In the nonlinear stage the island decreases the local electron temperature gradient, which in turn leads to the mode saturation at a small amplitude. When including the bootstrap current perturbation, the mode can further develop into a large amplitude. The saturated island width is found to decrease for large diamagnetic drift frequency. (b) The heat diffusion across a local stochastic magnetic field is studied numerically. With the increase of the ratio between the parallel and the perpendicular heat diffusivity, the enhanced radial heat diffusivity due to the parallel transport along the field lines is found to be determined first by the additive effect of individual islands and then by the field ergodicity. A quasi-linear analytical theory are developed, which agrees with the numerical result. (c) Numerical modelling on the stabilization of NTMs by localized RF current drive are carried out. When the RF deposition width is much larger than the island width, the modulated RF current drive to deposit the RF current around the island's o-point is found to have a stronger stabilizing effect than a non-modulated one. A more effective way for stabilizing NTMs is found by using both the RF wave and a resonant helical field. The helical field decreases the island rotation frequency, leading to a longer island rotation period comparing with the slowing down time of the fast electrons and therefore a larger stabilizing effect by the RF current.

TH/P3-14 · Plasma Rotation Braking and Driving in Tokamaks

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Abstract: The current work addresses certain issues related to plasma rotation in tokamaks.We derive an explicit expression for the error-field-induced torque and determine the torque in ITER for a given error field strength in the case of shear Alfvén resonance using the AEGIS code. We find that the liquid metal wall can shield the error field effect by eliminating the resonant amplification and reversing self-enhancing synergistic sequence in the rotation braking. We also give an alternative theoretical explanation for the spontaneous rotation observed during the cyclotron wave heatings. Our theory agrees with experimental observations for various characteristic features: such as the co-current rotation direction for ion cyclotron radio-frequency heating, the counter-current rotation direction for electron cyclotron heating, and the scale of rotation as the plasma stored energy normalized by the plasma current.

TH/P3-15 · MHD Simulations for Studies of Disruption Mitigation by High Pressure Noble Gas Injection

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Abstract: The problem of disruptions is a major challenge for ITER which establishes engineering and operational limits. Mitigation of disruptions by high pressure gas injections (HPGI) has been demonstrated experimentally and successfully modeled with the NIMROD code. The role of MHD in cases where impurity penetration is shallow is seen in a series of Alcator C-Mod simulations employing a simple radiation model to cool the tokamak edge with an assumed penetration ranging from well outside to just inside the $q = 2$ surface. Results indicate that this shallow penetration is sufficient to trigger a core thermal quench, and predict a relationship between the delay before the quench onset and the separation between the cooling front and the $q = 2$ surface. In the simulations a 2/1 modes destroys the outer flux surfaces and a 1/1 mode levels the core temperature. The simulations qualitatively reproduce several features of a gas-jet induced thermal quench on C-Mod, but a more sophisticated model is required for a truly predictive code. Improved modeling of the impurities is achieved through the coupling of the 3D NIMROD MHD code and the 0D KPRAD radiation code, to obtain accurate radiation rates and track all charge state populations. Preliminary results from the new code are presented, and comparison is made with C-Mod bolometry data.

TH/P3-16 · Effects of Energetic Beam Ions on Stability Properties of Field-Reversed Configurations

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Abstract: Stability properties of Field-Reversed Configurations (FRCs) formed by counter-helicity spheromak merging method have been studied numerically using the nonlinear hybrid and MHD simulation code HYM, including the effects of neutral beam injection. It is shown that the beam ions can have a stabilizing or destabilizing effect on the global modes in FRCs, depending on the toroidal mode number n, the mode polarization, and the beam parameters. Linear simulation results agree well with a qualitative analysis based on a generalized energy principle. Nonlinear simulations show that the beam-driven instabilities saturate nonlinearly due to changes in the distribution function of the beam ions. A new stability regime has been found for FRCs with elongation $E \sim 1$, which requires a close-fitting conducting shell and energetic beam ion stabilization. It is shown that the $n = 1$ and $n = 2$ MHD modes can be effectively stabilized by a combination of conducting shell and beam ion effects, and that the residual weakly unstable $n > 2$ modes saturate nonlinearly at low amplitudes. The resulting configuration remains stable with respect to all global MHD modes, as long as the FRC current is sustained.

TH/P3-17 · Current Drive with Oscillating Magnetic Fields and Helicity Injection and Neutral Beam Injection in a D-He^3 FRC Reactor

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Abstract: The use of oscillating helical magnetic fields to produce and sustain the toroidal and poloidal currents in a Reversed Field Pinch (RFP) is investigated. A simple physical model that assumes fixed ions, massless electrons and uniform density and resistivity is employed. Thermal effects are neglected in Ohm's law and helical coordinates are introduced to reduce the number of coupled nonlinear equations that must be advanced in time. The results show that it is possible to produce RFP like magnetic field profiles with pinch parameters close to the experimental values. The efficiencies obtained for moderate resistivity, and the observed scaling, indicate that this could be a very attractive method for high temperature plasmas. The effect of finite electron mass on the formation and sustainment of a FRC by rotating magnetic fields (RMF) is studied. The importance of inertial effects is measured by the ratio between the RMF frequency and the electron-ion collision frequency. When this ratio is very small previous results corresponding to massless electrons are recovered. When this ratio increases there are significant changes in the value of the minimum external rotating field needed to sustain the FRC and the time needed to reach a steady state. Since the collision frequency decreases with increasing temperature and decreasing density these effects are expected to become more important as fusion relevant temperatures are approached. The injection of high energy (1 MeV) neutral beams in a D-He^3 FRC reactor ($L = 17$ m, $B = 6.7$ T, $T = 87.5$ keV) is studied with a Monte Carlo code already used to study NBI in moderate size FRCs and Spheromaks. A 3D, finite volume, resistive MHD code is employed to study the formation and sustainment of a flux core spheromak by helicity injection through magnetized electrodes. The energy content of the different modes, the decay rates of the ideal invariants and their cascades are studied.

TH/P3-18 · Two Fluid Dynamo and Edge-Resonant m=0 Tearing Instability in Reversed Field Pinch

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Abstract: Current-driven tearing instabilities are believed to dominate magnetic relaxation in self-organized high temperature plasmas such as the reversed field pinch (RFP) and spheromak. In the Madison Symmetric Torus (MST) RFP experiments, tearing instabilities are observed in the form of magnetic field, flow velocity and current density fluctuations that follow a temporally cyclic sawtooth behavior. During a sawtooth crash, a surge occurs in the dynamo – a fluctuation-induced mean electromotive force in the generalized Ohm's law that combines the MHD $\mathbf{v} \times \mathbf{B}$ and $\mathbf{j} \times \mathbf{B}$ Hall dynamos. The dynamo modifies parallel electric field and plasma current profiles. This ultimately leads to current flattening in the core and current sustainment in the plasma edge. We report new analytic and numerical results on the physics of two-fluid dynamos as well as on the problem of spontaneous (linear) instability of edge-resonant $\mathrm{m} = 0$ tearing modes. The $\mathrm{m} = 0$ mode is of a special importance for RFPs because of its impact on mode coupling, ion heating, momentum and energy transport. The key findings are: (1) two fluid effects are critically important for dynamo through their influence on the phase between the fluctuations; two-fluid theory yields a non-zero flux surface averaged Hall dynamo, absent in resistive MHD; (2) the two fluid version of the NIMROD code confirms analytic results during the linear stage of the instability but exhibits significant broadening of the Hall dynamo profile on the longer time scales of nonlinear evolution; (3) improved modeling of force-free RFP equilibrium predicts a wide range of RFP parameters in which $\mathrm{m} = 0$ tearing mode is spontaneously unstable, a result that is consistent with recent MST experimental observations.

TH/P3-19 · Modeling and Interpretation of MHD Active Control Experiments in RFX-mod

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Abstract: The RFX experiment is a Reverse Field Pinch (RFP) that has been modified (RFX-mod) by adding a set of 4(poloidally) × 48(toroidally) feedback controlled saddle coils, which are able to accurately control the radial field at the conductive shell, providing a clean plasma magnetic boundary. This paper will present the results of a series of ongoing theoretical studies aiming at interpreting the experimental RFX-mod results. We successively consider Resistive Wall Modes (RWMs) active control and the interaction of an external applied field with the internal dynamo modes. The RWMs are slow MHD instabilities expected to play an important role in setting tokamaks beta limits and whose comprehension is therefore of key interest not only for RFPs. A linear response model is compared with the experimental data to interpret the observed stabilization. Other studies regard the so called Virtual Shell (VS) operation, where a zero normal magnetic field contour is created by using the large available number of active coils and sensors. This operation has been seen to affect both the non resonant modes (RWMs) and also the internal resonant tearing modes. A Newcomb solver constrained by the external measurements is employed to calculate the magnetic field components in the plasma region and to reconstruct particle and field line trajectories for transport studies. Finally a set of experiments have been carried out where a single helicity either resonant or non resonant is applied to the plasma by the external circuits. In the case of non resonant fields the phenomenon of Resonant Field Amplification (RFA) of marginally stable RWM has been studied by employing the linear response model, while for resonant fields a more complex interaction with the internal tearing modes is observed. This phenomenon is very interesting especially in the attempt of actively inducing the so called Single-Helical state and it is studied theoretically by employing 3D cylindrical nonlinear codes.

TH/P3-20 · Momentum Transport and Ion Heating from Reconnection in the Reversed Field Pinch

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Abstract: Tearing instabilities and reconnection events in fusion plasmas can have enormous impact on the macroscopic behavior of the plasma. This is true in tokamaks, and is especially true in reversed field pinch plasmas, in which multiple tearing instabilities can be present. Two particularly large effects are the sudden transport of toroidal angular momentum and ion heating that occurs during a sawtooth crash in RFP experiments. During the crash, the radial profile of the toroidal rotation flattens and the ion temperature doubles, both in a time period of about 100 microseconds. Neither effect can be explained by classical processes. We report here analytic and computational results on mechanisms that can cause

these effects through the action of tearing instabilities. The key findings are: (1) a single tearing mode can transport momentum through the Maxwell stress (the mean Lorentz force arising from tearing fluctuations), and the effect is greatly enhanced from the nonlinear interactions that accompany multiple tearing modes; (2) viscous damping of tearing modes can contribute significantly to ion heating, even in a plasma with low collisionality, given the strongly sheared flows that accompany tearing.

TH/P6-1 · Phase Space Gradient Driven Discrete Compressional Alfvén Eigenmodes in Tokamaks: Simulations and Observations

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Abstract: The spectrum of Compressional Alfvén Eigenmodes (CAE) driven by phase space gradient measured in NSTX and DIII-D plasmas is analyzed numerically for the first time. Advanced diagnostic capabilities made it possible to measure single mode polarization and toroidal mode numbers, which unambiguously identifies studied modes to be of compressional branch. CAE modes form the discrete spectrum with each mode having three (quantum) mode numbers (M, S, n), where M, S, and n are poloidal, radial and toroidal mode numbers, respectively. CAE mode frequency splitting corresponding to change of each of these mode numbers seem to be observed in experiments and is consistent with our numerical analysis. CAE mode structure is computed to be localized in both radial and poloidal directions and is shown to be consistent with the internal reflectometer diagnostic data.

TH/P6-2 · Destabilization of magnetosonic-whistler waves by a relativistic runaway beam

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Abstract: Magnetosonic-whistler waves may be destabilized by runaway electrons with strongly anisotropic velocity distribution. The unstable wave frequency is well below the non-relativistic electron cyclotron frequency but above the ion cyclotron frequency. The linear instability growth rate of the magnetosonic-whistler wave destabilized by an avalanche of relativistic runaway electrons through the anomalous Doppler-resonance is calculated in a local analysis using the homogenous plasma approximation. The perturbative stability analysis is complemented by numerical solution of the dispersion equation including the full hot plasma dielectric tensor. In the parameter range relevant to disruptions in large tokamaks, the growth rate is largest for nearly perpendicular propagation. By assuming that the dominant damping mechanism in the cold post-disruption plasmas is due to collisions, the local threshold of the instability can be shown to depend on the fraction of runaway electrons, the magnetic field and the temperature of the background plasma. The dependence on the magnetic field is consistent with the experimental observations suggesting that there is a critical toroidal magnetic field below which there is no runaway current after the disruption. One reason for the absence of the runaways may be that the instability scatters the runaways in pitch-angle and prevents the beam from forming. Indeed, the quasilinear analysis shows that the main result of the instability is pitch-angle scattering of the runaway electrons on a typical time scale of a microsecond.

TH/P6-3 · Synergetic effect of TF ripples and MHD modes on fast ion transport

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Abstract: The present paper investigates the combined effect of toroidal field (TF) ripples as well as MHD induced low frequency perturbations on fast ion confinement in tokamak plasmas. The transport coefficients of fusion alphas in presence of TF ripples and TAE modes are calculated using the simplectic method for the integration of Hamiltonian systems. It is shown that MHD induced modes can result in both, either a degradation or an improvement of fast ion confinement in a rippled tokamak magnetic field, which is in qualitative agreement with observations of charged fusion products confinement in TFTR.

TH/P6-4 · Particle Simulation Analysis of Energetic-Particle and Alfvén-Mode Dynamics in JT-60U Discharges

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Abstract: Two different types of bursting modes have been observed by MHD spectrography during negative neutral beam (NNB) auxiliary heating in the JT-60U tokamak. One of these modes has been dubbed fast frequency sweeping (fast FS) mode. It is characterized by a timescale of the order of few milliseconds and frequencies branching upwards and downwards. The other mode, called the abrupt large-amplitude event (ALE), has shorter timescale (order of hundred microseconds) and larger amplitude. On the occurrence of ALEs, a significant reduction of the neutron emission rate in the central plasma region is observed. Such a change can be attributed to a redistribution of the energetic ions, with a marked reduction of their on-axis density. In this paper we present the results of particle simulations of a typical NNB-heated JT-60U discharge, performed by the HMGC code. We observe that the linear-growth phase is dominated by a fast-growing mode, localized at half radius, where the maximum of the energetic-particle pressure gradient occurs, with a significant coupling with the Alfvén continuum, showing the energetic-particle-driven nature of the mode. The saturated phase presents a complex phenomenology. In the early stage, the configuration is dominated by a TAE-like mode, localized in the external part of the discharge. The original mode is replaced by a couple of nearly degenerate modes. A weak central mode also appears, with frequency well localized in the continuum gap. At later times, the central mode becomes the dominant one, while the external one still persists at lower amplitudes. As the nonlinear effects become important, a macroscopic outward displacement of the energetic particles is observed, producing a significant reduction of their density in the central region. Numerical results seem to reproduce quite well the relaxed fast ion profile and the time scale observed experimentally. As the energy content of the energetic ions is reduced in the simulation, a weaker radial displacement and a longer time scale of the phenomenon are observed. Also the splitting of the two nearly degenerate modes becomes more apparent, supporting the idea of identifying the strongly driven modes as the responsible of ALEs and the weaker ones as the FS modes.

TH/P6-5 · Fishbone Instability Excited by Circulating Electrons

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Abstract: Fishbone instability excited by the supra-thermal circulating electrons in tokamaks is investigated. It is found for first time that the procession of all the circulating electrons is in ion diamagnetic direction if magnetic share is neglected. The circulating electrons play bigger role on the modes than the barely trapped electrons. The analyses show that the mode frequency is close to procession frequency of the circulating electrons comparable with experiment observations. The correlation of the theory with experiments is discussed.

TH/P6-6 · Redistribution of Energetic Ions During Reconnection Events in NSTX

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Abstract: Experiments on the spherical torus NSTX show that reconnection events (sawtooth crashes, internal reconnection events et.) can result in strong, by a factor of two, drops of the neutron yield, which are an evidence of losses of energetic ions. In the present work, mechanisms of the energetic ion redistributions during sawtooth crashes are analyzed, and strong drops of the neutral emission in a particular NSTX experiment are explained. It is found that in NSTX shots with a low magnetic field the main mechanisms of the particle redistribution during the crash are the particle resonance with the 1/1 harmonic of the perturbation and the stochastization of the particle motion. The latter is considerably enhanced by the diamagnetic perturbation of the magnetic field strength. The conclusion is drawn that the loss is sensitive to the safety factor profile, the plasma pressure and the particle precession rate, which, in turn, depends on the particle energy and pitch angle, the plasma elongation etc. The NSTX shot #104505, where drops of the neutron yield by a factor of 2 were observed during reconnection events, was selected for detailed analysis. A semi-analytical model of the electromagnetic field evolution during a Kadomtsev-type reconnection, which takes into account the perturbation of the magnetic field strength, is suggested and incorporated into the code GYROXY, which calculates the particle motion without using the guiding-center approximation. It is found that the motion of a considerable fraction of 80-keV ions is stochastic during the reconnection, and this stochasticity can lead to a particle loss. The observed stochasticity results from the overlap of resonance islands, in particular, secondary islands near the separatrix of the 1/1 resonance and islands produced through a nonlinear mechanism. To calculate the post-crash distribution function of energetic ions and the loss of these ions, Monte-Carlo simulations of the particle motion in the evolving magnetic field are carried out. Using this distribution, the change of the neutron reactivity is calculated and compared with the experiment.

TH/P6-7 · Evolution of fusion-born alpha particles in JET tritium NBI discharges: measurements and modelling

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Abstract: The paper is devoted to the investigation of the evolution of gamma-ray emission induced by fusion alphas in tritium NBI experiments on JET. It demonstrates the possibility of evaluating the rates of alpha loss and slowing down from measurements of the delay of gamma emission against the fusion alpha production. Comparison of measurements with modelling indicates the existence of losses additional to first orbit loss of alphas, especially in low I/low B plasmas and current hole plasmas. The requirements for adequate gamma-ray diagnostics and best plasma and tritium NBI characteristics are identified in order to obtain utmost information on the confinement of fusion alpha particles.

TH/P6-8 · Analysis of Ion Cyclotron Heating Issues for ITER

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Abstract: ICRH is a major component of planned heating systems for ITER. Possible species mixes include H or D plasmas with a He^3 minority for initial operation, D-T with both thermal and fast alpha populations for Q $\sim$ 10 inductive operation at relatively high density, and D-T with both thermal and fast alpha populations for non-inductive, Q $\sim$ 5 operation at lower densities. Depending on the specific case, RF-plasma interactions can include mode conversion, parasitic absorption on fast ion components, and energetic-ion tail formation. We have begun analyzing the various options using the AORSA and TORIC full-wave simulation codes. In addition, we have computed quasilinear diffusion operators from the AORSA RF fields that are coupled to the CQL3D Fokker-Planck code. When the RF-driven energetic tails are of sufficient density and/or energy to affect RF propagation and absorption, these distribution functions can be passed back to AORSA, allowing iteration to convergence. Our principal results are consistent with trends found previously with less exact models, include: (1) At the higher range of densities ($>$4–6 $\times 10^{19}$ m^{-3}) planned for Q $\sim$ 10 inductive operation, RF-tails from either minority heating or RF absorption by fusion alphas are not significant. Lower densities and/or increased power will increase the possibilities of such tail formation. (2) For nominal Q $\sim$ 10 operation, absorption of the power by fusion alphas is in the 1–5% range of the total coupled RF power. We find the absorption by a non-Maxwellian slowing-down fusion-alpha distribution to be the same as a 1.2 MeV Maxwellian. (3) There is a Be resonance at the inner edge of the plasma (depending on details of magnetic field and wave frequency) that has the potential of unwanted edge absorption. (4) Mode conversion can be significant for densities in the 2–3 $\times 10^{19}$ m^{-3} range at high minority fractions.
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TH/P6-9 · Simulation of burning plasma dynamics by ICRH accelerated minority ions

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Abstract: The α-particle dynamics can be simulated in pure deuterium plasmas by ions accelerated by radiofrequency waves. The use of ICRH in the minority scheme (H or He^3) can indeed produce fast particles that, with an appropriate choice of the minority concentration, of the RF power and of the plasma density and temperature, can reproduce the dimensionless parameters, the fast particles radius and the fast particle beta characterizing the α-particles in ITER. Thus, a device operating with deuterium plasmas in a dimensionless parameter range as close as possible to that of ITER and equipped with ICRH as a main heating scheme would be able to reproduce the most important features of α-particles heated plasmas and therefore would be capable of assessing the relevant scenarios before their implementation on

ITER itself. The aim of the present paper is to determine the characteristic fast-ion parameters, by solving the coupled problems of ICRH propagation and quasi-linear absorption. The 2D full-wave code TORIC is used coupled to the SSQLFP code, which solves the quasi-linear Fokker-Planck equation in 2D velocity space. Using as reference parameters those considered for the FT3 conceptual study, the power deposition profiles on the ion minority, majority and electrons are first determined; then, the effective temperature of the minority ion tail and the fraction of fast ions is evaluated. Moreover, the quasi-linear analysis let know how much of the power absorbed by the minority will be redistributed by collisions on the main species of the plasma: electrons and majority ions.

TH/P6-10 · Integrated Full Wave Analysis of RF Heating and Current Drive in Toroidal Plasmas

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Abstract: Self-consistent and sufficiently accurate modeling of wave-plasma interactions is one of the key issues in producing and sustaining burning plasmas. We have carried out self-consistent full wave analysis taking account of the modification of momentum distribution functions by using the integrated modeling code TASK. The formation of energetic ions and broadening of absorption profile in the ICRF heating are discussed for ITER plasmas. The full wave analysis by TASK code was also applied to the propagation and absorption of the electron cyclotron waves in a small-size spherical tokamak. Integral formulation of the self-consistent full wave analysis was extended to include the finite gyroradius effects indispensable for describing the behavior of electron Bernstein waves and absorption of ICRF waves by energetic ions. The formulation of integral form of the quasilinear operator is also discussed.

TH/P6-11 · Electron Bernstein Wave Studies: Current Drive; Emission and Absorption with Nonthermal Distributions; Delta-f Particle-in-Cell Simulations

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Abstract: Electron Bernstein waves (EBW) are suitable for heating, current drive, and radiation temperature measurements of overdense plasma (plasma > cyclotron frequency) in spherical torus devices. In NSTX, design of a multi-megawatt EBW current drive system is supported by experimental measurements and computations of obliquely viewing, dual-polarization EBW emission (EBE) radiometry. Efficient EBW coupling, $80 \pm 20\%$ at 16.5 GHz, is demonstrated, in agreement with calculations. Ohkawa EBWCD is calculated to generate 40–50 kA/MW of off-axis current. Calculations for the Pegasus experiment are also presented. Studies of an ARIES-ST discharge show under a broad range of conditions that the EBW wave energy penetrates up to 30% of the distance into the plasma center. EBW damping and emission are calculated based on nonthermal distributions from a 3D bounce-averaged Fokker-Planck code. BXO emission from rf quasilinear modified nonthermal distributions due to 1 MW of injected EBW power gives intermediate temperature between the thermal and tail electron nonthermal temperature. First results of slab-model simulation of EBW by a delta-f particle-in-cell code are given. Delta-f enables low-noise simulations of the linear mode-conversion of injected X-mode radiation to EBWs in the edge region of an overdense plasma, which are in good agreement with analytic analysis. At higher input power, second through fourth harmonics of the fundamental EBW are nonlinearly generated, as well as decay into two EBWs at lower frequencies.

TH/P6-12 · Dynamics of Electron Cyclotron Current Drive

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Abstract: Modeling of electron cyclotron current drive (ECCD) is usually carried out using complex Fokker-Planck codes although they are not convenient for examining the time evolution of different operational scenarios. For this application simpler dynamical models are needed, based either on the moment's method or on the Green's-function formulation. The Green's method is limited to the linear regime, breaking down for high power densities when a high energy beam is created. Alternatively, the moments or fluid model depends on the adoption of a proper distribution function for the energetic electrons, avoiding the solution of a two-dimensional momentum space problem. It leads to results valid in the strong radio frequency (RF) regime. This paper presents preliminary results of a fluid model for the beam of current-carrying electrons. To keep the model simple, trapped particle and radial transport effects are neglected. These phenomena can considerably affect the efficiency of ECCD, but their elimination allows

studying the dynamical aspects of the problem by means of a zero-dimensional model. It is expected to include these effects in a future one-dimensional formulation. Beam electrons are fully-relativistic and both the effects of a time varying inductive electric field and quasi-linear diffusion by RF waves are included. The model is firstly developed for a general diffusion coefficient and then specialized for the case of EC waves. In particular, the small gyroradius limit of the diffusion coefficient is used. Assuming a distribution function for the energetic electrons, one takes moments of the Fokker-Planck equation for the rates of change of density, momentum and energy. The simplest approximation in this case is to assume a nearly mono-energetic beam, but a fully relativistic bi-Maxwellian representation for the energetic electrons is also being implemented. Then, the equations of motion can be easily solved looking for different equilibrium solutions or forms of time-dependent operation.

TH/P6-13 · Test of the Quasilinear ICRH Operator with a 6D Particle Simulation in a Toroidal Plasma and Development of a New 5-1/2D ICRH Particle Simulation Technique

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Abstract: Plasma heating by ion cyclotron range of frequency (ICRF) waves remains to be one of the most important sources of auxiliary heating in the ITER plasma. Its use may not be limited to the heating of the plasma to reach the ignition condition, but also be extended to the advanced plasma operation control. There are world-wide efforts in the ITER partnership countries to build large-scale simulation codes to understand, predict and control the plasma behavior under a high power ICRF heating. However, there has been growing concern that the widely-used quasilinear operator in describing the wave-particle interaction physics may not be well-justified for application to a toroidal confinement device. A six dimensional, numerical Lorentz-force simulation reveals that the quasilinear operator breaks down in many physical situations and can yield incorrect estimate of the heating rate. In this work, a detailed test on the validity of the quasilinear theory will be presented using the six dimensional numerical simulation under the assumed rf field profiles of practical interest. Development of a new 5-1/2 dimensional particle simulation technique will also be reported which can dramatically improve the numerical simulation accuracy of ICRH heated ion dynamics in magnetic fusion plasmas, which may also be used to extend in the future to improve the accuracy of the wave propagation simulations in a self-consistent manner.

TH/P6-14 · Integrated particle simulation of neoclassical and turbulence physics in the tokamak pedestal/edge region using XGC

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Abstract: The edge neoclassical ion particle code XGC-0 [C.S. Chang, S. Ku, et al., IAEA FEC conference, 2004; Phys. Plasmas 11, 2649 (2004)], which includes all the baseline neoclassical kinetic physics important to the edge plasmas, has been greatly extended to XGC-1 to simulate the integrated neoclassical-turbulence physics self-consistently. The simulation region is from the top of the pedestal to the scrape-off and divertor regions bounded by a material wall. The new version XGC-1 includes the full-f electron kinetics, in addition to the full-f ion and neutral dynamics inherited from XGC-1, to evaluate the baseline kinetic neoclassical phenomena. The large variation of electrostatic potential and plasma in the open field lines is also resolved by the full-f ion/electron dynamics. At the same time, XGC-1 currently uses the adiabatic electrons to analyze the electrostatic turbulence physics self-consistently with the baseline neoclassical physics. The mixed-f electron approach used in XGC-1 is new, successfully integrating the neoclassical and turbulence physics. XGC-1 reveals that the ambient electrostatic potential is negative in the H-mode pedestal layer and positive in the scrape-off layer. The plasma flow is counter-current in the pedestal and around the separatrix, with a co-current component in the scrape-off layer near the wall. Detailed analysis of the neoclassical and turbulence physics in the pedestal/scrape-off region will be presented, together with experimental validations. XGC-1 also includes the magnetic ripple and neutral beam heating effects. The code will be further developed in the future to include the more advanced electron turbulence responses and the electromagnetic turbulence physics.

TH/P6-15 · Critical Issues Identified by the ASDEX Upgrade Edge and Divertor Modelling

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Abstract: A detailed comparison between the ASDEX Upgrade (AUG) experimental data and results of the SOLPS 2D edge code simulations has recently been performed. High quality upstream profiles of electron density and ion and electron temperatures in the scrape-off layer (SOL) of AUG have been collected for two shots with different upstream collisionalities: a low density ELMy H-mode shot (low collisionality) and a medium density Ohmic shot (higher collisionality). A generally broad agreement, within a factor 2, considering basic parameters characterising the divertor, has been reached between simulations and experiment. In both Ohmic and H-mode shots, however, the tendency of SOLPS solutions to underestimate the divertor electron temperature and overestimate its density has been reliably established. Two main possible causes of the discrepancies have been considered: some deficiencies in the neutral modelling (e.g. missing atomic and molecular reactions in EIRENE, the Monte-Carlo neutral part of SOLPS), and the presence of a significant population of supra-thermal ions and electrons in the SOL and divertor plasma. The results of dedicated SOLPS runs where the sensitivity of the code solution to various assumptions of the neutral model and parallel heat transport of ions and electrons are described. A comparison between simulated and experimentally measured Mach numbers of the parallel ion flow in the SOL is presented, and conditions necessary for obtaining fast flows in the code are analysed.

TH/P6-16 · Emerging Chaos in Rotation Velocity Profile in Collisional Tokamak Edge Layer

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Abstract: Since Stringer had shown that any radial location in tokamak plasma is prone to a spontaneous poloidal spin-up with a concomitant rise of a radial electric field, plasma rotation and radial electric field gradually emerged as important factors in stability considerations for toroidal equilibrium and transport. As novel studies indicate, also tokamak turbulence instability due to the geodesic acoustic mode (GAM) or zonal flow have similar mechanisms involving perturbations of the plasma rotation velocities and the radial electric field. On the other hand, as plasma collisionality suppresses such instabilities, it is less likely to encounter rotational turbulence, for example, in high collisional plasmas with steep gradients. In the present extension of the neoclassical theory of rotation and electric field in high collisionality tokamak plasmas with steep gradients, however, it will be shown that in some radial interval the poloidal (and to some extent also toroidal) rotation velocity displays chaotic behaviour.

TH/P6-17 · Two-dimensionally steep structure of the electric field in tokamak H-mode

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Abstract: Researches on tokamak H-mode physics have been greatly developed, but there remain some unresolved problems. One of the problems is the rapid formation mechanism of a density pedestal on an L/H transition. Formation of a poloidal shock structure is predicted in the H-mode transport barrier, and existence of the poloidal electric field induces a convective particle flux in the radial direction. Therefore, consideration of the two-dimensional (2-D) structure both in the radial and poloidal direction is inevitable to clarify the formation mechanism of the H-mode pedestal. The analysis is carried out with edge plasmas in H-modes, which are induced either spontaneously or by electrode biasing in the plasma edge region. 2-D structures of the potential, density and flow velocity are calculated with the momentum conservation law and the Boltzmann relation. A set of equations is derived by considering the nonlinearity in bulk-ion viscosity and using a model of turbulence-driven shear viscosity. The validity of the 1-D L/H transition theory and the iterative process to obtain the 2-D structure is confirmed by our analysis. A steep electric field structure both in the radial and poloidal direction is obtained by using the plasma parameters in the electrode-biasing H-mode. A steep gradient in the poloidal direction (poloidal shock) induces a convective particle flux in the radial direction. The flux-surface averaged convective velocity is calculated to give the particle pinch velocity more than 1 m/s in the barrier region, which increases rapidly in 100 microseconds at the onset of the L/H transition. This convection is large enough to affect particle transport. The analysis is also performed in the case of the spontaneous H-mode driven by ion-orbit losses. In the 2-D analysis of the L/H transition, two basic mechanisms must be highlighted, i.e., (i) reduction of anomalous transport by the steep gradient of the radial electric field, and (ii) the onset of an inward pinch associated with the poloidal shock structure. Both effects from the radial and poloidal electric field influence the

rapid formation of the steep gradients in H-mode plasmas. A transport model including both effects is constructed to reveal the self-sustained mechanism of the density pedestal formation in the L/H transition.

TH/P6-18 · Modeling of Dust-Particle Dynamics, Transport, and Impact on Tokamak Plasma Performance

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Abstract: Dust has been identified as having a potentially large impact on ITER-scale plasma experiments. Understanding and predicting the role of dust require models of dust production mechanisms, dust interactions with plasma and surfaces, and dust transport within and outside the plasma. Here, recent theoretical studies and experimental observations of dust in fusion plasmas are reviewed and a physical model used to describe dust transport in fusion devices is described. The model includes the dynamics of dust due to dust-plasma, dust-turbulence, and dust-surface interactions. These processes have been incorporated into the DUST Transport (DUSTT) code that includes the following capabilities: a 2D curvilinear non-uniform mesh based on MHD equilibrium; plasma and neutral-gas parameters calculated by the edge plasma/neutrals transport code UEDGE; and tracking of test dust particles in 3D using the resulting force, particle and energy fluxes, and other parameters based on UEDGE data. Results of dust particle dynamics and transport simulation are shown for current tokamaks (NSTX, DIII-D, C-Mod) as well as for the ITER. These simulations demonstrate that dust particles are very mobile and accelerate to large velocities due to the ion drag force (cruise speed >100 m/s). Deep penetration of dust particles toward the plasma core is predicted. It is shown that DUSTT is capable of reproducing many features of recent dust-related experiments. The simulation of dust particle penetration into the ITER burning plasma shows that dust particles launched either from the main chamber walls or from the dome can penetrate all the way to the separatrix. Penetration of dust toward the core plasma can represent a significantly enhanced impurity source there. As follows from our UEDGE simulations, a feedback can occur where radiation from ionized dust-impurities can reduce the divertor temperature, which in turn allows further penetration of the dust and associated impurities. Such a feedback can result in strongly detached plasma conditions.

TH/P6-19 · Improvement of plasma confinement due to ion and electron heating at the edge of tokamak.

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Abstract: Improvement of plasma confinement due to ion and electron heating at the edge of tokamak. The behavior of turbulent fluxes in the vicinity of a resonace point $\mathrm{m/n} = \mathrm{q}(\mathrm{x}_{\mathrm{res}})$ in the plane wall plasma layer in tokamak is studied by numericaly analyzing the nonlinear MHD equations in four-field electrostatical model. It is shown that the injected auxiliary power into as ion so electron components produces the reduction of the turbulent fluxes, which looks like a L-H transition. Such behavior of the flux is found to be due to the stabilizing effect of the $\mathrm{E} \times \mathrm{B}$ drift velocity, which increases if ion or electron temperature increases.

TH/P6-20 · Nonlinear Simulation of Edge-Localized Mode in Spherical Tokamak

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Abstract: We propose a modeling for the dynamics of an Edge-Localized Mode(ELM) crash in the spherical tokamak with a consecutive scenario which is initiated by the spontaneous growth of the ballooning mode instability by means of a three-dimensional nonlinear magnetohydrodynamic simulation. The simulation result shows a two-step relaxation process which is induced by the intermediate-n ballooning instability followed by the $m/n = 1/1$ sawtooth crash, where m and n represent the poloidal and toroidal mode numbers, respectively. By comparing with the experimental observations, we have found hat the simulation result can reproduce several characteristic features of the so-called type-I ELM in an appropriate time scale: (1) relation to the ballooning instability, (2) intermediate-n precursors, (3) low-n structure on the crash, (4) formation and separation of the filament, and (5) considerable amount of convective loss, simultaneously. Furthermore, more realistic situations are examined by using the drift model. The ion diamagnetic drift terms are found to stabilize the low-n components linearly, and to have less considerable effect on the nonlinear dynamics.

TH/P6-21 · Blob Transport Models, Experiments, and the Accretion Theory of Spontaneous Rotation

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Abstract: Radial convective transport competes with classical parallel transport to determine the radial penetration of plasma into the tokamak scrape-off-layer (SOL), influencing recycling and wall damage. This motivates theoretical models and experimental verification of blob velocity scalings discussed in this paper. In a separate study, recent developments in the Accretion Theory of spontaneous tokamak rotation, and its relation to blob physics are also considered. Blobs are intermittent filamentary convecting plasma structures. The theory of curvature-driven blob transport regimes in a tokamak is developed and employed to make testable predictions for the magnitude and scaling of the blob radial velocity. The theory encompasses the sheath-connected, resistive X-point and resistive-ballooning-disconnected regimes, and is therefore applicable to both X-point and limiter configurations. Gas puff imaging (GPI), a fast-time-scale high-spatial-resolution imaging technique, presents a new opportunity, investigated here, for the comparison of blob theory with 2D data. The atomic and neutral physics aspects of interpreting GPI images are studied. It is shown that the single state collisional radiative (CR) model is able to resolve time variations slower than 1 microsecond and is, thus, suitable for present GPI experiments. Using GPI data from the NSTX tokamak, the blob velocity is analyzed and interpreted in terms of blob models. We find that the observed velocity is bounded by theoretically predicted minimum and maximum velocities corresponding to the sheath-connected and resistive-ballooning-disconnected limits respectively. It is also shown that the blob birth zone is near the local maximum of the edge logarithmic pressure gradient, suggesting blob generation by underlying edge instabilities. Finally, the extensive series of recent experimental observations that are consistent with the predictions of and the interpretation by the Accretion Theory of the spontaneous rotation phenomenon is analyzed. Relevant developments of this theory are presented. The process of angular momentum ejection from the plasma column to the surrounding wall is attributed to the formation of blobs which arise from modes excited at the edge of the plasma column.

TH/P6-22 · Turbulence Modeling of JET SOL Plasma

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Abstract: Understanding and modeling radial plasma transport originating at the transition from closed to open field lines is of vital importance for the development of fusion machines. This strongly intermittent turbulent transport fuels the scrape off layer (SOL) and dictates the density and temperature profiles across the SOL. It is responsible for plasma contact with the vessel wall, which is detrimental for the operation of fusion reactors. The two dimensional, electrostatic edge-SOL turbulence code ESEL [O.E.Garcia et al., Phys. Plasmas 12 (2005) 062309; O.E. Garcia et al., Plasma Phys. Control Fusion 48 (2006) L1] simulates the perpendicular dynamics of transport events in the SOL together with a self consistent development of the SOL profiles at the outboard midplane. ESEL simulations were performed for parameters corresponding to JET ohmic discharges. The simulations produce profiles of density and temperature similar to the ones inferred from experiment. The turbulent perpendicular energy fluxes are split up into convective and conductive parts, with the former exceeding the latter by two orders of magnitude. The radial turbulent particle flux profile is broader when compared to the turbulent heat flux profile, reflecting the faster parallel loss of electron energy. The effective particle diffusivity at the outer midplane, including the effects of both collisional and turbulent perpendicular transport, increases from the separatrix to a peak value at about 20 mm into the SOL, in both qualitative and quantitative agreement with experimental observation [K. Erents et al., Plasma Phys. Control Fusion 46 (2004) 1757; K. Erents et al., Nuclear Fusion 38 (1998) 1637]. The ESEL simulations for JET further predict parallel Mach numbers $M \sim 0.2$ in agreement with Mach probe measurements. By coupling a simulation domain situated at the low B-field side to one on the high B-field side, utilizing the parallel losses in the SOL, the dynamics on the high-field side can be inferred. Most importantly the direction of the turbulent perpendicular transport is determined to be radially inward on the high field side. On the basis of the reasonable agreement between simulation and experiment we conclude that the SOL radial transport in JET ohmic plasmas is dominated by electrostatic interchange turbulence.

TH/P6-23 · Edge Gyrokinetic Theory and Continuum Simulations

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Abstract: Results are presented from the development and application of TEMPEST, a fully nonlinear five dimensional (3d2v) gyrokinetic continuum code. The simulation results and theoretical analysis include studies of the H-mode edge plasmas neoclassical transport and turbulence in a real divertor geometry and its relationship to the plasma flow generation with zero external momentum input, including the important orbit-squeezing effect due to the large electric field flow-shear in the edge. Key results include: (1) In a large-aspect-ratio circular geometry, excellent agreement is found for a neoclassical equilibrium with parallel flows in the banana regime with zero temperature gradient. (2) The four dimensional (2d2v) version of the code produces first self-consistent simulation results of collisionless damping of geodesic acoustic modes and zonal flow (Rosenbluth-Hinton residual) with Boltzmann electrons using a full-F code. In divertor geometry, it is found that the endloss of particles and energy induces pedestal-like density and temperature profiles inside the magnetic separatrix and parallel flow stronger than the neoclassical predictions in the SOL. (3) As a test of the interaction of collisions and parallel streaming, TEMPEST has been compared with published analytic and numerical results for endloss of particles confined by combined electrostatic and magnetic wells. Good agreement is found over a wide range of collisionality, confining potential, and mirror ratio; and the required velocity space resolution is modest. (4) Our 5D gyrokinetic formulation yields a set of nonlinear electrostatic gyrokinetic equations that are for both neoclassical and turbulence simulations. (5) A set of generalized gyrokinetic Vlasov-Maxwell equations in the gyrocenter coordinate system valid for the edge plasmas has been derived by applying the Lie transform perturbation method to the Poincare-Cartan-Einstein 1-form and the pullback transformation for the distribution function. This formalism allows large-amplitude, time-dependent background electromagnetic fields to be developed fully nonlinearly in addition to small-amplitude, short-wavelength electromagnetic perturbations.

TH/P6-24 · Dynamics of edge oscillations and core relaxations in harness.

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Abstract: Resistive kink oscillations in tokamak plasmas are usually treated as core localized events, yet there there are several mechanisms by which they may harness the edge dynamics. In this work I investigate core-edge oscillatory entrainment through direct propagation of heat pulses, inductive coupling, and global higher order resonance effects.

TH/P6-25 · Fluid Simulations and Theory of Boundary Plasma Fluctuations

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Abstract: The fluid turbulence code BOUT is used for simulation of microturbulence in the edge plasma of Alcator C-Mod; the simulation data shows radial and poloidal correlation lengths at the outer midplane that are close to the C-Mod values, and autocorrelation times comparable to C-Mod values in the near SOL but not further out. BOUT is also used to study the linear and nonlinear evolution of ELMs (ballooning-peeling modes); analysis is done to quantify physical mechanisms of particle and energy transport, comparing radial advection of filaments and stochastic transport due to the magnetic perturbation. Self-consistent simulation of turbulence and transport in the tokamak edge is performed by coupling the 2D transport code UEDGE with BOUT. Two different regimes are observed, weakly intermittent turbulence when the flow shear is large, and strongly intermittent turbulence when the flow shear is small. We study, analytically and with BOUT, modes driven by sheath impedance, parallel shear of the $\mathrm{E}\times\mathrm{B}$ velocity, and curvature and confined to the divertor leg by x-point magnetic shear. At high-enough beta, evanescent modes localized near the divertor plate appear. The sheath-driven modes are strongly sensitive to the radial tilt of the divertor plate. Finally, we report on two new effects of x-point magnetic shear on "blobs" (large-amplitude intermittant structures in the SOL): the divertor-leg instabilities can grow into divertor-leg blobs; second, as main-SOL or divertor-leg blobs propagate radially, the x-point effects decrease, de-localizing the blobs and changing their propagation rates [R.H. Cohen, D.D. Ryutov, Contrib. Plasma Phys., to be published]. We examine experimental data for evidence of divertor-leg instabilities and the predicted changes in blob propagation rates.
* Work performed for U.S. DOE, by U.C. LLNL under Contract W7405-ENG-48.

TH/P7-1 · Analysis of Net Plasma Currents in Non-Axisymmetric Plasmas

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Abstract: Net plasma currents and those time evolutions are studied numerically for non-axisymmetric plasmas. Even in a tokamak plasma, there exists non-axisymmetry due to toroidal field (TF) ripples and it could reduce the bootstrap current. In this study, the effect of TF ripples on the bootstrap current in tokamaks is clarified. Though the net plasma current is not necessary for MHD equilibrium in helical plasmas, finite net toroidal current has been observed in actual experiments. We have developed a net plasma current simulation code for the integrated simulation of helical plasmas and analyzed net plasma currents observed in an LHD plasma.

TH/P7-2 · Shear flow generation in stellarators – configurational variations

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Abstract: Plasma momentum transport within magnetic surfaces plays a fundamental role in a number of toroidal plasma physics issues, such as: turbulence suppression, impurity transport, bootstrap current generation, and the shielding of resonant magnetic error field perturbations. Stellarators provide opportunities for improved understanding of plasma flow effects because (a) new forms of quasi-symmetry (e.g., helical, poloidal) can be produced that differ significantly from the tokamak; and (b) symmetry breaking effects (always present to some degree) remove the degeneracy between parallel and cross-field transport characteristic of symmetric systems. Furthermore, external control coils can be used to further enhance or suppress such effects. A method has been developed to evaluate the variation of neoclassical self-generated plasma flows in stellarators both within and across magnetic surfaces. This introduces a new dimension into both the optimization of stellarators and to the improved understanding of the existing confinement database. Application of this model to a range of configurations indicates that flow directionality and shearing rates are significantly influenced by the magnetic structure; flexibility variations within each configuration provide further control over flow characteristics. The stellarator confinement database contains evidence of machine-dependent effects that can be related to neoclassical transport physics. However, the measured cross-field transport rates are clearly anomalous. The possibility that neoclassical flow shearing effects are playing a role in these effects has become an important focus for applications of our model. For example, a recent analysis of a series of inwardly shifted LHD discharges has indicated that decreases of up to a factor of 10 in the neoclassical viscosity (allowing greater flow shearing) were correlated with the experimentally observed improved confinement times.
* Research sponsored by the U.S. Department of Energy under Contract DE-AC05-00OR22725 with UT-Battelle, LLC.

TH/P7-3 · Localized Breaking of Flux Surfaces and the Equilibrium Beta Limit in the W7AS Stellarator

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Abstract: We report on PIES three-dimensional equilibrium calculations for W7AS plasmas which exhibit degraded confinement at high beta with no indication that the confinement degradation is being caused by instabilities. The equilibrium calculations exhibit stochastic field lines in the outer region of the plasma, with the flux surfaces appearing to break only locally in the neighborhood of the outer midplane and to remain intact elsewhere. This conclusion follows from plots of field line trajectories, which show smooth, confined curves punctuated by rapid, erratic radial excursions appearing each time the trajectories cross the outer midplane. This result conforms with intuition and with conventional wisdom, which suggest that the flux surfaces should break near the outer midplane due to the strong compression of the three-dimensional flux surfaces there by the Shafranov shift. The results also conform with a WKB calculation (which is justified by the large mode numbers of the magnetic islands involved). This emerging picture, and the associated long connection lengths of the magnetic field lines, may explain why the impact of the predicted stochastic region on the pressure profile in the experiments may be modest. Although the pressure profile is modified in that region, a substantial pressure profile may be supported there.

TH/P8-1 · Effects of "Sharpness" of the Plasma Cross-Section on the Stability of Peeling-Ballooning Modes in Tokamaks

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Abstract: An ideal magnetohydrodynamic (MHD) mode destabilizing near the plasma surface is one of the causes of edge-localized modes (ELMs), which constrain the maximum pressure gradient in the pedestal at the tokamak edge region. The stability of ideal MHD modes, called a peeling, a ballooning and a peeling-ballooning modes, depends on the shape of the equilibrium, the ellipticity, the triangularity, the squareness and so on. In this paper, the effect of the new shaping parameter 'sharpness' on the stability of the peeling-ballooning mode is investigated numerically with the linear ideal magnetohydrodynamic (MHD) stability code MARG2D, where the sharpness is defined in terms of the curvature at the top and bottom of the outermost flux surface. Though the increase of the sharpness has little impact on the stability of current driven (peeling and kink) modes but the stable pressure gradient limit restricted by the stability of the peeling-ballooning mode significantly improves as the sharpness increases. The increase of the sharpness also makes broader the second stable region against an ideal ballooning mode on s-α diagram, where s is the magnetic shear and α is the normalized pressure gradient. This expansion of the second stability region is considered to be important to enhance the stable pressure gradient limit restricted by the peeling-ballooning mode, and the sharpness is an important parameter for high performance H-mode operations with high pedestal pressure.

TH/P8-2 · MHD Stability in X-point Geometry: Simulation of ELMs

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Abstract: Edge localised modes (ELMs) associated with the edge transport barrier in H-mode plasmas remain an important issue for ITER. It is generally accepted that the onset of an ELM is caused by MHD instabilities, notably ballooning modes driven by the edge pressure gradient and the external kink(peeling) modes driven by the bootstrap current. In order to study the non-linear evolution of the ballooning and kink modes in a full x-point/separatrix geometry, the code named JOREK is being developed. The current version solves the reduced MHD equations in toroidal geometry using either 3D finite elements or 2D finite elements with Fourier harmonics in the toroidal direction. The finite elements cover both the open and closed field lines and are aligned to the equilibrium flux surfaces. The JOREK code has been used to study the influence of the x-point on the linear stability of external kink(peeling) modes driven by an edge current gradient. The traditional peeling modes are found to be strongly stabilised by the presence of the x-point, both for ideal and resistive peeling modes. A resistive MHD instability is found to remain unstable in the presence of the x-point. This instability is much less sensitive to the specific value of q close to the boundary. Its mode structure is very similar to the conventional peeling mode except close to the x-point where the mode shows a phase inversion as a function of radius. The non-linear evolution of this so-called peeling-tearing mode shows a saturation of the mode amplitude and a local flattening of the density profile just inside the x-point. This could be consistent with the relatively long-lived low-n precursors ('Outer Modes') to the giant ELMs in JET hot-ion H-modes. The ELM crash is simulated by evolving a medium-n ballooning mode starting from an equilibrium which is linearly unstable to the ideal MHD ballooning mode. The non-linear simulations show the unstable ballooning mode to lead to a small amplitude ballooning perturbation of the flux surfaces. The temperature perturbation follows the magnetic perturbation due to the large parallel transport. The density, on the contrary, is strongly perturbed by the ballooning mode flow pattern. This leads to expulsion of high density 'blob-like' structure which are sheared of the main plasma by the poloidal flow.

TH/P8-3 · Extended Magnetohydrodynamic Simulations of Edge Localized Modes in Existing and Future Tokamak Devices

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Abstract: Studies of ELMs using the nonlinear, initial-value code NIMROD code are presented. Linear studies of ELMs show the the effects of diffusivities do not qualitatively change the linear mode spectrum as compared to linear ideal MHD studies. Two-fluid effects change the linear spectrum by stabilizing high-n modes. Nonlinear simulations presented in this work are the first to show significant plasma-wall interactions as a result of an ELM instability over a global computational domain. The nonlinear

evolution of the linear modes drives a rapid loss of internal energy with approximately 70 kJ (~10%) of the internal energy being lost within 60 ms. The computation finds that the primary loss channel is convective ($n\vec{V}T$) rather than conductive ($\vec{q}$), which is not inconsistent with laboratory measurements. The ability to reproduce these important laboratory measurements suggests that nonlinear fluid simulations have potential to provide significant insight into how ELMs evolve and deposit heat onto the wall. We also present how a combination of linear and nonlinear results can be used for integrated modeling of future reactors.

TH/P8-4 · Second Ballooning Stability Effect on H-mode Pedestal Scalings

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Abstract: Edge localized modes (ELMs) have a strong effect on the plasma at the edge of H-mode tokamak discharges. Robust models for triggering ELM crashes are needed for predictions of the pedestal height. In this study, the transition from the first ballooning mode stability to the second stability is included in the determination of scalings for the height of the temperature pedestal of type-I ELMy H-mode discharges. The ballooning threshold, derived from MHD stability calculations obtained using the HELENA and MISHKA codes, together with six theoretical-based models for the pedestal width, are implemented to derive scalings for the pedestal temperature height. These pedestal height scalings include the effects of plasma shaping, separatrix, bootstrap current, and the effect of access to the second ballooning stability. Each pedestal model is calibrated using experimental data obtained from the latest public version of the International Tokamak Physics Activity (ITPA) Pedestal Database (Version 3.2). In addition, an improved empirical model is developed for the height of the density pedestal. Statistical measures are used to show that the new models using the ballooning stability threshold that takes into account a possible access to the second stability region agree with experimental data about equally well and yield better agreement with experimental data than previous models. Results are presented for the pedestal height expected in ITER discharges based on the new scalings.

TH/P8-5 · Theoretical analysis and predictive modelling of ELMs mitigation by enhanced toroidal ripple and ergodic magnetic field

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Abstract: Ripple-induced transport and externally driven stochastic magnetic perturbations near the separatrix are both considered as prospective methods of ELM mitigation in present day tokamaks and ITER. Although these methods rely on completely different physics to generate extra transport, the influence of this transport on plasma dynamics and ELM mitigation is either similar or supplementary. ELM mitigation by ripple-induced transport relies on extra transport of thermal ions provided by the vertical drift of toroidally trapped particles. The parametric dependence of ion losses and their radial distribution as function of main dimensionless plasma parameters and magnetic field geometry is studied using Monte Carlo orbit following code ASCOT. This code also allows the study of the role of ripple losses in the formation of the edge radial electric field. The latter parameter influences L-H transition thresholds and transition from type-III to type-I ELMs. The results of ASCOT analyses were used in the 1.5D core transport code JETTO to predict how ripple-induced transport influence ELM dynamics. These results are compared with recent experimental observations from JET and JT-60U. Unlike ripple, stochastic magnetic field near the plasma edge induces mainly electron transport. This method was successfully used in DIII-D tokamak, where it was shown that externally driven Resonant Magnetic Perturbation (RMP) fully suppresses ELMs in low collisionality plasma. Two unconventional results of recent experiments are discussed in this report. The first relates to experimentally observed trend that the RMP increases particle transport more that electron thermal conductivity. The opposite result is expected from theoretical considerations. Secondly, experiment reveals that an observed level of stochastic transport is two orders of magnitude smaller than that given by the field tracing code. Possible explanations to both observations are presented. Relatively stronger manifestation of enhanced particle diffusion might be explained as a compensation of the opposite trend, observed in a conventional ELMy H-mode (where suppressed electron transport within ETB leads to uncontrolled density rise). A much lower level of stochastic transport might be a result of magnetic field screening by rotating plasma. Work conducted under EFDA and partly supported by Euratom and the UK EPSRC.

TH/P8-6 · ELM Simulations with M3D

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Abstract: Edge localized modes (ELMs) are important because they can control the loss of energy and particles at the edge of tokamak plasmas. We have carried out extended MHD simulations of ELM crashes and subsequent plasma relaxation, using DIII-D and ITER geometry and initial profiles. Gyroviscous stabilization was found to have a relatively small effect on ELMs, which are dominated by long wavelength modes. We are also performing simulations using a kinetic model to initialize bootstrap current and other profile data.

IT

ITER Activities

IT/1-1 · The Engineering Challenges of ITER

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Abstract: ITER is the most important step on the path to developing fusion energy using magnetic confinement. For the first time, reactor-grade plasma will be brought together with current technology to see whether a viable power source can be built. The main engineering challenge of ITER is therefore to produce it on time and within budget to have a timely decision on possible future energy sources and show the economic possibilities. On the way the aim will be to be flexible enough to take account of improved ways of implementing the design, or to take account of new plasma physics insights that have developed over recent years, if these can be accommodated within the schedule and cost, provided they give an appropriate benefit for the enhanced risk. Since the ITER design was completed 5 years ago, to the extent that a cost estimate could be agreed on by the then Participants, there have been a number of design modifications with a view to making the design and the cost estimates more realistic in practice, or actually to cut costs. Naturally there have also been some research and technical developments during that period, which now might allow better design solutions to be implemented. Thus, before ITER construction can go ahead, and before the licensing documents can be finalised, it is essential to carry out a detailed design review to assess whether the solutions now proposed still are valid, and can be accomplished in the planned timescale. The collaboration formed to build the ITER project is a powerful mix of many countries, cultures and institutions. With seven partners lining up to accept the Joint Implementation Agreement, with their different strengths and specialities, and with common interests to develop the key technologies, the procurement process for many components is highly interdependent, and involved. Experience in the large particle accelerator construction, other areas of science as well as in industry, however, shows that with a cooperative atmosphere between different suppliers of similar components there are great benefits to be had when problems inevitably arise. While it is too soon to state in detail any particular engineering challenges of ITER, especially as the design review is underway, this paper will elaborate on the above themes, highlighting where the experiences of the high energy physics field in the construction of large acc.

IT/1-2 · Review of ITER Physics Issues and Possible Approaches to Their Solution

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Abstract: ITER will demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes by nominal operation of 500 MW fusion power for 400 seconds. With its all superconducting coil technology, ITER will be capable of moving toward steady-state, high gain operation for fusion power. While the physics basis for ITER's nominal inductive operation is well established, the physics basis for steady-state is currently being developed (ref PIPB). Ongoing tokamak research programs must continue to contribute strongly during ITER construction to various physics issues whose resolution will improve both the inductive and steady-state operation of ITER.
* Work supported by US DOE under DE-FC02-04ER54698.

IT/1-3 · Edge pedestal physics and its implications for ITER

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Abstract: The H-mode pedestal plays a central role in achieving the required integrated plasma performance in ITER. The critical issues in the pedestal research are 1) understanding of the type I ELM trigger and energy losses, 2) identification of the processes determining the pedestal structure, 3) development of small/no ELM regimes and type I ELM mitigation methods, and 4) construction of integrated prediction codes. For all of these issues, remarkable progress has been achieved by integrating the results obtained in single- and inter-machine experiments (Alcator C-Mod, ASDEX-U, DIII-D, JET, JFT-2M, JT-60U, MAST and NSTX) with theoretical progress. It has been confirmed that the trigger of type-I ELM can be explained systematically by the peeling-ballooning modes. Nonlinear explosive evolution of ELM crash has been revealed by improved diagnostics and reproduced numerically. The ratio of the energy loss by an ELM to the pedestal stored energy increases with decreasing collisionality, and a simple projection to ITER shows that it exceeds the allowable level. It has been revealed that the transport between ELMs is close to the ion neoclassical transport and that the pedestal width is determined by the magnetic field structure and non-dimensional parameters. Neutral penetration depth also plays a role. Prediction of the fusion gain Q in ITER depends strongly on the pedestal temperature T_{ped}. Although some models predict

a $T_{ped} \sim 4$ keV with $Q > 10$, the range of predicted T_{ped} is still wide. The ripple loss of fast ions and the shift of plasma rotation into co-direction increase the pedestal height. The small/no-ELM regimes (QH, grassy-ELM, type II ELM, EDA/HRS) have been reproduced in multiple devices by matching the non-dimensional parameters and the plasma shape. Accessibility to these regimes has been categorized. The grassy-ELM and QH regimes have been achieved at collisionalities close to ITER. Type I ELM mitigation techniques, such as pellet injection and application of a resonant magnetic perturbation, have been developed and their applicability to ITER is under evaluation. The modeling capability for integrating the core, pedestal, SOL and divertor regions has achieved remarkable progress. Models based on turbulence suppression and peeling-ballooning stability have reproduced experiments and predicted ELMing edge evolution in ITER.

IT/1-4 · Plasma-surface interaction, scrape-off layer and divertor physics: Implications for ITER

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Abstract: This review of work by the International Tokamak Physics Activity (ITPA) SOL/Divertor group shows the considerable progress achieved in the last several years. There is a better understanding of the SOL profiles and relationship of transport to the underlying turbulence both in regions near and far from the separatrix. ELMs have been found to travel with high velocity far into the SOL, carrying pedestal characteristics as filaments of plasma with high poloidal/toroidal number. Present estimates of the ELM energy reaching non-divertor surfaces are ∼10–20% of the total ELM energy (large uncertainties). The impact can be substantial for main chamber tile lifetime and impurity sources affecting the core plasma. The broadening of the plasma footprint during disruptions leads to ∼50% of the plasma magnetic and thermal energy being deposited in the divertor. The disruption divertor energy deposition also appears to be reduced as the amount of thermal energy in the plasma increases. This again leads to predictions of reduced loading and erosion of the divertor but higher levels of loading/erosion for main chamber surfaces. However, such surfaces can absorb damage with less affect on steady state performance than the divertor. Since the plasma energy content is often reduced as the disruption approaches, a significant fraction of the stored energy (up to 80%) can be lost before the thermal quench. Disruption mitigation techniques, both through chemical and optical techniques, are being developed and tested on a variety of tokamaks. The present experience bodes well for ITER. The retention of tritium remains a serious concern for ITER where current experience show T to be mainly co-deposited with C on tile surfaces facing low-Te plasmas and areas shadowed from direct plasma contact. Tile-side retention is significant. In actively-cooled devices running long pulses, the D retention rate remains constant during the discharge, while the D recovery after the shot does not increase with the pulse length: the accumulated vessel inventory is then proportional to the discharge duration, reaching levels as high as 50%. Overall D retention in tokamaks with high-Z PFCs appears lower than for carbon. D/T recovery techniques are being tested but the scalability to the needs of ITER (rates and materials) is still to be shown.

IT/1-5 · The design and implementation of diagnostic systems on ITER

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Abstract: In order to meet the expected needs for first wall and plasma measurements, ITER will require about 40 different diagnostic systems drawn from all the main generic diagnostic groups – magnetics, neutron systems, optical and microwave systems, spectroscopic, bolometric, probes, pressure gauges and gas analysers. The design and implementation of the diagnostic systems is a major challenge because of the harsh environment in which many of the diagnostic components are located coupled with the restricted access and the need to meet stringent engineering requirements arising from the fact that ITER will be a nuclear device. It has stimulated an extensive design and R&D programme and the development of some novel approaches to diagnostic installation: for example, the use of plugs with custom modules at the upper and equatorial levels that serve both to support the diagnostic components and to provide the necessary shielding of the neutrons. In the paper, the difficulties of implementation will be summarized and the novel solutions described. An assessment of the performance of the diagnostic system relative to the specified target measurement requirements will be given.

IT/1-6 · EUROPEAN Contribution to the Design and R&D Activities in View of the Start of the ITER Construction Phase

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Abstract: The European effort in supporting the ITER design and R&D programme was maintained at a considerable level (about 70 M Euro/year in 2005 and 2006) in order to be ready to start the construction phase as soon as the ITER site is decided and the ITER Team is nominated. The main objectives of the activities performed in 2005 and 2006 are: (a) To continue the design and R&D effort towards the ITER procurement requirements in close collaboration with the ITER International Team. (b) To continue and complete manufacturing R&D to determine the most technically and cost effective manufacturing methods of the ITER components to be built in Europe. (c) To launch or to continue the preparation of the new facilities needed during ITER construction (DIPOLE, HELOKA, DTP-2, ECRH Test Facility, Fatigue Testing Facility). (d) To support the European site preparation process through an appropriate organization. (e) To develop the capabilities of the EU Associations in preparation of the procurement of ITER systems in the Heating and Current Drive and Diagnostic areas. (f) To maintain support to EU Industries in the fusion related work. The main achievements in the design and R&D have been: Divertor – small, medium and full-scale prototypes have been successfully tested at heat flux above the ITER requirements; Shield modules – alternative fabrication techniques are been developed to increase reliability, competition among industries and decrease fabrication costs; Vacuum Vessel – different welding techniques and distortion prediction models have been investigated; Magnets – advanced Nb_3Sn strands and 70 kA high temperature superconductor current leads have been developed and tested exceeding the ITER requirements; Test Blanket Modules – the design was completed; manufacturing processes using EUROFER are developed; Fuel Cycle – extensive and successful tests were performed with half size torus exhaust model cryopump; ICRF – design work of the antenna array is being advanced; Diagnostics – significant progress has been made on several of the diagnostic the EU is likely to supply to ITER; Safety and Environment – experimental R&D and safety assessment in support of ITER licensing was carried out. The paper will provide a detailed description about the status of the ITER design and R&D activities developed in Europe highlighting the major achievements.

IT/2-1Ra · New results and remaining issues in superconducting magnets for ITER and associated R&D in Europe

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Abstract: Both the CS (Central Solenoid) coils and the TF (Toroidal Field) coils in ITER will use Nb_3Sn as superconducting material, which proved very sensitive to applied strain and has a limited production, when 517 t will be required for ITER. PF and CC coils will use NbTi strands, although no conductors, carrying 45 kA and operating in pulse mode in a tokamak, have ever been produced. For Nb_3Sn conductors, the main milestones of the R&D programme were the manufacture and tests of two model coils, the Central Solenoid Model Coil (CSMC) and the Toroidal Field Model Coil (TFMC). Despite the achievement of their nominal operation in terms of current and magnetic field, these coils showed a reduced margin, compared to what was expected from strand measurements. Following a revision of the design of these conductors in 2003 a complementary R&D programme was launched to qualify the modified design. For NbTi conductors for PF coils a specific development is carried out and an important milestone will be the tests of a 50 m PF conductor, wound in a single layer solenoid and inserted in the CSMC bore. Although manufacturing techniques for TF and CS coils have been qualified by the construction of the model coils, nevertheless, several points require further development. The metallic screen inside the insulation of the PF conductor, aiming at control of the dielectric quality of the insulation through the life of the machine, is the first one. A second item is the insulation system of the TF coils. Whereas the TFMC used a classical multilayer glass-polyimide composite, impregnated by epoxyde resin, a specific development is being carried out, to demonstrate the feasibility of using in the TF coils radiation-resistant resins, such as cyanate-ester. Contrary to the model coils, the CS, TF coils and PF coils will be wound into multiple pancakes, which implies the insertion of helium inlets at the innermost turn. First results of this development will be presented, including as well mechanical qualification as hydraulic qualification. A dedicated development is also carried out to demonstrate the manufacturing feasibility of the radial plates into which the TF conductor will be wounded. Another development includes prototypes of the precompression rings. Endly, fatigue testing of a prototype mock-up is planned to demonstrate feasibility and effectiveness of a PF tail design.

IT/2-1Rb · Technology Development for the Construction of ITER Superconducting Magnet System

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Abstract: Engineering basis of the ITER superconducting (SC) magnets has been established through the development of the CS model coil (13T, 46 kA) and TF model coil (10T, 80 kA) in the ITER EDA. Based on this achievement, Japan Atomic Energy Agency (JAEA) has started activities in order to further develop the magnet technologies to a full scale level for the ITER construction. In the model coil development, JAEA has developed two types of Nb_3Sn strands: bronze process and internal tin process strands. The results of the model coil experiments, however, revealed a significant loss of superconducting performance in a cable-in-conduit conductor under operating conditions. In order to compensate the loss, new requirements have been specified for Nb_3Sn strands, and a trial fabrication (about 0.1 ton each supplier) was performed in Japan. Critical current densities of 1,000 A/mm^2 and 750 A/mm^2 were obtained in the internal tin process strands and in the bronze process strands, respectively, both of which satisfied the ITER requirements (700 A/mm^2 in bronze process, 800 A/mm^2 in internal tin process). TF coil manufacture consists of three major processes: (1) winding pack manufacturing, (2) structure manufacturing, and (3) encasing of the winding pack and final machining. Detailed procedures and tooling requirements are being finalized for these processes through the studies with industries. Based on these studies, demonstration at full scale level is also being performed. The TF coil uses several kinds of cryogenic steels, depending on the requirements of the mechanical strength. At the highest stress area, grade JJ1 will be used, and at the second highest stress area, grade ST316LN (strengthened SS316LN with nitrogen content of more than 0.17%) will be used. Trial fabrications of these materials are under way to establish the manufacturing route of large scale materials and the manufacturing processes for machining and welding. Database of material properties will also be established in these trials. Yield strengths of the forgings of JJ1 and ST316LN were confirmed to exceed the required values of 1,000 MPa and 850 MPa, respectively. These activities are being performed in extensive collaboration with industries and will provide a firm technical basis to realize the required performance of the magnets while maintaining both planned schedule and cost.

IT/2-2 · High Temperture Superconductors for Future Fusion Magnet Systems – Status, Prospects and Challenges

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Abstract: The application of High Temperature Superconductors (HTS) in fusion research has, until now, been limited to the use in current leads. Studies for a use in bus bar systems show a technological potential, too. The applicability of HTS in fusion magnets is presently not feasible but would be desirable for use in future fusion reactors, e.g., DEMO and beyond. The main driver to use HTS materials for fusion coils is to operate the magnets either at high magnetic fields, e.g. 20 T, or at medium or high operation temperatures (50–65 K). As the low temperature superconductor (LTS) layouts for a Tokamak or a Stellarator has been optimized by balancing the requirements for current distribution, AC losses and cooling capability, the situation is different in case of HTS, especially for YBCO coated conductors. While transient cooling is not important due to the much higher heat capacity of the materials, AC losses, thermal stability, hot spot temperature and current distribution will play a challenging role due to the structure of the coated conductor tape. This contribution will give an overview about status, promises and challenges of HTS conductors on the way to an HTS fusion magnet system for DEMO and beyond.

IT/2-3Ra · Production of High Power and Large-Area Negative Ion Beams for ITER

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Abstract: The present paper reports the progress of negative ion source developments for the ITER neutral beam (NB) system at Japan Atomic Energy Agency (JAEA) since the last conference. In the accelerator development, an H^- ion beam of 146 A/m^2 was successfully accelerated up to 836 keV, which is the first production of the high power density beam relevant to ITER requirements. To improve the spatial uniformity of the extracted beam intensities, the magnetic configuration in the ion source was modified and tested. The root-mean-square deviation of the beam intensities from the averaged value was reduced to about a half of that before the modification while the total beam current was kept to be the same level.

IT/2-3Rb · Technological aspects of the different schemes for accelerator and ion source of the ITER Neutral Beam Injector

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Abstract: In this contribution are reported the latest results of the revision of the ITER Neutral Beam (NB) design performed by Consorzio RFX in the framework of the European activities for ITER. The design and technological feasibility of the RF ion source and of the single gap (SINGAP) accelerator have been addressed . As a result, the design of both components has been brought to a level of detail comparable to that of the arc discharge ion source and of the multiple aperture, multiple gap (MAMuG) accelerator adopted in the NB reference design for ITER. The mechanical compatibility of the ion sources, the electrical insulation of the Bushing and of the Transmission Line and the Power Supply System have been studied identifying the main technological issues common or specific of each concept and in particular. a) The RF driven source has been newly designed focusing on key components as flexible connections, feedthroughs, insulators, electro-hydraulic junctions and coaxial cables. b) An optimized electrostatic design of the 1 MV bushing for SINGAP configuration has been studied focusing on layout and assembly. c) The design of the Transmission Line has been developed for the SINGAP concept. It is based on a coaxial scheme and is insulated in SF6. Inside the inner conductor, water cooled copper busbars are installed to feed the ion source power supply. In particular the issue of the coolant under 1 MV voltage has been addressed. The polarisation effects of the dc voltage on the insulating structure have been analysed. d) The Accelerator Grid Power Supply (AGPS) and the Ion Source Power Supply (ISPS) have been studied in SINGAP configuration and compared with those in MAMuG configuration. The most critical elements are found the step-up transformers, which must assure the extremely high insulation voltage level of -1 MV between the windings and to ground. In the paper the studies performed to assess the system feasibility and in particular that of the transformers, including the implications for the adaptation to the SINGAP concept, are presented. Finally a discussion on the implication of the above listed results on the decision concerning the concepts to be adopted in the Neutral Beam for ITER is given.

IT/2-3Rc · Progress of the development of the IPP RF Negative Ion Source for the ITER Neutral Beam System

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Abstract: For heating and current drive the ITER neutral beam system requires negative hydrogen ion sources capable of delivering above 40 A of D^- ions from a 1.5×0.6 m^2 source for up to one hour pulses with an accelerated current density of 200 A/m^2. In order to reduce the losses by electron stripping in the acceleration system and the power loading of the grids, the source pressure is required to be 0.3 Pa at an electron/ion ratio $\leq$1. The development of the source was initially concentrated on filamented arc sources as described in the ITER reference design. As an interesting alternative, IPP Garching is currently developing a RF driven negative hydrogen ion source. RF sources offer substantial advantages: they are basically maintenance-free in operation being quite beneficial for the remote handling requirements of ITER; in contrast to the arc sources having a limited filament lifetime no regular maintenance periods are necessary. In 2005 the IPP RF source has demonstrated its principal suitability for the ITER NBI system: current densities (330 A/m^2 H^-/230 A/m^2 D^-) in excess of the ITER requirements have been already achieved on the small test facility "BATMAN" (Bavarian Test Machine for Negative Ions) at the required source pressure (0.3 Pa) and electron/ion ratio (<1), but with only small extraction area (0.007 m^2) and limited pulse length (<5 s). The development concentrates now on the extension of source size and extraction area as well as on the extension of the pulse length. This is done at two other test facilities in parallel: at the large test facility "MANITU" (Multi Ampere Negative Ion Test Unit) the extraction area can be extended up to 0.03 m^2 and the pulse length up to 3600 s. In order to demonstrate the required homogeneity of a large RF plasma source as well as the operation of an ITER relevant RF circuit, a so called "half-size source" – with roughly the width and half the height of the ITER source – was designed and went into operation on a dedicated plasma source test bed ("RADI"). An extensive diagnostic and modelling programme is accompanying those activities. The paper presents the latest results of the RF source development, with an emphasis on deuterium operation and Cs dynamics, on the first results of the operation of the half size ITER source and on the status of the long pulse operation.

IT/2-3Rd · Review of the EU Activities in Preparation of ITER

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Abstract: The paper deals with the activities in Europe towards the development of the EC power sources and of the NB injectors for ITER. The European effort for the EC power sources is centred on the development of a 170 GHz, 2 MW, CW coaxial cavity gyrotron of Collector Potential Depressed (CPD) type: a gyrotron with higher unit power compared to the ITER baseline (1 MW) will reduce the cost of the installation and will allow a compact upper port launcher design. Coaxial cavity gyrotrons have the potential to allow high unit power to be achieved and 2.2 MW in single mode were demonstrated in short pulse operation. The construction of the first industrial 2 MW prototype is well underway. A new full performances EC test facility is also being established including the required power supplies. The procurement of the ITER NB system presents several challenges and a robust R&D programme is necessary in order to achieve the required ITER parameters. The paper will report the main experimental results achieved in EU on the MANTIS and SINGAP test facilities at CEA, Cadarache, and at IPP, Garching, were a RF driven ion source, alternative to the ITER baseline, is being developed. The experimental activities on SINGAP and the RF ion source have been supported by a substantial design effort aiming and reviewing the main engineering aspects of the ITER NB injectors. Finally, the limited performances of the existing test facilities do not allow a reliable extrapolation to the ITER scale. The establishment of a full scale test facility is therefore necessary in the ITER NB development plan and has become a centre piece of the European NB development strategy.

IT/2-4Ra · 170 GHz, 2 MW, CW Coaxial Cavity Gyrotron for ITER – status and experimental results

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Abstract: A 170 GHz coaxial cavity gyrotron with 2 MW output power in continuous wave (CW) operation is under development for ITER. In proof of principle experiments carried out at FZK Karlsruhe on a 165 GHz coaxial cavity gyrotron during the last years, the feasibility of manufacturing a 2 MW, CW coaxial gyrotron at 170 GHz has been demonstrated and information necessary for a technical design has been obtained [B. Pioscyk, et al., IEEE Trans. Plasma Science 32, 3 (2004) pp. 413–417 and 853–860]. Based on these results and on the experience acquired during the development of the 1 MW, CW, 140 GHz gyrotron for W7-X, the technical feasibility of a 2 MW, CW, 170 GHz coaxial cavity gyrotron has been studied before EFDA placed a contract at industry for procurement of a first industrial prototype of such a gyrotron tube. The development work on the industrial prototype tube is performed in cooperation between European research centers together with European industry (TED). Within this cooperation the physical specifications and the design of main gyrotron components have been done by the research institutions whereas TED considered the technological aspects and is performing the manufacturing of the prototype gyrotron which is expected to be delivered in summer 2006. A test facility for testing the 2 MW tube up to CW is under construction and will be available in time at CRPP Lausanne. Due to a delay in the delivery of the SC magnet the experiments are expected to start around the end of 2006.

IT/2-4Rb · Development in Russia of High Power Gyrotrons for Fusion

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Abstract: Electron cyclotron systems of fusion installations are based on powerful millimeter wave sources – gyrotrons, which are capable to produce now microwave power up to 1 MW in very long (hundred seconds) pulses. The paper presents the latest achievements in development at IAP/GYCOM of MW power level gyrotrons for fusion installations. During last years several new gyrotrons were designed and tested at IAP/GYCOM. Among them are a new version of 170 GHz gyrotron for ITER and multi-frequency (105–140 GHz) gyrotrons for Asdex-Up. All these gyrotrons are equipped with diamond CVD windows and depressed collectors. The most efforts were spent for development of ITER gyrotron. The tests were performed at specially prepared test stand in Kurchatov Institute. The following gyrotron output parameters were demonstrated so far in many pulses: 0.9 MW/20 sec and 0.7 MW/40 sec, 0.5 MW/100 sec. Also a gyrotron with a higher power 1.2–1.5 MW was designed. A model of this tube showed 1.2 MW in relatively short pulses 0.1 sec. The tests continue. In two tested long-pulse dual-frequency gyrotrons, power in the output Gaussian beam exceeding 0.9 MW at 140 GHz (radiated power over 1 MW) and

0.7 MW at 105 GHz (radiated power of 0.8 MW) was attained at specified 10-s pulse duration. The multi-frequency gyrotron should operate at least at four frequencies in the frequency range 105 GHz–140 GHz. Two window concepts for the gyrotron were considered: Brewster window and two-disc adjustable window. The Brewster window is very attractive because of very wide instant frequency band, however the quasi-optical converter design in this case is more complicated. A short pulse gyrotron equipped with a Brewster BN ceramics window was tested at 11 frequencies. The results are encouraging.

IT/2-4Rc · Development of the 170 GHz Gyrotron and Equatorial Launcher for ITER

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Abstract: The progress of the development of a 170 GHz gyrotron and a launcher for an equatorial port (equatorial launcher) is described. The 1000 sec oscillation has been attained at 0.2 MW using the preprogrammed control of a heater power. The stable 1.5 MW oscillation was demonstrated by a 170 GHz short pulse gyrotron with a higher order oscillation mode $TE_{31,12}$. In the development of the equatorial launcher, neutron irradiation tests have been done for key components of a movable mirror, and reveals the applicability to ITER neutron environment.

IT/2-4Rd · The 140 GHz, 10 MW, CW ECRH Plant for W7-X: A Training Field for ITER

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Abstract: Both, the W7-X Stellarator and the ITER Tokamak, will be equipped with powerful ECR-Heating and Current Drive systems, which are similar in frequency and have CW-capability (140 GHz, 10 MW, CW for W7-X and 170 GHz, 24 MW, 1000 s for ITER). As the physics and technology demands have many similarities in both devices, results from the W7-X ECRH may provide valuable input for the ITER plant. The W7-X installation served already as a high power test bed for ITER ECRH- components. W 7-X ($R = 5.5$ m, $r_{eff} = 0.55$ m) aims at demonstrating the inherent steady state capability of stellarators at reactor relevant plasma parameters. A 10 MW ECRH plant with CW-capability is under construction to meet the scientific objectives. The physics background of the different heating- and current drive scenarios is briefly reviewed. The ECRH plant consists of 10 RF-modules with 1 MW power each. The RF-beams are transmitted to the W7-X torus (typically 60 m) via two open multi-beam mirror lines with a power handling capability, which would already satisfy the ITER requirements (24 MW). Integrated full power, CW tests of two RF-modules (Gyrotrons and the related transmission line sections) are reported. 0.9 MW were transmitted from each Gyrotron for 30 min through 7 mirrors of the transmission system into a calorimetric dummy load. A front-steering quasi-optical launcher mock up was fabricated, which is capable of wide angle scanning (<36 deg), first cyclic tests demonstrated the viability of the concept. The ECRH-project has entered the phase of series installation and commissioning, seven SC-magnets and five Gyrotrons are in different states of installation and tests. The status is reported.

IT/2-4Re · Experimental Results of the 1 MW, 140 GHz, CW Gyrotron for W7-X

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Abstract: For the stellarator Wendelstein 7-X now under construction at IPP Greifswald, Germany, a 10 MW ECRH system is foreseen. A European collaboration has been established between Forschungszentrum Karlsruhe (FZK), IPP Garching/Greifswald, IPF Stuttgart, CRPP Lausanne, CEA Cadarache and TED Vélizy, to develop and build the 10 gyrotrons each with an output power of 1 MW for continuous wave (CW) operation. The dependence of the output power and efficiency of the first series tube versus the beam current will be shown in short pulse operation (without depressed collector) and in CW operation (up to 30 minutes, depressed collector operation). RF-field measurements have been performed in order to characterise the output field of the gyrotron and to minimise losses during the transmission to the W7-X device. Several parameters have been optimised (e.g. beam radii, magnetic field) to maximise output power and efficiency of the tube. At FZK site, long pulse tests up to 180 s have been performed (limited by the available power supply), at IPP site the pulse length could be extended to 30 min, both at a power level of 1 MW and high efficiency.

IT/2-5 · Progress Towards Steady State Systems For Fusion Devices

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Abstract: Developments during many years have produced RF systems which have supported the progress of plasma performances and opened up new areas of investigation. For the next step devices, long pulse operation is essential and it is a new challenge for the integration of the RF systems. The Euratom-CEA association has developed over the years its knowledge and skill on RF systems development and integration. Three RF systems are presently operated on TS: ICRH, LHCD, and ECRH. Recent results include combined operation of RF systems, to sustain the plasma current and improve the plasma performances by raising the bootstrap fraction and the density in conditions relevant to ITER, i.e. with actively cooled PFCs. High power combining ICRH and LHCD has been injected in plasmas performed at a density close to the Greenwald limit, with a loop voltage less than 0.1 V. So far, the highest injected energy was 470 MJ: 4 MW/65 s ICRH pulse during a plasma sustained by 3 MW of LHCD. The fraction of non-inductive current was close to 80%, including a bootstrap fraction of about 20%. These physics results depend strongly upon RF systems development. Tore Supra LHCD system is presently being upgraded within the CIMES project to extend the pulse duration to 1000 s. An in-depth transformation of the transmitter will use new CW klystrons specially developed to deliver 700 kW on any load presenting a VSWR of 1.4. This transmitter will feed a new antenna using the PAM concept as foreseen for ITER. This concept allows to bring the cooling channels in the walls between the active guides, thus providing neutron shielding and cooling where necessary. The concept has been briefly tested on FTU, in collaboration with CEA, and similar launcher was also proposed for JET. A prototype ICRF antenna has been built for Tore Supra using the conjugate-T concept proposed for ITER. It has been already tested in 2004 in moderate power and pulse length (600 kW/6 s). Improved diagnostic and control systems have been implemented for the automatic control loops and the matching experiments which are in progress. These experiments complement the JET-EP antenna experience on JET, and help to document choices for the ITER ICRF antennas. Parallel studies are also going on for the ITER ICRH antenna that will deliver 20 MW to the plasma. Examples of these steps towards steady state RF systems will be presented.

IT/2-6 · Development of ITER-Relevant Plasma Control Solutions at DIII-D

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Abstract: The requirements of the DIII-D physics program have led to the development of many operational control results with direct relevance to ITER. These include new algorithms for robust and sustained stabilization of neoclassical tearing modes (NTM) with electron cyclotron current drive (ECCD) [R.J. La Haye, et al., Phys. Plasmas 9 (2002) 2051], model-based controllers for stabilization of the resistive wall mode (RWM) in the presence of edge localized modes (ELMs) [C.M. Gransson, et al., Phys. Plasmas 10 (2003) 3961], coupled linear-nonlinear algorithms to provide good dynamic axisymmetric control while avoiding coil current limits, and adaptation of the DIII-D plasma control system (PCS) [B.G. Penaflor, et al., Fusion Eng. and Design 71 (2004) 47] to operate the next-generation superconducting tokamaks, KSTAR [M. Kwon, et al., Fusion Sci. and Tech. 42 (2002) 167] and EAST [Y.X. Wan, et al., Proc. 20th IAEA FEC, Vilamoura, Portugal (2004) FT/3-3]. Development of integrated plasma control, a systematic approach to model-based design and controller verification, has enabled successful experimental application of high reliability control algorithms requiring a minimum of machine operations time for testing and tuning. The DIII-D PCS hardware and software and its versions adapted for other devices can be connected to integrated plasma control simulations to confirm control function prior to experimental use. This capability has been critical to control system implementation for tokamaks under construction.
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IT/E-1 · ITER Site Preparation in Cadarache

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Abstract: After the ITER site decision in June 2005, the work for the site preparation intensified in all areas. The paper provides an overview of the technical tasks by EU (primarily through the instrument EFDA) and by France, specifically, to support the timely implementation of ITER project in Cadarache. The implementation of the work , as defined in a master schedule for site preparation activities, regularly discussed and reviewed with the ITER Team, covers safety and licensing, in-fence technical studies, off-site

technical studies and socio-economy aspects (public information, communication, economic studies). The most relevant aspects of the work in each area are presented and the paper provides the latest update of the overall status of the ITER site preparation.

IT/E-2 · Broader Approach Activities toward Fusion DEMO Reactors

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Abstract: At the time of ITER site decision in Moscow on 28 June 2005, EU and Japan jointly declared their intention to implement Broader Approach Activities on a time frame compatible with its construction phase. On the basis of this declaration, technical discussions have been made by the working groups composed of experts from EU and Japan for the possible area of implementation. Agreement has almost been reached by both Parties: The Broader Approach Activities should cover most of the key R&D activities needed for realization of DEMO reactors, either those in support of ITER or those as technical basis of DEMO rector design, and comprise the following three projects: 1) Engineering Validation and Engineering Design Activities for the International Fusion Materials Irradiation Facility (IFMIF/EVEDA); 2) the International Fusion Energy Research Center (IFERC), comprising: a) A DEMO Design, R&D coordination Center aiming at establishing a common basis for a DEMO design, b) A Computational Simulation Center composed of super-computer facilities for large scale simulation activities, and c) An ITER Remote Experimentation Center to facilitate broad participation of scientists into ITER experiments. 3) the Satellite Tokamak Programme in the upgrade of JT-60 Tokamak to an advanced superconducting tokamak and participation in its exploitation, to support ITER and research towards DEMO. The EU and Japan will establish a Steering Committee responsible for the overall direction and supervision of the activities. A Project Committee will be established to support Steering Committee. Each project is lead by the respective Project Leader supported by the Project Team. EU and Japan will nominate an agency to discharge its obligations for the implementation of these projects. Resources for the Broader Approach Activities will be equally shared by EU and Japan, contributed mostly in-kind, and allocation of responsibilities have been identified. The IFMIF/EVEDA and IFERC projects will be implemented at Rokkasho while the Satellite Tokamak Programme at Naka, Japan. These activities will be open to other ITER Members.

IT/P1-1 · ITER – Safety and licensing – One year after site decision

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Abstract: The safe design of ITER has been paramount since the beginning of the ITER studies and the safety analysis was included in the Generic Site Safety Report produced in July 2001. On June, 28th 2005 it was decided to implement the experimental fusion facility in Europe, close to Marseilles, south of France. The design now needs to be checked according to the local legal requirements and the licensing will have to comply with the French regulations. The safety analysis will be presented to the regulator and public hearings should lead to the "license". Construction will then start. From March to May 2006 a public debate was launched to present the facility during a set of 20 meetings mostly in the locality. This process is the French implementation of the Aarhus convention, signed on June, 25th 1998. The future ITER operator was represented by the French Atomic Energy Authority (CEA), the ITER team being present for all meetings. The French regulations are mostly non-prescriptive and request that design provisions be taken according to the level of risk. Any codes and standards can be used as long as their safety margins are in good agreement with the level of reliability requested by the analysis. Nevertheless a few areas must follow prescriptive design rules. Fire prevention thus requires putting in place fire and confinement sectors; any chemical and radioactive vessels must be protected against accidental spilling. Pressure vessels and equipment must comply with a European directive and a French order in case of nuclear inventory. Building design and construction have to comply with European rules. The ITER designers, in close contact with the Participant Teams, are proceeding with the upgrading of the design to comply with these requirements. Priority has been given to those inputs which could have high impact on the design, for instance the building layout. The codes and standards for all equipment are also under revision in order to fit with the expected requirements, taking into account the procurement sharing agreement. Finally, and may be primarily, the QA system of the future organization will have to comply with a French order for nuclear facilities set in 1984 and close to the IAEA 50-C/SG-Q derived from the previous well known 50-C-QA. The responsibility of the future ITER operator for procurements dealing with safety is emphasized in this order.

IT/P1-2 · Benchmarking of Lower Hybrid Current Drive Codes with Application to ITER-Relevant Regimes

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Abstract: Lower hybrid (LH) waves have the attractive property of damping strongly via electron Landau resonance on relatively fast tail electrons. Consequently these waves are well-suited to driving current in the plasma periphery where the electron temperature is lower, making lower hybrid current drive (LHCD) a promising technique for off-axis current profile control in reactor grade plasmas. In addition, the RF source frequency can be chosen high enough to minimize the parasitic interaction of LH waves with fusion-generated alpha particles. The relatively high phase speed also minimizes deleterious effects due to particle trapping which can become important in the periphery. Given these physics considerations, we have undertaken a detailed benchmarking exercise in which we compared the predictions of several advanced simulation models for LHCD using a test case based on a proposed steady state operating mode (Scenario #4) for the ITER-FEAT device [ITER Technical Basis Document (IAEA, Vienna, 2001) Doc. No. GAO FDR 1 00-07-13 R1.0, Section 4.3.3]. The most advanced models that we have used combine a 3D Fokker Planck calculation with a toroidal ray tracing package. The modules iterate to compute a self-consistent nonthermal electron distribution function. Preliminary estimates for ITER-FEAT [ITER Technical Basis Document (IAEA, Vienna, 2001) Doc. No. GAO FDR 1 00-07-13 R1.0, Section 4.3.3] indicate that 1.6–2.0 MA of LH current can be generated at r/a = 0.65–0.70, using 30 MW of LHRF power. The advanced Fokker Planck – ray tracing code predictions have also been compared with less sophisticated models that combine ray tracing results with a 1D parallel velocity solution of the Fokker Planck equation, where the collision operator has been corrected for 2D velocity space effects due to pitch angle scattering. We shall also present recent calculations of the LH wave-alpha particle interaction using a Monte Carlo orbit following code, and discuss their implications in terms of the choice of LH source frequency. Finally we shall discuss optimization studies of LH current drive for ITER that address issues related to spectral control of the spatial location of current generation as well as maximizing the current drive efficiency. These results strongly promote the scientific case for the use of LHCD on ITER, in that sophisticated and benchmarked tools are now available to optimize the use of LHCD in ITER scenarios.

IT/P1-4 · Study on Current Drive Capability of Lower Hybrid Waves and Neutral Beam in an ITER Steady State Scenario

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Abstract: Neutral beam current drive (NBCD) and bootstrap current are dominant non-inductive CD sources in ITER steady state scenarios with the initial investment. However, an additional CD source is necessary unless we expect a larger $H_{H98y,2}$ to provide a larger bootstrap current. Lower hybrid current drive (LHCD) is attractive for its high off-axis CD capability. A previous study with LHCD presented fully non-inductive CD scenarios with $Q > 5$, assuming an a-priori LH current drive efficiency. We report LHCD capability in an ITER steady state scenario assessed with a physics code that incorporates a relativistic one-dimensional Fokker-Planck calculation and a ray tracing code. The LHCD code was experimentally validated in JT-60U. For a LH power spectrum on the refractive index parallel to the toroidal magnetic field $n_\parallel$ calculated from the present ITER LH launcher design, the total LHCD for 20 MW injection is calculated to be 0.54 MA for the ITER steady state condition in the previous study. The corresponding current drive efficiency is less than half of the assumption in the previous study. This low CD capability results from the low directivity of 70% and relatively high $n_\parallel$ of 2. Therefore we need to improve LHCD by optimizing the LH power spectrum. LHCD increases with $1/n_\parallel^2$. Decreasing $n_\parallel$ below 1.9, however, LHCD becomes worse because the accessibility condition of LHWs approaches. By peaking the electron density profile with keeping the fusion power, LHCD improves down to a lower $n_\parallel$ since the accessibility condition shifts due to a lower density in the LH absorption region. In a moderately peaked profile case, the duration of the hybrid operation will be limited only by the cooling capability ($t < 3000\,s$). A scan of the spectrum width $\Delta n_\parallel = 0.055 - 0.25$ shows that a narrower width is favourable for CD. Two-dimensional effects in the electron velocity space, which are expected to increase CD, are also examined. We also investigate physics models in NBCD codes for example for the ionization process, fast ion behaviour, electron shielding effect, and so on, and we compare a Fokker-Planck code and a Monte Carlo code and evaluate NBCD capabilities in ITER steady state scenarios.

IT/P1-5 · Transport Physics of Hybrid Scenario Plasmas in the International Multi-Tokamak Database and Implications for ITER

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Abstract: Turbulence induced transport coefficients are calculated using the GYRO gyrokinetic turbulence simulation code for Hybrid plasmas contained in the international ITPA profile database. Hybrid plasmas address the ITER long pulse, high fluence mission. Common features are central safety factors near or above unity, with sustained stationary high β_n, high confinement, and reduced inductive current relative to standard H-mode plasmas of equivalent fusion performance. Credible prediction of ITER hybrid performance depends on the successful application of simulation codes on existing experiments. A key conclusion of this work is that off diagonal elements in the transport matrix strongly affect energy, particle and momentum transport. This runs counter to the conventional approach of existing predictive models which rely on dominant diagonal transport coefficients. Implications for ITER simulation studies and directions for future investigation are also presented. The nonlinear GYRO studies are used to compute the changes in the transport coefficients $\chi_j \equiv \{\chi_{main}, \chi_{imp}, \chi_e, D_{main}, D_{imp}, D_e, \chi_{momentum}\}$ resulting from varying the drive/damping terms $d_k \equiv \{a/L_{T_{main}}, a/L_{T_{imp}}, a/L_{T_e}, a/L_{n_{main}}, a/L_{n_{imp}}, a/L_{n_e}, T_{main}/T_e, \gamma_{E\times B}/\gamma_0\}$ from their measured values by $\pm 20\%$. The subscripts "main" and "imp" refer to the bulk and effective impurity ion species, both treated kinetically, $\gamma_{E\times B}$ is the $E \times B$ flow shearing rate, and γ_0 is its value inferred from measurements. These results give a 7×8 matrix estimating $\partial\chi_j/\partial d_k$. Besides the usual diagonal terms, many of the off-diagonal terms have significant and complicated contributions to the transport. Examples of results for one of the JET hybrid plasmas averaged over the region r/a between 0.4 and 0.8 indicates that χ_{main} is driven mainly by $a/L_{T_{main}}$ and damped mainly by T_{main}/T_e; χ_e is driven mainly by $a/L_{T_{main}}$ and $a/L_{n_{main}}$; D_e is damped by a/L_{T_e} and $a/L_{T_{main}}$, and increased by a/L_{n_e} and T_{main}/T_e; and D_{imp} is driven by $a/L_{T_{main}}$, a/L_{T_e}, $a/L_{n_{main}}$, $a/L_{n_{imp}}$, and damped by T_{main}/T_e.

IT/P1-6 · Characteristics of the H-mode Pedestal in Improved Confinement Scenarios in ASDEX Upgrade, DIII-D, JET and JT-60U

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Abstract: The scaling of the H-mode pedestal in improved confinement scenarios is an essential step towards predicting the performance of these scenarios in ITER. To this purpose, this paper compares global and pedestal parameters in conventional H-modes and improved confinement scenarios in ASDEX Upgrade, DIII-D, JET and JT-60U. In all four tokamaks the ratio of pedestal to total thermal stored energy is in the range 0.2 to 0.5 for the scenarios under study. Depending on the particular regime, the performance improvement – with respect to conventional H-modes – is due to a combination of increased pedestal height and changes in the core physics. The main aim is to clarify whether a) increased pedestal pressure leads to increased global confinement or if b) increased core pressure improves the pedestal stability and this leads to an increase in pedestal pressure. The multi-machine scaling of the pedestal pressure with main global parameters is also investigated.

IT/P1-7 · Simulation of the Hybrid and Steady State Advanced Operating Modes in ITER

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Abstract: The International Thermonuclear Experimental Reactor (ITER) project has identified three primary operating modes for demonstrating controlled burning plasmas, the ELMy H-mode, the Hybrid mode, and the Steady State Mode. Integrated simulations are done to establish a physics basis, in conjunction with present tokamak experiments, for the operating modes in ITER. Since it is not possible to reproduce all the physics parameters of ITER plasmas simultaneously in present experiments, simulations are used to project to the ITER regime using theoretically based physics models, that are being benchmarked on present tokamak experiments. Simulations of the hybrid mode are done using both fixed and free-boundary 1.5 D transport evolution codes including CRONOS, ONETWO, TSC/TRANSP, TASK, and ASTRA. The hybrid operating mode is simulated using the GLF23 energy transport model. The injected powers are limited to the negative ion neutral beam (NNBI, 33 MW), ion cyclotron rf heating (ICRF, 20 MW), and electron cyclotron (EC, 20 MW). Overall, results indicate that the pedestal temperatures required to access the β_{N} of 3 regime are in the 8–10 keV range. Simulations of the steady

state operating mode are done with the same 1.5 D transport evolution codes cited above. In these cases the energy transport model is more difficult to prescribe since the models, used for H-mode plasmas (like the ELMy H-mode and hybrid) that are dominated by $E \times B$ shear stabilization, have deficiencies when applied to reversed shear ($dq/dr < 0$), and high pressures (Shafranov shift). Therefore the energy confinement models will range from theory based to empirically based. The injected powers include the NNBI, ICRF, EC, and lower hybrid (LH, up to 40 MW). Results using NNBI, ICRF, and LHCD indicate that if an internal transport barrier is formed, with a T_{ped} of 3 keV, a β_N of 3.0 can be reached, giving H_{98} of 1.7. Another case using NNBI, ICRF FW, and EC with the GLF23 theoretical energy transport model and a T_{ped} of 7.5 keV, giving β_N of 3.0, and an H_{98} of 1.5. These simulations will be presented and compared with particular focus on code to code results when using the same energy transport model, and within the same code when using different energy transport models.

IT/P1-8 · EC Radiation Transport in Fusion Reactor-Grade Tokamaks: Parameterization of Power Loss Density Profile, Non-Thermal Profile Effects under ECCD/ECRH conditions

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Abstract: Electron cyclotron radiation (ECR) was shown [Albajar F., et. al., Nucl. Fusion, 45 (2005) 642] to contribute significantly to the local energy balance in the central part of the plasma column in steady-state scenarios of ITER operation. Strong sensitivity of the net ECR power loss density profile, $P_{EC}(r)$, to the presence of superthermal electrons was shown in [Cherepanov K.V., Kukushkin A.B., 20th IAEA Fusion Energy Conference. (Vilamoura, Portugal, 2004), TH/P6-56] for ITER scenario 2 (Inductive). Here we report on solving the following three tasks for ITER-like conditions: (1) parameterization of the profile $P_{EC}(r)$ for maxwellian plasmas, as inferred from calculations with the code CYNEQ [Cherepanov K.V., Kukushkin A.B., 20th IAEA Fusion Energy Conference. (Vilamoura, Portugal, 2004), TH/P6-56] (parameterization is to be used as a simple simulator during the transport calculations, in particular, in the ITER case); (2) modeling of deviations of the electron velocity distribution function (EDF) from maxwellian, caused by the ECCD/ECRH at low harmonics of the cyclotron frequency (e.g., O-mode $n = 1$), using the beam tracing code TORBEAM [Poli E., et al., Comp. Phys. Commun., 2001, 136, 90] and the Fokker-Planck code RELAX [Westerhof E., et al., Rijnhuizen Report RR 92-211 (1992)]; (3) modeling, with the code CYNEQ, of the profile $P_{EC}(r)$ for the non-maxwellian EDF of item 2 to evaluate the influence of ECCD/ECRH-produced superthermal electrons on the profile $P_{EC}(r)$, which, for the ITER case, is dominated by the transport of plasma's ECR at harmonics $n \sim 3$–10. The combined calculations with the codes TORBEAM+RELAX+CYNEQ for scenario 2 predict maximal impact of the ECCD-produced superthermal electrons on the profile $P_{EC}(r)$ (a $\sim 20\%$ rise in the core) for oblique launch with full power deposition in the center (e.g., for equatorial launch at 170 GHz, O-mode, $n = 1$, with toroidal injection angle $\beta \sim 20°$).

IT/P1-9 · Simulation of Impurities Behaviour for Basic ITER Scenarios

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Abstract: Results of simulation of transport and radiation of different impurities for the basic ITER scenarios are presented. Radial profiles of its concentrations and radiation are calculated. Possibility creating of re-radiating layer near plasma periphery without accumulation impurities in the plasma core is considered. Modelling of impurity behaviour was performed by the ZIMPUR code, which has been integrated with the ASTRA transport code for simulation of the bulk plasma parameters. Role of the neoclassical and anomalous impurity transport in different scenarios is considered. It is shown that the neoclassical thermal diffusion assists in screening of the plasma core from impurities in all considered regimes. The most appreciable effect is exhibited for cases where the neoclassical impurity transport is expected in the plasma core: for steady-state reversed shear scenario and for high-Z impurities in ELMy H-mode scenario. Noticeable effect is waiting also for the basic inductive ELMy H-mode scenario for impurities with smaller Z, which more influenced by anomalous transport. Simulations show that impurity radiation increases to the plasma edge as a result of a reduction of the impurity charge state due to charge exchange with hydrogen isotope neutrals. So, possibility of creating a re-radiating layer near the plasma edge without impurity accumulation in the plasma core is demonstrated. Lethal concentrations of different impurities (W, Ar, C, Be, He) are calculated for the basic inductive ITER scenario. At the exceeding of these concentrations, discharge transfers to the L-mode what results of plasma cooling.

IT/P1-10 · The Tortuous Route of Confinement Prediction near Operational Boundary. — Improvement of Analysis based on ITERH.DB4/L.DB3 Database.

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Abstract: For practical purposes, empirical confinement time scalings, e.g. ITER-89P (L-mode) and ITERH-98(y,2) (ELMy H-mode, thermal), have been used and are used to predict the confinement time in future fusion experiments such as ITER and accompanying satellite devices. For a variety of reasons, prediction intervals, especially for reactor-scale devices, are tainted with a considerable uncertainty, see e.g. [ITER Physics Basis, Nuclear Fusion 30 (1999) 2137; O. Kardaun, Nuclear Fusion 42 (2002) 841]. Perhaps unexpectedly, systematical improvement of such scalings turns out to be a long and even tortuous path. Scalings beyond simple power laws for H- and L-mode are developed that describe the confinement roll-over near the Greenwald limit as a function of magnetic shape and plasma fuelling. This approach, based on multi-machine databases, is motivated by reactor design issues and aims at improving physical understanding by providing a benchmark for various plasma-physical theories. The isotope effect is interesting for physical reasons and also motivated by the fact that during the first years of ITER operation the discharges are expected to be in L-mode and with hydrogen or helium working gas.

IT/P1-11 · A Method for Error Field Detection in ITER

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Abstract: The problem of the error field detection in tokamaks is discussed from the viewpoint of possible application in ITER. The analyzed algorithms rely on measuring the plasma dynamic response to the finite-amplitude external magnetic perturbations, both the intrinsic error fields and pre-programmed probing pulses. In ITER such pulses can be produced by the coils designed for the static error field correction and for stabilization of the resistive wall modes, the technique used now in several tokamaks, including DIII-D and JET. Here the analysis is based on the theory predictions for the resonant field amplification (RFA). To achieve the desired level of the error field correction in ITER, the magnetic diagnostics must be sensitive to signals of several Gauss. Therefore the measurements for error field detection should be performed near the plasma stability boundary, where the RFA effect is stronger. While the proximity to the marginal stability is important, the absolute values of plasma parameters are not. This advantage can be used by lowering the plasma stability threshold in diagnostic experiments and operating in a regime much below the nominal level. The discussed method of the error field detection is an extension of the ‘active MHD spectroscopy’ recently successfully employed in the DIII-D tokamak and the EXTRAP T2R reversed-field pinch. The estimates for ITER are presented. The method can be tested in dedicated experiments in existing tokamaks.

IT/P1-12 · Dependence of the H-mode Pedestal Structure on Aspect Ratio

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Abstract: We report on a set of experiments between NSTX, MAST, and DIII-D to determine the aspect ratio dependence of the pedestal. The dimensionless parameters of electron collisionality ($\nu_e^\star$) and normalized ion gyroradius ($\rho_i^\star$) were matched at the top of the outboard pedestal, and the widths and gradients were assessed. These experiments were motivated in part by the fact that many of the machines in the international database used for scaling the pedestal parameters for the International Thermonuclear Experimental Reactor (ITER) have different aspect ratios, and the experiments were conducted through the ITPA pedestal physics working group. A significant dependence of the pedestal on aspect ratio would not be surprising, because variation of the aspect ratio primarily affects the edge magnetic topology. A common double-null shape was developed for these experiments with triangularity ~0.5 and elongation ~2. The toroidal fields and plasma currents used were 0.45–0.55 T and 0.6–0.8 MA in all three machines. The dimensionless parameters $\nu_e^\star \sim 1$ and $\rho_i^\star \sim 0.01$ were matched at the top of the outboard pedestal by variation of the target density and neutral beam heating power while maintaining ELMy H-mode. The pedestal widths and gradients were analyzed in each machine using a ‘standard’ modified hyperbolic tangent function; the ranges of pedestal top parameters obtained in this manner were n_e^{ped}: 3–5 $\times 10^{19}$ m^{-3}, T_e^{ped}: 100–250 eV, and P_e^{ped}: 0.4–1.5 kPa. The pedestal n_e, T_e and P_e widths measured in DIII-D for these discharges were between 6–8% in Ψ_N (normalized poloidal flux), i.e. almost twice as large as the normal range of widths at the normal B_t = 2.1T. In comparison, the pedestal widths in MAST were between

1.5–4% in Ψ_N, and final assessment of the widths in NSTX is still in progress. Edge stability analysis has commenced and will be presented at the conference for all three machines.

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IT/P1-13 · Optimization of Sensor Signals for Resistive Wall Mode Control in ITER

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Abstract: Advanced tokamak scenarios, such as the ITER Scenario-4, may suffer from global ideal MHD instabilities – the low-n, non-axisymmetric resistive wall modes (RWM), which limit the operational space in terms of achievable plasma pressures under steady state operation. It is highly desirable to stabilize the RWM in ITER, especially for the $n = 1$ mode that gives the most severe pressure limit for the Scenario-4. Our previous work has indicated that, since rotational stabilization of the $n = 1$ RWM may not be robust in ITER, active control of this mode is necessary. Some recent work for the vertical stability control (the $n = 0$ RWM control) has suggested an idea of using a combined sensor signal, measured at various poloidal locations, as one way to improve the feedback system. The key element in this idea is to use as many as possible sensor signals, in order to extract the response purely from the full system unstable mode. In this work, we apply the similar idea to improve the $n = 1$ RWM control in ITER using radial sensors. The additional benefit of extracting purely unstable RWM is that, (in ideal case) only proportional feedback gains are required to stabilize the mode. No derivative gains are needed even at high plasma pressures. This is highly desirable since derivative gains require the time differentiation of sensor signals, which increases the noise level in the sensor signal. We use the single fluid MHD stability code MARS-F for the study of the RWM feedback stabilization. The plasma pressure in ITER is characterized by a parameter C_β, which scales linearly with the plasma pressure, such that $C_\beta = 0$ corresponds to the no-wall pressure limit, and $C_\beta = 1$ corresponds to the pressure limit with an ideal wall. Our calculations show that, by keeping the present design of the control coils in ITER (the super-conducting side correction saddle coils), and by using three sets of radial sensors along the poloidal angle, it is possible to find a single, optimal linear combination of the sensor signals, which leads to stabilization of the $n = 1$ RWM for C_β up to about 0.9. The stabilization is achieved by using proportional feedback only. For comparison, using the mid-plane radial sensors alone stabilizes the mode for C_β up to about 0.4.

IT/P1-14 · ELMs and disruptions in ITER:Expected Energy Fluxes on Plasma-Facing Components from Multi-machine Experimental Extrapolations & Consequences for ITER Operation

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Abstract: Operation of ITER in high fusion gain regimes comes associated with plasma conditions in which Plasma Facing Components (PFCs) can be subject to large energy fluxes by ELMs and disruptions. These loads can contribute significantly to the overall erosion rate and lifetime of these components. Significant progress in the characterisation of the ELM and disruption transient loads in divertor tokamak experiments has taken place in the last few years. The measurements obtained have provided a physics-based framework on which the expected transient energy loads on ITER PFCs can be estimated. The expected ELM power fluxes depend on the mechanism dominating the energy loss from the plasma during ELMs: conduction and convection. Convective ELMs are associated with small normalised ELM energy losses, which would be compatible with the required lifetime of the ITER PFCs, but are obtained in conditions which are not compatible (pedestal collisionality or normalised energy confinement) with the ITER reference scenario. Studies of the energy balance and power fluxes during disruptions have shown that the thermal energy of the plasma at the thermal quench is a factor of 2–4 lower than that of the full performance plasma for most plasma conditions. Most of the plasma thermal energy during disruptions is deposited on PFCs by conduction/convection during the thermal quench, onto an area which is a factor of 5–10 larger than for normal plasma operation. On the basis of these results, the expected fluxes on the ITER PFCs during transients are : (1) Divertor target. Type I ELM energy fluxes: 0.5–4 MJ/m^2 in timescales of 300–600 microseconds, Thermal quench energy fluxes of 2–13 MJ/m^2 in timescales of 1–3 ms. (2) Main wall. Type I ELM energy fluxes: 0.5–2 MJ/m^2 in timescales of 300–600 microseconds. Thermal quench energy fluxes of 0.5–5 MJ/m^2 in timescales of 1–3 ms. Mitigated disruption radiative loads of 0.1–2 MJ/m^2 in timescales of 0.2–1.0 ms. The physics models used to perform the extrapolations from present devices to ITER and the results of the experimental and modelling studies carried out to determine the associated erosion of the

divertor and main chamber PFCs in ITER and the implications for the plasma discharge and the operation of the device for transient loads in these ranges will be discussed.

IT/P1-15 · Laser Methods Development for in situ ITER Walls Detritiation and Deposition Layers Characterisation

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Abstract: Laser methods for tokamak surface detritiation and characterisation have been under thorough multiaspect experimental and theoretical study in LILM laboratory (CEA Saclay, France). The graphite surface characterisation and cleaning with the pulsed repetition rate Nd-YAG laser systems were successfully realised. Heating and ablation regimes were distinguished by ablation threshold fluence. It was demonstrated that due to the significant difference in ablation thresholds for graphite (25 kJ per square meter) and a deposited layer (4 kJ per square meter), it was possible to decontaminate the graphite surface without its damage, if the laser fluence is lower than the ablation threshold for graphite. The ablation rate of the TEXTOR deposited layer was determined as 0.2 micrometer per laser shot. Thus, with the pulsed repetition rate lasers of 250 W mean power, a deposited layer of 20 micrometers can be removed with the rate of one square meter per hour. The developed laser facilities are very flexible and can be easily fixed on a robot for in situ surface characterisation and detritiation. The developed laser system will be installed onto AIA (Articulated Inspection Arm) robot on TORE SUPRA. The ablated matter will be aspirated by the nozzle fixed on AIA and collected on the aspirator filters. The integrated pyrometer system will be applied to record the surface temperature in laser heating regime (laser fluence below 4 kJ per square meter) with a high repetition rate Nd-YAG laser. The flexibility of the developed laser system is an important advantage. The same laser system (by adjusting appropriately the laser beam energy and spot) may allow to switch from heating regime (deposited layer depth estimation by pyrometer method) to ablation (where the layer depth is directly measured from the total ablation time) and to Laser Induced Breakdown Spectroscopy (LIBS) with laser plasma plume formation. A good agreement was demonstrated between the experimental results and the developed theoretical 3D model of surface heating (graphite + layer) that allowed to determine the deposited layer depth with micrometric accuracy. The preliminary studies have shown that analysis by LIBS method might be suggested to estimate tritium concentration in the material. Further goals and tasks to satisfy ITER requirements are discussed.

IT/P1-16 · Effect of pumped gas reflux on divertor operation in ITER

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Abstract: The paper describes the latest results of B2-Eirene modeling of ITER divertor operation. The operational window is further explored with improved model of the neutral transport, the effect of gas leaks between the divertor cassettes is assessed, and the initial results on impurity seeding are presented.

IT/P1-17 · Methane Formation under Charcoal Interaction with Atomic Hydrogen and Deuterium at 77 K

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Abstract: Charcoal is a working material of sorption cryopumps in the ITER project. An interaction of thermal hydrogen (deuterium) atoms and molecules with charcoal has been analyzed by adsorption measurements at 77 K and TDS (77–300 K). A stream quartz tube of 100 cm length, 1 cm radius with an $H_2(D_2)/CH_4$ RF discharge was used for the production of H(D) atoms and CH_3 radicals. The charcoal was an effective sink for atomic hydrogen. Under charcoal interaction with the $H(D)/H_2(D_2)$ mixture (77 K) the methane or deuteromethane (CD_4) appeared in thermodesorption spectrum. Only methane was recorded in mass spectrum at annealing of sample up to 300 K. Methane was observed after a critical fluence depending on charcoal mass. The numbers of methane molecules formed and hydrogen atoms in the tube cross-section with the samples were close to each other. It means that the sticking coefficient of atomic hydrogen for the charcoal surface is close to unity at 77 K. At constant H/H_2 flow the methane formation rate was higher for the samples with lesser mass. The methane yield increased with the exposure time. The CD_4 yield some exceeded of CH_4 one. After increasing temperature (up to 300 K) of the samples exposed in $H(D)/H_2(D_2)$ mixture remained hydrogenated. Charcoal contained the chemically bound hydrogen

in quantities of tenths parts of H atoms fluence. After transfer of the sample to adsorption setup the decreasing of hydrogen sorption capacity (T = 77 K) made up several percent when carbon atoms (∼0.1%) transformed into methane.

IT/P1-18 · Studies on Behavior of Tritium in Components and Structure Materials of Tritium Confinement and Detritiation Systems of ITER

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Abstract: The confinement and removal of tritium are the key subjects for safety of ITER. The ITER buildings are confinement barriers of tritium. In a hot cell building, tritium is often released, as vapor and is in contact with the inner walls. Also those of an ITER tritium plant building will be exposed to tritium in an accident. However, the data are scarce, especially on the penetration of tritium into the concrete of the wall materials. The tritium released in the buildings is removed by the Atmosphere Detritiation Systems (ADS), where the tritium is oxidized by catalysts and is removed as water. Special gas of SF_6 is used in ITER, and is expected to be released in an accident such as fire. Although the SF_6 gas has the potential as a catalyst poison, the performance of ADS with the existence of SF_6 has not been confirmed yet. Tritiated water is produced in the regeneration process of ADS, and is subsequently processed by the ITER Water Detritiation System (WDS). One of the key components of WDS is an electrolysis cell. The electrolysis cell is made of organic compounds, and there is no data on the durability of the cell exposed to tritium. To overcome these issues in a global tritium confinement, a series of experimental studies have been carried out as an ITER R&D task: 1) tritium behavior in concrete; 2) effect of SF_6 on performance of ADS; and 3) tritium durability of electrolysis cell of ITER-WDS. 1) The tritiated water vapor penetrated into the concrete up to 2 cm from the surface only in two months' exposure. The penetration rate of tritium in the concrete was thus appreciably large, so that it is required to evaluate the effect of the lining on the penetration rate quantitatively from the actual tritium tests. 2) The SF_6 gas decreased the detritiation factor of ADS. Since the effect of the SF_6 depends on its concentration closely, the amount of SF_6 released into the tritium handling area in an accident should be deduced by some ideas of the arrangement of components in the buildings. 3) It was expected that the electrolysis cell of ITER-WDS could endure 3 years' operation under the ITER design conditions. Measuring the concentration of the fluorine ions could be a promising technique for monitoring the damage of the electrolysis cell.

IT/P1-19 · Disruption scenarios, their mitigation and operation window in ITER

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Abstract: Representative disruption scenarios in ITER calculated with a numerical code, DINA, based on physics guidelines derived from experimental database analysis, are examined to check the robustness of the design of the vacuum vessel (VV) and in-vessel components, e.g. blanket modules (BM) against various EM loads. Detailed examinations of the newly available experimental data have been performed using the quench time (Δt) between 0.8 and 0.2 of plasma current before disruption. Available data indicate that the minimum value for the quench time normalized by the poloidal cross section area (S) is $\Delta t/S \sim 1$ ms/m^2, which predicts a full current quench time of ∼36 ms with a linear waveform or ∼16 ms of time constant for an exponential waveform in ITER. Both waveforms are used for major disruptions (MD) and up/down-ward VDEs to examine the EM load. It is confirmed that the EM load on the VV and BM are within a design target value, though the margin is not large. Massive noble gas injection is investigated to prevent reduction of the availability caused by large thermal loads during disruptions. Impurity species and its amount are specified to optimize the mitigation capability. Resulting EM loads and the response time to trigger the radiation collapse are important key features in the optimization. Regarding the EM load, the current quench time after the injection must not be too short. A disruption code based on the DINA code has been developed, in which impurity rate equations are simultaneously solved. Calculations with this code show that neon is the most effective impurity since the current quench time could be longer than those at argon injection and unmitigated disruptions by a factor of ∼2. Neon is also superior to argon and heavier impurities regarding the response time. A recent neural network prediction scheme has been applied and it has been found that the success rate can be significantly high (∼80%) with an acceptable pressure on the gas inlet valve.

IT/P1-20 · Beryllium containing plasma interactions with ITER materials

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Abstract: A beryllium-seeded deuterium plasma is used in PISCES-B to investigate mixed-material erosion and redeposition properties of ITER relevant divertor materials. The beryllium containing plasma simulates the erosion of first wall material into the ITER sol plasma and its subsequent flow toward the carbon divertor plates. The experiments are designed to quantify the behavior of plasma created mixed Be/C and Be/W surfaces. Developing an understanding of the mixed material surface behavior is crucial to accurately predicting the tritium accumulation rate within the ITER vacuum vessel. The temporal evolution of the plasma interactions with the various mixed surfaces are examined to better understand the fundamental mechanisms in play at the surface and to allow scaling of these results to the conditions expected in the ITER divertor. A new periodic heat pulse deposition system is also installed on PISCES-B to simulate the transient temperature excursions of surfaces expected to occur in the ITER divertor during ELMs and other off-normal events. These periodically applied heat pulses allow us to study the effects of transient power loading on the formation, stability and tritium content of mixed-material surfaces that are created during the experiments.

IT/P1-21 · Disruption Characterization and Database Activities for ITER

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Abstract: 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, Control and Disruption are described. Accomplishments during 2005–2006 include: 1) formation of an International Disruption Database (IDDB) Working Group, 2) implementation of an MDS plus-based IDDB infrastructure for collection and retrieval of disruption-relevant tokamak data, and 3) collection of a "version-1" data set from seven elongated-plasma-cross-section tokamaks. Database content available as of March 2006 comprises some 3500 disruptions from four tokamaks: Alcator C-Mod, DIII-D, JET and NSTX. Analysis of the current quench data from this provisional data set has allowed the IDDB Working Group to provide an updated recommendation, $t_{CQ}/S = 1.67\,\mathrm{ms/m^2}$, for the lower bound on the expected plasma current decay time in ITER. Here t_{CQ} is the linear-basis current decay time and S is the before-disruption plasma cross-section area. This recommendation, derived from C-Mod, DIII-D, and JET data, applies for plasmas with aspect ratio A in the range $2.5 \leq A \leq 3.5$. Consideration of the low-A (NSTX) data presently in the IDDB shows that a further modification of this formula is required to accurately predict the lower bound on t_{CQ}/S for low-A plasmas. Incorporation of additional low-A data and definitive development of low-A guideline is in progress. Plans for further current quench data analysis and for future expansion of the scope and content of the IDDB to include halo current, thermal energy, runaway electron and 2-D and 3-D plasma equilibrium evolution data have been identified.
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IT/P1-22 · First Mirrors for Diagnostic Systems of ITER

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Abstract: About half of all diagnostics presently foreseen for ITER will use in-vessel mirrors as plasma-viewing components. Mirrors are used for observing the plasma radiation in a very wide wavelength range: from about 1 nm up to a few mm. In the hostile ITER environment, mirrors will be subject to erosion, deposition, particle implantation and other adverse effects which will change their optical characteristics affecting the entire performance of the respective diagnostic systems. The Specialists Working Group (SWG) on first mirrors was established under the wings of the International Tokamak Physics Activity Topical Group on Diagnostics to coordinate and guide the investigations on diagnostic mirrors towards the development of optimal, robust and durable solutions for ITER diagnostic systems. The results of tests of various ITER-candidate mirror materials, performed in Tore-Supra, TEXTOR, DIII-D, TCV, T-10 and LHD under various plasma conditions, an overview of laboratory investigations of mirror performance and mirror cleaning techniques along with the first modeling data on the optical characteristics of mirrors under

plasma exposure will be presented in the paper. The current tasks in the R&D of diagnostic mirrors will be addressed.

IT/P1-23 · Progress in Diagnostic ITER-relevant Technologies at JET

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Abstract: The driving elements in JET diagnostic developments during the last campaigns were the operation in tritium (TTE) and the project of the new ITER like RF antenna. These aspects of JET activity promoted a lot of ITER relevant developments in the field of measurements for the fusion products. Neutron spectrometry was upgraded with the implementation of: (i) a Time of Flight neutron spectrometer, (ii) an upgraded Magnetic Proton Recoil spectrometer, with new scintillator solution to measure also the 2.45 MeV neutrons; (iii) compact NE213 liquid scintillators and diamond detectors. The neutron cameras are also being upgraded with fully digital electronics to improve the potential of this system. New data communication technologies will also allow testing feedback schemes based on the neutron emission for burn control applications. Detection techniques for all the various phases of fast particle life, from the first slowing down to the thermalisation and losses, were implemented and tested. The other major area of investigation at JET on the route to ITER is the one of plasma wall interactions. To assess the problems linked with the erosion and redeposition, a series of innovative detectors potentially applicable to ITER, like the rotating collector, was installed. Significant progress in understanding the dependence of the redeposition on the temperature is expected from the new cooled and heated quartz microbalances. Improved information about power loads on the first wall is derived from a new infrared wide angle camera. The installation of an ITER-like Be wall and a new divertor will result in qualitatively different spectroscopic needs. A significant upgrade of spectroscopic diagnostics, from the visible to X-rays, is ongoing to cope with the future metallic wall. Another important drive for diagnostic upgrades is the need of improved profile determination of several kinetic quantities. A better diagnosis of the electron fluid is obtained with: i) an improvement of the ECE radiometer, to increase its resolution and operational space up to 4 T; ii) the implementation of an innovative sweeping reflectometer, with electronic delay to compensate for the long waveguides; iii) upgrades of the Thomson Scattering diagnostics based on the LIDAR method, to hopefully reach spatial resolutions very close to ITER requirements.

IT/P1-24 · High Priority R&D Topics in Support of ITER Diagnostic Development

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Abstract: The development of diagnostics ITER is a major challenge because of the harsh environment, strict engineering requirements and needs for advanced control. Within the International Tokamak Physics Activity (ITPA), a Topical Group (TG) specialises in diagnostics and aims to support the development and design of the needed systems. Five R&D tasks have been identified as 'high priority' and form the focus of current work: 1) Review the requirements for measurements of the neutron/alpha source profile and assess possible methods of measurement; 2) Support the development of methods to measure the energy and density distribution of confined and escaping alpha-particles; 3) Assess the effects of radiation on coils used for measurements of the plasma equilibrium and support the development of new methods to measure steady state magnetic fields accurately in a nuclear environment; 4) Determine the life-time of plasma facing mirrors used in optical systems; 5) Develop the requirements for measurements of dust, and assess techniques proposed for measurement of dust and erosion.

IT/P1-25 · Progress in Development of Thomson Scattering Systems for ITER

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Abstract: The electron temperature and density are important indicators of the plasma performance as well as key components in transport analyses. The central part of the burning plasma has the highest temperature (up to $T_e \sim 40$ keV for some ITER conditions). Scattering spectra must be obtained in a core region with a LIDAR system for all electron temperatures of interest. Relativistic effects have a large effect on the blue wing of the spectra. Since it is difficult to cope with the full temperature range of an ITER plasma using a single laser wavelength, the spatial profile along the chord may have to be merged from two profile measurements, one for the high temperature core and the other for the "low temperature" edge with use of a dual wavelength laser for plasma probing. If suitable detectors are developed for the

wavelength region 850–1060 nm, all spectral measurements could be done with a single laser wavelength at 1064 nm. In order to meet the demanding specification for spatial resolution in the edge (equivalent to 5 mm at the mid-plane), a conventional Thomson scattering system is planned for the upper edge region. It is proposed that the high-resolution measurements would extend into the plasma, to r/a of ∼0.9. T_e, n_e measurements at the edge along a line-of-sight passing about 50 cm above the X-point in the reference H-mode plasma are to be performed with use of a LIDAR system with a laser beam tilted to the flux surfaces in the space of a significant flux expansion. It is expected that the system will provide measurements with a spatial resolution ∼10 cm along the beam in the required temperature and density ranges with a lower temperature limit of 10 eV. The difficulty for Thomson scattering in the divertor is associated with access and survivability of optical components that have to be placed in the vicinity of this region. Significant effort has been focused on developing a Thomson scattering system for the outer divertor leg with a lower temperature limit of about 1 eV. Resolutions of 5 cm along the leg and 0.3 cm across are foreseen with a repetition rate of 20–50 Hz. The measurement lines up with the flux lines in the divertor leg, which facilitates the establishment of proper divertor operation, especially the location of the ionization front.

IT/P1-26 · Review of Beam Aided Diagnostics for ITER

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Abstract: A reassessment of the proposed package of beam aided diagnostic systems for ITER has led to a number of new developments. As a result of a joint development effort and comprehensive optimisation studies (c.f. [M. Von Hellermann, "Active Charge Exchange Spectroscopy (CXRS) and Beam Emission Spectroscopy (BES+MSE) with Diagnostic Neutral Beam (DNB)" (EFDA contract: 01.649, 2003; not publicly available, contact: mgvh@rijnh.nl)]) two separate CXRS (Charge Exchange Recombination Spectroscopy) and BES (Beam Emission Spectroscopy) systems both based on a dedicated Diagnostic Neutral Beam (H0, 100 keV/amu, 22 A) injected radially into the plasma have been proposed: A 'Core CXRS' system (EU) which covers dominantly the inner half of the confined plasma, and an 'Edge CXRS' system (RF) which ensures a high radial resolution in the outer half of the plasma. The Core system utilises an upper port periscope, and two periscopes located in an equatorial port are envisaged for the Edge CX system. The equatorial port is shared with a dedicated MSE (Motional Stark Effect) system for the measurement of magnetic field pitch angles making use of the Heating Neutral Beams. A complementary MSE approach is proposed for the exploitation of intensity line ratios as measured by the BES data from upper and equatorial ports. The optical layouts of the two systems have been recently reviewed and alignment and instrumentation schemes were developed. A comprehensive multi-parameter spectral modelling effort based on an ITER reference scenario ($n_e(0) = 10^{20}\mathrm{m}^{-3}$, $T_e(0) = T_i(0) = 20$ keV, B = 5.2 T, $Z_{eff} = 1.6$) the actual observation geometry, and modelling of atomic processes, has enabled an update of expected performances in terms of measurement accuracies, time and spatial resolution. The key message is that the measurement of radially resolved profiles of ion temperature, plasma rotation and ion densities, including that of the helium ash, appear to be feasible and, moreover, Signal-To-Noise numbers well above 10 are expected for the main part of the confined plasma, that is for $0.2 < r/a < 1$ for the Upper port and $0.4 < r/a < 1$ for the E-port. Pitch angle measurement based on line ratios in the MSE multiplet spectrum are predicted with error margins below 10% and total magnetic field values deduced from line separations with error of less than 1% for the region $0.2 < r/a < 1$.

IT/P1-27 · Requirements for Fast Particle Measurements on ITER and Candidate Measurements Techniques

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Abstract: Recent results on JET and JT-60U [M. Mantsinen, et al., PPCF 47 (2005) 1439]; [K. Shinohara, et al., PPCF 46 (2004) S31–S45] have underlined the need of looking into the parameter measurement requirements of fast particles diagnostics for ITER. Not only alpha particles present in the plasma,but all fast ions (He^4, p, D, T, He^3): ions used for minority heating, should be diagnosed. The electron diagnostics monitoring the deviations from a Maxwellian distribution function, play a central role in the Electron Cyclotron Heating and Current Drive. In Present measurement requirements only alpha particles and He ions are mentioned: space resolution is $\delta r = a/10$ (a is the minor radius), time resolution is $\delta\tau = 100$ ms, accuracy 10–30%. Proposed measurement requirements for fast particles are: spatial resolution ∼a/20 (10 cm on ITER); time resolution (minimum) ∼100 t Alfvén (∼300 microsec for ITER); density of fast ions: the minority ions could be 4–10% of the plasma density. The candidate diagnostic techniques considered

for ITER are:gamma-ray spectroscopy; Collective Thomson Scattering (CTS); Charge Exchange Recombination Spectroscopy (CXRS). Gamma-ray spectroscopy detects ions in the energy range $1 < E < 5$ MeV, and R&D is needed to demonstrate the feasibility of these measurements in the presence of a neutron background. Techniques using ultraviolet spectroscopy of ions like Krypton are being tested , for the low energy part of spectrum of fast particles. A design study of CTS shows that the requirements are close to be met, the system includes a high power gyrotron operating in a region not yet tested in current applications (i.e. using a gyrotron at a frequency below the first ECE harmonic). For CXRS the spatial resolution and accuracy are not achievable for $r/a < 0.3$, and R&D is needed to assess the minimum figures in terms of accuracy possible for measurements close to the centre.

IT/P1-28 · Studies On The Characteristic Of Titanium-Tritium Reaction

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Abstract: The p-c-T curves and the relation between equilibrium pressure and temperature of tritium absorption and desorption in Ti are measured in the centigrade temperatures of 250, 500, 525 and 550. Equilibrium pressures of the absorption and desorption at different temperature of Titanium are given and then the thermodynamics parameters and the tritium absorption capacity are determined according to Van't Hoff equation. There are different changes of enthalpy and entropy when titanium changing to different phases. The change of enthalpy in the first and second plateau are -101.5 kJ mol^{-1} and -179.6 kJ mol^{-1} respectively and the change of entropy in the first and second plateau are -165.3 J K^{-1} mol^{-1} and -290.3 J K^{-1} mol^{-1} respectively. The lagging effect is not found when titanium desorbing tritium. P-t curves of titanium tritiation were investigated at the centigrade temperatures of 550 to 750 by using the method of the reaction rate analysis in a constant volume system. The rate constants of titanium tritiation reaction are determined and the activation energy value obtained by this analysis is 155.5 kJ mol^{-1}. Desorption P-t curves of titanium tritide were investigated at centigrade temperatures of 350 to 550.The desorbing rate constants of titanium tritide at different temperatures are determined and the activation energy of desorbing is 62.1 kJ mol^{-1}.

IT/P1-29 · Modeling of Edge Control by Ergodic Fields in DIII-D, JET and ITER

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Abstract: The reduction of the heat and particles loads on the Plasma Facing Components (PFCs) due to Type I Edge Localized Modes (ELMs) remains a significant research problem for the international tokamak program. Among the possible approaches of ELMs control is the use of an external perturbing coils to ergodise the edge magnetic fields, thereby increasing the thermal and particle losses in the region of the H-mode barrier and ensuring that the pressure gradient required to trigger Type I ELMs is never attained. A resonant magnetic perturbation (RMP) generated by I-coils in DIII-D has been proven to be able to suppress the type I ELMs reliably without significant degradation of plasma confinement [R. Moyer, et al., Phys. of Plasmas 12, 05611 (2005)]. The paper presents recent modeling results of the Type I ELMs control using stochastic boundaries for the possible future ITER and JET control systems using the realistic experiment in DIII-D as a reference case. A number of possible designs of the external and in-vessel coils ITER are analyzed taking into account physical, technical and space constraints. For typical H-mode scenario and six upper and six lower coils ($n = 3$) configuration $\sim$400 kA current is needed for external coils and $\sim$20 kA for in-vessel coils mounted on the blanket modules in order to generate DIII-D-like magnetic perturbation spectrum in ITER. The heat transport and Type I ELMs suppression modelling in the presence of the external magnetic perturbation was done using non-linear code TELM [M. Becoulet, et al., Nucl. Fusion, 45 (2005) 1284] for DIII-D, JET and ITER. The mechanism of the increased heat transport in ergodic fields is the appearance of a radial component of the heat flux parallel to the field lines, due to the radial component of the magnetic field. The first results based on the non-linear MHD modeling in X-point geometry [G. Huysmans, Plasma Phys. Control Fus. 47 (2005) 2107] taking self-consistent plasma response on the external magnetic perturbation are presented. The most important mechanism of the pressure reduction in the presence of the magnetic perturbation identified is the plasma density transport due to the $E \times B$ drift. It was demonstrated that an amplification of the external magnetic perturbation is possible near the threshold for tearing modes.

IT/P1-30 · Radiation Damage Modeling of Fused Silica in Fusion Systems

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Abstract: Amorphous Silica is one of candidate materials for both final focusing optics of lasers for NIF and future inertial fusion reactors and diagnostics of the Safety and Control Systems of the ITER machine as well as DEMO magnetic fusion reactors. In operation, these materials will be exposed to high neutron irradiation fluxes and it can result in point defect and vary the optical absorption, that is, degradation of the optical properties. In this paper we present molecular dynamic simulation of displacement cascade due to energetic recoils in amorphous silica trying to identify defects formation. We have made a statistics of the different kind of defects at different energy of primary knock-on atoms (PKA). The range of studied PKA energies are from 400 eV to 10 keV and it is made to both component of this material Silicon and Oxygen. We measure how vary the concentration of different kind of defects that appear in the lattice at different recoil energies and we will try to catalogue the defect depending for its final energy potential, its morphology, calculating the coordination of all atoms, etc. These defects can be generated directly by irradiation, as mentioned before, or by the presence of impurities in the material. Hydrogen is a ubiquitous impurity in this material and, moreover in a fusion reactor environment this material will be exposed to energetic hydrogen isotopes. Hydrogen isotopes will be deposited also on the surface of the fused silica components coming from the reaction chamber. On the other hand, some experimental results show that radiation damage can be different depending on hydrogen content, indicating that a detailed knowledge of the hydrogen role in fused silica should be fully understood. In this work we present molecular dynamics simulations to study the effects of different hydrogen isotopes in this material and their interaction with defects. The interatomic potential developed by Feuston and Garofallini will be used in these studies. The diffusion coefficients and mechanisms of H mobility in fused silica will be calculated and compared with those existing in the literature.

IT/P2-1 · System Engineering and ITER Integration of the EU HCPB Test Blanket Module System

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Abstract: The Helium Coolant Pebble Bed Blanket (HCPB) is a concept of solid breeder blanket that is supported by the EU as candidate for the DEMO reactor. This concept will be tested in ITER in to gain information on in-rector behaviour under typical fusion conditions. The Test Blanket Module (TBM) will be located to occupy a half of a horizontal port of ITER; the TBM will be supported by sever auxiliary systems, dedicated to supply the coolant helium, to extract the tritium produced in the solid breeder beds, to purify the losses of tritium in the main coolant and to perform several measurement campaigns. This paper gives an overview on the recent progress on the engineering of this test systems and its integration in the ITER machine. The Testing Programme foresees the testing of 4 TBM during the first 10 years of ITER operation; dedicated experiments will be performed with the TBM systems to study and compare their response in a relevant fusion environment with the analyses in the fields of neutronics, thermo-mechanics of the solid breeder and multiplier and tritium handling. The engineering design of the HCPB TBM systems and its ancillary loops is mostly concluded and the priority is now on the development and qualification of the fabrication technologies. From calculations point of view, the last modelling efforts related to the thermal-hydraulic and the thermal and mechanical resistance in accidental conditions of the first wall are presented. In particular the arrangement of the components in the ITER building is critical; the design has to provide for the integration of several components, instrumentation, control systems in different locations (Vacuum Vessel, Port Cell, TCWS vault, Tritium Building, etc.) and to cope to the remote handling requirements for the Port Plug and Hot Cell. Furthermore, all these systems require mechanical, hydraulic and electrical connections to the supporting structures, the water and power supply lines in ITER. The development of diagnostics and measurement systems is a strategic task to gain the required information from the tested systems. Their design, the compatibility with ITER conditions (electromagnetic, neutronic, thermal compatibility), as well the integration in the TBM design are issues. Finally, safety and licensing analyses necessary to include the TBM systems in the ITER preliminary safety report (RPRS) are discussed.

IT/P2-2 · ITER Shield Blanket Design Activities At SWIP

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Abstract: Design works on ITER shield blanket started at SWIP in the end of 2003 conducted by ITER task agreements. In this context 2 shield blanket modules located in the Neutral Beam (NB) port region were studied. One has a regular shape like the other outboard modules in normal places. Another is a special module besides NB opening, which has two Beryllium protected surfaces in both NB duct and main vacuum vessel subject to NB as well as plasma thermal loads. The interfaces of the modules were defined by ITER International Team (IT). The detailed design of the cooling channels in the modules were carried out at SWIP according to the cooling structure proposed by the ITER IT and specifications in the ITER DDD and SDC-IC. Hydraulic and thermal/stress analysis were also studied at SWIP to optimize the structure.

IT/P2-3 · Hydraulic and Thermal Analysis of ITER Standard NB Blanket Module

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Abstract: Based on the fabrication methods of forging, drilling and welding, the cooling channels in ITER shield block are drilled radial holes with flow drives. In the old design of FDR 2001, the pressure drop in the poloidal hole was very high and it was difficult to achieve uniform flow distribution in the radial holes. In recent years, great improvements in the blanket design were made by ITER international team. Hydraulic and thermal studies on ITER shield blanket module was also carried out by SWIP to assess the hydraulic performance and cooling efficiency, and the flow drives was optimized to achieve "uniform" flow distribution. When some improvements and optimizations were done, the current blanket design was confirmed to satisfy the design requirements according to the results from the analyses.

IT/P2-4 · Tungsten and Beryllium Armour Development for the JET ITER-like Wall Project

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Abstract: The operational behaviour and the interplay of the ITER plasma facing material choice has never been investigated in a tokamak experiment. This motivated the ITER-like Wall Project at JET in which the present main chamber CFC tiles will be exchanged with Be tiles and in parallel a fully tungsten-clad divertor will be prepared. Among the scientific objectives of the ITER-like Wall project are general questions of plasma operation with a low melting Be wall, compatibility of all envisaged ITER scenarios with a W divertor, tritium retention and removal and mixed materials effects, erosion behaviour and lifetime investigations. For the tungsten divertor, two R&D programs were initiated: Forschungszentrum Jülich, Germany, developed a conceptual design for a bulk W horizontal target plate, based on an assembly of tungsten blades. For all other divertor parts five Euratom Fusion Associations performed R&D to provide the technology to coat the 2-directional CFC material used at JET with thin tungsten coatings. In addition, beryllium coatings for the first wall inconel steel are developed. In a first screening, the tungsten coated CFC tiles were subjected to heat loads with power densities ranging from 6 MW/m^2 to 22 MW/m^2 with surface temperatures exceeding 2000°C. In a second step, a selection of coatings was exposed to cyclic heat loading for 200 pulses at 10 MW/m^2 for 5 s corresponding to surface temperatures of about 1600°C. All coatings tested developed cracks perpendicular to the CFC fibres due to the stronger contraction of the coating upon cool-down after the heat pulses. For the bulk tungsten, a design with an an assembly of tungsten blades was developed. To minimise electromagnetic forces the design consists of stacks of tungsten blades of 6 mm width that are insulated in toroidal direction. High heat flux tests of a test module were performed on the electron beam facility JUDITH at a nominal power and duration of (7 MW/m^2, 10 s) for 100 pulses and finally with increasing power loads leading to surface temperatures in excess of 3000°C. No macroscopic failure occurred during the test while SEM showed the development of microcracks at grain boundaries.

IT/P2-5 · Progress towards a Better Be/Cu Joining for ITER First Wall in China

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Abstract: One of the in-vessel components that China will fabricate for ITER is the shield blanket, which consists of first wall (FW) panels and shield blocks. The panel is a joint of Be/Cu/SS, bonded by hot isostatic pressing (HIP). One of the key issues for the bonding is the formation of intermetallic phases that increase the brittleness of the joint. In the present paper, optimization of interlayer materials, HIP parameters and the base metals has been investigated. Much progress has been made towards a good Be/Cu bond by HIPing. Chinese HIP-Be of 98.5% in purity and CuCrZr were used. Interlayer of Ti, Al, AlSiMg alloy and Cu in forms of foil and coating were investigated. HIP was conducted at 500-850°C with pressure of 120–140 MPa. Ultrasonic NDT was performed and the interface was analyzed by optical microscope and SEM observation. The tensile and shear strength was measured at room temperature. Results showed that the joining with interlayer of Ti and AlSiMg alloy are relatively promising. In the case of Ti, the tensile strength was high up to 113 MPa though NDT and SEM observation showed defects at interface. Using pure Cu as the interlayer seemed to be a failure since detachment often occurred. A high heat flux test of the Be/Ti/Cu joint fabricated by HIP at 850°C/140 MPa was conducted. The specimen withstood 1000 thermal cycles at 3.2 MW/m^2 without off-normal temperature change at the Be surface and near the interface. SEM observation showed that the interface was very thick, containing several layers in microstructure. There would be high residual stress at the interface after cooling down from the HIP temperature due to the different thermal expansion of Be and Cu. To improve the quality of the joint, the thickness and the stress should be reduced. A post-HIP annealing was conducted at 425°C for 4 hours. The tensile strength of the joint reached up to 113 MPa from the previous level of 83 MPa. The thickness was reduced by shortening the HIP duration and lowering the HIP temperature. In one case for the HIP at 830°C, duration was reduced to 1 h. The tensile test results should an increase in strength by 31%. In another experiment, the HIP temperature was decreased to 580°C. The shear strength of the joints at room temperature has reached up to 143 MPa. Besides, it was found that using CuCrZr alloy in a softening state could improve the property.

IT/P2-6 · High heat flux tests of small-scale Be/Cu mock-ups for ITER

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Abstract: Two kinds of Be/Cu mock-ups were tested, one is Be/Cu joints with Ti film of 50 mm in thickness as interface at high bonding temperature of 850°C, and the other one with Ti and Cu coating made by physical vapor deposition (PVD) as interlayer at low bonding temperature of 580°C. The bonding strength of the Be/Cu joints is up to 100 Mpa, and non-destructive tests (NDT) indicate that all of the samples have good bonding. All of the tested samples show good thermal fatigue resistance capabilities, they can sustain 1000 cycles without visible failure at absorbed power density of about 2.5 MW/m^2 and 15 s pulse duration. The speed and pressure of cooling water are 0.6 m/s and 0.2–0.4 Mpa, respectively. Maximum surface temperature of the specimen is in the range of 500–600°C and their bulk temperature of Cu plates is in the range of 260–300°C. After thermal fatigue experiments, NDT and morphological observation were carried out. Most parts of the interface of Be/Cu joints kept a good bonding but some local cracks appeared at the interface, in particular in the central region. Maximum width and length of cracks are about 3 mm and 2–3 mm, respectively. Temperature and stress analyses have been performed by ANSYS code and a good agreement between simulated calculations and experimental results was obtained. Since it is lack to measure in-situ the spatial distribution of surface temperature of beryllium tiles and its evolution with respect to time in the present experiment, it is not available to detect the local failure of Be/Cu joints due to the formation of interface cracks, especially at the situation of fine cracks. Furthermore, simulation analysis indicates that the change of surface temperature could be only several degrees at the conditions of fine cracks. Therefore an infrared thermography with high resolving power is strongly recommended. Meanwhile a new heat load facility with higher power and high scanning frequency is required for larger mock-ups. All these are being seriously considered and will be used in the next stage experiments.

IT/P2-7 · Design Concept and Testing Strategy of a Dual Functional Lithium Lead Test Blanket Module for ITER and EAST

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Abstract: Liquid lithium lead (LiPb) breeder blankets, including Quasi-Static Lithium Lead (SLL) breeder blanket and the Dual-cooled Lithium Lead (DLL) breeder blanket, are considered by several parties of ITER program as a design option of DEMO blankets for fusion power reactors due to their potential attractiveness of economy, safety and relatively mature technology base. Considering the confliction between the limited ITER resources for TBM (Test Blanket Module) testing and the requirement of various DEMO blanket concepts proposed by Parties, a Dual Functional Lithium Lead (DFLL) TBM concept has been proposed for testing in ITER to demonstrate the technologies of both SLL and DLL blankets concepts, which emphasizes the balance and the consistence between the risk and the attractiveness of blanket technology development. The DFLL-TBM consists of two specific blanket schemes, namely the DLL and SLL schemes, with as similar as possible basic structure and auxiliary systems except for including FCIs and quicker flowing LiPb as coolant in DLL-TBM to achieve a high outlet temperature. The DLL scheme is the major target of the concept. The SLL scheme may be an optional target if the critical issues (e.g. MHD effects and FCI technology) for the DLL scheme could not be solved. The proposed test strategy will allow consecutively validating SLL/DLL blanket concepts, technologies, and design tools. It covers three phases: materials R&D and small-scale out-of-pile mockup testing in loops, middle-scale TBMs testing in EAST, the superconducting tokamak in China, severing as a pre-testing platform for TBM prior to ITER, because the major parameters of EAST are comparable to those of ITER (for EAST, BT = $3.5 \sim 4$T, D-D neutron rate = $10^{15} \sim 10^{17}$ n/sec, average heat flux on the FW = $0.1 \sim 0.2$ MW/m^2, port size = 0.97×0.53 m^2, plasma pulse length = ~ 1000 sec), and full-scale consecutive TBMs corresponding to different operation phases of ITER during the first 10 years. Design details on TBM concept, such as structure scenarios, auxiliary systems, performances analyses and testing strategy combining TBMs testing, in sequence and in parallel, in EAST and ITER, are presented in this contribution.

IT/P2-8 · Neutronics Analysis for the Test Blanket Modules proposed for EAST and ITER

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Abstract: The Dual-Functional Lithium Lead – Test Blanket Module (DFLL-TBM) system, which is designated to demonstrate the integrated technologies of both He single coolant (SLL) blanket and He-LiPb dual coolant (DLL) blanket, is proposed for test in ITER to check and validate the feasibility of the Chinese LiPb blankets. So far, the construction and operation of ITER will still take a period of ten years, but EAST, the superconducting tokamak device, in China, has been in operation. In EAST D-D phase, the neutron yield is about $10^{15} \sim 10^{17}$ n/s and about $10^{17} \sim 10^{18}$ n/s in ITER D-D phase. Therefore, EAST is expected to serve as a valuable pre-testing platform for TBMs, which is not only for electro-magnetics (EM) and thermo-mechanics but also for neutronics. The neutronics analysis for the TBMs is performed by using the coupled three-dimensional (3D) Monte Carlo – Deterministic code MCSN and the nuclear data library FENDL2.1. The activation calculations will be carried out with the home-developed multi-functional neutronics analysis code system VisualBUS and multi-group data library HENDL. The real 3D neutronics calculation model of the middle-scale (1/3 size-reduced) TBM testing in the EAST super-conducting tokamak and full-scale consecutive TBM testing in the ITER machine have been developed with the Chinese home-developed CAD/MCNP interface code MCAM, which can be used as a converter of large complex 3D CAD models into MCNP models and vice versa as well as an analysis tool of MCNP models by the way of visualization to contribute the QA of neutronics analysis. Neutronics calculations, which include neutron spectra and flux distributions, tritium generation, nuclear energy deposition and D-D phase activation, of the TBMs in EAST are carried out and be made an analogy to those in ITER for the close extent of the neutron yield in D-D phase. Further, the foreseen D-D operations in ITER can be treated as an initial nuclear phase including D-T operation. So the presented nuclear performance estimates for TBM in EAST are important for radiation safety, maintenance and staged ITER construction for it in ITER.

IT/P2-9 · ITER limiters moveable during plasma discharge and optimization of ferromagnetic inserts to minimize toroidal field ripple

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Abstract: Two important design updates have been made in the ITER VV and in-vessel components recently. One is the introduction of limiters moveable during a plasma discharge, and the other is optimization of the ferromagnetic insert configuration to minimize the toroidal field ripple. In the new limiter concept, the limiters are retracted by ∼8 cm during the plasma flat top phase in the divertor configuration. This concept gives important advantages: (i) the particle and heat loads due to disruptions, ELMs and blobs on the limiters will be mitigated approximately by a factor 1.5 or more; (ii) the gap between the plasma and the ICRH antenna can be reduced to improve the coupling of the ICRH power, with the ICRH antenna in a protected position flush with the FW. A flexible support of thin plates is used for the limiter system and there is no sliding support inside the vacuum, to keep the reliability of the system. Driving mechanisms are also located outside the vacuum boundary. During the plasma shutdown phase, the limiters are again moved to the same location as that during the startup phase. The time scale of the movement is ∼5 s. The limiter position and angles can be precisely adjusted between plasma shots. The location and the filling factor of the ferromagnetic inserts have been optimized based on the field ripple calculation. The ferromagnetic inserts are located exactly periodically in the toroidal direction. The ferromagnetic inserts have previously not been planned to be installed in the outboard midplane region between equatorial ports due to irregularity of tangential ports for NB injection. The result is a relatively large ripple (∼1%) in a limited region of the plasma, which nevertheless seems acceptable from the plasma performance viewpoint. However, toroidal field flux lines fluctuate ∼10 mm due to the large ripple in the FW region. To avoid problems due to the large TF flux line fluctuation, additional ferromagnetic inserts are now planned to be installed in the equatorial port region. The additional inserts near the NB ports are not identical with those between regular equatorial ports. Detailed analysis and assessment will be reported in the paper.

IT/P2-10 · Design and Analysis of the ECH Upper Port Plug Structure at ITER

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Abstract: The ECH Upper Port Plug at ITER is designed for controlling plasma instabilities, with a major emphasis on the stabilisation of neoclassical tearing modes, based on the injection a total of 20 MW mm-wave power at 170 GHz into the plasma. The required targeting of flux surfaces will be achieved by angular steering in the poloidal direction. The paper describes the integration of the mm-wave system into the upper port plug structure with a special focus given to the front steering mm-wave launcher variant. The launcher structure consists of the blanket shield module closing the gap in the blanket at the port; the port plug frame which houses the internal shield; the closure plate forming primary vacuum boundary; and the launcher back-end following the closure plate up to the final flange for the door placed for transfer to the hot cells. The design principles for the main structure are discussed with respect to baking conditions, maintenace and remote handling aspects. The shielding structure is essentially formed by the blanket shield module and the internal shield. For these subsystems, the design methodology is presented which includes a specially adapted first wall panel welded to a double-walled housing, dedicated shield blocks formed in encased and/or solid configurations according to space requirements, and the internal shield integrated to the port plug frame. Thermo-mechanical stresses in the first wall panel and the housing of the blanket shield module were analysed by FEM ("ANSYS") calculations with stationary loads for the complete BSM housing and with transient loads for a typical plasma burn of 400 s using a simplified slice structure. The nuclear shielding performance was analysed on the basis of 3D Monte Carlo calculations with the MCNP code for the radiation transport simulation and activation calculations with the FISPACT inventory code. It was shown for a fusion power of 500 MW and an operation over 0.5 full power years that all sufficiency criteria were fulfilled with safety margins of at least 5.

IT/P2-11 · Assessment of the Mechanical Properties of ITER Magnet Insulation Candidate Systems before and after Neutron Irradiation

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Abstract: The ITER design fluence of fast neutrons (E > 0.1 MeV) is 1×10^{22} m^{-2} at the level of the TF magnet insulation. At this fluence conventional glass-fibre/Kapton insulations impregnated with DGEBA epoxy resins as were used for the ITER TF Model Coil (TFMC) are at the edge of disintegration. This was the starting point for engaging in investigations of the mechanical properties of sample materials from magnet manufacturers and alternative materials under static and cyclic load at ambient and low (77 K) temperature. A research program, comprising about 10 systems of conventional and advanced insulation systems, was set up in agreement with specialists of chemical and magnet industry as well as with EFDA and ITER. The sample materials were produced with boron-free glass-fibre tapes and Kapton foil. Plates were VPI impregnated according to specific prescriptions with (a) DGEBA resin systems. (b) cyanate ester AroCy-L10, and (c) with blends of various compositions of AroCy-L10 and DGEBA resins. After cutting the plates into samples suitable for tensile and shear tests, they were irradiated in the Vienna TRIGA reactor and investigated at 77 K in screening tests. The most promising systems were selected for more detailed investigations. The results can be summarised as follows: (a) The DGEBA based insulation systems show a significant, some of them a dramatic, degradation of their mechanical properties after irradiation to the ITER design fluence. (b) Samples impregnated with pure cyanate ester do not show any measurable degradation. However, this material is quite expensive and delicate in its application. (c) Following a suggestion by the resin supplier Huntsman, new series of samples were impregnated with several blends of cyanate ester with DGEBA resins in proportions from 40:60 to 20:80. It turned out that these blended systems also did not show significant degradation after irradiation. As their application is much simpler than that of the pure cyanate ester and because of their reasonable price, these systems are interesting and very promising candidates for the ITER magnet system.

IT/P2-12 · Development of a Simulation Code for ITER Vacuum Flows

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Abstract: The ITERVAC code, newly developed at FZK, has become an essential tool for the proper design of the ITER vacuum pumping systems (NBI, torus exhaust, cryostat) which are characterised by transitional flow conditions. By recent modelling studies performed for the ITER torus exhaust vacuum pumping system several weaknesses could be revealed. In the field of neutral beams, the gas density distribution along and radial to the beamline components could successfully be calculated. Because of the importance attached to the calculation results of ITERVAC, the code must be extensively benchmarked. A theoretical benchmarking was done in the field of laminar and purely molecular flow, however, it has been found that literature data for (intermediate) transitional flow range, which is of major relevance for the ITER conditions, are scarce and even not existing for geometries as complex as for the ITER vacuum pumping ducts. In order to validate proposed design modifications, an experimental confirmation of the modelling results is required. Consequently, a new test rig, denoted TRANSFLOW facility, is described to provide a broad and relevant range of well defined experimental data which can be used to benchmark the code in transitional flow range. The experimental aim is to measure the conductance of different interchangeable test pieces in a broad range of Re, Kn and Ma numbers, and to compare the results against predictions from ITERVAC analysis as well as from other calculations. In this context, a very promising approach is based on kinetic theory which has been successfully used to model vacuum flows in channels of various shapes. The programme will start with simple circular geometries, continue with more complex channels (trapezoidal, triangular) and conclude with a 1: 5 model of the ITER torus pumping duct critical components. The scale-down will be based on dimensionless similarity. This paper will summarize the status of the current ITERVAC simulation work done in the field of NBI and torus exhaust pumping, and highlight the main conclusions which were drawn from them. In the second part, the code will be described to some detail. The final section will focus on the benchmark validation efforts and present first experimental results.

IT/P2-13 · Assessment of current drive efficiency and synergetic effect for ECCD and LHCD in ITER steady state and hybrid scenarios

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Abstract: Current drive efficiencies in ITER steady state and hybrid scenarios have been assessed for the case of ECCD and combined ECCD and LHCD. Analysis is carried out for plasma parameters corresponding to the ITER scenario 4 with plasma current I_p = 9 MA and moderate, $H_{H98y,2} = 1.3$, and high, $H_{H98y,2} = 1.7$, confinement improvement. A possibility of ECCD efficiency improvement due to the synergetic effect in the presence of LHCD is analyzed. The modelling is based on the OGRAY code originally designed for ECCD modelling. Absorption of microwave beams is calculated from the Fokker-Planck equation with the Coulomb collisional term in the linearized weakly relativistic approximation. The code was modified to incorporate the quasilinear diffusion term caused by absorption of LH waves. The Fokker-Planck equation takes into account the existence of trapped electrons as well. The EC beam-tracing procedure is based on extended geometrical optics. The beam-tracing of LH waves and LH wave spectrum at the location of absorption is based on the Fast Ray-Tracing Code combined with the ASTRA 1.5 D transport code calculation. The initial parameters for the beam tracing correspond to the ITER EC and LH launching systems. The calculations are carried out for 33 MW of the tangential NB injection, 20 MW of the LHCD power with a frequency of 5 GHz, and 20 MW of ECCD power with frequency of 170 GHz. According to calculations, the equatorial EC launcher will provide current sufficient for SS operation with $H_{H98y,2} = 1.7$. For the case $H_{H98y,2} = 1.3$, the upper launch of the EC waves will generate current at the location $0.6 < r/a < 0.7$ of LHCD. The assessed synergetic effect is not sufficient to provide SS operation, but hybrid operation with $Q > 5$ and a burn time limited only by the cooling capability (3000 s) is possible. Steady state operation will require further confinement improvement to $H_{H98y,2} \sim 1.5$.

IT/P2-14 · Design of the ITER Electron Cyclotron Wave Launcher for NTM Control

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Abstract: Neoclassical Tearing modes (NTM) are potentially unstable in plasmas with a positive shear, above a certain marginal beta, and ECCD is the method of choice in ITER for NTM control. A parameter for evaluation of NTM stabilisation efficiency is the ratio eta of the driven current density, j_{cd}, to the bootstrap current density, at the mode location. A fit to experimental data provides quantitative criteria that were used for the launcher design evaluation, where sufficient stabilisation is achieved for $\eta > 1.2$. Physics analysis for ITER NTM stabilisation was based on 3 selected H-mode scenarios having considerably different q profiles and bootstrap current. Only 3/2 and 2/1 modes were included in the analysis, that was carried out with beam tracing codes calculating j_{cd} and the P_{EC} to achieve $\eta > 1.2$. An EC launcher for ITER has to satisfy the functional requirements for NTM stabilisation, the strict environmental constraints, as well as ensuring reliability and availability over the whole lifetime of the ITER device. Concerns about the attainable reliability of a "standard" front-steering (FS) launcher (steerable mirror in the ITER vacuum) led to the initial choice of a Remote Steering (RS, all moving parts outside the primary vacuum) launcher concept. Although the RS design work showed that a launcher could be integrated into the ITER environment, the analysis also identified intrinsic limitation in the achievable beam focussing in the plasma (i.e low j_{cd}) for the required steering range ($\sim$25–28°), resulting in j_{cd} insufficient to stabilize NTMs over the designated range of ITER plasma scenarios. On the basis of these results, the design of an FS launcher was resumed, achieving very narrow beams in the plasma for all scenarios and high figures of merit, in excess of requirements for NTM stabilisation. Experience gained in the port plug integration of the RS launcher was transferable to the FS design. Detailed analysis has been performed on the steering mechanism design to demonstrate resilience to EM forces, nuclear heating, radiation damage and cycling fatigue. On this basis, FS has been adopted as the reference concept for the ITER upper launcher. The high performance margins achievable with the FS design may open the way to further exploitation of EC waves for ITER, especially sawtooth control.

IT/P2-15 · The ITER ECH FS Upper Launcher mm-wave Design based on a Synergy Study with the Equatorial Launcher

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Abstract: The ITER ECH heating and current drive system delivers 24 MW at 170 GHz, which can be directed to either the equatorial (EL) or upper (UL) port launching antennas (launchers) depending on the desired physics application. The UL reference design uses a front steering (FS) mirror that sweeps the beam in a poloidal plane providing co-ECCD over the outer third of the plasma cross section. A novel frictionless, backlash-free steering mechanism has been developed for an increased reliability. The design avoids components such as bearings and push-pull rods, which tend to grip in conventional launching systems in use on present ECH systems. Flexure pivots replace of bearings and a pneumatic seal-less actuator using pressurised helium integrated into the rotating mirror assembly offers a fast and precise response avoiding push-pull rods, linkages representing sliding bearings and remote actuators. The result is a complete self-contained frictionless kinematic assembly rotating the steering mirror up to $\pm 7°$ ($\pm 14°$ for RF beam). The launcher has a single dedicated purpose of stabilising the neoclassical tearing modes (NTM), with the launcher steering range accessing the region in which the $q = 3/2$ or 2 flux surfaces are expected to be found for scenarios susceptible to NTM. The performance of the FS launcher far exceeds (by a factor of 1.5 to 3) that required by the physics to stabilize the NTM. The two mirror (focusing and steering) system of the FS launcher essentially decouples the steering and focusing functions of the FS launcher, which offers the flexibility to increase the access range beyond that required by the NTM stabilization such that the launcher can access further inward for sawteeth control. Extending the range of the UL can relax the EL steering range, and optimize the launcher for enhanced physics performance with an optimised central deposition and potential for counter ECCD.

IT/P2-16 · Pellet Fueling Technology Development for Efficient Fueling of Burning Plasmas in ITER

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Abstract: Pellet injection is the primary fueling technique for efficient core plasma fueling of ITER burning plasmas, which is necessary for achieving high fusion gain. Injection of pellets from the inner wall has been shown on present day tokamaks to provide efficient fueling and is planned for use on ITER. Here we present progress on the development of the pellet technology to provide reliable high throughput inner wall fueling, the validation of physics models for pellet ablatant drift, and apply the results of these studies to ITER burning plasmas. Tests of a mockup of the ITER inner wall injection line indicate that 300 m/s pellets will remain intact under realistic operating conditions. A high throughput extruder is under development that can supply the 0.3 g/s ($\sim$1.5 cm^3/s solid DT) needed by ITER for pellet formation. Acceleration to the relatively slow pellet speeds can be accomplished easily with a centrifuge or light gas gun with low propellant gas throughput. Modeling of the projected mass deposition with the Pressure Relaxation Lagrangian model has been preformed for the ITER burning plasma pellet fueling scenario with realistic pellet sizes and speeds dictated by the guide tube tests. The modeling shows that the pellet mass can reach well beyond the pedestal region. Scaling of the ablatant mass drift in ITER from the modeling has strong pellet size and pedestal temperature dependence. Pellet injection results from many present day tokomaks indicates that frequent small pellets may make an attractive ELM mitigation scheme. The ITER pellet injection system will have the flexibility to be employed as an ELM mitigation tool if needed.

IT/P2-17 · ORE Assessment in ITER: a Proposal for the Methodology Approach and an Example of Application

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Abstract: A licensing process for the proposed ITER machine is currently on going in France. Among the more complex items to be addressed, from the safety point of view, is the assessment of the occupational radiation exposure. Its complexity is related to the fact that the functional systems, including: plasma heating, machine cooling, power supply and plasma diagnostics, are prototypes. Component design is on going and design integration will occur only after completion of the design. Moreover, maintenance procedures will be completely defined only after some years of machine operation. In the meantime, estimation of occupational dose and continuous checks of the application of the ALARA process have to be done for the licensing of the plant. In this paper a systematic approach for the ITER ORE assessment is

presented. The ORE-Code software tool has been developed in Excel spreadsheet format to facilitate data handling and updating. To date, this systematic approach has been applied to the ITER port interfacing systems (NBI, ECH&CD, ICH&CD, LHH&CD, Diagnostics, TBM).

IF

Inertial Fusion Experiments and Theory

IF/1-1 · Compression and Fast Heating of Liquid Deuterium Targets in FIREX Program

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Abstract: The purpose of the first phase of the FIREX (Fast Ignition Realization Experiment) program is to demonstrate fast heating of a fusion fuel up to the ignition temperature of 5–10 keV. Such high temperature never be achieved with deuterated plastic targets used in the previous experiments because of the extremely high radiation loss associated with high concentration of carbon ions. Using liquid deuterium targets is the obvious solution to significantly lower the radiation loss. Compression and heating characteristics of deuterium targets are under active investigation. Preheating temperature of the deuterium target was found to be about or less than the Fermi temperature, implying that the energy required for the compression is close to the lowest energy required for compressing a perfectly degenerated Fermi gas. We have also performed fast heating of planar deuterium targets for the first time. It has been found that hot electrons are transported through a plastic barrier layer and deposited their energy in the deuterium target, generating a significant amount of DD neutrons.

IF/1-2Ra · Studies of electron and proton isochoric heating for fast ignition

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Abstract: Isochoric heating of inertially confined fusion plasmas by laser driven MeV electrons or protons is an area of great topical interest in the inertial confinement fusion community, particularly with respect to the fast ignition (FI) proposal to use this technique to initiate burn in a fusion capsule [M. Tabak et al., Phys. of Plasmas 1, 1626 (1994)]. Experiments designed to investigate electron isochoric heating have measured heating in two limiting cases of interest to fast ignition, small planar foils and hollow cones. Data from Cu K_α fluorescence, crystal X-ray spectroscopy of Cu K shell emission, and XUV imaging at 68 eV and 256 eV are used to test PIC and Hybrid PIC modeling of the interaction. Isochoric heating by focused proton beams generated at the concave inside surface of a hemi-shell and from a sub hemi-shell inside a cone have been studied with the same diagnostic methods plus imaging of proton induced Kalpha. Conversion efficiency to protons has also been measured and modeled. Conclusions from the proton and electron heating experiments will be presented. Recent advances in modeling electron transport and innovative target designs for reducing igniter energy and increasing gain curves will also be discussed.
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IF/1-2Rb · Plasma Photonic Devices for Fast Ignition Concept in Laser Fusion Research

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Abstract: One of the most attractive paths to fusion energy is now being explored in the fast ignition scheme of the laser fusion using a plasma photonic device such as a re-entrant cone. In the fast ignition scheme, ultra intense laser light as an "ignitor" pulse with an energy of tens of kilo Joules must be efficiently deposited to the imploded core plasma, which is one of the most critical issue in this concept. The plasma photonic device such as a hollow cone can efficiently guide the ultra-intense laser light into the core avoiding nonlinear interaction processes of the heating laser pulse in long scale-length plasmas surrounding the compressed core plasma. The heating pulse only interacts with high density plasmas. The cone plasma device plays two interesting roles on efficiently heating of compressed plasmas. One is guiding of laser light into a small tip as manner of focusing optics. Another is collimation of energetic electrons in the cone geometry like electron beam collimation optics. These roles of the cone plasma device were experimentally cleared on the fast heating process.

IF/1-2Rc · Relativistic Electron Generation and Its Behaviors Relevant to Fast Ignition

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Abstract: Our integrated fast ignition experiment showed a significant success using a gold cone guide inserted CD target showing more than 103 neutron increase with a PW laser fast heating. This heating indicated that hot electrons were coupled to the highly compressed core with more than 30% efficiency.

The energy transport of the hot electrons is under intensive study in order to understand the function of hot electron behavior within the cone, electrostatic potential formation and its affect on hot electrons as well as the mechanisms of the electron energy deposition in a high density low Z fuel plasmas. We address these key physical issues especially that (1) the relativistic electron created on the inner surface of cone could be actually guided along the cone surface with more than 30 MGauss self-generated B field and (2) the potential formation around the target can be controlled by the return current duration. These understanding will lead to an accurate target design of fast ignition integrated experiment planned within a few years with multi-kJ PW laser systems.

IF/1-3 · Radiating Z-pinch Investigation and "Baikal" Project for ICF

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Abstract: The implosion of radiating Z-pinches is considered as one of methods of ICF target ignition. For development of installation with parameters close to an ignition the cooperation from Efremof Institute, TRINITI, Kurchatov Institute and VNIITF investigate a X-rays generation in Z-pinches and developes the project of generator with a current more than 50 MA. The results of experiments on multiwire arrays implosion on installation "Angara-5-1" are described for a current up to 4 MA and impulse of a X-rays power ~7 TW. The X-ray radiography and the data of a X-ray emission of plasma Z of a pinch together with miniature magnetic probes give informations on a current and mass distribution in a pinch during implosion. The data on density and magnetic fields distribution are used to verify two-dimensional numerical RHMD codes taking into account the model of the plasma production at wires. The basic points of a powerful current generator of "Baikal" developed for application as the driver for experiments on an ignition of targets for ICF are described. The experimental results of investigation the installation "MOL" is described. "MOL" is testbed for testing in actual size basic elements of the module of the generator of "Baikal".

IF/P5-1 · Development of the Foam Cryogenic Target for the FIREX Project

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Abstract: The Fast Ignition Realization Experiment (FIREX) project was launched at the Institute of Laser Engineering (ILE), Osaka University. Key technologies are developments of high-energy lasers and solid fuel targets, also called a cryogenic target. ILE and the National Institute for Fusion Science (NIFS) have started a cooperative study on the target. It has a unique appearance designed for the project. To realize its specification, a foam shell method is applied as fuel layering technique. Subjects of the target development are fabrication of a foam shell, assembling the foam shell and other parts to form the target and fuel layering. In this paper, accomplishments and some remaining issues of the target development are described.

IF/P5-2 · Development of 10-kJ PW Laser for the FIREX-I Program

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Abstract: In laser fusion research, the concept of fast ignition by an ultra-intense laser holds great interest because high gain is feasible with relatively small driver energy. On the basis of experimental demonstrations of fast heating of deuterated plastic plasma by a PW laser, we have started phase I of the FIREX (Fast Ignition Realization Experiment) program, in which fuel imploded using the existing GEKKO XII will be heated by a high-energy PW laser. The missions for FIREX-I are achievement of a heating temperature of 5–10 keV and the development of laser technology for phase II of FIREX. The high-energy PW laser under construction for FIREX-I (LFEX: Laser for Fast Ignition Experiment) is a Nd:glass laser aiming for a 10 kJ output energy. To achieve the 5–10 keV temperature by overcoming the energy loss due to plasma expansion, the required pulse shape is a 10 ps trapezoid with a rise time less then 2 ps. The focal spot size should be smaller than 30 μm with an encircled energy larger than 50%. In accord with these specifications, the system design has been undertaken in terms of a multi-pass amplifier for efficient energy extraction, operational conditions of chirped pulse amplification (CPA), which meets a realistic pulse compressor size, and a minimized wavefront distortion.

IF/P5-3 · Hyper-velocity Acceleration of Foil Targets for Impact Fast Ignition

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Abstract: A new ignition scheme, impact fast ignition (IFI), is studied, in which the compressed DT main fuel is to be ignited by impact collision of another fraction of separately imploded DT fuel, which is accelerated in the hollow conical target. As the first step toward the proof-of-principle of IFI, we have conducted preliminary experiments under the operation of GEKKO XII/HYPER laser system to achieve a hyper-velocity of the order of 10^8 cm/s. As a result we have observed a highest velocity, 6.5×10^7 cm/s, ever achieved. Two-dimensional hydrodynamic simulation results in full geometry in use of plastic instead of DT fuel are presented, in which some key physical parameters for the impact shell dynamics such as 10^8 cm/s of the implosion velocity, 300–400 g/cm^3 of the compressed density, and the converted temperature beyond 5 keV are demonstrated.

IF/P5-4 · Fast Ignition Integrated Interconnecting Code (FI3) – Integrated Simulation and Element Physics

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Abstract: The fast ignition (FI) scheme is one of the most fascinating and feasible ignition schemes for the inertial fusion energy. The numerical simulation plays an important role in studying detail mechanism of fast ignition, demonstrating the performance, designing the targets, and optimizing laser pulse shapes for the scheme. In order to study the physics of FI, we have developed "Fast Ignition Integrated Interconnecting code" (FI3). In the result of the latest integrated simulation by FI3, we find that density gap between cone tip and core plasma is important for the heating efficiency from hot electron to core plasma. We perform individual simulation of each element process also, and some important progresses are obtained. In the 2D simulation of laser plasma interaction by collective PIC code, we find that the electrons accelerated at the oblique inner surface of the cone cause three Maxwellian distributions in energy spectrum, which effects to the heating efficiency. About the formation of the high density fuel core plasma, we simulate the 2D non-spherical implosion using radiation hydrodynamic code. In result, we find non-spherical implosion is robust over hydrodynamic instability which is critical phenomena for spherical implosion. These two latest results are favorable facts for FI, and they will be reflected to the FI3 simulation as feedback.

IF/P5-5 · Recent results on fast ignition jet impact scheme

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Abstract: We present recent developments in the design of the fast ignition jet impact concept [Velarde, 2004, 1005]. This scheme, unlike the previous designs proposed so far, need only one driver for the whole process to ignition. In the jet impact case, the ignition of the compressed core is produced by hypervelocity jets [Zababakhin 1990, Velarde 1997] generated during the process. The collision of jets converts their kinetic energy into thermal energy of the nuclear fuel, which is expected to produce ignition under proper design. Recently we have improved the design, increasing the efficiency of the jet production process and we have explained theoretically the production of jets with the small angle liner cones, seen in some numerical simulations. Now we use low Z materials for the jets, with better properties for the interaction between the jet and the compressed core.

IF/P5-6 · Reactor-scaled cryogenic target formation: mathematical modeling and experimental results

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Abstract: In filling, layering and delivery research of inertial fusion targets, attention now focuses on the elaboration of efficient and reliable methods of large target production. To achieve the implosion conditions, the fusion fuel contained inside a hollow polymer shell must be formed as a highly uniform and smooth solid layer. Furthermore, the fusion fuel must have an isotropic structure to withstand the environmental effects: excess of heat and mechanical load during target delivery to the target chamber. To meet these requirements, an approach to fuel layering based on conduction cooling of a batch of moving spherical targets has been developed at the Lebedev Physical Institute (LPI). The approach demands to use free-standing targets (FST) in each production step, and the targets must move in the layering

module to bring about a random walk of a thermal contact area onto the outer shell surface. Currently, a set of new FST-layering investigations is underway to elucidate the physical fundamentals underlying the efficiency of the FST-layering method in the case of reactor-scaled targets. Thereto, a special rotating and bouncing (R&B) cell was proposed and examined. Constructively, the R&B cell is placed into the cryostat. A vibrating membrane is an integral part of the cell. The couple "membrane & target" is driven by an input signal generated due to either alternative magnetic field or inverse piezoelectric effect. The R&B cell operates at cryogenic temperatures, which are controlled within the rates of 0.001–60 K/min. If we hold fixed temperature, it is controlled in to ± 0.05 K. Modulation of the input signal impresses information on the carrier frequency and amplitude. This allows placing the target in trajectory with a given mode: rotating (R-mode), bouncing (B-mode) or mixing (R&B-mode). In this report, we discuss the results of mathematical modeling and fabrication advantages of the FST layering technique using the R&B cell as a layering module. Special attention will be paid to: 1) FST-theory (description of the heat transfer and layer symmetrization mechanisms); 2) Experimental study (optimization of operational factors.)

IF/P5-7 · Advances in Target Design and Materials Physics for IFE at DENIM

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Abstract: New analysis is presented for a neutronic p-B11 in degenerate plasmas with ignition at 20 keV and densities of 10^{31} electrons/cm^3 (30 Mg/cm^3). We study proton beamlets of very high intensity to trigger the fusion onset, but extremely high currents are needed in few picoseconds. Moderately high compressions in classical plasmas offer an interesting design in stagnation-free implosion with jets heating. New scheme for fast ignition has been proposed, using matter accelerated to high velocity for heating the compressed DT fuel, avoiding the use of additional high intensity laser to produce ignition. ARWEN package has been improved with modules of new EOS fitting package and a NLTE package that is still under testing on the multigroup radiation diffusion. Line photon transport in non-homogeneous plasma using radiative coupling coefficients is implemented. The implementation of sparse matrices techniques and iterative solvers in the calculation of level populations for NLTE plasmas has been concluded. The influence of excited configurations and plasma interaction in NLTE plasmas is studied with analytical approaches and the radiative opacity of laser-produced plasmas using a relativistic-screened hydrogenic model for ions including plasma effects has been calculated. Improvements of our computational methodology ACAB have been developed for a reliable prediction of activation responses. ACAB system now is likely the world-wide most powerful tool to deal with cross section uncertainties in activation analysis. Applications of ACAB system have been focused on IFE, but also in IFMIF, MFE and even waste transmutation systems. The last two applications have been very useful for IFE, since activation XS needs in those fields are also closely related to some of the IFE needs. Defects diffusion and microstructure in irradiated hcp materials and pure Fe have been studied with new Kinetic MonteCarlo model. Dislocation Dynamics has been used to study interaction of defects and partial dislocation in fcc metals. Primary damage and displacement cascades analysis has been developed to describe defects in SiO_2 in fusion reactors. We have developed a methodology that includes tritium diffusion and deposition processes in the soil and vegetables, penetration in the underground, re-emission and later conversion to organic tritium to simulate the behaviour of different chemical forms.

IF/P5-8 · Modelling the Nonlinear Saturation of the Parametric Instabilities Generated by Laser-Plasma Interaction

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Abstract: This paper is concerned with the fluid-type modelling of the nonlinear evolution of the SBS (Stimulated Brillouin Scattering) and SRS (Stimulated Raman Scattering) instabilities that can take place in an underdense plasma illuminated by a powerful laser beam. Concerning SBS, we have designed a new multi-dimensional numerical code consisting in the coupling of several modules: (i) the transverse waves are decomposed into the incident and scattered ones, each of them being described within the paraxial approximation; a so-called harmonic decomposition is made, in which the fast spatial variation of the ion acoustic wave (IAW) is factorized analytically, so that the low frequency response of the plasma is decomposed into (ii) the hydrodynamics equations describing the plasma global motion, and (iii) envelope equations describing the IAW evolution within the paraxial approximation. The emission of harmonics and of sub-harmonics of the fundamental IAW component generated by SBS is taken into account, and the kinetic effects are modelled by nonlinear frequency shifts in the envelope equations describing the

various IAW components. We have carried out extensive simulations using a hybrid code (kinetic ions described in the PIC limit, fluid electrons) on one hand, and our fluid-type code on the other hand. The comparison between their results will be discussed, and scaling laws of the SBS reflectivity will be presented. Concerning SRS, we have designed another multidimensional numerical code in which the transverse waves are described within the paraxial approximation and the SRS excited Langmuir wave is coupled to the IAWs as in the Zakharov equations; (iii) the overall plasma density profile can be modified by the transverse waves ponderomotive force giving rise to self-focusing. Special attention has been given to the boundary conditions in order to be able to describe the case of a linear density profile corresponding to spatial amplification limited by the Rosenbluth gain factor. We have also carried out extensive simulations using a full PIC code in order to determine the validity of these fluid-type descriptions, in the cases of a homogeneous and inhomogeneous plasmas. Comparisons between the results of these various codes will be discussed, and scaling laws of the SRS reflectivity will be presented.

IF/P5-9 · Effects of Laser Radiation, Nanostructured Porous Lining and Electric Field on the Control of Rayleigh-Taylor Instability at the Ablative Surface of the IFE Target

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Abstract: This paper presents the effects of electric field, laser radiation and porous lining of nanostructure smart material on electrohydrodynamic Rayleigh-Taylor instability (ERTI) at an ablative surface of thin IFE target shell using linear stability analysis. A simple theory based on electrohydrodynamic approximations is used. Also, the assumption on densely packed porous lining made up of smart material of nanostructure is needed to maintain laminar flow using the Darcy equation with slip condition. The growth rate of RTI including the effect of laser radiation is derived analytically. The cutoff and maximum wavenumbers and the corresponding maximum frequency are obtained. We found that a combined effect of the electric field and the porous lining reduces the growth rate of ERTI considerably compared to the classical growth rate.

IF/P5-10 · Investigation of characteristics of laser source of ions from the targets of different densities

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Abstract: It is well-known that a heavy ion fusion (HIF) scenario employs very intense heavy ion beams, which can deliver energy of several megajoules to a fusion target during a time of about 10 ns [N.A. Tahir et al., Laser and Particles Beams 22, 485-497 (2004)]. One of the main problems facing HIF is to find a source of ions having characteristics suitable for ICF [M. Ogawa et al., Laser and Particles Beams 21, 633-638 (2003)]. The laser source of ions can provide the highest intensity of multiply charged ions for injection into practically any accelerators. For the practical use of these sources it is desirable to have high momentum without reduction in intensity and charge of ions. One of the ways to increase the momentum of ions is to use multi-element targets [R.T. Khaydarov et al., Laser and Particles Beams 23, 521-526 (2005)]. In this work mass-charge and energy spectra of multiply charged ions of plasma, generated from targets of different density under the action of laser radiation are investigated. Experiments were carried out in a laser mass-spectrometer with Neodymium glass laser. Ho_2O_3 targets with diameter 1.0 cm and thickness 0.5 cm are used in the experiment, which have density $\rho_0 = 1.2$ g/cm^3 (initial condition-powder), $\rho_1 = 1.4$ g/cm^3, $\rho_2 = 2.8$ g/cm^3, $\rho_3 = 3.2$ g/cm^3, $\rho_4 = 3.5$ g/cm^3, $\rho_5 = 3.7$ g/cm^3. Experimental results have shown that at low energy part of the spectra maximal charge for oxygen ions is reached at low densities (ρ_1, ρ_2), while maximal charge of Ho ions is obtained at higher target densities (ρ_3). This effect is the results of non-equilibrium ionization processes in the plasma due to the changing of the volume, which absorbs laser radiation. Physical processes at the interaction of laser radiation with two-element targets of different density and leading to the change of charge and energy spectra of ions are discussed.

IF/P5-11 · Heavy-Ion-Fusion-Science: Summary of U.S. Progress

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Abstract: Over the past two years noteworthy experimental and theoretical progress has been made towards the top-level scientific question for the U.S. program in Heavy Ion Fusion Science and High Energy Density Physics: “How can heavy ion beams be compressed to the high intensity required to create high energy density matter and fusion conditions?”. New results in transverse and longitudinal beam compression, beam-target interaction, high-brightness transport, and beam production, as well as a new scheme for beam acceleration, will be reported. Central to this campaign is final beam compression. With a neutralizing plasma, we demonstrated transverse beam compression by an areal factor of 200, and longitudinal compression by a factor of 50. Low energy ion beams of a few MeV are shown to produce highly uniform temperature profiles essential for the study of warm dense matter. High beam brightness is key to high intensity on target, and detailed experimental and theoretical studies on the effect of secondary electrons on beam brightness are reported. We have also demonstrated a compact high-brightness ion injector suitable for heavy ion fusion. Finally, a new accelerator concept for near-term low-cost target heating experiments was invented, and the predicted beam dynamics validated experimentally. For all components of our high intensity campaign, the new results have been obtained via tightly coupled efforts in experiments, simulations, and theory.

IF/P5-12 · Fast Z-Pinch Experiments at the Kurchatov Institute Aimed at the Inertial Fusion Energy

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Abstract: Fast implosion of Z-pinches is considered as possible way to the generation of X-ray pulse on the level of some dozens MJ aimed at the Inertial Fusion Energy (IFE) Program. The series of IFE experiments is carried out on the S-300 high-current generator (3 MA, 100 ns, 0.15 Ohm) at the Kurchatov Institute. In this talk, the experimental results on the S-300 machine are presented, related to the study of operation of co-axial magnetically self-insulated transporting line, by the linear current flow density on the inner electrode surface up to 7 MA/cm. The specific parameters of this current-carrying line fairly correspond to those of the Sandia Laboratories conceptual project of IFE reactor based on the fast Z-pinch. Another experimental series was carried out to study the magnetized Hohlraum, which is the case typical of the next generation of pulsed power machines aimed at the IFE target ignition. The radiative temperature of the inner wall of Hohlraum has been obtained on the S-300 machine as high as 40–50 eV. The theoretical model has been elaborated that predicts the same effect in the multi-megaampere experiments, on the level exceeding 200 eV.

IF/P5-13 · Linear analysis of sheared flow profiles and compressibility effects on Z-pinch MRT instability

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Abstract: A linear analysis of the ideal magnetohydrodynamic (MHD) stability of the compressible Z-pinch plasma with axial flow is presented. The effects of magnetic field and compressibility, which can not be detected in an incompressible model, are investigated. Results indicate that, in the early stage of the implosion the compressible model is much more suitable than the incompressible one. With the cooperation of sheared axial flow, magnetic field, and plasma compressibility, the stability of the Z-pinch plasma is improved markably. The analysis of different axial flow profiles shows that, the mitigation effect of the axial flow on the MRT instability is caused by the radial velocity shear, and it is highly susceptible to the shear value nearby the plasma outer surface. By adjusting the flow profile, the mitigation performance can be promoted evidently.

IF/P5-14 · Fast Ingnition by Laser Driven Particle Beams of Very High Intensity

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Abstract: Since laser pulses in the order of picosecond duration and powers of TW up to few PW are available, the fast ignition (FI) of DT fuel for controlled fusion is studied extensively. Initially only the interaction of laser beams was considered [M. Tabak, et al., Phys. Plasma 1, 1626 (1994)] but modifications followed for using laser produced intense proton beams for fast ignition (PFI) [M. Roth, et al., Phys. Rev. Letters, 86, 436 (2001)] irradiating pre-compressed DT of about 1000 times the solid state. Using 10 TW-ps laser pulses should produce very high intensity fife MeV electron beams igniting in nearly uncompressed solid DT of larger volume controlled fusion reactions with gains above 10^4 [3]. We present here experimental and theoretical results how laser driven very high intensity DT ion beams [4] to be used for strongly improving the PFI scheme [M. Roth, et al., Phys. Rev. Letters, 86, 436 (2001)] and may be applicable also to ion beam ignition similar to electron beam ignition [J.H. Nuckolls et al., The future of Inertial Confinement Fusion, LLNL Report UCRL-JC-149860 (2002)]. We report on several new results we have gained with “clean” laser pulses where relativistic selffocusing is suppressed by using contrast ratios

higher than 10^8 reaching ion current densities exceeding $10^{10} \sim$ Amp/cm^2 producing plasma blocks (or pistons) by a skin layers acceleration dur to nonlinear (ponderomotive) forces. The 1000 times higher DT current densities with ballistic plasma blocks of increased thickness may be sufficient to verify the PFI scheme of Roth et al. [M. Roth, et al., Phys. Rev. Letters, 86, 436 (2001)]. The resulting subrelativistic plasma blocks may represent an ion beam ignition [H. Hora., J. Badziak, et al., Laser and Particle Beams, 23, 423 (2005)] similar to the electron beam ignition concluded by Nuckolls et al. [J.H. Nuckolls et al., The future of Inertial Confinement Fusion, LLNL Report UCRL-JC-149860 (2002)].

IC

Innovative Confinement Concepts

IC/P7-1 · Project Epsilon – the Way to Steady State High β Fusion Reactor

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Abstract: Present day level of hot plasma magnetic confinement physics understanding have let a possibility to formulate magnetic field geometry quality specifications, which could provide, in principle, realization of all the magnetic confinement potential. It is possible to realize high b hot plasma confinement at steady state operational mode with a "tokamak" confinement time level. That permits, in particular, to increase magnetic field use efficiency and operational flexibility or to examine a possibility of alternative fuel cycles application. The quality specifications are based on magnetic field module level lines topography (isomagnetics lines) at equilibrium magnetic surfaces. Analysis shows that it is possible to eliminate superbanana drift trajectories, which defined a deterioration of magnetic system confinement properties if one provide a longitudinal magnetic invariant level lines closure around magnetic or torus axis. There are two types of pseudosymmetry: poloidal and toroidal ones. Only the poloidal one provides high beta operation. It is very important to note that the two types of pseudosymmetry are topologically incompatible. It is proposed to check that principle in frame of project EPSILON (Experimental Pseudosymmetrical Trap) – closed rippled trap without rotational transform and with poloidal pseudosymmetry. It is combination of few axially symmetrical mirror traps (two as a minimum) closed by curved connectors. MHD stability and plasma cleaning will be provided with symmetrical toroidal divertors displaced along the mirror traps with zero magnetic field at separatrix. Reactor perspectives of the EPSILON are preliminary examined.

IC/P7-2 · Experimental Studies in a Gas Embedded Z-pinch Operating at Mega Amperes Currents

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Abstract: A gas embedded Z-pinch has been implemented using the SPEED2 generator (4.1 μF equivalent Marx generator capacity, 300 kV, 4 MA in short circuit, 187 kJ, 400 ns rise time, dI/dt $\sim 10^{13}$ A/s). Initial conditions to produce a gas embedded Z-pinch suitable to be driven by the SPEED2 and with enhanced stability by means of resistive effects and by finite Larmor radius effects were obtained using a 0-D model. Thus, electrodes were constructed in order to obtain a double column Z-pinch and a hollow discharge. Experiments were carried out in deuterium at mega amperes currents. The diagnostics used are: current derivative and voltage signals, neutron detections using silver activation counters, and He3 detectors; scintillators with photomultilier; and interferograms using a pulse Nd-YAG laser (8 ns FWMH at 532 nm). A plasma column apparently stable is obtained and neutrons have been detected.
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IC/P7-3 · Measurements of Rotational Velocity Shear and Interchange Stabilization in the Maryland Centrifugal Experiment

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Abstract: The Maryland Centrifugal Experiment (MCX) produces supersonically rotating plasmas in a mirror geometry with a radial electric field produced by a coaxial core biased at high voltage. MCX has achieved high density ($n > 10^{20}$ m^{-3}) fully ionized plasmas rotating supersonically with velocities of $\sim$100 km/sec for times exceeding 8 ms under a wide range of conditions. Ion temperatures are 30 eV and confinement times $\sim$100 microseconds. Sonic mach numbers are 1–2 and Alfvén mach numbers somewhat less than 0.5 for standard discharges. MCX has also demonstrated an enhanced mode of operation with sonic mach numbers greater than 3, confinement times of several hundred microseconds and Alfvén mach numbers near one. Plasmas remain grossly stable, or steady, for many milliseconds, much longer than MHD instability timescales for MCX, though significant magnetic fluctuations are measured with magnetic probes. Measurements of rotational velocity profiles are made employing a five channel high resolution spectroscopy system, viewing across the plasma. Profiles for impurity emission lines and neutral Hydrogen are measured. Abel-like inversions of the emissions from the five chords clearly show that the profile of angular velocity is parabolic type. The peak and average velocities are fully consistent with the direction and the magnitude of the rotation velocities inferred from the voltage and B field. Rotational velocities for different charge states (C^+ and C^{++}) agree within error bars, which should be the case since the $E \times B$ drift dominates other drifts. The velocities measured for H_α are lower than those for the charged states, as expected. The inferred emissivity radial profiles all show a hollow structure with the plasma interior substantially depleted. Most importantly the shear in the rotation profile is calculated. In a

database consisting of both O and HR modes, the HR mode shear exceeds the theoretically required shear over most of the radial profile and exceeds the threshold by up to factors of 5. For the O mode, the observed shear is of the same order as and somewhat exceeds the theoretical threshold. These observations are essential in establishing the physics of velocity shear stabilization of MHD interchanges in rotating plasmas.

IC/P7-4 · Self Creation of Toroidal Field as a Merging of Conventional Field Systems at the Coilless STPC-M for Convenient ST Design

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Abstract: The aim of this study is to identify the physical bases of an alternative self organization mechanism that exist on the coilless STPC-M machine. In mentioned device, the necessary toroidal magnetic field is generated simultaneously by both the plasma center post (PCP) and the external center post (ECP) methods. In this context; At the PCP mode, the average toroidal field and the standard deviation (SD) are 206.54 G and 49.09 G respectively. The average value of displacement on the radius of ST at this phase is 13.50 cm and SD has been calculated 1.86 cm. This value has not effected the axial merging synchronization. Considering the priliminary experimental data taken, evaluating the STPC-M machine, it is understood that the part of 55% of needed toroidal field for instance 378.37 G of 206.54 G can be obtained by PCP method.In this case hybrid method of PCP+ECP can be accepted more preferable with respect to the conventional one.

IC/P7-5 · Cross-Field Resitivity Scaling With Density and Temperature For Steady-State FRCs Under Rotating Magnetic Field Current Drive

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Abstract: FRCs can be sustained in steady state by Rotating Magnetic Fields at densities relevant to fusion reactors. The obtainable density scales linearly with the RMF magnitude B_ω and inversely with the plasma resistivity. The RMF power required to sustain the FRC current and flux is also directly proportional to the total toroidal current and the average resistivity. It has been found in TCS experiments that as the plasma temperature is increased by using anti-symmetric RMF drive, or by reducing the impurity level, that despite higher currents and magnetic fields, the RMF power required to maintain the configuration does not increase significantly. This is due to most of the added diamagnetic current being carried in a low resistivity core near the FRC field null, and also to a resistivity that appears to scale as 1/B. Past experiments have been limited in temperature by high impurity levels and low Z radiation barriers, and experiments at higher temperatures will be reported in a new device, TCS/upgrade, significantly upgraded to produce clean plasmas.

IC/P7-6 · Spheromak Formation by Steady Inductive Helicity Injection

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Abstract: A spheromak is formed, for the first time, by a new steady state inductive helicity injection method. Using two inductive injectors with $n = 1$ symmetry and oscillating at 5.8 kHz, a steady state spheromak with $n = 0$ symmetry is formed and sustained through non-linear relaxation. A spheromak with about 13.5 kA of toroidal current is formed and sustained using about 3 MW of power. This is a much lower power threshold for spheromak production than required for electrode based helicity injection methods. Internal magnetic probe data, including oscillations driven by the injectors, agree with the plasma being in the Taylor state. The agreement is remarkable considering the only fitting parameter is the amplitude of the spheromak component of the state.

IC/P7-7 · First Experiments to Test Plasma Confinement by a Magnetic Dipole

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Abstract: We report the first production of high beta plasma confined by a laboratory superconducting dipole using neutral gas fueling and electron cyclotron resonance heating (ECRH). The pressure results from a population of anisotropic energetic trapped electrons that is sustained by microwave heating provided sufficient neutral gas is supplied to the plasma. The trapped electron beta was observed to be limited by the low-frequency hot electron interchange (HEI) instability, but when the neutral gas was programmed so as to maintain the deuterium gas pressure near 0.2 mPa, the fast electron pressure increased by more than a factor of ten and the resulting stable high beta plasma was maintained quasi-continuously for up to 14 seconds. The levitated dipole experiment (LDX) is a new research facility that is investigating plasma confinement and stability in a dipole magnetic field configuration as a possible catalyzed DD fusion power source that would avoid the burning of tritium.

IC/P7-8 · Properties of an Optimized Quasi-Isodynamic Stellarator with Poloidally Closed Contours of the Magnetic Field Strength

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Abstract: It was demonstrated earlier that the quasi-isodynamic stellarator with poloidally closed lines B = const. on magnetic surfaces [Nucl. Fus. Vol. 42 (2002) L23] can exhibit excellent fast-particle confinement. In the present report the results are presented of integrated physical optimization comprising MHD and neoclassical theory of $N = 6$, $A = 12$ quasi-isodynamic stellarator. The configuration is shown to conserve excellent fast-particle confinement and possess nearly zero bootstrap current and small effective ripples. These properties are attainable for high MHD-stable beta limit ($\langle\beta\rangle \sim 0.085$).

IC/P7-9 · Spatial Distribution of D-D/D-^{3}He Advanced Fuels Fusion Reactions in an Inertial Electrostatic Confinement Device

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Abstract: We have revealed experimentally the spatial distribution of D-^{3}He reaction in an Inertial Electrostatic Confinement (IEC) fusion device, and consequently the fractions of volumetric and embedded fusion contributions, both of which are essential for understanding the IEC mechanism and accordingly for innovative IEC-based concepts coming up. An IEC device basically consists of a highly transparent gridded cathode concentrically held at the center of a spherical anode. Ions are accelerated to the center and the spherical focusing of ions results in high fluxes of neutrons and protons up to 10^{10} sec^{-1} so far by D-D in pulsed operation. An IEC device could be thus an transportable neutron/proton source ideal for versatile applications. In IEC researches aiming at further enhanced neutron/proton yields, understanding the spatial distribution of fusion reactions is no doubt one of the most intensive interests. A localized volume of fusion reactions within the transparent cathode is desirable and expected, while a diverse distribution is also predicted resulting from the potential formation by the converging ions' and/or the ion-loss processes with the residual gas molecules. Furthermore, recent experimental results imply, unlike D-D, a strongly localized birthplace of D-^{3}He protons on the cathode grids, i.e. fusion reactions between deuterium beams and embedded ^{3}He, which is undesirable considering the advantages over the existing beam-target type neutron sources, i.e. the long lifetime without maintenance owing to the use of 'plasma target'. In this study, in order to make these essential issues clear, the 14.7 MeV protons from D-^{3}He reactions were counted as functions of collimation geometry by use of movable collimator masks. Then, the Maximum Likelihood Expectation Maximization method was applied to reconstruct the proton yield distribution. As the results, we found the volumetric contribution of 57% within the transparent cathode in the conventional glow-discharge-driven IEC, which is very encouraging compared with the aforementioned predictions. The results also imply an enhancement in the neutron/proton yield by a new IEC-based scheme utilizing a larger diameter cathode. Further experiments are being carried out to find out the D-D distribution as well, and their dependences on the applied voltage and on the operational schemes.

IC/P7-10 · Improved Stability and Confinement in a Novel High-β Spherical-Torus-Like Field-Reversed Configuration

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Abstract: A novel spherical-torus- (ST-) like field-reversed configuration (FRC), with an extremely high-β (over 85%), has been produced in the translation, confinement, and sustainment (TCS) experiment by highly super Alfvénic translation of a spheromak-like plasmoid [H.Y. Guo, A.L. Hoffman, K.E. Miller et. al, Phys. Rev. Lett. 92, 245001 (2004)]. Such a compact FRC-ST carries predominantly a diamagnetic current with the toroidal field magnitude much smaller than that of the poloidal field. However, when combined with the high elongation and small aspect ratio, it results in a safety factor exceeding unity over much of the configuration with a significant magnetic shear near the edge. Relaxation has been demonstrated, for the first time, in such a high-β plasma state, i.e., clearly not a Taylor state. Modeling using the newly developed nearby-fluids theory [L.C. Steinhauer, and H.Y. Guo, "Nearby-fluids equilibria – II: zonal flows in a high-β, self-organized plasma experiment", Phys. Plasmas 13 (2006); in press] shows that a broad core of the FRC-ST resembles a two-fluid minimum energy state. This FRC-ST state exhibits significantly reduced transport with up to four times improvement in confinement [H.Y. Guo, A.L. Hoffman, L.C. Steinhauer et al., Phys. Rev. Lett. 95, 175001 (2005)] over the scaling of conventional θ-pinch formed FRCs. It also exhibits remarkable stability to global low-n modes such as the normally lifetime terminating $n = 2$ centrifugally driven interchange modes present in θ-pinch FRCs. This is explained, for the first time, by a simple stability model, accounting for the magnetic shear of the unique FRC-ST configuration. Comparisons will also be made with the calculations from a 3D two-fluid NIMROD code.

IC/P7-11 · Equilibrium Evolution in the ZaP Flow Z-Pinch

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Abstract: The ZaP Flow Z-pinch experiment at the University of Washington investigates the innovative plasma confinement concept of using sheared flows to stabilize an otherwise unstable configuration. The stabilizing effect of a sheared axial flow on the m = 1 kink instability in Z-pinches has been studied using linearized, ideal MHD theory to reveal that a sheared axial flow stabilizes the kink mode when the shear exceeds a threshold. The ZaP experiment generates an axially flowing Z-pinch that is 1 m long with a 1 cm radius with a coaxial accelerator coupled to a pinch assembly chamber. Magnetic probes measure the fluctuation levels of the azimuthal modes m = 1, 2, and 3. After assembly the plasma is magnetically confined for an extended quiescent period where the mode activity is significantly reduced. Time-resolved Doppler shifts of plasma impurity lines are measured along 20 chords to determine the plasma axial velocity profiles showing a large, but sub-Alfvénic, sheared flow during the quiescent period and low shear profiles during periods of high mode activity. The plasma has a sheared axial flow that exceeds the theoretical threshold for stability during the quiescent period and is lower than the threshold during periods of high mode activity. The sheared flow profile is coincident with a plasma quiescent period where magnetic mode fluctuations are low. The value of the velocity shear satisfies the theoretical threshold for stability during the quiescent period and does not satisfy the threshold during high mode activity. Multichord and holographic interferometers measure a Z-pinch plasma with a peaked radial profile during the quiescent period. Internal magnetic fields have been recently determined by measuring the Zeeman splitting of impurity carbon emission. The measurements are consistent with a well-confined pinch plasma.

IC/P7-12 · On Heat Loading, Divertors and Reactors

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Abstract: We show that the relatively low thermal power handling capacity of the standard divertors (SD), used in current as well as projected machines, forces extremely high (∼95%) radiation fractions f_{Rad} in power reactors with characteristically large heating powers (much larger than ITER-FEAT). Independent of how one apportions this radiation (in the SOL or in the core), such high values of f_{Rad} have severe consequences on core confinement and stability. Low-z impurity seeding does not scale well to reactor power levels so that ∼85% radiation will come from core radiation. To compensate for radiation loss, the confinement time τ_{E} due to plasma transport must significantly exceed the "H-mode" energy confinement time τ_{H} predicted by the ITER98H(y,2) gyro-Bohm scaling law. For a burning plasma, simple robust

models also indicate that high core radiation leads to 1) rapid thermal instability, and 2) core He buildup and radiation collapse. These analyses indicate that even operation in the ITB mode would not lead to an acceptable power reactor. With the SD, robust and favorable scaling with machine size and power level is lost at power levels much beyond ITER-FEAT due to increasing f_{Rad}. By designing divertors with considerably enhanced thermal capacity (through a flaring of the field lines) we have proposed a way out of the disabling core confinement and stability problems caused by high f_{Rad}. We suggest a possible class of experiments which could lay the foundation for an efficient and attractive path to practical fusion power. Realistic coils for EAST, PEGASUS, MAST, NSTX, proposed ST Component Test Facilities, and reactors (ARIES, CREST, EU-B, and EU-C) are designed using the TEQ code. The flux expansion increases to >20 and field line length are roughly doubled. The power-handling capacity of the new XD and standard SD divertor geometries are quantified using UEDGE simulations for free-boundary TEQ equilibria. The new divertors cure recognized severe heat flux problems for proposed CTFs. If they work as expected, XD divertors will lower the core confinement requirement to an extent that even scaling somewhat poorer than gyro-Bohm will be sufficient for a reactor. The robust, and favorable scaling with machine size and power level becomes available once again – even for machines with power levels much beyond ITER-FEAT.

IC/P7-13 · Studies of Free-Boundary Field Reversed Configurations with Improved Stability in the Magnetic Reconnection Experiment

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Abstract: A recent campaign of field reversed configuration (FRC) studies in the Magnetic Reconnection Experiment (MRX) has produced MHD-regime oblate FRCs with improved stability to low-n MHD modes, with an associated extension of the plasma performance. These FRCs are produced by the merging of two spheromaks with anti-parallel toroidal fields. The plasma shape in controlled through three independently energized pairs of equilibrium field (EF) coils; we parameterize the shape of the equilibrium field by its mirror ratio. A conducting center column has been used in some cases to provide important stabilization of the spheromaks during formation and translation before merging, and provides some stabilization of the final FRC state. The FRC without a conducting center column is observed to be unstable to $n = 1$ tilt/shift motions. The dangerous $n = 1$ tilt is suppressed by the center column except at very low mirror ratio, but $n = 2$ & 3 modes often remain. These modes can be reduced by forming plasmas with very large EF mirror ratio. These very oblate plasmas have a minimal amplitude of low-n perturbations, and the longest lifetime. Note that both shape control and passive stabilization are required: plasmas whose $n = 1$ activity is reduced by the central conductor may still be quickly terminated by $n = 2$ activity if the mirror ratio is not sufficiently large. Free-boundary equilibria for these FRC plasmas are computed with a new Grad-Shafranov solver, which constrains the equilibrium solution to the available data. The calculations indicate that as the EF mirror ratio varies from 2 to 4.2, the elongation (κ) varies from 1 to 0.6 and the triangularity (δ) varies from 0.5 to -0.4. The equilibria are used in a model for rigid-body tilt/shift motions, which indicates that these oblate FRCs are in the MHD tilt-stable regime. Initial 3D MHD calculations with the HYM code indicate improved stability to both radially and axially polarized low-n instabilities.

IC/P7-14 · Magnetosphere-like Plasma Produced by Ring Trap 1 (RT-1) – a new approach to high-beta confinement

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Abstract: The Ring Trap 1 (RT-1) is a novel plasma device constructed to explore ways to the advanced-fuel fusion that requires a very high-beta value. The RT-1's mechanism of plasma confinement is based on the theory of self-organized states in flowing plasmas, which predicts that the hydrodynamic pressure in a fast plasma flow can balance the thermal pressure (Bernoulli's law) creating a relaxed state with very high-beta value. The equilibrium state produced by the RT-1 device simulates Jupiter's magnetosphere, as well as wider variations, by using a levitated superconductor ring. The RT-1 will be devoted for investigations of a variety of flow-induced phenomena, including self-organization of flow-field coupling, boundary layer (plasma edge) formation and transport, waves and instabilities in shear flow, and creation of singularities. The RT-1 device employs a high-Tc superconductor (Bi-2223) ring that is levitated in the vacuum chamber. It produces a magnetic field that traps high-temperature plasma, creating a magnetosphere-like configuration. Giving a radial electric field yields a strong flow whose hydrodynamic pressure can balance

the thermal pressure. The first experiment (performed in January 2006) succeeded to produce plasma, using an X-mode 8.2 GHz microwave, in a dipole magnetic field generated by the levitated magnet.

IC/P7-15 · A Gyrotron-Powered Pellet Accelerator for ITER

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Abstract: A new gyrotron-powered, gun-configuration pellet accelerator is proposed for ITER. The large size of ITER opens the possibility that a gun-configuration with a barrrel located between high-field-side blanket/shield modules can have a straight barrel 0.5 m long. The resulting straight trajectories will be normal to the HFS separatrix and hence optimize fuelling efficiency. The gun works as follows: A composite pellet is introduced into a waveguide/guide-tube outside the main tokamak structure at a modest speed $\sim$10 m/s so that it will remain intact as it propagates through waveguide bends. When the pellet has reached a location just outside the HFS vacuum vessel with a trajectory pointing along an outward major radius vector, the gyrotron is energized to a $\sim$1 MW level, the power is launched into the waveguide, and it propagates to the pellet location. A composite three-layer pellet structure with diamond/deuterium/DT-fuel elements absorbs the gyrotron radiation in the deuterium pusher gas element after it has first propagated through the diamond millimeter-wave window. The diamond window has a mass large compared to the DT fuel pellet and acts to absorb recoil. Fixed and free designs are being considered. Conductivity of the deuterium pusher gas is controlled by seeding with graphite (or lithium) powder to absorb mm-wave power and vaporize, accelerating the DT pellet. Dual use of millimeter waveguides as pellet guide-tubes as well a diamond millimeter wave window as a recoil tamper are innovative design features. A theoretical analysis [P.B. Parks and F.W. Perkins, "A Gyrotron-Powered Pellet Accelerator for Tokamak Refueling", General Atomics Report GA-A25270, submitted to Nucl. Fusion (2005)] of the accelerator leads to a third order differential equation with analytical asymptotic solutions. These project the velocity will be $V = 1.0(PL/M_{pellet})^{1/3} \approx 3000$ m/s (all SI units) – an order of magnitude above the value currently planned for the ITER pellet velocity. Standard formulas for penetration against ablation project for ITER project an increase from 0.2 m to 0.6 m, assuming a pedestal temperature of 3 keV.

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FT

Fusion Technology and Power Plant Design

FT/1-1 · Critical Physics Issues for Tokamak Power Plants

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Abstract: Analysis of tokamak fusion power plants such as the EU power plant conceptual study (PPCS) makes clear that, to ensure that tokamaks can be exploited to generate electricity at economically attractive rates, improvements beyond the ITER baseline performance levels are required. Recent studies within the EU which extend the results of the PPCS analysis have provided a more quantitative characterization of the physics of tokamak reactors which will follow ITER and have highlighted the key issues which must be resolved to establish a convincing physics basis for these devices. The power plants investigated typically have net electrical power in the range 1–1.5 GW in steady-state operation, and the aim of the studies has been to develop an improved characterization of physics performance in reactor-scale devices by validating the 0-D systems code analysis used previously against 1-D and 2-D modelling of the plasma core and edge transport and 2-D mhd stability analysis. The PPCS study showed that high beta and high density appear explicitly in the derived scaling for the cost of electricity (CoE). The need for high plasma beta is well established, while high density – typical cases studied have a Greenwald fraction above unity – is required to allow efficient use of the plasma beta and efficient radiation of exhaust power to the reactor walls. Additional critical factors include the ability to handle the power exhausted from the plasma while maintaining high energy confinement quality, high beta and acceptable plasma contamination, achievement of sufficiently high current drive efficiencies to maintain steady-state operation, and the exploitation of auxiliary heating and current drive systems to maintain the current profiles required for high confinement and mhd stability. Steady-state operation is considered to be the preferred operating scenario for fusion reactors following ITER and has formed the focus of these studies, but the development of the "hybrid" scenario, with enhanced plasma confinement and mhd stability characteristics, offers a fall-back approach to power plant operation. Our analysis indicates that trade-offs among the principal power plant parameters could allow steady-state operation in the hybrid regime. The paper will present the results of the latest modelling analysis and discuss the implications for tokamak physics R&D.

FT/1-2 · Power Plant Conceptual Studies in Europe

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Abstract: The European Power Plant Conceptual Study (PPCS) has been a study of conceptual designs for commercial fusion power plants. It focussed on five power plant models, named PPCS A, B, AB, C and D, which are illustrative of a wider spectrum of possibilities. They are all based on the tokamak concept and they have approximately the same net electrical power output, 1500 MWe. These span a range from relatively near-term, based on limited technology and plasma physics extrapolations, to an advanced conception. All five PPCS plant models differ substantially from the models that formed the basis of earlier European studies. They also differ from one another, which lead to differences in economic performance and in the details of safety and environmental impacts. The main emphasis of the study was on system integration. Systems analyses were used to produce self-consistent plant parameter sets with approximately optimal economic characteristics for all models. In the PPCS models, the favourable, inherent, features of fusion have been exploited to provide substantial safety and environmental advantages. The broad features of the safety and environmental conclusions of previous studies have been confirmed and demonstrated with increased confidence. Two key innovative developments made within the PPCS study are worthy of a special note. One is the development of a scheme for the scheduled replacement of the internal components which shows the potential for an overall plant availability in excess of 75%. The other is a conceptual design for a helium-cooled divertor, which permits the toleration of heat loads of at least 10 MW/m^2. The PPCS study has highlighted the need for specific design and R&D activities, in addition to those already underway within the European long term R&D programme, as well as the need to clarify the concept of DEMO, the device that will bridge the gap between ITER and the first-of-a-kind fusion power plant.

FT/1-3 · Next Phase Activity of the International Fusion Materials Irradiation Facility under a New Framework

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Abstract: IFMIF has been pushed forward so far under the Implementing Agreement on Fusion Materials Development in the frame of IEA collaborating agreements. The KEP (Key Element technology Phase) was successfully completed in 2002, and the two subsequent years were devoted to follow-up activities and preparation for the EVEDA phase. In the past five years, major accomplishments have been made in each of the four areas of IFMIF development, i.e. a) accelerator system, b) target system, c) test facility and d) design integration and conventional facilities. These outcomes are compiled in the Comprehensive Design Report and will also be briefly reported in this presentation. In the next phase (which will last six years), the content of EVEDA as described in the CDR has been modified and fabrication and tests of 1:1 prototypes of the low energy part of the IFMIF accelerator are planned. The present prototype accelerator concept consists of ECR ion source, Radio Frequency Quadrupole, matching section and the first tank of the Drift Tube Linac and will be tested with beam. After completion of the EVEDA mission, the use of the prototypes is foreseen in the construction of the final IFMIF accelerator or as spare parts for cost reduction. In the target system area, a model loop with its lithium inventory 1/3 of the IFMIF full scale will be constructed and tested for more than 10000 hours. As for the test facility tasks in EVEDA, mock-ups of the test assembly will be fabricated and the intended performance will be demonstrated, including reliability tests of the rig assemblies in existing fission reactors. The IFMIF EVEDA activity will be conducted under the ITER-BA framework as a Bilateral Agreement between Japan and Europe. In this framework, Japan and EU, each will contribute 8% of the ITER construction cost and approximately 20% of this fund will be allotted to EVEDA. Participation of other parties in the BA is foreseen. The incorporation of the expertise of other countries will be beneficial for the IFMIF project, including IFMIF users.

FT/1-4Ra · Mechanical Properties of Reduced Activation Ferritic/Martensitic Steels after European Reactor Irradiations

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Abstract: The timely development of First Wall and Blanket materials which are capable of withstanding many years the high neutron and heat fluxes, is for material scientists a critical path to fusion power. In an energy generating fusion reactor, structural materials will be exposed to very high levels of irradiation damage up to about 150 dpa. Regarding the radiation damage resistance of the considered Reduced Activation Ferritic/Martensitic steels like the European EUROFER and the Japanese F82H mod., as international reference material of this kind, broad European reactor irradiation programs cover several steps from up to 5 dpa for ITER Test Blanket Modules, till up to 80 dpa for First Wall and Blanket of a DEMO fusion reactor. The lower irradiation damage conditions until 15 dpa can be realized in European fission reactors like the HFR at JRC, Petten, but higher damages – in reasonable times – in fast reactors only. For this purpose the fast reactor BOR 60 of the State Scientific Centre of Russian Federation Research Institute of Atomic Reactors, Dimitrovgrad, has been utilized for irradiations up to 80 dpa. Results from the lower damage irradiations like SIENA, MANITU and HFR-Ib, up to 2.4 dpa had been reported in the past frequently. Recent results of mechanical properties, like Ductile to Brittle Transition Temperatures from instrumented impact-V tests with sub size specimens and stress and strain values from tensile tests with miniaturized specimens will be presented from specimens of the HFR Phase-IIb (SPICE) irradiation project up to 15 dpa at different irradiation temperatures between 250 and 450°C. The fast reactor irradiation project ARBOR 1 reached at temperature $\leq$340°C an irradiation damage of 33 dpa. In the post irradiation instrumented impact-V tests a significant increase in the Ductile to Brittle Transition Temperature as an effect of irradiation has been detected. During tensile testing the strength values are increased and the strain values reduced due to substantial irradiation hardening. Post irradiation heat treatments revealed novel and very encouraging results: After first thermal recovery tests with 550°C annealing for three hours it could be demonstrated on EUROFER specimens irradiated at $\sim$340°C up to 15 dpa, that nearly virgin conditions in Ductile to Brittle Transition Temperature and tensile behavior could be achieved.

FT/1-4Rb · Status and Key Issues of Reduced Activation Martensitic Steels as the Structural Materials of ITER Test Blanket Module and Beyond

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Abstract: The status of research and development (R&D) of reduced activation martensitic steels (RAMs) in Japan are reviewed and key issues suggested from recent achievements in Japan since the last conference are highlighted, with the aim of the fabrication for the ITER Test Blanket Module (TBM) and application for the DEMO reactor. It was pointed out that international collaboration would be desirable for research on key issues such as precipitate stability under irradiation or Ta effects which are common for all RAMs and require an extensive research effort.

FT/1-5 · Advanced Qualification Methodology for Actively Cooled High Heat Flux Plasma Facing Components

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Abstract: High heat flux plasma facing components (PFCs) in steady state fusion devices require high reliability. This can be only guaranteed by a very high level of qualification obtained with a rigorous acceptance inspection protocol. These components have to withstand heat fluxes from the plasma in the range of 10–20 MW/m^2 involving a number of severe engineering constraints: (i) the armour materials must be refractory and compatible with plasma wall interaction requirements; (ii) the heat sink should have a high thermal conductivity, high mechanical resistance and sufficient weldability behaviour; (iii) the cooling system, which is generally based on a circulation of pressurized water in the PFCs heat sink, must offer a high thermal efficiency; (iv) the joint of the refractory armour material onto the metallic heat sink,. To meet the power exhaust needs of PFCs during plasma operation requires control of their thermal and mechanical integrity. The first step is to detect defects in the element, such as material discontinuities like cracks and debondings. These will cause hot spots on the armour materiel and may even lead to the destruction of the PFC e.g. critical flux event. As the heat exhaust capability and the PFCs lifetime during plasma operation will stem from the manufacturing quality, a set of qualification activities should be performed during the component development and subsequent manufacturing phases. The major progress brought by this methodology stems from the combination and the correlation of three techniques: thermomechanical modelling, high heat flux testing and advanced non-destructive techniques, such as active infrared thermography. The scheme is applied during all the qualification activities: research and development phase, prototype manufacture including damage study for high heat flux, first series fabrication to define acceptance criteria and commissioning of the series fabrication. The paper describes the qualification route, which has been followed in order to define an acceptance criterion for CFC based flat tile target elements for the TORE SUPRA series production and more recently for the W7-X component first series production. First investigations for the ITER Divertor elements, which are more challenging due to the thicker armour material and the more complex monoblock geometry, are also presented.

FT/1-6 · Alcator C-Mod Ion Cyclotron Antenna Performance

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Abstract: Ion cyclotron range of frequency (ICRF) heating is expected to be an important auxiliary heating source for ITER and fusion reactors. One of the keys to successful ICRF heating is the antenna performance and a number of issues can limit the antenna performance including poor voltage and power handling, impurity production, strong RF plasma edge interactions, poor RF coupling, and localized heating of the antenna structure. High power density antenna operation, with all metal protection tiles and plasma facing components (PFC), present significant challenges to ICRF antenna operation. High performance plasmas can be achieved with all metal PFC's after boronization. In Ohmic H-mode discharges, the plasma performance degradation occurs at a rate 3–4 times slower than RF heated discharges with the similar input energy (discharge integrated). The erosion process also appears to be accelerated for weaker single pass absorption heating scenario, D(^{3}He) on C-Mod. The C-Mod H minority single pass absorption is stronger and is similar to that expected in ITER implying similar RF induced erosion in ITER as on C-Mod. Since Faraday screen-less antenna operation has a number of advantages; the J antenna was operated without a Faraday screen. The voltage and power handling were unaffected by the screen removal. However, the heating effectiveness was 15–20% less and the influx of Cu was identified as the likely cause

of the decreased performance. On C-Mod, high density discharges can yield neutral pressures at which antenna operation is prohibited. This neutral pressure limit may be related to phenomena associated with antenna ELM (edge localized mode) interactions. Experiments showed that multipactor can cause a glow discharge at neutral pressures two orders of magnitude below the Paschen breakdown limit. In the presence of a 0.1 T B-field, measurements on the C-Mod antennas showed the presence of a glow discharge at a neutral pressure similar to the observed operational neutral pressure limit, suggesting the neutral pressure limit is a result of multipactor induced discharge. We plan to modify the J antenna to have materials with <1 secondary electron coefficient for all electron energies and initial results will be presented.

FT/2-1 · SST-1 Commissioning and First Plasma Results

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Abstract: The fabrication and assembly of the SST-1 tokamak, has been completed at the Institute for Plasma Research. Commissioning of SST-1 tokamak is currently in progress. A 10 kA DC power supply and associated energy dump system, has been commissioned for the superconducting (SC) toroidal field (TF) magnets of SST-1. A self-protecting, quench detection system has been developed and commissioned. A DC bus system, with appropriate switches, has been installed and tested for powering the Ohmic and vertical field coils of SST-1 from power supplies of ADITYA tokamak. Remote operation of the system from SST-1 control system has been established. One pair of current leads for operating current of 10 kA at 4.2K has been designed and manufactured indigenously. The pair of leads has been tested successfully for required ramp rate and steady currents. The leads have been integrated with the TF magnets. The cryogenic systems, at 4.2K and 80K, have been commissioned. The pumping systems for both the cryostat and the vacuum vessel have been installed. After completion of the assembly, initial leak tests and necessary repairs, the cryostat has been pumped down to a base pressure better than 10^{-5} torr. Leak tests in the main vacuum vessel have been completed and welding leaks repaired. Base vacuum better than 10^{-6} torr have been achieved without baking and discharge cleaning of the vessel. The thermal shields of the SST-1 have been successfully cooled down to 80K. Subsequently simultaneous cool down of the thermal shields and SC magnets has been taken up. The SC magnets and thermal shields were cooled to 70K when cold leaks developed in the helium circuit. Leaks have been identification and repaired. Cool down of the SC magnets and current leads, is underway. First phase diagnostics have been installed on SST-1. Data acquisition and control system have been tested and commissioned. A pair of radially movable poloidal limiters has been installed in the vacuum vessel preparatory to production of circular plasma through Ohmic discharge. The operational experience of cooling down of the thermal shields and SC magnets as well as performance results of the cryogenic system of SST-1 will be presented along with the results on Ohmic discharges.

FT/2-2 · KSTAR Assembly

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Abstract: Since all of the sixteen Toroidal Field (TF) magnets completely assembled in February 2006, the site assembly of the KSTAR is being actively accelerated to meet the assembly finish in middle of 2007. According to the assembly plan, all the SC magnets and most of the in-cryostat components such as SC buslines, helium piping system, thermal shields, and cables of the various kinds of the sensor will be assembled in the cryostat within 2006. Moreover, the integrated commissioning plan before shot for the 1st plasma is being also prepared to keep the phase with rapid progress change in the site assembly of the KSTAR tokamak. In this paper, overview of the KSTAR assembly including brief history and result for the assembled system, progress in the assembly hall until October 2006, remained works, and general commissioning plan will be reported.

FT/2-3 · Experience gained during fabrication and construction of Wendelstein 7-X

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Abstract: During the design and fabrication of the main components for the W7-X stellarator a number of technical challenges and severe problems had to be overcome many of which are considered relevant also for other machines of similar size and complexity, like ITER. Massive supporting structures are used to sustain the large electromagnetic loads. Finite element analyses (FEA) were used to design the main

support structures. However, when the original FEA models were improved and refined, weaknesses showed up requiring reinforcement after components had already been fabricated. Some highly loaded elements are qualified by testing of instrumented prototypes. This proved to be very important to eradicate weaknesses. The coils for W7-X have been the subject of considerable delivery delays due to a variety of problems. In addition to the reinforcement of the coil support blocks on the non-planar coils, refined FEA showed that the planar coils undergo excessive out-of-plane bending. This has been corrected by introducing a large number (several hundred) of shear pins to stiffen the joint between the lateral plates and the internal and external circumferential plates of the casings. In an early stage it was decided that all 70 coils for W7-X should be cold tested. The cold tests have shown that of all the coils tested thus far the specified superconductive behaviour was according to specification. However, a number of faults have been found that otherwise would almost certainly not have been detected. These include cold leaks, poor performance of instrumentation, some of which is considered important for machine operation, as well as electrical faults. After cool-down and warm-up in a test cryostat all coils are tested under Paschen conditions. Initially many non-planar coils did not pass this test. After returning to the factory detailed investigations have been carried out, repair procedures have been qualified and applied. The W7-X experience has shown that (i) detailed FEA using benchmarked codes and models should be complemented with tests on full or partial prototypes to qualify highly loaded structural elements, (ii) instrumentation must be qualified for its intended use by testing, (iii) cold testing of coils may show some faults that otherwise may not be detected, and (iv) Paschen tests after at least one cool-down/warm-up cycle are highly recommended.

FT/2-4 · Progress in the Construction of NCSX

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Abstract: The National Compact Stellarator Experiment (NCSX) is currently being constructed at PPPL in partnership with ORNL. Its motivation is the need for physics solutions to the problems of achieving steady-state operation and avoiding disruptions in MFE confinement systems. Technological challenges in the construction of NCSX are: 1) realizing the complex component geometries with the accuracy required to obtain attractive physics properties and 2) providing for the physical measurements needed to diagnose those properties in experiments. Manufacturing solutions were developed through a manufacturing R&D program and the initial stages of production. For the modular coil winding forms (MCWF), a custom stainless steel alloy was developed to provide attractive properties for both manufacture and operations. Processes were developed for constructing the modular coils with the current center positioned to the required ± 0.05 mm accuracy. The coils are wound with flexible copper rope conductor installed, together with insulating materials and copper cooling strips, on accurately machined surfaces of the winding form and then encapsulated in epoxy. An R&D coil protototypical of the actual coil winding pack cross section and of the most difficult twists and bends was constructed and tested to demonstrate the process. The vacuum vessel shell geometry realized within 5 mm accuracy. Inconel panels are pressed formed on machined dies at room temperature, and then assembled onto accurately machined skeletal fixtures where the seams are welded to form the shell. The process was demonstrated by constructing a prototype 20-degree shell sector with an installed port. An array of magnetic flux loops will be mounted on the surface to make measurements needed for equilibrium reconstruction. The design was optimized using singular value decomposition (SVD) techniques applied to simulated magnetic signals from a data base of 2,500 NCSX free-boundary VMEC equilibria. Production of NCSX components began in 2004 with the placement of manufacturing contracts for the MCWF and the vacuum vessel. The first modular coil has been fabricated.

FT/2-5 · Overview of Modification of JT-60U for the Satellite Tokamak Program

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Abstract: Modification of JT-60U to a superconducting device has been planned as a national device to establish high beta steady state operation of tokamak toward DEMO. Recently, this modification is re-assessed as a satellite tokamak in the Broader Approach project to contribute and supplement ITER toward DEMO. This paper addresses overview of physics and technical designs of modification of JT-60U, named as “JT-60 Super Advanced (JT-60SA)” provided by the assessment of JA-EU satellite tokamak working group and JT-60SA design team.

FT/P5-1 · Recent Results from Real-Time Active Control of MHD Modes in RFX-mod

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Abstract: The paper focuses on the recent results achieved using at best the technology developed for the RFX-mod real time control system of MHD modes, which provides control features unique in the worldwide fusion experimental program. The paper aims at highlighting the successful technological aspects that have led to the outstanding scientific achievements obtained in the machine recent operation, including a dramatic increase in plasma duration and a significant decrease in magnetic perturbations. RFX-mod addresses the MHD mode control by acting on the magnetic field configuration at the boundary. The approach is not limited to the stabilization of Resistive Wall Modes (RWMs) in RFX-mod, but it also aims at acquiring an experimental knowledge on the MHD mode control that can be relevant for high beta discharges in tokamaks. To the purpose, an ad-hoc field correction coil system has been installed, consisting of 192 saddle coils covering the whole torus surface. The coils are arranged in 48 poloidal arrays, each made up of 4 equispaced coils. Each coil is fed independently. By means of this actuating system, magnetic field poloidal and toroidal harmonic components (m, n) can be generated with $m = 0$, $n = 0$ to 24 and $m = 1$, $n = -23$ to 24. The control system is based on a modular hardware and software infrastructure, capable to fulfill the tight system requirements in terms of input/output channels (>700/>250), real time data flow (>2 Mbyte/s), computation capability (>1 GFLOP/s), and real time constraints. The control system, which has been extensively and routinely used in the operation of RFX-mod since the commissioning, provides a very powerful environment for interaction with MHD modes and for their control. Two basic control strategies have been developed: the Selective Virtual Shell (SVS) and the Mode Control (MC). The former aims at canceling all modes (Virtual Shell concept) except for selected preset ones (Selectivity concept). The Selectivity concept allows operating without interfering with the vertical equilibrium field ($m = 1$, $n = 0$), allows studying the growth of selected unstable RWMs, and facilitates producing Quasi-Single Helicity plasmas. The MC scheme aims at interacting with single or multiple modes. In this strategy, the regulators act directly on the modes by means of complex gains that can account for mode rotation.

FT/P5-2 · Development of Strongly Focused High-Current-Density Ion Beam System and Its Application for the Alpha Particle Measurement in ITER

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Abstract: Strongly focused and high-current density ion beam system was successfully developed using large concave electrodes. The beam with a diameter of 345 mm at the electrode is focused to a diameter of ~36 mm at the focal point with a small total divergence angle of about 1.6 deg. In the cases of H^+, He^+ and HeH^+ beams of ~25 keV, it is estimated that the extracted ion current density achieves as high as ~190, ~86 and ~13 mA/cm^2, respectively. The HeH^+ beam of ~25 keV can be considered as a primary beam of a diagnostic helium neutral beam of ~1–2 MeV for the alpha particle measurement in ITER, and it is demonstrated for the first time that the current density level of HeH^+ beam is sufficient for this purpose.

FT/P5-3 · Achievement of High Availability in Long-term Operation and Upgrading Plan of the LHD Superconducting System

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Abstract: The Large Helical Device (LHD) that has been demonstrating high performance of heliotron plasma is the world's largest superconducting system. Availability higher than 98% has been achieved in a long-term continuous operation both in the cryogenic system and in the power supply system. It will be owing not only to the robustness of the system but also to efforts of maintenance and operation. One big problem is shortage of cryogenic stability of a pair of pool-cooled helical coils. Composite conductors had been developed to attain the sufficient stability at high current density. However, it was revealed that a normal-zone could propagate below the cold-end recovery current by additional heat generation due to the slow current diffusion into a thick pure aluminum stabilizer. Besides, a novel detection system with pick-up coils along the helical coils revealed that normal-zones were initiated near the bottom of the coil where the field is not the highest. Therefore, the cooling condition around the innermost layers in the high field is considered to be deteriorated at the bottom of the coil by bubbles gathered by buoyancy. In order to raise the operating currents, methods for improving the cryogenic stability have been examined, and stability tests have been carried out with a model coil and small coil samples. The coil temperature is planned to be lowered from 4.4 K to 3.5 K, and the operating current is expected to be increased from 11.0 kA to 12.0 kA that corresponds to 3.0 T at the major radius of 3.6 m.

FT/P5-4 · High Performance Operation of Negative-Ion-Based Neutral Beam Injection System for the Large Helical Device

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Abstract: Development of high performance negative-ion-based NBI (N-NBI) heating system is one of key technical issues for ITER and fusion reactor. The N-NBI systems of LHD and JT-60U are two facilities that are operating for high power plasma heating and current drive in the world. Because handling of negative hydrogen/deuterium ions is amateur technology, it has taken long time to improve their performances. In LHD, we succeeded to improve the performance of one of three beam lines dramatically in 2004 by adopting a multi-slot grounded grid for the accelerator of ion source. It has become more reliable by these two-year operational experiences. The total port-through beam power of 10–13 MW was injected successively throughout four-month long experimental campaign. The averaged negative ion beam current density at the exit of ion source, which was evaluated from the port-through injected power, was achieved up to 350 A/m^2. This value is larger than the required value of ITER NBI for hydrogen beam operation. Pulse length at high beam power level has also been extended by increasing the transparency of grounded grid due to reduction of heat load on it, which means that the heat load on the grid is ascribable to accelerated electrons, halo ions or neutrals. These results (increase in power and pulse length) have contributed to expand the operation region of LHD and to improve LHD plasma parameters. We found an associated problem of multi-slot grounded grid system, that is, unmatched optimum conditions of beam optics in vertical and horizontal directions. As a result, averaged beam divergence was worse compared with the old accelerator. This phenomenon comes from the geometrical asymmetry of electric field in the accelerator. The characteristic of accelerator was investigated at the NB test stand, and a solution has been found. Therefore, further improved performance will be expected.

FT/P5-5 · Tokamak Fusion Neutron Source Requirements for Nuclear Applications

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Abstract: A series of conceptual design studies has established that the tokamak D-T fusion neutron source needed to drive a sub-critical, fast-spectrum nuclear reactor for the transmutation of spent nuclear fuel 1) could be designed on the ITER physics and technology databases, 2) could be constructed on the basis of ITER operation as a prototype, and 3) would be less demanding than a DEMO or commercial fusion reactor. The nominal tokamak parameters are ($R < 4$ m, $\beta_N < 2$, $H(y,2) = 1$, $Q_p < 3$, $P^{fusion} < 200$ MW, $I = 7$–8 MA, $\gamma_{cd} = 0.6$, $B < 6$ T). The sub-critical nuclear reactor designs are based on concepts and technologies being developed in the GEN-IV studies to insure realism.

FT/P5-6 · Innovation in Design and Fabrication of Compact Stellarators

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Abstract: The Quasi-Poloidal Stellarator (QPS) is being developed to test key physics issues at very low plasma aspect ratio, 1/2-1/4 that of existing stellarators. Engineering innovation is driven by both the complex 3-D design requirements and the need for reduced cost and risk in fabrication. Complex, highly accurate stainless steel modular coil winding forms are cast and machined; conductor is wound directly onto the modular coil winding forms; a vacuum-tight cover is welded over each coil pack; the coils are vacuum pressure impregnated; and the completed coils are installed in an external vacuum vessel. As a result, QPS differs significantly in design and construction from other toroidal devices. A full-size prototype of the largest and most complex of the modular coil winding forms has been cast by machining sand molds and a high temperature pour. The resulting casting required more than an order-of-magnitude fewer major weld repairs than similar sand castings using conventional patterns. A high-current-density cooled conductor that can be wound into complex 3-D shapes was developed. Compacted cable conductor consisting of stranded copper filaments wound around an internal copper cooling tube was successfully fabricated. The tube is filled with a low-melting-temperature eutectic during processing, which allows the copper tubing to be wound in a small radius without distortion or buckling. The eutectic is flushed from

the cooling tube with hot water prior to vacuum impregnation. A mockup weldment of a vacuum-tight coil can and a section of prototypical casting showed negligible weld distortion and benign temperature rise at the location of the windings. Vacuum testing on prototypical cast material was satisfactory. The high-temperature cyanate ester resin (CTD 403) used for potting has several advantages over the usual epoxy. While the mechanical properties are similar for both, CTD-403 can be used up to 150°C; it doesn't absorb water, which provides another barrier against water leaks; and it is easier to work with since it has the viscosity of water and an essentially unlimited pot life at room temperature. A simple external vacuum tank is used rather than a highly shaped internal vacuum vessel between the plasma and the modular coils, which avoids the need to slip complex-shaped nonplanar coils over a complex-shaped vacuum vessel.

FT/P5-7 · Calculation of Neutronics for a New CH HCSB NT-TBM with 3×3 Sub-modules

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Abstract: By using three-dimension MCNP code, radioactivity code FDKR and FENDL2.0 data library, the neutronics calculation for a HCSB (Helium Cooling Solid Breeder) TBM (Test Blanket Module) with 3×3 sub-modules has been performed. Under conditions of neutron wall loading of 0.78 MW/m^2 and duty factor of 22%, it is given for the tritium breeding ratio (TBR) of 0.907, total tritium generation rate of 0.0175 g/day, peak power density of 9.27 MW/m^3 and total power production of 0.422 MW/m^3. The total radioactive inventory in the TBM (Ci) at shutdown is 0.543 MCi and drops to $\sim$0.535 MCi within a minute. At shutdown, the total afterheat is $\sim 8.77 \times 10^{-3}$ MW, which is attributed mainly to the structure. After 1 hour, 1 day, 1 year, 10 years, and 100 years, the total afterheat are 4.59×10^{-3} MW, 1.64×10^{-4} MW, 6.00×10^{-5} MW, 4.70×10^{-6} MW. For the EUROFER structure, ^{59}Ni ($T_{1/2} = 75$ ky), ^{93}Zr ($T_{1/2} = 1.5$ My), ^{94}Nb ($T_{1/2} = 20$ ky) are the main contributors.

FT/P5-8 · Thermal Hydraulic and Mechanical Analysis of CH HCSB TBM

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Abstract: Based on the structure design and the result of neutronics analysis of the CH HCSB TBM (Chinese Helium Cooled Solid Breeder Test Blanket Module), the thermal hydraulic analysis and thermal mechanical analysis of the CH HCSB TBM have been carried out to confirm the feasibility for the normal conditions and extreme conditions in ITER NT operation phase using FE code ANSYS. It is confirmed that the design of the CH HCSB TBM is reasonable and acceptable.

FT/P5-9 · Neutronics and Nuclear Data for Fusion Technology — Recent Achievements in the EU Programme

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Abstract: The nuclear design of any kind of fusion device relies upon the results of neutronics calculations. Neutronics and nuclear data thus play a key role in the strategic approach to develop fusion as a future energy source. A well-qualified nuclear database and validated computational tools are required for quality assured neutronics and activation calculations including uncertainty assessments. In the framework of the European Fusion Technology Programme, the efforts focus accordingly on the development and qualification of computational tools and data required for reliable design analyses of ITER, in particular the layout of the Test Blanket Modules (TBM), and the IFMIF neutron source facility. A major effort is devoted to integral experiments with the objective to provide the experimental data base for testing the nuclear data and validating neutronics design calculations. This paper presents an overview of the progress achieved in the EU over the past two years in the field of neutronics and nuclear data for fusion technology. Significant progress has been achieved in making available new data evaluations for neutron transport and activation calculations satisfying the needs for both ITER and IFMIF applications, in developing advanced computational tools such as the CAD interface programme for the MCNP code, extending the capabilities for Monte Carlo based calculations of sensitivities/uncertainties by using the track length estimator, and, in conducting a benchmark experiment on a breeder blanket mock-up.

FT/P5-10 · Development of Advanced Tritium Breeders and Neutron Multipliers for DEMO Solid Breeder Blankets

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Abstract: With a target to realize attractive breeding blankets for DEMO reactors, advanced tritium breeders and neutron multipliers, operational at temperatures and neutron fluence higher than those of conventional materials, have been developed. As for tritium breeders, lithium titanate (Li_2TiO_3) pebbles are one of the primary candidate breeding materials mainly from viewpoints of easy tritium release at lower temperatures and chemical stability at high temperatures. However, degradation of tritium release property caused by grain growth at higher temperatures is an issue for this material. Therefore, effects of additives to Li_2TiO_3 have been investigated, and a good prospect has been obtained by using oxide additives such as TiO_2, CaO and Li_2O. The influence of Li evaporation on the crystal structure was examined by adding CaO or Li_2O to Li_2TiO_3, which indicated that the CaO and Li_2O additives are able to control not only the growth of the grain size but also the amount of lithium defects. As for neutron multiplier, development of a real-size electrode fabrication technique and the characterization of beryllium based intermetallic compounds such as Be-Ti and Be-V have been performed. The growth rate of the reaction layer for the Be-Ti alloys decreased with increasing the Ti content up to 5at% in compatibility tests between Be-Ti alloys and structural materials such as SS316LN and F82H. Tritium inventory and irradiation effects of Be-Ti alloys have been evaluated and the properties of the Be-Ti alloys are better than those of beryllium metal. Furthermore, steam interaction of a Be-Ti alloy was about 1/1000 as small as those of beryllium metal. These results indicate a possibility to reduce a risk of a water or air ingress accident and to realize a blanket with high efficiency of electric power generation. These activities have given bright prospects to realize the water-cooled DEMO breeder blanket by the application of these advanced materials.

FT/P5-11 · High Availability Remote Maintenance Approach for the European DEMO Breeder Blanket options

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Abstract: An advanced blanket maintenance concept for DEMO was developed which implies vertical multi-module blanket segments (MMS) and a small number of vertical remote handling ports. As a common feature the MMS blanket consist of reasonable small blanket modules which are mounted onto a vertical back-plate that contains the coolant channels and in case of liquid metal blankets also the breeder manifolds. Depending on the breeder blanket option (HCPB, HCLL or DCLL) different radial build, sub-module and attachment concepts are considered preferable. In the HCPB MMS blanket configuration the blanket sub-modules are attached flexibly to the backplate and the blanket modules in front of the manifolds are acting as a thermal and neutron shield which allows to operate the MMS manifold almost steady-state even in pulsed plasma regimes. The clearly defined temperature and deformation of the backplate during heat up allows to connect it rigidly to a permanent shield structure. The shield structure is operated at the same temperature as the manifold will be build from vertical segments that a strongly connected in both toroidal and poloidal direction. In the DCLL MMS configuration a flexible attachment system and a permanent frame structure is used to mount the MMS onto a low-temperature shield or the vacuum vessel. For all blanket concepts a common remote handling strategy was developed which in addition to a crane-like machine involves a remote transport machine located in the divertor region to transport the MMS in toroidal direction. The transport machine is equipped with sliding carriage systems for the movement of the inboard and outboard MMS in transverse radial/poloidal direction. In the final position the MMS are either bolted onto a permanent self-supporting hot shield structure or are mounted onto a frame which allows for relative thermal expansion of the MMS and the vacuum vessel. While the first option seems to be possible for the HCPB and HCLL concept the later could allow for integration of a new DCLL MMS blanket option.

FT/P5-12 · Neutronics Investigation of Advanced Self-Cooled Liquid Blanket Systems in Helical Reactor

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Abstract: Neutronics performances of advanced self-cooled liquid blanket systems have been investigated in design activity of the LHD-type helical reactor FFHR. In the FFHR design, DT plasma is covered with

four separated blanket layers running helically in the toroidal direction. Therefore, neutron transport in the complicated blanket components, leakage through opening between the blanket layers and reflection from components such as support structures, vacuum vessels etc. are important factors to be simulated in the neutronics evaluation with three-dimensional (3-D) geometry data. In the present study, a new method of 3-D neutronics calculation system has been developed for the neutronics investigation in the helical reactor. The system is focusing on quick feedback between design modification and neutronics evaluation by generating the geometry data according to numerical equations defining the helical structures. Using this new calculation system, recent modified self-cooled Flibe or Li blanket systems for FFHR have been evaluated to make clear design issues for enhancement of neutronics performance. In particular, the possibility of tritium breeding without adding neutron multiplier such as Be has been tested for the limited blanket space. Evaluation of the tritium breeding ratios (TBRs) performed with the calculation system indicated that both of Flibe+Be/JLF-1 (RAFS) and Li/V-alloy (Vanadium alloy) blanket systems would achieve adequate tritium breeding performance in the FFHR design. Evaluation of TBR has been underway also for Flibe cooled spectral-shifter and tritium breeder blanket (STB) system. For further understanding of neutronics characteristics in the helical reactor, the development of the 3-D neutronics calculation system has been conducted by adding the functions to simulate the helical configuration of neutron source and to import geometry data from CAD system. Especially for issues of radiation shielding, neutron behavior through the helical structures is expected to be understood by 3-D visualization of neutron flux distribution using the system. This work has been supported by National Institute for Fusion Science NIFS05UCFF007 and Japan/US collaborative program JUPITER-II task3-1.

FT/P5-13 · Experimental study on nuclear properties of water cooled pebble bed blanket

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Abstract: For the first time, nuclear properties are experimentally examined on the water cooled pebble bed blanket by using DT neutrons in this study. In order to evaluate tritium production rate (TPR), neutronics studies are performed on the multi-layered blanket mockup experiment with water panel, and pebble bed layer experiment. The multi-layered blanket mockup is composed of a set of slabs of first wall panel, two Li_2TiO_3 layers, two beryllium layers and four partition panels. The first wall and partition panels are composed of F82H and water. Diagnostic pellets of Li_2CO_3 were embedded inside the Li_2TiO_3 layers. Tritium activities produced in these irradiated pellets were measured with a liquid scintillation counter. Numerical analyses were conducted by using MCNP-4C with FENDL-2. Average ratios of the calculated results to the experimental values (C/Es) are 0.99 and 1.04 on the first and second layers, respectively, on TPRs. The prediction uncertainties were clarified on TPRs for the blanket with water from this experimental study. The pebble bed layer mockup is composed of Li_2O pebble with 1mm in diameter, beryllium block and F82H. The Li_2O pebbles were also applied as the detectors for TPR. A calculation method by the heterogeneous geometry is proposed in this study, and the differences between TPRs for the heterogeneous and homogeneous geometries are discussed. The hexagonal close-packed models were assumed in the heterogeneous geometry, and all pebbles and void among the pebbles were simulated using the repeated-structures modeling method. It is proposed that uniform annular gaps are installed at the boundaries between adjacent pebbles in the calculation to adjust the packing fraction in the experiment in this study. Average C/Es are 0.97 and 0.99 in the homogeneous and heterogeneous geometries, respectively. Total TPR obtained by the homogeneous geometry is smaller than that by the heterogeneous one. With the increase of the ^{6}Li enrichment, the difference between TPRs for the heterogeneous and homogeneous geometries becomes larger. In the case of the 90% enriched ^{6}Li, the difference is more than 5%. The heterogeneous geometry is important for the detailed design calculation of blankets. It can be concluded that the proposed method is very useful for the evaluations of TPR and TBR in the pebble bed layer.

FT/P5-14 · He-cooled Divertor Development: Technological Studies and HHF Experiments for Design Verification

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Abstract: In the course of the He-cooled divertor development at Forschungszentrum Karlsruhe, which was aimed at reaching an HHF of at least 10 MW/m^2, two design concepts based on the use of multiple-jet cooling (HEMJ) and a slot flow promoter (HEMS), respectively, have been found to be most promising. Both concepts use small finger units, consisting of a W tile, as thermal shield that is brazed to a pressure-carrying W-alloy thimble. The W finger units are connected to the steel structure via a transition piece

with Cu filling and conical lock. Technological studies related to the joining of W-W and W-steel parts and the fabrication of W parts have been performed successfully. Based on these results, a series of W 1-finger mock-ups have been manufactured for full HHF tests in a helium loop (10 MPa, 600°C) at Efremov using the TSEFEY EB facility for simulating the heat load. The first test results have confirmed the design feasibility and shown good agreements with the predicted divertor performances. Post-examination of the first W mock-up tested is underway. The test series will last until mid-2006. Discussion of these issues and of the state of the art of divertor development shall be the subjects of this report.

FT/P5-15 · Assessment of the Shielding Efficiency of the HCLL Blanket for a DEMO-type Fusion Reactor

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Abstract: The work summarizes the results of a neutronic study on the HCLL blanket for a DEMO-type fusion reactor. The analysis is based on 3-d Monte Carlo calculations with the MCNP-4C code and comprises the assessment of the shielding efficiency in terms of the radiation loads to the superconducting TF-coils and the neutron induced production of He and H gas in the steel structure of the blanket and at locations where the welding of pipe connections is assumed. The analysis is performed utilizing three-dimensional 9-deg. sector model representing the modular HCLL breeder blanket, developed previously by FZK. The shielding performance of the blanket/shield/vacuum vessel system is assessed for the radiation loads to the superconducting TF-coils at the torus mid-plane considering three variants of a two-component shield, arranged in between the central inboard modules and the VV. Estimates of radial distributions of the He and H gas production in the steel structure of the blanket are obtained and the radial depth where the re-weldability criterion for the He production <1 appm is fulfilled is identified.

FT/P5-16 · An Innovative Concept of High Temperature Liquid Blanket for Hydrogen Production

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Abstract: Hydrogen is considered as the most potential energy carrier in the future, and the nuclear power would be used as a provider of high temperature heat for hydrogen production. An innovative concept of the high temperature lithium lead blanket with "ultilayer flow channel inserts" has been presented in this paper. The maximum outlet temperature 950°C of lithium lead coolant is not restricted within engineering limit of 550°C for RAFM structural material. The theoretical analyses and numerical calculations have been performed to validate the feasibility of this concept. And further technology issues, such as tritium permeation and material corrosion on high temperature condition, are to be clarified for improvement of this innovative blanket concept.

FT/P5-17 · Influence of High Magnetic Field on Fusion Reactor Blanket Processes

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Abstract: During operation of a fusion reactor, materials and processes of blanket zone will be subject to high temperatures, intense fast neutron radiation and action of intense magnetic field. MF changes kinetics of chemical reactions of high energy as a result of spin transformation, diffusion parameters of charged particles. An alternating MF can induce e.m.f. in materials. Changeable processes of this kind in the blanket may be radiolysis of ceramic materials (colloid Li and O_2), tritium release from ceramic and Be pebbles, corrosion processes between ceramics and structural materials, tritium permeation. Up to now, influence of MF was studied for the most part on previously irradiated materials (PIE). The following research results are described in this study: influence of MF on radiolysis of Li_4SiO_4 and Li_2TiO_3 pebbles, tritium release from irradiated ceramic and Be pebbles, preliminary results about mutual corrosion of ceramics and EUROFER 97 at high temperature under radiation. Irradiating the ITER and DEMO relevant Li_4SiO_4 (FZK) and Li_2TiO_3 (CEA) pebbles at different temperatures with fast electrons (5 MeV), MF of 1.4 T increases the radiolysis by 20%–25%. The effect is related to the change of spins of primary excitons of the matrix in MF, resulting in an increase in the efficiency of formation of radiation-induced defects and products of radiolysis. MF of 2.4 T delayed by 40% the tritium release at thermoannealing to 1120 K of the Li_4SiO_4 pebbles (FZK) irradiated in the EXOTIC-8/8 experiment. The effect may be related to lengthening of the diffusion path of T^+, as a main form of localised tritium, in the volume of ceramic grains (the grain size 30–40 μm). In the Be pebbles irradiated in the BERYLLIUM experiment, tritium is localised in the T_2 and T^0 chemical forms. As, for the most part, diffusion of T^0 ensures the

tritium transfer in the Be grain volume, MF delays the tritium release. The delaying effect of MF is related to the decrease of T^0 as a result of spin change of an uncorrelated T^0 ... T^0. On the other hand, under the simultaneous action of ionising radiation, MF increases considerably the tritium release by up to 80%. The facilitating effect of MF may be caused by spin change from singlet to triplet state in a correlated pair of radicals generated as a result of T_2.

FT/P5-18 · Preliminary Design of China ITER TBM with Helium-Cooled and Solid Breeder Concept

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Abstract: ITER will play a very important role in first integrated blanket testing in fusion environment. Some DEMO blanket relevant technologies, such as tritium self-sufficiency, extraction of high-grade heat, design criteria, safety requirements and environmental impacts will be demonstrated in ITER test blanket modules (TBMs). China has planed to develop own-favored ITER TBM modules based on China's fusion energy development strategy and DEMO definition for testing during ITER operation period. He-Cooled Solid Breeder with FS is main stream for the fusion reactor blanket design and has foundation of the world R&D database. The helium-cooled solider pebble bed (HC-SB) concept has been adopted as one of option of China ITERTBM design. The preliminary design and analysis have been carried out recently.

FT/P5-19 · Test Strategy and Development Achievements of ITER Solid Breeder Test Blanket Modules in Japan

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Abstract: This paper presents progresses of the test strategy development, design and supporting R&Ds of solid breeder TBMs in Japan. Japan is proposing to test its unique designs of water cooled solid breeder (WCSB) and helium cooled solid breeder (HCSB) TBMs from the beginning of the ITER operation. Water cooled solid breeder blanket has the better capability of heat removal and applicability to the more compact and economical fusion reactors. On the other hand, if it uses beryllium pebble beds as the neutron multiplier, it has the potential safety concern of thermo-chemical excursion of water and beryllium reaction in the case of the coolant water ingress in the beryllium pebble beds. Safety analysis of the WCSB TBM showed the possible preventive measures to such a case in TBM tests in ITER. Also, to void such potential safety concern, the advanced multiplier material, Be-Ti alloy, is under development in Japan because it has very low reactivity with water even in high temperature. Helium gas coolant has less heat removal capability but higher safety to thermo-chemical excursion phenomena. As for the design work, structure design showed steady progress and clarified detailed structure taking into account the fabrication procedure. As for supporting R&Ds, the corrosion characteristics of the structural material by high temperature and pressure water was clarified as one of critical structure integrity issues. Also, important design data of the breeder pebble bed has been clarfied. Along with the development progress, the test strategy has been investigated to obtain the most effective results of test blanket module test program.

FT/P5-20 · Progress in the Design of a Tritium Breeding Blankets for Testing in ITER

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Abstract: Korea has been developed the design of DEMO relevant blanket concepts for testing in ITER. Helium Cooled Solid Breeder (HCSB) blanket uses He as a coolant, Beryllium as a neutron multiplier, and Ferritic/Martensitic Steel (FMS) as a structural material. The ceramic breeder, Li_2SiO_4, and Be are used in a pebble-bed form and the amount of Be is reduced by using graphite as a reflector. He Cooled Molten Lithium/FS (HCML) blanket uses He as a coolant, Li as a tritium breeder, and Ferritic/Martensitic Steel (FMS) as a structural material. Graphite is used as a reflector in order to minimize the neutron leakage and the position of the graphite reflector is determined to maximize the Tritium breeding ratio. They will be tested in ITER if their acceptability for an installation is proved by technical feasibility and safety validation.

FT/P5-21 · Tritium Well Depth and Tritium Well Time

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Abstract: Somewhat similar to, but quite different from the xenon poisoning effects resulting from fission-produced iodine during restart-up process of a fission reactor, a complete new concept of “tritium well depth and tritium well time” is first time introduced in fusion energy research area by authors. To show what the least required amount of tritium storage is to start up a fusion reactor and how long time the fusion reactor needs to be operated for achieving the “tritium break even” during the initial start-up phase due to the finite tritium breeding time, the time dependent on the tritium breeder, specific structure of breeding zone, layout of coolant flow pipes, tritium recovery scheme and extraction process, the tritium retention of reactor components, unrecoverable tritium fraction in breeder, leakage to the inertial gas container, and the natural decay etc. Authors pointed out this new phenomenon and answered this problem by setting up and solving a set of equations, which describe a dynamic subsystem model of the tritium inventory evolution in a Fusion Experimental Breeder (FEB). Two different simulation models give almost the same results, “the tritium well depth” is about 317–319 g and “tritium well time” is approximately 240 full power days for reference case of the FEB designed detail configuration, also found that after one-year operation the tritium storage reaches 1.18 kg that is more than sufficient one to start up three of FEB-like fusion reactors.

FT/P5-22 · Concept of compact low aspect ratio demo reactor, SlimCS

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Abstract: A new concept realizing a compact demonstration reactor is presented. The reactor design named “SlimCS” adopts a reduced-size center solenoid (CS), which enables us to introduce a slender toroidal field (TF) coil system. Such a TF system decreases the reactor weight, eventually contributing to reducing the construction cost. The CS size (an outer radius of 0.7 m) is determined to have the capability of plasma shaping (triagularity of $\sim$0.4) enough to obtain high confinement in high density region and possibly to avoid giant edge-localized modes. The SlimCS concept expands the design window of fusion reactors to lower aspect ratio (A) of around 2.5 which facilitates higher elongation and higher beta access with reasonable design margins. As a result, SlimCS is as compact as advanced commercial reactor designs such as ARIES-RS and CREST, even with the assumption of relatively conservative plasma parameters. Another merit of low-A is that the first wall area on the low field side, where smaller electromagnetic (EM) force acts on disruptions, is wide compared with that of conventional-A. This means that tritium can be efficiently breeded with large blanket modules on the low field side. As a result, the demand for tritium breeding on the high field side is comparatively reduced so that small blanket modules, being robust to stronger EM force but less efficient for tritium breeding, can be arranged on the side. SlimCS is designed to produce 2.95 GW with a major radius of 5.5 m, aspect ratio of 2.6, normalized beta of 4.3, Greenwald-normalized density of 1.05, bootstrap current fraction of 0.77 and maximum field of 16.4 T. SlimCS uses technologies foreseeable in 2020’s such as Nb_3Al superconductor, water-cooled solid breeder blanket, and low activation ferritic steel F82H as the blanket structural material. Average neutron wall load is designed to be 3 MW/m^2. A major technical issue on SlimCS is non-inductive plasma current ramp because the plasma current must be raised using an overdrive with a combination of bootstrap current and non-inductive external current drive. This technique is considered to be a continuation of the steady state operation of tokamak. Another issue is a physics basis on plasma around $A = 2.6$, which can be resolved by the maximum use of a satellite tokamak NCT designed to cover $A = 2.6$–3.1.

FT/P5-23 · Transport and Stability Study of a Fusion Power Plant Scenario

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Abstract: Confinement and stability properties of plasmas in a fusion power plant have been studied for both pulsed and steady state modes of operation. The transport analysis is based on the first principle transport models. Stability analysis has been performed for ideal and resistive modes with account of ideally conductive and resistive wall stabilization. It has been found that predictions for a conventional pulsed scenario are quite conservative and either lead to a large-size device or require high pedestal temperature. These requirements can be relaxed if an advanced scenario with a reversed shear is considered. In this scenario, first a reversed shear configuration is created by an external off-axis current drive and then an

internal transport barrier (ITB) is formed. The bootstrap current driven in the ITB causes extention of the reversed shear zone with subsequent broadening of the ITB and further increase of the bootstrap fraction. A feedback control algorithm is proposed that prevents uncontrollable bootstrap growth and stabilizes it at a prescribed level. The process opens a route to a high-performance regime with steady state operation. The conditions for appearance and stability of this non-inductive scenario will be explored.

FT/P5-24 · Minimization of the External Heating Power by Long Fusion Power Rise-up Time for Self-ignition Access in the Helical Reactor FFHR2m

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Abstract: Minimization of the external heating power to access self-ignition is advantageous to increase the reactor design flexibility and to reduce the capital and operating costs of the plasma heating device in the LHD type helical reactor (FFHR2m) with the major radius of 14 m, the minor radius of 1.73 m, magnetic field strength of 6 T, and the fusion power of 1.9 GW. In this work the confinement enhancement factor of 1.92 over the ISS95 scaling and the factor of 1.5 larger than the Sudo density limit are used in one dimensional approach. Temperature and density profiles are assumed to be parabolic, and the ratio of the helium ash to the energy confinement time is assumed to be 3. During the ignition access phase the external heating power is applied to increase the density limit, and the fusion power is linearly increased by controlling the fueling rate. The peak density at the steady state is $2.67 \times 10^{20}\,\mathrm{m}^{-3}$, peak temperature is 15.8 keV, and average neutron wall loading is $1.5\ \mathrm{MW\,m}^{-2}$. Beta value is 3% which is smaller than the achieved value in LHD. As the density limit was set to 1.5 over the Sudo density limit which is less than the observed value in LHD experiments, the confinement factor required for ignition can be reduced from 2.16 to 1.92 due to expansion of the operating density regime. When the set value of the density limit margin is reduced, the feedback controlled external heating power can be reduced from 100 MW for the preprogrammed heating power of 70 MW to 50 MW for the preprogrammed heating power of 40 MW even for the shorter fusion power rise-up time of 120 sec. When the fusion power rise-up time is further extended to two hours, the external heating power can be reduced to 27 MW for the preprogrammed heating power of 25 MW. This is because the long fusion power rise-up time provides the small temporal variation of the plasma energy in the power balance equation. Thus we have discovered that a larger density limit itself, lower density limit margin together with the smaller preprogramming heating power, and over 300 s of the fusion power rise-up time minimize the external heating power to reach self-ignition. These results may open flexible design windows on the operation of the blanket, power supply, and ports for the heating power device.
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FT/P5-25 · Engineering Design of Demo-CREST and Analysis on Critical Development Issues toward Advanced Tokamak Power Plant CREST

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Abstract: CREST (Compact REversed Shear Tokamak) is one of the cost competitive power plant concepts based on a reversed shear high beta plasma and water cooled ferritic steel component. Recently, to assess the critical development issues of CREST in detail, the conceptual design of a demonstration power plant Demo-CREST has been carried out. In this report, we have discussed on engineering aspect in Demo-CREST design, and analyzed the critical development issues for advanced tokamak CREST. The power flow and power plant system are investigated for the improvement of the thermal efficiency in Demo-CREST, and based on this result and the previous report, the major design analyses for Demo-CREST and CREST are completed. Our development scenario is attractive due to early realization of electric power generation and two steps development from ITER to CREST shown, but in compensation for the attractiveness, there are critical issues to be resolved after ITER. In the demonstration phase of Demo-CREST, there are additionally clear technological gaps from ITER on the divertor performance, B_{tmax}, NBI beam energy, and neutron fluence to first-wall material. In the development phase of Demo-CREST, improvement of beta, plasma density and thermal efficiency will be critical issues in comparison with the present ITER experimental plan. Those critical issues are quantitatively analyzed.

FT/P5-26 · The ARIES-CS – A Compact Stellarator Power Plant

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Abstract: Stellarators have many desirable features as fusion power plants. A detailed and integrated study of compact stellarator configurations, ARIES-CS, was initiated recently to advance our understanding of attractive compact stellarator power plants and to define key R&D areas. The first goal of the S study was to develop configurations which are similar in size with advanced tokamaks as earlier stellarator power plant studies led to devices with large sizes ($\langle R \rangle$ = 14–20 m). This was achieved by a) developing configuration with a low plasma aspect ratio, and b) minimizing the space between plasma and the coils. We focused on quasiaxisymmetric (QA) configurations as they are able to operate at a relatively low plasma aspect ratio ($\sim$ 4–5). Three distinct classes of QA configuration is considered. First is NCSX-class configuration. Three variants of this configurations have been developed to explore trade-off between confinement of fast á particles and linear MHD stability. Progress has been made to reduce loss of á particles to $< 5\%$. It appears that the introducing a bias in the principle mirror term in the magnetic spectrum, plays an important role in reducing á particle losses. Second is MHH2 which aims that developing a very low aspect ratio geometry ($\sim$2.6) with relatively simpler coils. Third, SNS, is aimed at a configuration with excellent flux surface quantity and nearly flat rotational transform. In latter two cases, strict adherence to linear MHD stability is deemphasized. To reduce the blanket-coil spacing, a novel approach was developed in ARIES-CS in which the blanket at the critical areas of minimum stand-off is replaced by a highly efficient WC-based shield. In principle, by utilizing the shield-only region in strategic areas, we have been able to reduce the minimum stand-off, $\Delta_{\min}$, by -30% compared to a uniform radial build that was assumed in previous studies. The reduced $\Delta_{\min}$, together with the lower aspect ratio plasma lead to power plants that have similar size as advanced tokamak designs ($\langle R \rangle$ = 7–8 m). The device configuration, assembly, and maintenance procedures appear to impose severe constraints: Two distinct approaches were developed. Modular coils are designed to examine the geometric complexity and to understand the constraints imposed by the maximum allowable field, desirable coil-plasma separation, coil-coil spacing, and other coil parameters.

FT/P5-27 · Highlights of the Physics and Technology for the Ignitor Experiment

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Abstract: The highlights of the most recent developments in the physics and technology of the Ignitor program are presented. These include the investigation of quasi-stationary regimes close to ignition obtained with reduced plasma parameters with high duty cycles and of ignition achieved with “divertor” configurations and the H-regime (most recent scalings). The ICRH system, the plasma control system, the high speed pellet injection construction, the remote handling system, the fabrication procedures of toroidal magnet plates, the plasma chamber and first wall system design advances and the site analysis are among the engineering highlights.

FT/P5-28 · Component Testing and Materials Development for Fusion Applications using Materials Test Reactors

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Abstract: Fusion power plant operation will strongly depend on the economy and reliability of numerous components such as tritium breeding blankets and divertors. They must be fabricated in large quantities based on demonstrations with a limited amount of test beds. 14 MeV source such as IFMIF and a volumetric source will support the demonstration, but such sources have limitations in irradiation volume and control. Materials Test Reactors, MTSR's, are presently used for the development and demonstration, but the fission neutron spectrum is not completely representative for all fusion conditions. Nevertheless the MTR's offer today and in the future room for the demonstration of the fusion power plant components and advanced materials not directly exposed to 14 MeV neutrons alone. New MTR technology might even extend the test bed function for components and materials for fusion applications.

FT/P5-29 · Helium Permeability of SiC/SiC Composite Used for Blanket First Wall

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Abstract: Low activation materials such as ferritic steel, vanadium alloy and SiC/SiC composite are used for structure materials of blanket modules. High energy conversion efficiency is expected in the SiC/SiC composite blanket since the operation temperature is high, 1100 K. Helium gas is employed as the coolant in the SiC/SiC composite blanket, so that the concern is a leak of helium gas into plasmas. The leak rate of helium gas has to be lower than a helium production rate by fusion reactions. For this purpose, the helium permeability was measured at different helium gas pressures for SiC/SiC composites made by several methods, by using a vacuum device consisting two chambers. The helium gas permeability of SiC/SiC composite made by NITE process was very low, so that the helium leak can be suppressed if a small pumping system is attached to the blanket module. The other concern is an increase of the permeability due to heat loads during start-up and shut-down phases. The helium gas permeability was measured for the SiC/SiC composite after the heat cycles. The highest temperature, heating rate and cycle number were 1400 K, 10 K/s and 120, respectively. No increase in the permeability was observed when the temperature was lower than 1200 K at the highest heating rate and cycle number. The present results show that the blanket module can be made by using the SiC/SiC composite only, from a view point of helium gas permeability.

FT/P5-30 · Interaction of Beryllium Oxide with Hydrogen Plasma

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Abstract: Beryllium is considered a candidate material for the first wall of nuclear fusion plasma devices, e.g. ITER. Therefore, the interaction of beryllium oxide with a D/T plasma is important with respect to beryllium as oxygen getter and the influence of beryllium oxide on the retention of D and T in the first wall. Since it is not yet fully clear if and how much oxide will develop during operation, we investigated the interaction of beryllium samples with hydrogen plasma. Exposure of beryllium samples with native oxide layers and pre-oxidized samples to hydrogen plasma resulted in an oxide layer of similar thickness, suggesting the development of an equilibrium state during exposure. In order to determine the relation between equilibrium oxide layer and oxygen concentration in the plasma, the samples were exposed to hydrogen plasma with various additions of water vapour. Oxygen content was monitored with an optical spectrometer. Further experiments will aim at the dependence of hydrogen retention on the oxide layer thickness.

FT/P5-31 · Tokamak KTM Complex for Material Investigation

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Abstract: The experimental complex on the basis of spherical tokamak KTM is intended to study and test the first wall materials and designs of divertor plates constructions of reactor-tokamak, to try out the methods for reducing loads at a divertor and various methods of heat- and energy removal, as well as the methods of fast exhaust of divertor volume and development of the methods to prevent failures of intrachamber components. Parameters of energy loads and wide range of the used methods and diagnostics allow for studies and testings in the divertor space and at the first wall, which will be of great importance for the study of plasma facing materials in ITER and DEMO project, as well as for other experimental and power fusion reactors. The paper contains basic parameters of spherical tokamak KTM, test bench for simulations, plasma-physical and material test tasks and KTM project status activity. The activities of 2007–2008 are aimed at completion of the tokamak construction, and commissioning of the facility, including delivery of the equipment to KTM site, installation of the complex technological systems, mounting of impulse supply system, and commissioning of the facility. The potentials are being discussed to use the tokamak KTM as a physical prototype for the compact fusion power unit with warm electromagnetic system.

FT/P5-32 · Ferritic Insertion for Reduction of Toroidal Magnetic Field Ripple on JT-60U

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Abstract: Ferritic steel plates have been installed to improve the energetic ion confinement through reducing a toroidal magnetic field ripple. Aiming at cost-effective installation, orbit following calculations of energetic ions were performed for a design of ferritic installation on JT-60U by using the Fully three Dimensional magnetic field Orbit-Following Monte-Carlo (F3D OFMC) code, which was developed for ferritic insert experiments on JFT-2M and can treat complex magnetic field structure produced by ferritic inserts. The installed ferritic steel adds the non-linear magnetic field on magnetic sensors for a plasma control and an equilibrium calculation. We have successfully carried out a real-time plasma control which takes the magnetic field by ferritic steel into account for the first time. The heat load measurement indicates the improved confinement of energetic ions. These results are important for practical application of ferritic steel which is a leading candidate of a structural material on a demo reactor.

FT/P5-33 · Radiation Damage in Reduced Activation Ferritic/Martensitic Steels for Fusion Reactors: a Simulation Point of View

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Abstract: Reduced Activation Ferritic/Martensitic (RAFM) are good candidates for the future fusion reactors, for, relative to austenitic stainless steels, their lower damage accumulation and moderate swelling under irradiation by the 14 MeV neutrons produced by the deuterium-tritium fusion reaction. Irradiation of these steels is known to degrade mechanical properties, starting at the lowest doses. In this paper an overview on simulations studies of irradiation-induced damage accumulation and their impact on the mechanical properties is presented. In general these simulations describe so-called model alloys, as available theoretical tools cannot at this point describe real materials with their actual chemical composition. RAFM steels are typically modelled by pure Fe, as it constitutes their basis and all share the same crystallographic structure. Molecular dynamics simulation results indicate that He, either in the form of He bubbles or in solid solution, does not have a direct effect on the mechanical properties, but an indirect one, by promoting the formation of other types of defect clusters. This information is then carried to the next scale, dislocation dynamics simulations. Multiscale approach proves to be fruitful in its ability to broaden our understanding of materials response to irradiation, but care should be taken by accounting for approximations at all implied space and time scales and methods.

FT/P5-34 · Development of V-Cr-Ti Type Alloys with Small Additives for Advanced Fusion Applications

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Abstract: Advanced approach to develop a practically tough material was demonstrated in recent research activity of low activation vanadium alloys for fusion applications. It was shown that modification by small addition of yttrium to the V-4Cr-4Ti alloy improved integrity of their material performance. Vanadium alloy has been considered as a blanket structure material for fusion reactor applications because of their promising properties such as low induced radioactivity, high-heat-loading capability and compatibility with coolant materials like lithium. The critical issues to use vanadium alloys as blanket structure material in lithium self-cooled and breeding system include development of insulator coating for reducing magneto-hydrodynamic (MHD) pressure drop and filling up material performance database of the vanadium alloys. Significant progress has been made in development of the MHD coating and material database of vanadium alloys. In this paper, successful development of the vanadium alloys by means of advanced melting process and their material performance database will be described. Gaseous impurities such as oxygen in vanadium alloy affect hardening very much especially when radiation induced defects exist. Initial impurity levels were successfully reduced by modification of melting process. Purified V-4Cr-4Ti alloy have demonstrated improvement in the material properties including workability, high temperature strength, ductility and weldability. It has been shown that the impurity levels of the V-Cr-Ti alloys can be controlled by small additives such as yttrium by means of their scavenging effect. Tensile properties after neutron irradiation of the modified V-Cr-Ti alloys showed remarkable performance. Yttrium addition will reduce oxygen level by oxide slug formation on the melting ingot surface. Two possible mechanisms of the reduction were proposed, that is, suppression of oxygen penetration into molten metal at the surface and removal of oxygen from inside by forming Y_2O_3. A high-purity 15 kg-ingot of V-4Cr-4Ti-0.15Y alloy was made

by a levitation melting method. It was demonstrated that the melting process successfully completed. Concentration of interstitial impurities in the ingot reached adequately low levels.

FT/P5-35 · Free-Surface Fluctuation at High Speed Lithium Flow for IFMIF

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Abstract: An intense neutron source like IFMIF, requires very high speed, and very stable free surface lithium flow as the beam target. The experimental study of the flow target was initiated in 2002 with installing the test section to the lithium loop facility at Osaka University. The free surface test section, which is stainless steel 304 made 70 mm-width flow channel, is aligned horizontally and produces Li jet flow of 10 mm-depth at temperature of 300°C. Flow characteristic of the present loop was tested, and velocity of 15 m/s was attained under 0.1 MPa of Ar cover gas. The flow velocity in lower pressure condition is limited to a half by cavitation at the pump. Surface fluctuations, mainly caused by the waves, were measured with using a high-dynamic range ccd cameras and with scanning by an electro-contact probe along the designed beam axis. It was scanned along the perpendicular direction to the flow for the velocity range of 1–15 m/s. Frequency of the contacts were measured. It was found that the contact frequency increased with decreasing the probe height, and reached maximum near the normal thickness of 10mm. And then the frequency decreased down to 0, i.e. full contact of the needle and the liquid. The average thickness was defined as the height that the contact frequency had the maximum, and the wave amplitude as the height difference between no-contact and full-contact. The amplitude was found to grow with an increase of the velocity, and reached 2.2 mm at 15 m/s. At this velocity, however, the histogram shows that waves with larger amplitude than 1mm, which is IFMIF specification, were limited less than 10% in probability. These surface wave property was compared with the linear stability theory on a shear layer underneath the surface, and shows good agreement. Variation of the average thickness to velocity is examined from the point of view of surface wakes generated at the corners between the nozzle edge and side walls. Considering these results, engineering design of the IFMIF Li target is in progress.

FT/P5-36 · IFMIF Target and Test Cell – Towards Design Integration

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Abstract: The International-Fusion-Material-Irradiation-Facility (IFMIF) is an accelerator driven neutron source for irradiation tests of candidate fusion reactor materials. Two 40 MeV deuterium beams of 125 mA each will hit a flowing liquid lithium jet target, producing high energy neutrons up to 55 MeV at a rate of about $1 \times 10^{17} s^{-1}$. Those neutrons will penetrate the target back wall made of a thin Eurofer plate. In the attached High Flux Test Module (HFTM), a testing volume of 0.5 litres filled by qualified small scale specimens will be irradiated at displacement rates of 20–50 dpa/fpy in structural materials. The HFTM will also provide helium and hydrogen production to dpa ratios that reflect within the uncertainties the values expected in a DEMO fusion reactor. The Medium Flux Test Module (MFTM) comprises devices for in situ creep-fatigue and tritium release experiments, as well as tungsten spectral shifter or reflector plates. Farther down-stream the low flux region will provide irradiation tubes for additional material irradiation at lower fluence levels. The objective of the present paper is to present the progress achieved in the design integration of the Target and Test Cell of IFMIF. First, work is reported on collecting and harmonizing the CAD designs provided by various international groups involved in the IFMIF Target and Test Cell development. Second, further efforts devoted to the general nuclear layout of the Target and Test Cell are described, taking into account nuclear calculations of responses such as the nuclear heating, the activation inventories, and dose rates based on most advanced nuclear data and calculational procedures. Finally, results of an extensive study are presented on the cooling capabilities of the Target and Test Cell by natural convection.

FT/P5-37 · Development of DEMO Divertor with Reduced Activation Ferritic/Martensitic Steel (F82H) in JAEA

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Abstract: JAEA has been developing the divertor for DEMO reactors with Reduced Activation Ferritic/Martensitic steel, F82H, which has been developed by JAEA. This paper reports recent three remarkable R&D achievements on the DEMO divertor: 1) High performance divertor cooling tube, so called a screw tube, made of F82H instead of Cu-alloy achieves 1.5 times higher incident critical heat flux (CHF) than a F82H smooth cooling tube, 2) Thermo-mechanical analyses reveal that a dovetail divertor support structure can reduce thermal stress by 30% in a cooling tube with a full scale divertor mock-up, 3) A divertor with Ni-coated tungsten pin armors, which is newly developed bonding technique, withstands more than 3000 thermal-cycles at a heat load of 5 MW/m^2.

FT/P5-38 · Structural Materials for Fusion Power Reactors - the RF R&D Activities

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Abstract: Resent progress in the RF Structural Materials and Technologies R&D road map toward Fusion Demonstration Power Reactor DEMO via fast breeder reactors (FBRs) tests (BOR-60, BN-600) and test blanket modules (TBMs) DEMO in ITER is overviewed. The functional properties of the RF reduced activation ferritic-martensitic steel (RAFMS) RUSFER-EK-181 (Fe-12Cr-W-Ta-V-B) and V-4Ti-4Cr alloys are presented. The RAFMS, vanadium alloy and their semi-finished products can be manufactured on industrial scale with sufficient quality and their welding is also established. The structure materials and process technologies are mostly ready for the TBMs applications (300–500 kg/TBM). The next important steps to DEMO reactor include further studies of high dose irradiation and high temperature effects on mechanical properties of base structural materials and joints. The RF nuclear data library ACTDAM-AB is presented and the activation and transmutation behaviour of the RUSFER-EK-181 and V-4Ti-4Cr alloy in BN-600 and DEMO-RF (Kurchatov institute project) neutron spectra irradiations are calculated (IAEA code FISPACT-3.0(5)) and compared (dpa, H/dpa, He/dpa). Further progress is anticipated for the design of ITER TBMs. The results of the application of the internal friction method as undestroyed method to research the mechanisms of the low temperature embrittlement (the DBTT shift) of structural materials are presented and discussed in the correlation with results of the destroyed impact method. The important influence of the boron on the heat resistance properties of the structural materials and the concentration levels of boron and helium under irradiation are calculated and discussed. The special regimes of the heat treatments of the RAFMS and vanadium alloys are suggested to widen the temperature windows of applications. The results of the BOR-60 examinations (microstructure and mechanical properties) of RUSFER-EK-181 (irradiation temperature 320–330°C and doses up to 15 dpa) are presented. The BN-600 projects for the high doses (up to $(3\text{–}6) \times 10^{23}$ n/cm^2, E>0) and high temperature irradiation tests of the RAFMS and vanadium alloys are presented and suggested for international collaboration.

FT/P5-39 · Conceptual Design of Laser Fusion Reactor KOYO-F Based on Fast Ignition Scheme

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Abstract: We conceptually designed the Laser Fusion Power Plant KOYO-F 1) to make a reliable scenario for the fast ignition power plant basing on the latest knowledge of elemental technologies, 2) to identify the research goal of the elements and 3) to make the critical path clear. We have examined the design windows and the issues of the fast ignition laser fusion power plants. KOYO-F has 4 modular reactors driven by a 1.1 MJ compression laser and a 0.1 MJ heating laser operated at ∼16 Hz. The total electric power is 1.2 GW. In this the activity, we newly evaluated the gain curve for FI basing on latest simulation technology. Cooled Yb-YAG ceramic that had a potential to construct the electricity to optical efficiency of >15% was newly chosen as the laser material for the compression and heating lasers. New reactor scheme for a liquid wall, which yielded no stagnation point of ablated gas, and a rotary shutter system to protect the final optics were proposed. We have proposed the free fall cascade liquid chamber for cooling surface quickly enough to several Hz pulses operation by short flow path. The chamber ceiling and laser beam port are protected from the thermal load by keeping the surface colder to enhance condensation of LiPb vapor.

FT/P5-40 · New Concept of Laser Fusion Energy Driver Using Cryogenic Yb:YAG Ceramics

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Abstract: Laser material development is one of the most significant research issues for laser fusion energy driver developments. We propose cryogenic Yb:YAG ceramics as a novel laser driver material instead of conventional Nd:glass. Our characterization research on the cryogenic Yb:YAG revealed that both of the stimulated emission cross section and the thermal conductivity can be tuned well by controlling the material temperature. Also, using our diode-pumped oscillator with a cryogenic Yb:YAG disk, the highest optical-optical slope efficiency of 90% has been demonstrated, which has resulted in less heating of the laser material. Using the obtained laser parameters in our experiments, a 1.3 MJ, 16 Hz diode-pumped laser system has been conceptually designed with the cryogenic Yb:YAG ceramics. The overall electrical-optical conversion efficiency is numerically calculated as high as 16%. The compact main amplifier cluster with $20 \times 20 \times 8\,\mathrm{m}^3$ volume size would be realized by using the active mirror architecture.

FT/P5-41 · Integrated Modeling of DEMO Scenarios by the CRONOS Suite of Codes

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Abstract: The CRONOS suite of codes, which is being developed at CEA-Cadarache and that includes self-consistent kinetic computation of the fusion-born alpha particle distribution, time evolution of the various current sources (bootstrap, rf-driven, NBI-driven) and precise calculation of radiation losses, is used to make a full analysis of the former general scenarios established by EFDA for the design of DEMO. Since high temperatures are expected in these plasmas, electron cyclotron (EC) radiation may be an important cooling mechanism, and therefore a model which takes into account the non-local effects of this radiation, i.e. emission and re-absorption by the plasma itself, has been used. Results show how the expected inductive scenario of DEMO is well simulated, leading to $\mathrm{T_{e0}} = \mathrm{T_{i0}} = 50$ keV with pedestal temperature $\mathrm{T_{ped}} \sim 10$ keV. The fusion power obtained is of the order of 4 GW, leading to 1 GW net electrical power. In spite of the fact that in these scenarios the operating density is above Greenwald density, $\mathrm{N_e/N_{gw}} = 1.2$, and that Bremsstrahlung radiation is enhanced in this regime owing to relativcistic effects, EC radiation tends to be the main radiation cooling mechanism for the electrons in the plasma core, even when the optimistic reflection coefficient $\mathrm{R} = 0.8$ is used throughout the simulation. Therefore, the inclusion of a satisfactory model for the EC radiation is important for the correct analysis of DEMO scenarios. In fact, a local treatment of EC wave power losses based on Trubnikov formulae, as is generally adopted, does not correctly account for the EC radiation profile, underestimating the re-absorption of the outer part of the plasma, which may even lead to a net energy transfer from the core to the edge.

FT/P5-42 · Modelling of DEMO Core Plasma Consistent with Edge Parameter Scaling from Edge Modelling for DEMO, ITER and JET

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Abstract: Simulations of prototypical DEMO parameters are presented, for which the plasma core is calculated self-consistently with the scrape-off layer/divertor region using the Integrated Core Pedestal SOL (ICPS) Model in Astra. The model has previously been calibrated against JET and AsdexUG discharges and shown reasonable agreement. It has been applied to simulations of the ITER opreating space. Special emphasis is here given to operation with seeded impurities so that enhanced radaiation lowers divertor conditions to tolerable levels.

FT/P7-1 · Development Progress of the KSTAR Superconducting Magnet and Magnet Interface

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Abstract: There has been remarkable development progress on the superconducting (SC) magnet and the systems, helium refrigeration system (HRS) and current feeder system (CFS), those are required for KSTAR SC magnet operation. Total 18 toroidal field (TF) coils, including a prototype coil (TF00) for the performance test and one another spare coil (TF17), were already fabricated and assembled 16 coils on site. Among 7 pairs of poloidal field (PF) coils, 4 pairs were fabricated and the others have near at hand. All the fabricated coils were certified the quality through the inspections, critical current density measurement of

the SC strand specimens, three dimensional measurement after the vacuum pressure impregnation (VPI), electrical leakage current test under high electrical potential, helium channel flow rate test, and so on. It was launched the 9 kW HRS to Samsung Corporation and Air Liquid. Based on the KSTAR requirement, it was completed the engineering design and under the status of fabrication processing. It will be completed the assembly and commissioning of HRS itself within middle of 2007. Another system, the CFS, is needed in charging SC magnet. It consists of current leads (CL), current lead box (CLB), SC buslines, and those own vacuum pumping system (VPS), helium control system (HCS), and I&C. There are two CLBs and buslines for PF coils and TF coils, respectively. Two CLBs and VPS are already developed and assembled on site. Thermal shields of the two boxes were cooled down below 100 K using liquid nitrogen for the certification. Two pairs of prototype leads for PF coils and TF coils had developed and tested up to the currents two times higher than optimum design values (26 kA for PF leads and 35 kA for TF leads). It was launched 9 pairs of leads, HCS, and SC buslines for the first plasma experiments and it will be fabricated and assembled within February, 2007, at least.

FT/P7-2 · Progress in the Heating System Development Towards a Long Pulse Operation in KSTAR

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Abstract: Assembly of the KSTAR tokamak will be completed in 2007, followed by a series of first plasma experiments and the main long pulse operation period. The ICRH and NBI systems are expected to play important roles during a long pulse operation period, through a selective heating of ions and electrons, controlling the pressure and current profiles, a core fueling, and a beam diagnostics for the KSTAR. In addition, ICRH contributes to the first plasma experiments by way of a discharge cleaning and an assisted tokamak startup. In this presentation, recent achievements in the development of ICRF and NBI heating systems are shown with an emphasis on the eventual long pulse operation as well as the first plasma experiments. The four-strap ICRF antenna, which has the long pulse capabilities up to 300 s, has been successfully tested up to 41 kV at the vacuum test chamber. A preliminary experimental study on ICRF assisted discharge cleaning system and startup experiments will be carried out during 2006–2007 on ASDEX-U which is similar in its torus dimensions to KSTAR. The neutral beam injector was designed to extract a 120 keV, 65 A deuterium beam with a rectangular beam size of 12×45 cm^2 and we obtained the preliminary result of hydrogen ion beams of 100 keV, 24 A, for a 3 sec pulse mode. In addition, a long pulse operation for 200 seconds at a 1 MW beam power has also been tested successfully as one of the important milestones and new records are still being complied from recent experiments. Technical descriptions on the developed prototype KSTAR neutral beam heating system and up-to-date experimental results will be reported.

FT/P7-3 · Present status of the tokamak T-15 and further plans of its update

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Abstract: At present time the tokamaks T-10 and T-15 located in the Nuclear Fusion Institute of Russian Research Centre “Kurchatov Institute” form the basis of fusion experimental researches in Russian Federation. Tokamak T-15 being possessed by the unique superconductive coils is one of the biggest world tokamaks. Design objectives and characteristics allow the scientific and engineer programs aimed at solving of the ITER problems to be realized. Performed earlier experiments on the T-15 tokamak contributed significantly to perfection of the superconductive technology, development of the plasma diagnostics methods and the powerful plasma heating systems including the superhigh frequency and neutral beam injection heating. Modernization of the tokamak T-15 with the purpose of creation of the ITER-like magnetic configuration and installation of the divertor system into the vacuum chamber is assumed to be able to provide for further successful participation of RF in the experimental investigations in the tokamak field. Divertor configuration makes specific demands of the magnetic system of tokamak. Obtaining of the most effective physical regimes with high plasma performance is possible in a magnetic configuration with high plasma elongation K and triangularity. For creation and control of the elongated magnetic configuration of the T-15 tokamak, a modification of existing system of the poloidal magnetic fields is provided. Tokamak T-15 is supposed to be the Russian fusion researches center joining a scientific and technical potential of different scientific laboratories and providing for a wide investigation spectrum during 10-15 years. The experimental investigations on the tokamak T-15 will keep and develop Russian scientific-research fusion school and will provide manpower training for ITER and later on for DEMO.

FT/P7-4 · Design Optimization for Plasma Control and Assessment of Operation Regimes in JT-60SA

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Abstract: Design of modification of JT-60U, the JT-60SA device with superconducting coils, has been optimized in viewpoint of plasma control, and operation regimes have been evaluated. JT-60SA is characterized by enhanced flexibility in the aspect ratio (A) and the plasma shape. The beam lines of negative-ion NBI (10 MW, 500 keV) were shifted downward for off-axis current drive, in order to obtain a weak or reversed shear plasma that is required for high stability and high confinement. A stablizing plate and active feedback coils are prepared for the resistive wall mode (RWM) control to maintain high normalized beta exceeding the no-wall stability limit. The position of feedback coils has been determined by the RWM analysis, so that they will be located along the edge of large port holes in the stabilizing plates for efficient coupling. The regime of full current drive operation has been extended with upgraded heating and current drive power (41 MW for 100 s). Full current drive operation for 100 s with reactor-relevant high values of normalized beta (4.4) and bootstrap current fraction (70%) is expected in a highly-shaped low-aspect-ratio configuration (A = 2.65, plasma current of 2.4 MA, electron density of 86% of the Greenwald density, $HH_{y2} = 1.3$). In an ITER-like plasma shape with A = 3.1, high normalized beta (3.1) ELMy H-mode plasmas with an ITER-relevant high density of 9.1×10^{19} m^{-3} are expected at a plasma current of 3.5 MA. A large fraction of electron heating power (∼55%) is obtained with high power N-NBI and ECRF (7 MW) heating.

FT/P7-5 · Engineering Feature in the Design of JT-60SA

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Abstract: The construction of the JT-60U modification (JT-60SA) is now being planned by both parties of Japan and EU as a part of the ITER Broader Approach. It means that a new function of the ITER satellite tokamak is clearly defined as a mission on the JT-60SA program, and it has a big impact on the engineering design of machine, plasma heating and current drive devices and power supplies. The magnetic coils and vacuum vessel were designed to ensure flexibility in the plasma shape (aspect ratio A = 2.6–3.1). The NbTi superconductor will be adopted to generate the central magnetic field of 2.7 T at R = 3.0 m. A large flux swing capability of about 38 Wb is necessary to sustain an ITER-relevant high density plasma at fGW of about 0.85, 3.5 MA for 100 s, though about half of the plasma current will be driven by non-inductive current drive method. A new CS conductor was designed to generate 10 T maximum so as to sustain such plasma for 100 s. An optimization of neutron and radiation shielding was done on the vacuum vessel and cryostat structures under the neutron yield of about 2×10^{19}/shot produced by the plasma heating systems (NBI and ECRF) of 41 MW, 100 s. The estimated nuclear heating values at the inboard and outboard legs of TF coil conductor using 1D code of THIDA-2 are 0.23 mW/cc and 0.15 mW/cc, respectively. The semi-closed vertical divertor with flatter dome was adopted for flexibility of plasma shaping. The mono-block type CFC divertor armor is being planned to withstand heat load of 10–20 MW/m^2. Each of cryo-panels installed under the flatter dome and the outer baffle plate has a strong pumping capability of about 50 m^3/s. A new AC power system combined with a power grid and the existing MG set was deigned to satisfy the total energy of 12.9 GJ for the heating systems of 41 MW, 100 s. This paper intends to clarify the latest design option of the JT-60SA.

FT/P7-6 · Integrated Software Development for Wendelstein 7-X

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Abstract: Steady-state fusion devices, such as Wendelstein 7-X, pose a new challenge to the software capabilities for control, data acquisition and data analysis. These modules must cope with huge amounts of data. Proof-of-principle demonstrations of important steps in a physically motivated framework are shown to be feasible. This includes integrated data analysis from different diagnostics and coupling of modeling codes, even for advanced device control. The present approach makes use of a strict modular design and already allows the use of modules in present design and optimization studies of device components and diagnostics setups. A striking benefit is provided by embedding validated legacy codes within a service oriented software architecture, thereby ameliorating incompatibility issues.

FT/P7-7 · Design, Analyses and R&D for EAST In-vessel Components

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Abstract: In-vessel components are important parts of EAST superconducting tokamak. It is include plasma facing components, passive plates, cryo-pumps, in-vessel coils, etc. The structure optimized design, analyses and related R&D has been complete. Divertor is up down symmetric to accommodate both double null and single null operation. Passive plates can supply around 100 ms time constant during plasma vertical movements. In-vessel coils are used for plasma vertical movements active control. Each cryo-pumps can supply around 45 m^3/s pumping speed for particle exhaust. Analyses shows when 1 MA plasma current disrupt in 3 ms EM loads caused by halo current ($I_{\rm halo} = 25\% I_{\rm p}$, TPF = 2) will not bring unacceptable stress on divertor structure. Graphite tiles bolted to heat sink. Cooling channel consist of holes on heat sink. The bolted divertor thermal structure can sustain 2 MW/m^2 up to 60 s operation if we limited first surface temperature 1500°C. Thermal testing and structure optimised testing have been made to demonstrate the analyses result. All the in vessel components are under fabrication and components for first plasma will be complete around July 2006.

FT/P7-8 · The Design and Testing Result of TF Power Supply System of EAST Tokamak

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Abstract: This paper analyses the design project of control system and protection system of toroidal field (TF) power supply system of EAST tokamak, and the main parameters of TF power supply system are selected. Through the first experiment of EAST tokamak, the testing curves of control system and protection system demonstrate the validity of this design project.

FT/P7-9 · Design, fabrication and testing results of vacuum vessel, thermal shield and Cryostat of EAST

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Abstract: The EAST (Experimental Advanced Superconducting Tokamak) is an advanced steady-state plasma physics experimental device, which has been approved by the Chinese government and being constructed as the Chinese national nuclear fusion Research project. Vacuum vessel is the location for the operation of plasma as one of the key component for EAST device. During it operation the vacuum vessel will not only endure the electromagnetic force due to the plasma disruption and Halo current but also the pressure of boride water and the thermal stress owing to the 250°C baking out by the hot pressure nitrogen gas or the 100°C hot wall during plasma operation. The cryostat is a large single walled vessel surrounding the entire Basic Machine with central cylindrical section and two end enclosures, a flat base structure with external reinforcements and dome-shaped lid structure. It provides the thermal barrier with the base pressure of 5×10^{-4} Pa between the ambient temperature testing hall and the liquid helium cooled superconducting magnet. The thermal shields comprise the vacuum vessel thermal shield (VVTS), between the vacuum vessel and the cold TF coil structures, the cryostat thermal shield (CTS), covering the walls of the cryostat, thereby preventing direct line of sight of the room temperature walls to the cold structures, the vacuum port thermal shields (VPTS) that enclose the port connection ducts. The thermal shields are made of double-wall panels, sandwich structure consist of two stainless steel panels and weld quadrate cooling pipe in between the total surface of the thermal shields is about 351 m^2. This paper is a report of the structure design and mechanical analyses on the vacuum vessel, thermal shield and cryostat. According to the allowable stress criteria of ASME, the maximum integrated stress intensity on these key components is less than the allowable design stress intensity 3 Sm. The fabrication for these components was completed in 2004 and has been installed in the position since the end of 2005. The first cooldown of the tokamak was carried out recently. In this report some key R&D and testing results have been presented, which included supporting system and the assembly of the whole vacuum vessel, thermal shield and cryostat.

FT/P7-10 · Superconducting Toroidal Field Magnet System for EAST Device

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Abstract: The EAST (HT-7U) toroidal field (TF) Coil is a large (3.5 meters high, 2.5 meters wide and 0.37 meters thickness), approximately “Dee” shaped , superconducting (NbTi/Cu) coil, and it is wound using the square cable-in-conduit (CIC) type conductor, which is based on UNK NbTi wires made in Russion and cooled with supercritical helium. At present, the design, fabrication and testing for the TF coils in EAST device have been finished. Meanwhile, all of TF coils have been installed in the position, and the first cooling down of the EAST device and the first discharge test for the TF system has been carried out. This paper describes the design and fabrication of TF conductors, the design and manufacture of TF coil and the assembly of TF coils, and besides, the acceptance test of the TF coil and the first discharge test for the TF system are given in detail.

FT/P7-11 · The Engineering commissioning of EAST Superconducting Tokomak

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Abstract: The assembling of EAST device was completed by the end of 2005. Just after that the first commissioning started. The highest vacuum in cryostat reached 3.8×10^{-5} Pa. The 2 kw/4.5 K cryogenic system is tested successfully and all of coils were cooled down to 4.5 K. After that total 260 shots have been energized. The longest TF current duration was 5000 seconds and the highest TF current was 8200 A (2 T). All the testing results showed that the EAST machine and its sub-systems have been successfully built. The installation of in vessel components and improvement of sub-system were started after the device warm up and will be completed in the middle of 2006. It is planed that the second commissioning operation will be started this autumn. The main items to be carried out in the second run experiment are the full current excitation of each coil and obtain the first plasma. The successful EAST commission gives EAST team the confidence: EAST will be successfully constructed and start operation in this year when it will provide fusion community a very good research facility for steady state divertor plasma research.

SE

Safety, Environmental and Economic Aspects of Fusion

SE/P2-1 · Is Fusion Research Worth It?

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Abstract: Long-term energy R&D such as fusion needs to be valued as a "real option". The R&D itself does not provide energy, but rather the option to construct and operate energy-producing systems. An initial analysis of this problem [Goldenberg, Linton, Nuclear Fusion R&D, Energy Risk, 2006] used an inversion of the Black-Scholes formula to take explicit account of fluctuations in the real value of energy. That study concluded that for reasonable assumptions about the operating cost of fusion power plants, the fusion option was cost effective. Here we use a simpler estimate of the future value of energy, but look more carefully at the question of the opportunity cost of engaging in fusion R&D. We find again that fusion research is a good investment.

SE/P2-2 · Safety and Economical Requirements of Conceptual Fusion Power Reactors in Co-existing Advanced Fission Plants

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Abstract: An EPR fission plant is expected to operate from 2010 to 2070. In this time range a new generation of advance fission reactors and several stages of fusion reactors from ITER to DEMO will emerge. Their viability between the competitive environment and also their possible synergy benefits are discussed in this paper. The studied cases involve the Finnish EPR, Generation IV, and the EFDA Conceptual power plant study Models A-D. Cross-cutting issues of technologies are investigated including also supercritical water for cooling. The main concentration is on economic and safety assessments. We have reviewed several studies on the economic potential of both conceptual fusion power plants and those of Generation IV candidates. As a reference level we have used the present Finnish EPR. Comparison using various pricing methods are being studied for fusion and Generation IV: the mass flow analyzes together with engineering, construction and financial margins exhibit one method and another one on simple scalings between components or structures with common technology level. In all these studies fusion competitiveness has to be improved concerning plant availability and internal power recirculation. Present best fission plants have plant availability well above 90% and internal power circulation of the order of 3–4%. The operation and maintenance solutions of Model C and D show the right way. A remarkable rise of the fuel costs of present LWRs would make at first the Generation IV breeder options more competitive and thereafter fusion plants. We shall also discuss the costs of safety related components, like containment and equipment for severe accident mitigation such as the core catcher in a LWR, and to what extent the inherent fusion safety features could compensate such expenses. For an overall assessment of the various nuclear options considered we have used the results including both internal and external costs. Due to the long lead times we have used fuzzy optimization methods for assessing their viability. The biasing of various risk factors and human needs, indicates proper directions towards more acceptable NPP candidates.

SE/P2-3 · C-14 Production in CTR Materials and Blankets

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Abstract: A consecutive study of the source terms, specific and total production of C-14 as the major contributor to the external costs of fusion was performed by neutron activation analysis of low activation structural materials, coolants and breeders suggested for future power fusion reactors. It shows that the specific C-14 activity induced in the materials of interest is significantly dependent upon the assumption for nitrogen content. Gas-cooled, water-cooled and lithium self-cooled blanket concepts were considered from the C-14 production point of view. A comparison of the C-14 activity induced by CTR blankets and by natural and artificial sources as nuclear tests and power fission reactors is given in the report. It is recommended to minimize the nitrogen content in beryllium and in the low activated structural materials below 0.01 wt %. Then due to environmental and waste disposal reasons C-14 generation in CTR will have negligible impact on the cost.

Author Index