22nd IAEA Fusion Energy Conference

Book of Abstracts

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*** Magnetic Confinement Experiments ***

EX/1-1 · Development of Reversed Shear Plasmas with Large Bootstrap Current Fraction towards Reactor Relevant Regime in JT-60U

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Abstract: This paper reports the development of reversed magnetic shear (RS) plasmas with a large bootstrap current fraction (fBS) towards a reactor relevant regime, especially lower q_{95}. By utilizing large volume configuration close to the conductive wall for wall stabilization, β limit is significantly improved. As a result, high confinement RS plasmas exceeding non-wall β limit with large f_{BS} are obtained for the first time in a reactor relevant regime, where $\beta_N \approx 2.7$, $\beta p \approx 2.3$ is achieved in RS plasma with $q_{min} \approx 2.4$, and then $HH_{98y2} \approx 1.7$, $n_e/n_{GW} \approx 0.87$ and $f_{BS} \approx 90\%$ are obtained at $q_{95} \approx 5.27$. In addition we found a strong linkage between pressure and current profiles with local current diffusion time scale.

EX/1-2 · High β_N Regimes at JET: Progress Towards Steady-State Operation

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Abstract: High β_N plasmas, with or without an ITBs, have been studied in JET AT scenarios at high triangularity and high density at $I_p = 1.2 - 1.5$ MA / $B_T = 1.8 - 2.4$ T. The current profile is tailored via a fast current ramp, ohmic or with Lower Hybrid Current Drive (LHCD), and early application of Neutral Beam (NBI) or NBI + Ion Cyclotron Resonance Heating (ICRH) power. The initial q profile has low or weakly reversed shear in the core. LHCD is used in some of the discharges to further broaden the current profile, but the accessibility and current drive efficiency are limited at this low toroidal field. $\beta_N \sim 2.8$, $\beta_N \sim 3$ transiently, has been achieved with $H_{IPB98(y,2)} \sim 1.0-1.2$. The development of an ITB contributes by $20 - 25\%$ to β_N, the best performance being obtained when an ITB forms in both ion and electron temperature channels. Discharges are routinely obtained with total β_N above the no-wall limit, but the measured β-limit and the achievable β decrease with increasing q_{min}, as do the global confinement and the core pressure. The total non-inductive current INI reaches transiently 5% of I_p, at the maximum values of β_N and $> 60\%$ I_p in a more stationary phase. The thermal Boostrap current is $\sim 50\%$ of INI, while the rest is beam driven. The experiments described above confirm that a viable route towards sustainable high β_N operation exists, with different q profiles, with or without ITBs, if strong pressure gradients are avoided. Interpretative modelling of these discharges highlights the role of the various non-inductive components of the current profile and predictive modelling can, thus, be employed to optimise the scenario for JET and ITER. Of particular importance for JET is the preparation of the AT scenario in view of the upgrade of the JET heating power, NBI and ICRH: with extra power available, modelling suggests that operation at high β will be achievable at higher toroidal field which, in turn, will improve the LH wave accessibility and increase the capability to tailor the current profile and sustain negative shear for times comparable to the resistive diffusion time.

EX/1-3 · Demonstration of ITER Operational Scenarios on DIII-D

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Abstract: The DIII-D program has recently initiated an effort to provide suitably scaled experimental evaluations of the primary ITER operational scenarios. New and unique features of this work are that the plasmas incorporate as constraints leading operational features of the ITER scenarios, such as the design values for the ITER plasma cross-section and aspect ratio, and that all four primary ITER scenarios have been evaluated on a single device, enabling direct cross-comparisons. Key aspects of the ITER baseline or reference scenario (Scenario 2), steady-state (Scenario 4), hybrid (Scenario 3), and "advanced inductive" plasmas have been replicated successfully, providing an improved physics basis for transport and stability modelling and performance extrapolation to ITER. In all four scenarios performance equals or closely approaches that required to realize the physics and technology goals of ITER. Utilizing a version of the ITER plasma scaled by a factor of 3.7, and with aspect ratio of 3.1, conventional ELMy H-mode baseline scenario plasmas have been operated at the target I/aB value of 1.415, corresponding to $q_{95} \sim 3$, with normalized β of $1.8 - 2.0$. Operation at the higher normalized β level provides normalized fusion performance close to the level required for $Q = 10$ operation on ITER. For the steady-state scenario, plasmas were run with the same ITER-like shape and aspect ratio, but with $q_{95} \sim 4.7$, $q_{min} \sim 1.5$ and normalized β of $2.7 - 3.0$. At the higher beta, normalized performance at the level required for $Q = 5$ operation on ITER was obtained. Hybrid plasmas with $q_{95} \sim 4.1$ and normalized $\beta \sim 2.8$ achieve normalized performance close to the level required for $Q = 10$ operation on ITER. Finally, the advanced inductive scenario, which targets the ITER $Q = 30$ physics goal, was operated with $q_{95} \sim 3.3$ and normalized $\beta \sim 2.8$, resulting in performance well above the level required for $Q = 10$ operation in ITER. A new and significant issue is the fact that the value of $l_i(3)$ in all scenarios is below 0.7, outside the present ITER plasma shape control system design range of $0.7 - 1.0$. The demonstration discharges also provide more realistic experimental profiles to use in transport and stability modelling for ITER, which will be presented at the IAEA meeting.

Work supported by the US Department of Energy under DE-FG02-01ER54615.

EX/1-4Ra · Development of the "Hybrid" Scenario in JET

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Abstract: In the recent 2006-2007 campaigns, JET has made a detailed experimental exploration of the plasma scenario, now known as "the ITER hybrid scenario". Extending the parameter space in terms of q_{95}, density and β_N in a large machine like JET is essential for the development of this scenario closer to the parameters of ITER at lower normalised Larmor radius ρ^* and collisionality ν^*. Experiments in JET recent campaigns, focused on extending the edge q profile range (q_{95}) of the "hybrid" scenario and making systematic comparisons of these discharges with a reference baseline H-mode scenario. No differences in thermal confinement have been so far observed between the JET scenario and a reference H-mode in contrast to other devices. Also the confinement dependence with β shows a similar dependence to that of the IPB98(y,2) scaling at high triangularity ($\delta = 0.45$) for ELMy H-mode in contrast to previous experiment executed with a low shape at $\delta = 0.2$. Dedicated experiments with preformed target q profile close to unity have extended the scenario operations at higher total normalised pressure ($\beta_N = 3.6$) without significant 2/1 NTM activity in contrast to other devices such as DIII-D or JT-60U. The JET scenario has also been extended to higher density using gas fueling in the X-point private area reaching the Greenwald density limit at $\beta_N \sim 2.7$ while keeping an $IPB98(y, 2)$ factor above 0.9.

EX/1-4Rb · Advances in the Physics Basis of the 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 future burning plasma devices. While this "hybrid scenario" has a type-I ELMy H-mode edge, like the standard H-mode scenario, hybrids differ by having suppressed sawteeth, a higher β limit to the 2/1 NTM, and remarkably good transport properties. Recent experiments on DIII-D have advanced our physics understanding of the hybrid scenario by extending the plasma conditions towards the burning plasma regime by separately incorporating strong electron heating, low torque injection, and ELM suppression from resonant magnetic perturbations (RMPs). Recent hybrid experiments have replaced some of the co-NBI with 2.4 MW of ECH to reduce the toroidal rotation and raise T_e closer to T_i. At fixed beta, raising T_e/T_i reduces the confinement factor by 10% and increases turbulence. Additional experiments on DIII-D found that the beneficial characteristics of the hybrid scenario (no sawteeth, benign NTMs, high beta) are maintained when shifting from co-NBI to nearly balanced-NBI. As the toroidal rotation velocity is reduced by an order of magnitude, the confinement factor is reduced from $H_{98y2} = 1.4$ to $H_{98y2} = 1.1$. modelling using TGLF shows that the increase in transport with lower NBI torque is consistent with changes in the $E\times B$ flow shear. For the first time, type-I ELMs have been suppressed in hybrid plasmas by applying edge RMPs with toroidal mode number $n = 3$. This is an important advance in developing hybrids as a baseline-operating scenario for ITER since such ELMs may cause unacceptable divertor erosion. In ITER-shaped plasmas with normalized β of 2.5 and edge safety factor of 3.6, the confinement factor drops from $H_{98y2} = 1.3$ to $H_{98y2} = 1.0$ during RMP ELM suppression owing to an increase in both the electron and ion thermal diffusivities. Additionally, high performance hybrid operation has been demonstrated with reduced frequency of wall conditioning. In both the 2006 & 07 campaigns on DIII-D, over 7000 s of plasma operation were conducted with no intervening boronizations. The maintenance of good wall conditions over these campaigns is seen in the impurity line emission and fueling/exhaust data from daily reference shots.

This work was supported by the US Department of Energy under DE-FC02-04ER54698.

EX/1-4Rc · Development of Advanced Operation Scenarios in Weak Magnetic-Shear Regime on JT-60U

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Abstract: Fully non-inductive discharge having relaxed current profile and high bootstrap current fraction $f_{BS} = 0.5$ has been realized in the high-β_p ELMy H-mode discharge with weak magnetic-shear satisfying $q_{95} = 5.8$, $q_{min} = 2.1$, and $q(0) = 2.4$, where q_{min} and $q(0)$ are the safety factor q at the minimum and the plasma centre, respectively. The rest of the plasma current is externally driven by neutral beams (NBs) and lower-hybrid (LH) waves. The safety factor profile evaluated by the motional Stark effect (MSE) diagnostic is kept unchanged during 0.7 s at the end of the full current drive (CD) sustainment for 2 s (1.5 times the current relaxation time). The loop voltage profile is spatially uniform at 0 V at the end of the sustainment. This demonstration contributes to the ITER steady-state operation scenario development. Another ITER advanced operation scenario, the hybrid operation scenario, has also been developed in the high-β_p ELMy H-mode with $q_{95} = 3.2$ and $q_{min} \sim 1$. Period of sustainment at high $\beta_N = 2.6$ has been almost tripled (to 28 s), and the JT-60U operation regime has extended towards longer sustainment and higher β_N. The development of advanced scenarios in JT-60U in these weak magnetic-shear regime leads development of advanced scenarios in ITER.

EX/1-5 · Compatibility of ITER Scenarios with Full Tungsten Wall in ASDEX Upgrade

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Abstract: In the last two years, ASDEX Upgrade has studied the compatibility of standard H-modes and advanced scenarios with a reactor relevant, tungsten first wall. Presented are results for ITER relevant scenarios: (1) The current ramp-up phase of the discharge, (2) ELMy H-modes at low $q_{95} \sim 3$ and (3) improved H-modes scenarios that are candidates for hybrid operation in ITER. Documented are the confinement properties of the plasma, the MHD stability and the stability against impurity accumulation. This is done for a range of plasma densities ($< n_e > /n_{GW} = 0.4 - 0.9$) by using different plasma fueling techniques (including pellet injection) and by optimising the three different heating schemes available at ASDEX Upgrade (neutral beam heating, ECRH and ICRH). Specific studies are performed in a tungsten machine without boronisation, and contrasted to results directly after boronisation.

EX/2-1 · Results of the Variable Toroidal Field Ripple Experiments in JET

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Abstract: The main goal of ITER is to produce plasmas with high fusion gain ($Q_{DT} > 10$), where a large fraction of the plasma heating is supplied by the α-particles from fusion reactions. To first order, this requires both thermal and fast ion confinement to be at least as good as predictions. In all tokamak devices, the finite number and toroidal extension of the toroidal field (TF) coils causes a periodic variation of the toroidal field from its nominal value, called toroidal field ripple $\delta_{BT} = (B_{max} - B_{min})/B_{av}$. It is well known that ripple in the toroidal field adversely affects fast ion confinement, and in the case of ITER, this has been accounted for by including in the design Ferritic Insets (FI) compensation, reducing δ_{BT} from approx 1.2% to approx 0.5%, to reduce first wall power loads. In these conditions, the main α-particle loss mechanism in ITER will be ripple banana orbit diffusion, and that the magnitude of these losses is expected to be in the 1% region, therefore negligible in terms of alpha-particle confinement. Experimental results from JT60-U and H-mode dimensionless H-mode experiments in JET and JT-60U have indicated that ripple may also affect the H-mode confinement, pedestal height, ELM size and plasma rotation. Although the physics mechanisms at the root of the reduced energy confinement with δ_{BT} was not identified unambiguously, the implication of a reduction of energy confinement on projected ITER performance due to ripple stimulated a series of experiments at JET. The experiments were carried out in the ELMy H-mode regime with $q_{95} = 3$, and investigated the effect of δ_{BT} on the pedestal and core properties of the plasma. Plasma density pump out and reduction of the global energy confinement was found for δ_{BT} around 0.5, in some experimental conditions. Ripple was also found to affect the size and frequency of ELMs. Plasma toroidal rotation was also strongly affected by ripple: the toroidal velocity is reduced for increased ripple and becomes negative at the edge for $\delta_{BT} = 1\%$. This paper discusses these results and their modelling including the possible influence of ripple on thermal ion transport, and outlines implications for the ITER design.

EX/2-2 · Transport Studies in the MAST Spherical Tokamak

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Abstract: Spherical tokamaks (STs) provide a challenging regime for the study of transport processes in magnetised plasmas and an understanding of these is crucial for prediction of the performance of future devices. Data from a suite of state-of-the-art diagnostics on MAST are combined to form an integrated self-consistent data-set using a new interactive data management tool prior to analysis using TRANSP. The tangential NBI heating drives strong toroidal rotation and the resulting $E\times B$ flow shear is sufficient to suppress ion scale turbulence in the core plasma. Ion thermal transport in this region is thereby reduced almost to the neo-classical level and the electron thermal diffusivity is of similar magnitude. A prototype beam-emission spectroscopy (BES) system has been used to characterise the density turbulence. Relatively high-amplitude fluctuations ($\sim$ few%) have been observed in the outer regions of L-mode plasmas and ELM precursor oscillations in the pedestal region have also been observed in extreme single-null diverted (SND) discharges. Transport analysis has shown that momentum and ion thermal transport are closely related, with the Prandtl number ~ 0.5 over the full profile. This is much higher than that expected from neo-classical transport alone. High-resolution measurements of the radial electric field (E_r) in the pedestal region of H-mode plasmas have been obtained. In L-mode, fields of only a few kV/m are observed with the flow parallel to the B-field, whereas in H-mode, a strong perpendicular flow develops with poloidal and toroidal velocities ~ 20 km/s. A deep negative electric field well forms, with $E_r \sim -15$ kV/m. Analysis of a database of 200 L-mode discharges reveals a correlation between electron density profile peaking and the normalised current density, which is common to that found in the conventional tokamaks TCV and JET. Transport studies on MAST are supported by a number of non-linear turbulence simulation codes covering spatial scales from the ion to the electron Larmor radius. At the low toroidal field of STs, simulations of ion temperature gradient (ITG) turbulence must incorporate both $E\times B$ flow shear and the radial variation of the equilibrium profiles. The former has now been incorporated in GS2 and is being tested in non-linear simulations and the first global ITG simulations for MAST are also being performed using the ORB5 code.

EX/2-3 · Full Bootstrap Discharge Sustainment in Steady State in the TCV Tokamak

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Abstract: The Advanced Tokamak scenario, one of the modes of operation being considered for ITER, relies on the attainment of a high bootstrap current fraction. These scenarios are typically characterized by an internal transport barrier (ITB). An internal feedback loop then governs the current profile, which strongly affects the confinement and thus the properties and location of the high-gradient region, where in turn the bootstrap current component is localized. The bootstrap current fraction can reach 100% only if the bootstrap current profile can be exactly and stably aligned with the high-gradient region it engenders. Recent work on the TCV tokamak has shown that such an alignment is indeed possible. We have produced discharges in which the current is entirely self-generated by the plasma in conditions of intense electron cyclotron resonance heating (ECRH, up to 2.7 MW), by employing two different methods. In one scenario, high-power, second-harmonic ECRH waves are launched with no current drive component, during the initial plasma current ramp-up. A strong ITB is generated by the transient negative central magnetic shear that develops in the current penetration phase. The Ohmic flux swing is zeroed immediately after breakdown, cutting off the external plasma current source. The plasma can then evolve spontaneously towards a stationary and quiescent state, characterized by a narrow ITB with a confinement enhancement of $2.5-3$ over L-mode. The discharge remains stationary over the time scale of a TCV pulse ($1-2$ s), which is significantly longer than a typical resistive current redistribution time ($\sim 150-300$ ms) and orders of magnitude longer than the confinement time ($\sim 3-6$ ms). Following a different approach, we have also succeeded in achieving a 100% bootstrap fraction by annulling the total EC-driven current in pre-existing stationary conditions. Standard stationary non-inductive eITBs are first generated by off-axis co-ECCD; counter-ECCD is then added gradually until the total driven current density is nominally zero everywhere. In some cases a quasi-stationary state is indeed established. In this case the barrier remains broad with a standard confinement enhancement factor of the order of $4-5$. As a result, the discharge is not truly quiescent, because of significant MHD activity causing oscillations and jumps both in the confinement and in the total current.

EX/2-4 · Experimental Study of the Ion Critical Gradient Length and Stiffness Level and the Impact of Rotational Shear in JET

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Abstract: The paper reports experiments in JET aimed at the identification of the threshold for the onset of ITG driven ion heat transport and the determination of the ion stiffness level. The experiments are performed in ICH dominated, low rotating plasmas, by shaping the ion heat flux profile using on- and off-axis power deposition. The identified ion threshold compares well with linear predictions of the gyrokinetic code GS2, but the experimental stiffness level appears higher than foreseen by non-linear GS2 simulations. Comparison with high rotation JET plasmas with similar parameters indicates a significant increase of ion temperature inverse gradient length with increasing rotation. Various experimental observations, including comparison of co- and counter-NBI plasmas, allow to conclude that such increase is mainly due to a decrease of stiffness level in high rotation plasmas, in addition to the foreseen smaller effect of threshold up-shift due to rotational shear. The stiffness levels measured in high rotation plasmas are in line with gyro-kinetic simulations. The implications for the interpretation of present experiments and the extrapolation to future devices are discussed.

EX/2-5 · Neoclassical Currents and Transport Studies in HSX at 1 Tesla

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Abstract: With quasihelical symmetry and a high effective transform, calculations indicate that the Helically Symmetric Experiment (HSX) has small equilibrium currents and low neoclassical transport. Previous results at 0.5 Tesla indicated that particle, momentum and heat transport were all reduced compared to a configuration that mimics a conventional stellarator, in agreement with neoclassical estimates. First experimental results from 1 Tesla operation are presented showing that the small magnitude of the Pfirsch-Schlüter (PS) and bootstrap currents agree with calculations and that the PS current is helical due to the lack of toroidal curvature. With the same ECRH power, electron temperatures up to 2.5 keV at the plasma core were measured in the quasihelically symmetric (QHS) configuration compared to 1.5 keV when the symmetry is intentionally broken (termed 'Mirror'). These results at the higher field continue to demonstrate the expected neoclassical benefits of the quasihelical configuration. In this paper we also present an initial assessment of the effect of quasihelical symmetry on anomalous transport. A modified Weiland ITG/TEM tokamak model shows good agreement with the measured electron temperature profile outside the plasma core, as well as with the global energy confinement time.

EX/3-1 · Formation Mechanism of Toroidal Rotation Profile and Characteristics of Momentum Transport in JT-60U

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Abstract: The diffusive term and the non-diffusive term of the momentum transport and the intrinsic rotation have been evaluated separately using the original transient momentum transport analysis. Thanks to the separation, characteristics of these terms are found for the first time: (i) the toroidal momentum diffusivity in the core region roughly scales linearly with neutral beam (NB) heating power, and decreases with increasing the plasma current (ii) the toroidal momentum diffusivity increases with increasing the heat diffusivity, (iii) the convection velocity correlates with the toroidal momentum diffusivity. (iv) We have also found that the intrinsic rotation with NBI, which is not explained with the toroidal momentum diffusivity, the convection velocity and the external momentum input, increases with increasing ion pressure gradient and its direction is always counter (CTR) direction. (v) The toroidal rotation velocity in the BAL-NB injected plasma changes in the CTR-direction with electron cyclotron heating (ECH). The change in the toroidal rotation velocity does not increase with the ion pressure gradient but increases with increasing ECH power.

EX/3-2 · Momentum Transport in Electron-Dominated Spherical Torus Plasmas

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Abstract: The Spherical Torus (ST), or low aspect ratio tokamak, operates at low toroidal fields, leading to high $E\times B$ rotational shearing rates. The National Spherical Torus Experiment (NSTX) operates between 0.35 and 0.55 T, which, when coupled to up to 7 MW of neutral beam injection, leads to central rotation velocities in excess of 300 km/s and $E\times B$ shearing rates up to 1 MHz. This level of $E\times B$ shear can be up to a factor of five greater than typical linear growth rates of long-wavelength ion (e.g., ITG) modes, at least partially suppressing these instabilities. Evidence for this turbulence suppression is that the inferred diffusive ion thermal flux in NSTX H-modes is often at the neoclassical level, and thus these plasmas operate in an electron-dominated transport regime. Recent studies indicate that in this regime the momentum transport properties can be different than those at higher aspect ratio, with the ion thermal diffusivity much greater than the momentum diffusivity. Despite this, the value of the momentum diffusivity as inferred from steady-state momentum balance is much larger than the neoclassical value. Analysis of perturbative experiments that used applied $n = 3$ magnetic fields to brake the plasma rotation indicated inward pinch velocities up to 40 m/s and perturbative momentum diffusivities larger by a factor of several than those values inferred from steady-state analysis with a zero pinch velocity assumed. The inferred pinch velocity values are consistent with values based on theories in which low-k turbulence drives the inward momentum pinch. Thus, in STs, the momentum transport can be a better probe of low-k turbulence than the energy transport. While the neoclassical ion energy transport effects can be relatively high and dominate the ion energy transport, the neoclassical momentum transport effects are near zero. This means that any residual low-k turbulence dominates the momentum transport . Understanding the source of the momentum transport, and how it scales to larger devices operating at lower collisionality, is critical to the performance of future ST-based Fusion Energy Development devices such as an ST-based Component Test Facility, as well as to conventional aspect ratio devices such as ITER, which is also expected to operate in electron-dominated transport regimes.

EX/3-3 · Experimental Evidence on Inward Momentum Pinch on JET and Comparison with Theory and Modelling

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Abstract: Dedicated experiments have been carried out in the JET tokamak in order to determine the diffusive and convective momentum transport separately. The power and torque, injected by the neutral beam heating system, are modulated to create a periodic perturbation in the toroidal rotation velocity. Transport analysis shows that the magnitude and profile shape of the momentum diffusivity are very similar to those of the ion heat diffusivity, resulting in the Prandtl number being around one throughout the whole plasma radius. Furthermore, a significantly large inward momentum convection term, up to 20 m/s, has been found. Its profile shape is roughly proportional to the momentum diffusivity. Both these results are consistent with recent developments in momentum transport theory and with linear gyro-kinetic simulations. These findings have significant implications on predictions for the magnitude and profile shape of the toroidal rotation velocity in future tokamaks. The existence of an inward momentum pinch will result in a centrally peaked toroidal velocity profile even in the absence of any central momentum source.

EX/3-4 · Developments in Predictive Understanding of Plasma Rotation on DIII-D

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Abstract: Recent experiments using DIII-D's capability to vary the injected torque at constant power have focused on developing the physics basis for rotation, through detailed studies aimed at understanding momentum sources, sinks and transport. These investigations have resulted in several surprising discoveries, including empirical evidence for an effective anomalous torque source to the plasma, observation of a spin-up of the plasma following application of non-resonant "braking", and distinct dependences of the momentum confinement on the applied torque depending on plasma conditions. Since rotation is known to affect a broad range of fundamental issues concerning fusion plasmas, the performance of future burning plasma devices including ITER will depend on the attained rotation profile. Consequently, obtaining predictive understanding of rotation, and ultimately exploiting such knowledge to generate a desired rotation profile will result in a significant payoff for fusion. The presence of an anomalous momentum source has been identified by varying the input torque from neutral beam injection at fixed β_N, until the plasma rotation across the entire profile is approximately zero. This torque profile is largely peaked at the edge, but also with a significant amount of torque present in the core. Non-resonant braking due to neoclassical toroidal viscosity (NTV) has generally been considered a sink of momentum, however, recent results from DIII-D suggest that it may also act as a source. The torque applied by the field depends on the unperturbed rotation, and interestingly is minimized at an initial rotation different from zero. In studies of momentum transport, the momentum confinement time τ_ϕ is found to have a strong dependence on the applied neutral beam torque, which may be favorable for improved momentum confinement and larger rotation on ITER. Perturbative studies of the rotation using combinations of co and counter neutral beams have uncovered the existence of a momentum pinch in DIII-D H-mode plasmas, which shows qualitative similarity to theoretical predictions resulting from consideration of low-k turbulence.

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EX/4-1 · Plasma Performance in DIII-D ELM-Suppressed RMP H-modes with ITER Similar Shapes

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Abstract: Fast energy transients, incident on the DIII-D divertors due to Type-I edge localized modes (ELMs), have been eliminated using small dc currents in a simple set of non-axisymmetric coils that produce edge resonant magnetic perturbations (RMP). In ITER similar shaped (ISS) plasmas with electron pedestal collisionalities below those expected in ITER, the stored energy, β normal and H-mode quality factor initially drop for several hundred ms during the ELM-suppression in DIII-D phase but recover within $600 - 700$ ms. A well-defined resonant window in the safety factor at the 95 percent normalized poloidal flux surface i.e., $q_{95} = 3.6$ is required for ELM suppression. The size of the resonant window can be increased by a factor of 4 in ISS plasmas by increasing the magnitude of the current in an $n = 3$ coil set (the DIII-D I-coil) along with the current in an $n = 1$ coil set (the DIII-D C-coil). While the details of the resonant ELM-suppression window are highly reproducible for a given plasma shape, coil configuration and coil current, we find that isolated resonant windows exist at other q_{95} values with different RMP coil configurations. For example, when the I-coil is operated in an $n = 3$ up-down asymmetric configuration (referred to as odd parity) rather than an up-down symmetric (even parity) configuration a resonant window is found near $q_{95} = 7.4$. Vacuum field line modelling has demonstrated that there is a strong statistical correlation between the Chirikov island overlap parameter and ELM suppression. These results indicate that with an optimized coil design ELM control can be obtained over a wide range of q_{95} (e.g., in ITER). This criterion has proven to be a valuable guide for RMP coil design studies in various ITER operating scenarios. Issues of specific importance to RMP ELM suppression in ITER such as: 1) RMP effects on energy confinement, 2) compatibility of RMP fields with pellet fueling, 3) RMP effects on density pump-out, and 4) separatrix splitting during ELM suppression with RMP fields will be discussed in detail and the viability of this approach for ITER will be considered.

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EX/4-2 · Active Control of Type-I Edge Localized Modes with $n = 1$ and $n = 2$ fields on JET

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Abstract: Recent experiments on JET have shown that type-I edge localized modes (ELMs) can be controlled by applying static low $n = 1$ external magnetic perturbation fields (EMPFs) produced by four external error field correction coils (EFCC) mounted far away from the plasma between the transformer limbs. When an $n = 1$ EMPF with an amplitude of a few Gauss at the plasma edge ($\rho > 0.95$) is applied during the stationary phase of a type-I ELMy H-mode plasma, the ELM frequency rises from ~ 30 Hz up to ~ 120 Hz and follows the applied perturbation field strength. The energy loss per ELM normalised to the total stored energy, $\Delta W_{ELM}/W_p$, decreased from 7% to below the resolution limit of the diamagnetic measurement ($\sim 2\%$). Transport analysis using the TRANSP code shows no or a modest reduction of the thermal energy confinement time because of the density pump-out, but when normalised to the IPB98(y,2) scaling the confinement shows almost no reduction. Stability analysis of mitigated ELMs shows that the operational point moves from intermediate n peeling-ballooning (wide mode) boundary to low-n peeling (narrow mode) boundary with $n = 1$ perturbation fields. The minimum $n = 1$ EMPFs amplitude above which the ELMs were mitigated increased with a lower q_{95} but always remained below the $n = 1$ locked mode threshold. Active control of type-I ELMs with $n = 1$ EMPFs has been developed for more ITER-relevant configurations and parameters in a wide operational space of plasma triangularity (δ_U up to 0.45), $q_{95}(4.8 - 3.0)$ and β (β_N up to 3.0) on JET. The first results of ELM mitigation with the $n = 2$ EMPFs on JET demonstrate that the frequency of ELM can be increased by a factor of 3.5, only limited by the available EFCC coil current. During the application of the $n = 1$, 2 EMPFs, a reduction in the ELM size (ΔW_{ELM}) and ELM peak heat fluxes on the divertor target by roughly the same factor as the increase of the ELM frequency has been observed. The reduction in heat flux is mainly due to the drop of particle flux rather than the change of the electron temperature. Similar plasma braking effect has been observed with $n = 1$ and $n = 2$ external fields when a same EFCC coil current was applied. Active ELM control by externally applied fields offers an attractive method for next-generation tokamaks, e.g., ITER.

EX/4-3Ra · ELM filament Heat Loads on Plasma Facing Components in JET and ITER

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Abstract: It is now widely recognized that Type-I ELMs generate filamentary structures which travel radially well into the far Scrape-Off Layer (SOL). The localised heat loads on plasma facing components (PCFs), including the divertor, limiter and upper dump plate tiles, associated with such ELM filaments are one of the critical issues for ITER; specifically, main chamber PFCs in ITER are presently being redesigned with the aim of handling such plasma related heat loads. Since their earliest observations, the kinematics, dynamics and dissipation of Type-I ELM filaments (i.e., their radial motion, driving forces and parallel loss mechanisms), as well as the resulting heat loads on the vessel wall, have been a topic of active research on JET. In this contribution, the findings of this activity are briefly reviewed with an emphasis on most recent results. These include filament visualisation using fast visible cameras and high resolution Thomson scattering, measurements of radial filament velocities using time of flight techniques, measurement of ion and electron energies at the vessel wall, and measurements of main chamber heat loads using wide angle infra-red thermography. The JET results are drawn together to form a coherent picture of the exhaust phase of the ELM. In particular, it is shown that the presently available JET data is both qualitatively and quantitatively consistent with a model of ELM filament dynamics in which the evolution of filament density and temperature proceeds via a competition of radial advection and parallel convective and conductive losses. The model successfully reproduces the radial e-folding lengths of density, electron temperatures and energy, inferred from dedicated outer gap-scan experiment for moderate ($\delta W/W \sim 5\%$) Type-I ELMs. It is also in good agreement with observations of far SOL ELM ion energies on JET, measured using a retarding field analyser probe head on fast scanning assembly. The same model is then applied to predict the average ELM filament heat loads on the first wall in ITER. In the reference ITER scenario (4 keV pedestal), it predicts that for moderate Type-I ELMs, roughly $\sim 8\%$ of the ELM energy would be deposited on the upper dump plate ($\delta r \sim 5$ cm) and $\sim 1.5\%$ on the outboard limiters ($\delta r \sim 15$ cm).

EX/4-3Rb · Divertor Heat Loads Due to Edge Localized Modes in ASDEX Upgrade and JET

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Abstract: Transient power loads due to ELMs are a major concern for the life time of divertor tiles in ITER, irrespective of the choice of plasma facing material. Large ELMs cause enhanced material erosion (ablation, evaporation) possibly leading to unacceptable impurity influx, radiation and fuel dilution in the plasma core. Significant progress has been made towards a physics-based model describing Type-I ELM energy transport to the divertor and the first wall. Specifically, the temporal evolution of infra-red measured divertor heat loads due to Type-I ELMs on ASDEX Upgrade (AUG) is well reproduced by an analytic formula representing the arrival of ions originating at the outer midplane with a Maxwellian velocity distribution characterized by only the pedestal. For large Type-I ELMs in JET ($100-600$ kJ, $5-20\%$ pedestal energy) and ASDEX Upgrade ($6-20$ kJ, $3-10\%$ pedestal energy) the ELM power deposition time on the inner/outer divertor targets is dependent only on the corresponding parallel ion transit time. In particular, for low pedestal collisionality (as expected in ITER), it predicts ELM power deposition time of 1.2 times the parallel transport time which yields a power deposition time of 280 μs at the ITER pedestal temperature. The analysis of JET and ASDEX Upgrade data also reveals the temporal hierarchy of ELM energy release, parallel transport and power deposition time. On JET, the fraction of energy of deposited on the target within the duration of the power deposition time varies between 20% for the largest ELMs (lowest pedestal collisionality) and to 35% for smallest ELMs (highest pedestal collisionality). It is noteworthy in that respect that the found temporal hierarchy of ELM energy release, parallel transport and power deposition time is broken for the much smaller (and more collisional) Type-III ELMs. For these ELMs, both quantities ELM energy release time and parallel transport have to be taken into account when estimating the resulting ELM peak heat fluxes. The above results confirm the previous, purely empirical scaling, on the type-I ELM power deposition times. However, combined with recent material test, the formerly calculated limit of a maximum ELM energy loss in ITER is revised and a significantly lower value of about 1˘2 MJ is found. Therefore, some form of ELM mitigation techniques appears to be mandatory for ITER.

EX/4-4Ra · Investigations of Impurity Seeding and Radiation Control for Long-pulse and High-density H-mode Plasmas in JT-60U

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Abstract: Reduction of heat loading appropriate for the plasma facing components (PFCs) such as the divertor and the first wall is crucial for a fusion reactor. Impurity gas seeding is one of techniques to decrease peak heat flux to the divertor both in steady-state and transient phases. Power handling by large radiation power loss has been studied in the ELMy H-mode plasmas on JT-60U with argon (Ar) gas seeding since good confinement ($HH \geq 0.85$) was obtained up to high density ($n_e/n_{GW} \sim 0.8-0.9$, n_{GW} is the Greenwald density). On the other hand, it was not clearly understood to sustain the good confinement plasma with the large radiation power under the wall saturated condition where particle recycling flux changes during the long discharge. In this paper, control of the large radiation in the good energy confinement plasma was, for the first time, investigated. Total radiation fraction of $P_{rad}/P_{abs} = 0.8-0.9$ was maintained during Ar gas puffing (up to 18 s so far), and control of radiation power in the main plasma has been investigated during outgasing condition from the PFCs.

EX/4-4Rb · Integrated Scenario with Type-III ELMy H-mode Edge: Extrapolation to ITER

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Abstract: One of the most severe problems for fusion reactors is the power load on the plasma facing components. The challenge is to develop operation scenarios, which combine sufficient energy confinement with benign heat loads to the plasma facing components. The radiative type-III ELMy H-mode seems a possible solution for such an integrated ITER scenario. Most notably the transient heat loads due to type-III ELMs are acceptable with even the most stringent boundary conditions. For instance, on JET the transient heat loads due to type-III ELMs onto the outer divertor target were reduced to 2 kJ/m^2. Scaled to ITER, type-III ELMy H-modes are expected to have a power load of approximately 0.3 MJ/m^2 transiently. This was achieved in experiments carried out with nitrogen seeding to mitigate the transient and steady state heat flux to the divertor. Typically the confinement is reduced by $\sim$ 8-20% compared to the type-I ELMy H-mode base scenario. However, increasing the plasma current to 17 MA on ITER and hence reducing the edge safety factor to 2.6, would allow $Q = 10$ operation at a reduced confinement enhancement factor. This operation scenario was demonstrated at JET up to plasma currents of 3 MA. At the highest plasma current the effective charge Z_{eff} can be as low as 1.5, mainly due to the increased absolute density and reduced carbon erosion. A large database of highly radiative type-III ELMy H-modes on JET is used for extrapolations to ITER. The data set shows no apparent dependence of the confinement enhancement factor on collisionality. The scaling of the confinement time with respect to the ion gyro radius is close to gyro-Bohm scaling. The "hybrid" regime, designed for high β stationary scenarios, has been extended recently at JET to the type III-ELMy H-mode operation by nitrogen seeding (at a plasma current of 1.7 MA) up to a normalized pressure (β) of 2.6. Similar to the standard ELMy H-mode the confinement enhancement factor is reduced by about 20%. The "hybrid" type III-ELMy H-mode scenario shows improved edge plasma condition without significant modification of the q-profile (stabilized near unity in the plasma core in order to reduce the sawteeth activity), indicating it is compatible with high β operation (optimized for current drive sources). Extrapolations to ITER are done with an integrated core/edge model.

EX/4-5 · Investigating Pellet Physics for ELM Pacing and Particle Fueling in ITER

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Abstract: Experimental investigations on the onset dynamics for spontaneous and triggered ELMs in a broad range of ELM regimes (type III, ELM-free, radiative cooled and hot edge type-I) revealed pellets force an ELM at any time and always with the shortest observed growth time while frequency and growth time of spontaneous ELMs vary with changing plasma parameters. In Ohmic and L-mode plasmas pellets do not trigger ELMs but drive create magnetic perturbations significantly larger than those observed during an initial ELM phase. The magnitude of the pellet driven perturbation depends only the plasma but not on the pellet parameters. This indicates that pellets provoke a strong non-linear modes growth during any phase of the natural ELM cycle via a driving mechanism already saturating in the accessible pellet parameter regime. For ITER thus ELM pacing seems feasible by injecting small high speed pellets from the Low Field Side of the machine, causing only little impact on particle inventory and confinement. Pellet fueling appears also compatible with scenarios relying on ELM suppression or avoidance by e.g., edge ergodisation since experimental evidence suggests that in regimes with stable pedestals pellets no longer trigger ELMs. Deep fueling pellet deposition leads to long particle confinement times and allowed high density operation with good energy confinement. Convective losses introduced by the fueling has been identified on AUG and JET to play the key role for the accessible density-confinement operational window. Sufficient pumping was required to prevent excessive rise of neutral and edge densities. In addition, abrupt, strong plasma cooling had to be avoided by gentle density ramps. Ion gyroradii shrinking with temperature cause – according to the ion polarization model – a reduced onset pressure making the plasma vulnerable to pellet triggered NTMs.

EX/5-1 · Advances in Global MHD Mode Stabilization Research on NSTX

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Abstract: Stabilizing modes that limit plasma β and reducing their deleterious effect on plasma rotation are key goals for efficient operation of a fusion reactor. NSTX has demonstrated both passive stabilization and active control of global kink/ballooning modes and resistive wall modes (RWM) accessing high toroidal and normalized β of 39% and 7.2 respectively. Active $n = 1$ control experiments with an expanded sensor set, combined with low levels of $n = 3$ field phased to reduce error fields, reduced resonant field amplification and maintained plasma rotation, producing record discharge durations at 0.9 MA plasma current. Discharges with rotation reduced by $n = 3$ magnetic braking suffer β collapse at normalized $\beta = 4.2$ approaching the no-wall limit, while > 5.5 has been reached in these plasmas with $n = 1$ active control, in agreement with single-mode RWM theory and an upgraded system model in the VALEN-3D code. Advanced state-space control algorithms proposed for RWM control in ITER yield significant stabilization improvements. Values of relative phase between the measured $n = 1$ mode and the applied correction field that experimentally produce stability/instability agree with theory. Loss of active control by poloidal deformation of the RWM is examined using multi-mode theory. The physics of RWM passive stabilization is a key outstanding question for confident extrapolation to ITER. Recent experiments have created marginally stable plasmas with normalized β over the no-wall limit and with zero rotation at the $q = 2$ surface, challenging the idea of a scalar critical rotation speed for stabilization defined at this surface. Kinetic modifications to ideal stability of this plasma are computed and can lead to stabilization with the trapped ion resonance playing the largest role. Physics understanding of non-axisymmetric field-induced plasma viscosity is key to ITER, especially if magnetic ELM mitigation techniques will be used. Recent experiments using new device capability have demonstrated magnetic braking by non-resonant $n = 2$ fields. The observed rotation damping profile is broader than that found for $n = 3$ fields, consistent with theory.

EX/5-2 · Dynamics and Stability of Resistive Wall Mode in the JT-60U High-β Plasmas

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Abstract: The dynamics and stability of the resistive wall mode (RWM) are investigated in the JT-60U high-β plasmas, and the sustainment of high-β plasmas above the ideal no-wall β-limit has been demonstrated as a new high-β scenario using a plasma rotation. We have successfully sustained $\beta_N \simeq 2.8$ for ~ 2 s, while the ideal no-wall β-limit is $\beta_N^{no-wall} \sim 2.4$, with suppressing the RWM by the plasma rotation. Moreover, the plasma rotation braking owing to the RWM and a precursor just before the RWM have been observed. It is found that the RWM distortion can clearly induce the plasma rotation braking inside the $q = 2$ surface. The observed precursor can reduce the plasma rotation shear around the $q = 2$ surface and finally trigger the RWM.

EX/5-3Ra · Effect of Resonant and Non-resonant Magnetic Braking on Error Field Tolerance in High-β Plasmas

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Abstract: High β tokamak plasmas become less tolerant to $n = 1$ error fields as β increases, due to resonant interaction of the error field with a weakly stable resistive wall mode (RWM). Non-resonant error fields may further reduce the tolerance. The limit is seen as a bifurcation in the rotation balance, followed by error field-driven locked modes and severe confinement degradation or a disruption. The error field tolerance is, therefore, largely determined by the braking torque resulting from the non-axisymmetric magnetic field. DIII-D experiments clearly distinguish between the effect of externally applied fields that are resonant with rational surfaces inside the plasma, and fields that are non-resonant. While only resonant braking is seen to lead to a rotation collapse, modelling shows that non-resonant components can lower the tolerance to resonant components. The strong reduction of the error field tolerance with increasing beta, which has already been observed in early high β experiments in DIII-D [1], is linked to an increasing resonant field amplification (RFA) resulting from a marginally stable resistive wall mode (RWM), which a small error field can easily drive to large amplitude [2]. Magnetic pick-up coils detect the amplification of externally applied $n = 1$ "error" fields at normalized β values as low as 1.0. The amplification ratio is seen to increase with β, with the increase accelerating above the no wall ideal MHD stability limit. The β dependence was not previously appreciated, and was not included in the empirical scaling of the error field tolerance reported in the 1999 IPB [3] leading to overly optimistic predictions for low torque, high β scenarios and, in particular, for advanced tokamak scenarios, which rely on operation in the wall-stabilized regime. However, the measurable increase of the plasma response with β and the rigid response to externally applied fields can be exploited for "dynamic" correction (i.e., with slow magnetic feedback) of the amplified component of the error field using correction coils with a simple geometry.

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[1] R.J. La Haye, et al., Nucl. Fusion **32**, 2119 (1992); [2] A.H. Boozer, Phys. Rev. Lett. **86**, 5059 (2001); [3] ITER Physics Basis, Nucl. Fusion **39**, 2137 (1999)

EX/5-3Rb · New Understanding of Tokamak Plasma Response to 3D Magnetic Fields

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Abstract: The performance of both present and future tokamaks such as ITER can be greatly degraded or enhanced by small deviations from axisymmetry produced by externally driven 3D magnetic fields. An important new tool for understanding 3D magnetic field effects in tokamaks is the Ideal Perturbed Equilibrium Code (IPEC), which computes 3D perturbed tokamak equilibria including plasma response effects such as poloidal harmonic coupling, shielding, and amplification. These improved calculations of the coupling of external vacuum field perturbations to the plasma core represent a new paradigm in error field (EF) analysis. Recent error field correction results on DIII-D and NSTX indicate that inclusion of plasma response effects is essential to explain the observed locked mode (LM) behavior. Further, IPEC results are consistent with DIII-D Edge Localized Mode (ELM) mitigation results as a function of Resonant Magnetic Perturbation (RMP) parity, and IPEC has been used to design and interpret NSTX experiments varying RMP toroidal mode number. Based on these results, IPEC has also been used for ITER by optimizing EF correction and coil designs for LM and ELM mitigation.

EX/5-4 · Neoclassical Tearing Mode Control with ECCD and Magnetic Island Evolution in JT-60U

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Abstract: Results of stabilization of neoclassical tearing modes (NTMs) with electron cyclotron current drive (ECCD) in JT-60U are described with the emphasis on effective stabilization: (1) identification of the minimum electron cyclotron (EC) wave power for complete stabilization and (2) clarification of the effect of ECCD deposition width on NTM stabilization. Minimum EC wave power for complete stabilization of an $m/n = 2/1$ NTM has been experimentally identified as $0.2 < j_{EC}/j_{BS} < 0.4$ for $W_{sat}/d_{EC} \sim 3$ and $W_{sat}/W_{marg} \sim 2$ from JT-60U experiments. Here, m and n are poloidal and toroidal mode numbers; j_{EC} and j_{BS} are EC-driven current density and bootstrap current density at the mode rational surface; W_{sat}, W_{marg} and d_{EC} are full island width at saturation, marginal island width at which the island spontaneously decays and full width at half maximum of ECCD profile, respectively. Simulation of magnetic island evolution using TOPICS code with the modified Rutherford equation also supports this result and indicates that the threshold value is $j_{EC}/j_{BS} \sim 0.4$. Effect of ECCD deposition width on stabilization has been experimentally compared between $W_{sat}/d_{EC} \sim 1.5$ and 3. The result indicates that similar dependence of stabilization effect on ECCD location is obtained by normalizing the misalignment by ECCD deposition width.

EX/6-1 · MHD Induced Fast-Ion Losses on ASDEX Upgrade

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Abstract: In magnetically confined fusion plasmas, fast (i.e., suprathermal) ions must be well confined until they transfer their energy to the plasma. This is of crucial importance for next step fusion devices like ITER where an alpha-particle loss below 5% of their nominal power is mandatory to avoid grave consequences. MHD instabilities can be driven by a population of fast ions but they also can lead to an enhancement of the fast-ions radial transport. Therefore, the interplay between MHD instabilities and fast-ions must be well understood. A detailed knowledge of the underlying physics can be gained from direct measurements of MHD induced fast-ion losses (FILs). The ASDEX Upgrade team has made considerable progress in this research field over the last years, as reported in this paper. Time resolved energy and pitch angle measurements of FILs correlated in frequency and phase with low-frequency (e.g., NTMs) and high-frequency (e.g., TAEs) MHD instabilities have been obtained with a scintillator based FIL-detector. A detailed analysis of FILs due to TAEs has revealed the existence of a new core localized MHD perturbation, the Sierpes mode. The Sierpes mode is a non-Alfvénic instability which dominates the transport of fast-ions in ICRF heated discharges. The internal structure of both, TAEs and Sierpes mode has been reconstructed by means of highly-resolved multichord soft X-ray measurements. A spatial overlapping of their eigenfunctions leads to a fast-ion loss coupling, showing the strong influence that a core-localised fast-ion driven MHD instability may have on the fast-ion transport. On the modelling side, we have identified the FIL mechanisms due to NTMs as well as due to TAEs. The drift islands formed by fast-ions in particle phase space are responsible for the loss of NBI fast ions due to NTMs. In ICRF heated plasmas, a resonance condition fulfilled by the characteristic trapped fast-ion orbit frequencies lead to a phase-matching between fast-ion orbit and NTM or TAE magnetic perturbation. The banana tips of a resonant trapped fast-ion bounce radially due to an $E \times B$-drift in the TAE-case. The NTM radial bounce of the fast-ion banana tips is caused by the radial component of the perturbed magnetic field lines. Finally, we will discuss the implications of these results for MHD induced fast-ion transport in ITER.

EX/6-2 · Alfvénic Instabilities and Fast Ion Transport in the DIII-D Tokamak

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Abstract: Future burning plasma experiments such as ITER may be subject to the excitation of Alfvén eigenmode (AE) instabilities by 3.5 MeV fusion born alpha particles that are needed to sustain the thermonuclear burn as well as neutral beam ions critical for current drive, heating, and momentum input. If allowed to grow unabated, these instabilities have the potential to cause fast ion redistribution or loss and possible serious damage to first wall components. Recent experimental results from the DIII-D tokamak have validated several key elements of AE theory including the linear coupling of multiple AEs, excitation of AEs by sub-Alfvénic beam ions via higher order resonances, and spatial eigenmode structure of various AEs in reversed shear plasmas. These studies have also yielded several surprising findings. Localized electron cyclotron heating (ECH) near the mode location stabilized reversed shear Alfvén eigenmode (RSAE) activity and resulted in significantly improved fast ion confinement relative to discharges with ECH deposition on axis. In these discharges, RSAE activity was suppressed when ECH was deposited near the radius of the shear reversal point and enhanced with deposition near the axis. During the beginning of the discharge when both cases have similar mode amplitudes, the fast ion density (n_{FI}) as measured by the Fast Ion D_{alpha} spectroscopy diagnostic is similar. Later, when the two levels of AE activity differ significantly, n_{FI} is almost 50% larger in the case with stabilized RSAEs. As a direct result of this improved fast ion confinement the central rotation and neutron emission are also enhanced. The most detailed orbit-following simulations to date utilizing multiple experimentally validated eigenmodes failed to reproduce the large measured fast ion transport. NOVA calculations of the largest ($\Delta B/B \leq 7 \times 10^{-4}$) eleven 3D eigenmode structures for a single well-diagnosed discharge were matched with experimental measurements and used in combination with the ORBIT guiding centre following code to simulate neutral beam ion redistribution. While ORBIT simulations predict transport in the correct locations, the change in the distribution function is far smaller than the empirically observed level, even with 5 times the measured eigenmode amplitude.

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EX/6-3 · Toroidal Alfvén Eigenmode Avalanches

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Abstract: While single Toroidal Alfvén Eigenmodes (TAE) are not expected to cause substantial fast ion transport in ITER, multiple modes, particularly if they strongly interact and become non-linear, can affect ignition thresholds, redistribute beam-driven currents and damage internal components. Experiments on the National Spherical Torus Experiment have found the fast ion β threshold for excitation of TAE. A further increase in beam heating power is seen to push the TAE into a repetitive cycle of increasingly stronger bursts, each cycle culminating in a large, multi-mode burst and a drop in the neutron rate of $\approx$ 10%. These strong bursts are identified as TAE avalanches, a nonlinearity introduced through perturbations to the fast ion distribution [1]. The plasma conditions are fully diagnosed, with MSE measurements to constrain the q-profile. Eigenmode profiles and amplitudes are measured with a five channel reflectometer array and show good agreement with the NOVA simulations. Redistribution, transport and loss of fast ions is seen with Neutral Particle Analyzer (NPA) diagnostics, which provide measurements of the population of confined fast ions, as well as fast neutron rate diagnostics and D_{alpha} monitors, which respond to fast ions lost from the plasma and hitting limiter surfaces. Direct, nonlinear three-wave interactions between Compressional Alfvén Eigenmodes (CAE) and both energetic particle modes (EPM) and TAE have also been conclusively identified. These nonlinear interactions occur simultaneously with other three-wave interactions between TAEs and EPMs. The nonlinear interactions occur during fast-ion loss events and lead to a spatial redistribution of CAE fluctuation energy that will modify the effect of the CAEs on fast-ion orbits. The modes are studied with an array of reflectometers, as well as with arrays of Mirnov coils.

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[1] Nucl. Fusion 35 (1995) 1661

EX/7-1Ra · High β Plasmas Exceeding Dual Stability Thresholds in the MST RFP

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Abstract: High-beta MST plasmas have been produced that exceed the stability thresholds for both local and global pressure-driven modes, but these plasmas also exhibit improved energy confinement [1]. Total beta, the average plasma pressure normalized to the total edge field pressure, reaches 26%, the largest value yet attained in the ohmically-heated RFP. Toroidal beta, normalizing to the edge toroidal field pressure, is about 100%. high β is achieved by injecting deuterium pellets into MST plasmas with inductive current profile modification and improved confinement. The pressure gradient in these plasmas exceeds the Mercier criterion for local interchange, but there is no experimental indication of interchange modes affecting plasma performance. This is similar to recent results in helical devices such as LHD. The pressure gradient is also large enough to affect the stability of core-resonant tearing modes, representing a new regime for the RFP. These modes cause the bulk of energy transport in standard RFP plasmas and are normally driven entirely by a gradient in the current profile. This drive is reduced by current profile modification.

[1] M.D. Wyman, B.E. Chapman et al., Phys. Plasmas 15, 010701 (2008).

EX/7-1Rb · Effects of Lowering the Aspect Ratio on MHD Behaviour in a Reversed Field Pinch

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Abstract: The low-aspect ratio (A) reversed field pinch (RFP) appears to offer attractive physical properties such as enhanced neoclassical bootstrap current and simpler MHD mode dynamics. The RELAX (ReEversed field pinch of Low-Aspect ratio experiment) machine with the world's lowest A of 2 ($R/a = 0.5$m/0.25m) has been constructed to explore the RFP properties in low-A regime. In flat-topped low-A RFP discharges in RELAX, plasma current of $\sim$ 50 kA has been attained with discharge duration of 2 ms. Neoclassical equilibrium analysis using a newly developed reconstruction code has shown that the bootstrap current fraction is lower than $\sim$ 5% in the present plasmas. In round-topped discharges with plasma current of $\sim$ 70 kA, quasi-periodic growth and decay of the dominant $m = 1/n = 4$ mode has been observed. When the dominant mode grows, the toroidal mode spectrum appears to be that of the quasi-single helicity (QSH) RFP state. A fast camera diagnostic has also revealed the appearance of a simple helical structure. These MHD phenomena resulting from the low-A characteristic of RELAX will be described.

EX/7-2Ra · Heat Load on Plasma Facing Components during Disruptions on JET

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Abstract: When a plasma disrupts, the thermal and magnetic energies are quickly lost (time scales about 1 ms and 20 ms, respectively, on JET) leading to high heat loads on plasma facing components (PFCs). Experimental results from AUG and JET indicate that the plasma may loose a significant part of its thermal energy prior to the thermal quench, resulting in reduced heat loads during the thermal quench phase. In addition, it has been observed that only $10-50\%$ of the plasma thermal energy is deposited on the divertor targets, suggesting that the remaining $50-90\%$ may be deposited onto the main chamber PFCs. This hypothesis was supported by preliminary observations using a wide angle infrared camera on JET, which indicated that a significant fraction of this energy is indeed deposited onto the inner wall and the upper dump plate. Here, recent power load measurements onto PFCs during disruptions on JET (using the above mentioned IR system) are reported. Different types of disruptions including vertical displacement events (VDE) due to loss of control, density limit disruptions (DLD) and radiation limit disruptions (RLD) are investigated. The heat load profiles on the upper dump plate, calculated from the toroidally-averaged surface temperature using a finite elements code, are compared. The peak heat load exhibits 3 distinct phases: the pre-disruption phase, the thermal quench and the current quench. For the VDE, the heat load prior to the thermal quench is comparable with the one during the thermal quench and is believed to be due to the rapid shrinking and upward displacement of the plasma. The heat loads measurements show that up to 60% of the thermal energy is released onto the upper dump plate during the disruptions (starting from the thermal quench). Using the new fast bolometer measurements, the radiated energy is also measured distinguishing the three phases. Most of the energy is radiated during the current quench and correspond to about $30-40\%$ of the total available magnetic energy. The energy balance presented here is still not complete, but is the most complete ever achieved for JET.

EX/7-2Rb · Progress in Understanding Halo Current at JET

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Abstract: In the $2004-5$ shutdown, 4 arrays of halo current sensors with a toroidal spacing of 90 degrees have been installed behind the JET upper dump plate (DP). Each array has 1 to 2 toroidal field pick up coils and up to 8 Rogowski coils. This set supplements the one installed in 2001 behind the Improved Upper Inner Wall Protection (IUIWP), which comprises only 3 toroidal locations, each equipped with one large Rogowski coil and one toroidal field pick up coil. Additionally, during the 2007 shutdown protective covers have been added in front of the in-vessel cable bundles to reduce charge collection and diminish the drift observed in the Rogowski coils. The Rogowski coils were used to collect JET halo current footprint data during dedicated experiments in 2007. A typical deliberate upward Vertical Displacement Event (VDE) of a 1.5 MA ITER-like configuration leads to halo current entering the outboard of the DP and exiting partially in the inboard of the DP and partially in the IUIWP, with the plasma usually tangent on row $2-3$ of the DP. This poloidal distribution of the halo current is consistent with the picture provided by the plasma boundary reconstruction. At least initially, the DP and the IUIWP data agree: the DP net halo current is equal (and opposite) to the IUIWP halo current. During these VDEs the current density peaks at ~ 50 kA/m^2 on row $4-5 \sim 15$ ms after contact, within the JET design criteria (< 300 kA/m^2 in a 6 MA event). The DP set can accurately calculate the Toroidal Peaking Factor (TPF) of a pure $n=1$ mode, while the IUIWP set could underestimate it by 25%. The TPF may still be underestimated if higher toroidal mode numbers are dominant. Halo current TPF and fraction from the IUIWP and the DP sets are in good agreement. In particular the TPF seen by the DP is not larger than that seen by the IUIWP. This supports the validity of the IUWP data. In addition, the 2007 dedicated experiments reinforce the statement that JET does not observe events with TPF higher than 2, as they include a set of discharges where the vertical kick strength and duration was scanned to find the combination with the highest halo fraction and TPF in a highly shaped plasma at low boundary safety factor. Results from both 2007 and 2008 dedicated experiments will be presented in detail and their portability to ITER discussed.

EX/7-2Rc · Study of Current Decay Time during Disruption in JT-60U Tokamak

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Abstract: Disruption is one of the most crucial issues for the tokamak fusion reactors. In order to estimate the electromagnetic force acting on in-vessel components during disruption, precise prediction of plasma current decay time is important. In the design of the ITER, so called, L/R model is used to predict the decay time. In this model, the plasma inductance is assumed to be constant in time, and the electron temperature and effective charge during disruption are important parameters. We evaluate the plasma current decay time during various disruptions of β collapse, density limit and resistive wall mode in JT-60U tokamak experimentally. The features of this study are as follows; (i) the fast time evolution of the electron temperature during disruption is estimated based on the measurement of the He I line emission intensity ratios, and the plasma inductance and cross-section area are estimated by the CCS (Cauchy-Condition Surface) method with the magnetic signals. (ii) We suggest a modified L/R model, in which the time evolution of the plasma inductance is taken into account, and it is clarified that the change of the plasma internal inductance associated with current profile plays an important role to predict the current decay time during disruption.

EX/7-3Ra · Disruption Control on FTU with ECRH

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Abstract: The use of ECRH has been investigated in FTU (Frascati Tokamak Upgrade) as a promising technique to mitigate some of the problems associated with disruptions (current quench, runaway production). Disruptions are produced in 0.5 MA and 0.35 MA deuterium plasmas by injecting Mo through laser blow-off or by puffing deuterium gas above the Greenwald limit. The toroidal magnetic field ($B_t = 5.3$ T) is kept fixed and the ECRH launching mirrors are steered before every discharge in order to change the deposition radius. The loop voltage signal is used as disruption precursor to trigger the ECRH power before the plasma current quench. The injected ECRH power has been varied in the range $0.4 - 1.2$ MW. The application of ECRH modifies the current quench starting time depending on the power deposition location. Various fine scans in deposition location have shown that the direct heating of one of the magnetic islands produced by magnetohydrodynamic (MHD) modes (either 3/2, 2/1 or 3/1) prevents its further growth and also produces the stabilization of the other modes (indicating that those modes are toroidal sidebands of each other and their harmonics) and current quench delay or avoidance. Disruption avoidance and complete discharge recovery is obtained when the ECRH power is applied on rational surfaces. The Rutherford equation has been used to reproduce the evolution of the MHD modes. The modes involved in the disruption are found to be tearing modes stabilized by a strong local ECRH heating. The model results are compared with the experimental data and are in agreement with the minimum absorbed power values found for disruption avoidance (0.4 MW at 0.5 MA with deposition on the $q = 2$ surface).

EX/7-3Rb · Fast Plasma Shutdowns Obtained With Massive Hydrogenic, Noble and Mixed-Gas Injection in DIII-D

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Abstract: Massive gas injection (MGI) experiments conducted in DIII-D with hydrogen, deuterium, helium, neon and argon and mixed (hydrogen + argon and deuterium + neon) gases injected into "ITER-similar" 1.3 MA ELMy H-mode plasmas are described. Injection duration is typically less than 2 ms; injected quantities reach 10^{23} atoms or molecules. Gas species, injected quantity, delivery rate and intrinsic and added impurities (i.e., gas mixtures) are found to affect the disruption mitigation [1] and runaway electron avalanche mitigation [2] attributes of the resulting fast plasma shutdowns. Low-Z injection does not produce runaway electrons. With sufficient injected gas quantity, effective disruption mitigation is obtained for all species. Optimal results for simultaneous disruption and runaway avalanche mitigation are obtained with massive pure helium injection, 3×10^{22} helium delivery in less than 2 ms. This short-pulse injection scenario yields a promising combination of moderately fast current quench, record free-electron densities, up to 2×10^{21} electrons per cubic meter, gas assimilation fractions that approach 0.4 and "Rosenbluth density" ratios [2] that approach 0.1 at the time of plasma current quench onset. Indications of favorable scaling of gas assimilation fraction with increasing gas quantity are seem for all low-Z gases. The results of the present experiments provide a rich source of validation data for emerging MHD/radiation simulation models and new insight about design of effective gas injection systems for disruption and avalanche mitigation in ITER. Gas delivery pulse durations that are less than the corresponding time for onset of radiative thermal collapse are indicated. The quantity of injected gas must also include allowance for finite assimilation at the time of current quench onset.

[1] Progress in the ITER Physics Basis, Chapter 3, Nucl. Fusion **47** (2007). [2] M.N. Rosenbluth, et al., Nucl. Fusion **37** (1997) 955.

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EX/7-3Rc · Effect of the MHD Perturbations on Runaway Beam Formation during Disruptions in the T-10 Tokamak

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Abstract: Acceleration of electrons to high energies (up to 3−30 MeV) is a typical feature of the disruption instability in tokamaks. The electron acceleration is generally connected with enhanced longitudinal electric fields ($E \sim 2{-}5$ V/m) formed in cooled post disruption plasma. Generation of the powerful runaway beams depends critically on amplitude of the electric field and presence of the initial ("seed") population of the nonthermal electrons. Production of the "seed" runaway electrons can be facilitated due to strong electric fields ($E \sim 10 - 50$ V/m) induced temporarily during magnetic reconnection at the initial stage of the disruption. Essential new result of the present experiments in the T-10 tokamak is identification of the beam birth at the initial stages of the disruption and its amplification during subsequent bursts of the MHD modes at series of minor disruptions. Analysis of the non-thermal x-ray bursts indicated that electron beams are initiated within the plasma core. Numerical modelling confirms indirectly connection of the "seed" runaway electrons with magnetic reconnection. Beams generation depends critically on rate of the reconnection and amplitude of the MHD perturbations. Production of the nonthermal electrons is enhanced in plasmas with strong induced electric fields (during fast changes of the perturbed helical magnetic fluxes), while it is less pronounced in presence of the continuous MHD perturbations with saturated amplitude. Auxiliary plasma heating using electron cyclotron waves are used in T-10 to prevent abrupt growth of the MHD modes at various stages of the disruption. Experiments have demonstrated that elimination of the bursting energy quenches (possibly connected with the reconnection) and formation of the saturated MHD modes at the initial stage of the disruption can delay and in some cases prevent formation of the primarily runaway beams. New saddle coils system is designed in the T-10 tokamak for generation of the resonant magnetic perturbations and analysis of their effect on the non-thermal electrons.

EX/8-1Ra · High Density High Performance Plasma with Internal Diffusion Barrier in Large Helical Device

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Abstract: An attractive high density plasma operational regime has been discovered in the intrinsic helical divertor configuration on the Large Helical Device (LHD). An Internal Diffusion Barrier (IDB) which enables core plasma to access a high-density/high-pressure regime has been developed. It was revealed that the IDB could be reproducibly formed by pellet fueling in the magnetic configurations shifted outward in major radius. Attainable central plasma density exceeds 1×10^{21} m^{-2}. Central pressure reaches 1.5 times atmospheric pressure and the central β value becomes fairly high even at high magnetic field, i.e., 5.5% at $B = 2.57$ T. No significant impurity contamination has been observed. The IDB is an encouraging finding and it demonstrates the potential for an alternative path to a fusion energy reactor at relatively high-density/low-temperature in a helical device.

EX/8-1Rb · Two Approaches to the Reactor-relevant High-beta Plasmas with Profile Control in the Large Helical Device

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Abstract: In order to realize economical fusion reactor, high-beta operation of plasma is required. Volume averaged $\beta \sim 5\%$ has been achieved in the LHD; the potential of the Heliotron type configuration as a reactor has been thereby demonstrated. This high-beta regime is realized by the increase in the heating power and by the optimization of the magnetic configuration. The heating efficiency of NB in the lower magnetic field is better when the magnetic axis remains unshifted. We adjust the aspect ratio of the plasma so that the Shafranov-shift of the magnetic axis is reduced. The edge MHD instabilities remain. However, the magnitude of them are saturated in a low level and do not make a serious deterioration of the confinement. They are identified as the resistive interchange modes. Therefore, the effect will be reduced when the plasma parameters approach reactor-relevant values. From an MHD point of view, there are many advantages in the peaked profile plasmas; the magnetic well is deeper in the core region from the larger Shafranov-shift and the pressure gradient in the edge region is smaller. A peaked profile is formed in the recovery phase after sequentially injected hydrogen pellets. While the electron density decreases after the pellets, the electron temperature recovers quickly. In this recovery phase, the pressure profile becomes peaked; high-central-beta plasma is formed in this phase. Though the final plasma with peaked pressure is stable, MHD stability is important in the process of the formation. When the vacuum magnetic axis Rax-vac is located inward (e.g., Rax-vac = 3.6 m), sawtooth-like instabilities are activated when the pressure profile is being peaked. The peaking of the plasma is thereby disturbed and the degree of peaking is small. On the contrary, in the outward-shifted cases (Rax-vac > 3.7 m), the achieved electron density is higher; a higher central-beta β_0 can be achieved. However, the increase of β_0 is limited by the so-called core density collapse events when the magnetic axis position exceeds 4.1 m. The magnetic axis should be then kept between 3.7 m and 4.1 m in order to avoid these two unstable regions. The highest β_0 is obtained with $B_t = 0.65$T. The central β ($\sim 9.9\%$) is comparable to the value in the highest averaged-beta discharge (Rax-vac = 3.6 m and $B_t = 0.425$ T) with a higher toroidal magnetic field.

EX/8-2Ra · Characteristics of High-Ion-Temperature Plasmas Heated by Neutral Beams in the Large Helical Device

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Abstract: Improvement of ion heat transport in Heliotron plasmas has been realized by upgrade of ion heating power by using low energy neutral beam (NB) in the Large Helical Device (LHD). The positive-ion-based NBI (p-NBI) with low energy of 40 keV was newly installed for ion heating experiments and was also utilized to measure T_i profile using charge exchange spectroscopy (CXS) measurement, which consequently leads to the extension of heliotron plasmas in the high-T_i regime and significant progress of ion transport study in LHD. The peaked profile of ion temperature (T_i) with steep gradient has been formed in the core region of high-power NB heated plasmas in LHD. The Ti strongly depends on direct ion heating power of NBs and is proportional to $P_i^a; a = 1.0 - 1.3$, while the prediction on scaling basis gives $a \sim 0.4$. The central T_i of 6.8 keV has been achieved in hydrogen plasma with the electron density of 2×10^{19} m^{-3}. The T_i increase is a transient phenomenon, and the duration time of the high-Ti phase is about 120 ms, which is a few time longer than the energy confinement time. The electron temperature is 3.5 keV and no improvement of electron transport was observed in these discharges. The experimental ion thermal diffusivity significantly decreases to the neoclassical level, indicating reduction of anomalous transport, which is the first observation in LHD. The analysis of neoclassical ambipolar diffusion showed the enhancement of the negative radial electric field ($E_r < 0$) associated with the T_i increase, indicating that the anomalous transport is significantly suppressed by the enhanced negative E_r. This improved mode is characterized by the reduction of ion heat transport associated with T_i increase, that is; $\partial\chi(i)/\partial T_i < 0$. Therefore, it is considered to be possible to extend the helical plasma in the further high-T_i regime by utilizing this feature.

EX/8-2Rb · Dynamics of Ion Internal Transport Barrier in LHD Heliotron and JT-60U Tokamak Plasmas

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Abstract: Dynamics of ion internal transport barrier (ITB) formation and impurity transport both in the Large Helical Device (LHD) heliotron and JT-60U tokamak are described. Significant differences in the formation of ITB between heliotron and tokamak are observed. The ITB formation, which is characterized by the abrupt appearance of large gradient region (ITB region) of temperature interior of the plasma, has been observed both in LHD and JT-60U plasmas. Although the mechanism of turbulence suppression and the reduction of thermal diffusivity can be explained by the $E\times B$ shear, the mechanism determining the magnitude and size of ITB (magnitude of the maximum ion temperature gradient and region of large ion temperature gradient) is not well understood. Therefore it is important to compare the magnitude and size of ITB and also the formation characteristics (how the ITB region develops) between the ITB plasma in LHD and ITB plasma in JT-60U in order to understand ITB physics more comprehensively. The formation of ITB in the LHD plasma is characterized by the expanding ITB to the wide region of the plasma and that in the JT-60U plasma with negative magnetic shear is localized ITB at the narrow region of interior plasma and does not extend to the plasma core where a rotational transform is small. Simultaneous achievement of high ion temperature and low concentration of impurity is crucial for the high fusion triple product, because the impurity causes dilution of fueling particle. Since the particle diffusion in the plasma with ITB is relatively low because of the suppression of the turbulence, the sign of the convection, which appears as an off-diagonal term of the transport matrix, becomes important in the ITB plasma. The sign differences of convection of impurity transport are observed in LHD and JT-60U ITB plasmas. The inward convection in JT-60U is predicted by the neoclassical impurity transport because of the large density gradient. This is in contrast to that no outward convection is predicted in the neoclassical impurity transport in LHD because of the negative radial electric field predicted in the ITB plasma, where the ion temperature is much larger than the Te by a factor of two. The outward convection of the impurity transport in the ITB plasma is considered to be beneficial for the fusion relevant plasma in future.

EX/8-3 · Internal Transport Barrier Dynamics with Plasma Rotation in JET

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Abstract: In advanced Tokamak scenarios internal transport barriers (ITB) are considered to be the prime candidate for enhancing the confinement. Both magnetic shear and the rotational shear have been identified as playing a determining role in the initial onset of an ITB (triggering) and its subsequent development. In this context, the understanding of the ITB dynamics has to be explored by trying to decouple the effects of heating on one hand and torque on the other with the ultimate objective of identifying the minimum torque required for the formation and development. Decoupling these two effects should also assist in separating the physics of the ITB triggering with the physics of its sustainment. Several experiments have been developed at JET to study these aspects. First of all, JET has the unique ability to modify the amplitude of the toroidal magnetic field ripple. By increasing the ripple, discharges with a similar absorbed power but different rotational shear could be created. ITB triggering events were observed in all cases and it is thought that this process is more determined by the q-profile characteristics and less by the rotational shear. However, the increase in pressure gradient following the ITB trigger was reduced in the presence of a lower rotational shear. This suggests that toroidal rotation and its shear plays a role in the growth of the ITB once it has been triggered. Secondly, ITBs were studied in discharges with a low toroidal torque using predominantly ion cyclotron heating in the ^{3}He minority ion scheme. These discharges showed ITB trigger events, but no further growth of the ITB. Again this suggests that the rotational shear rotation plays a small role in the ITB trigger mechanism while strong torque looks necessary to establish strong ITBs. In the presence of well established ITBs, local accelerations of the poloidal rotation, considerably higher than neoclassical values, have been measured, that could also be a component in creating the conditions leading to the ITB sustainment. These experiments could provide ultimately a minimum rotation profile requirement for developing ITBs in steady state ITER scenarios. This study already suggests that ITB triggering events are likely to be observable in ITER plasmas even at low input torque.

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EX/8-4 · Edge Pedestal Control in Quiescent H-Mode Discharges in DIII-D Using Co Plus Counter Neutral Beam Injection

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Abstract: Quiescent H-mode (QH-mode) plasmas in DIII-D with co plus counter neutral beam injection have demonstrated active control of the edge pedestal that can be used to optimize the edge conditions in future burning plasma devices such as ITER. Edge localized mode (ELM)-free QH-mode plasmas have demonstrated that the three reactor requirements of: (1) steady operation at maximum stable pedestal pressure, (2) ELM-free operation and (3) rapid particle transport for helium exhaust can be met simultaneously in discharges which operate with constant density and radiated power. Altering the torque input to QH-mode plasmas allows continuous adjustment of the pedestal density, pressure and particle transport over a range of about a factor of two while maintaining the ELM-free state. This active control capability allows operation near but below the ELM stability boundary. These plasmas exhibit edge particle transport more rapid than that produced by ELMs while operating at reactor relevant pedestal $\beta \sim 1\%$ and collisionality ~ 0.1; pedestal densities up to 1/2 the Greenwald density have been achieved. The essential feature, which distinguishes QH-mode from standard ELMing H-mode is the presence of an edge-localized electromagnetic mode, the edge harmonic oscillation (EHO). The EHO provides extra particle transport which prevents ELMs by keeping the edge pressure below the peeling-ballooning mode boundary. The EHO is spontaneously generated by the plasma itself and requires no external coils to generate a perturbed magnetic field as is necessary, for example, for ELM suppression via resonant magnetic perturbations. Edge stability calculations using the ELITE code show that the QH-mode operating point is near the peeling boundary. Much of the physics of the EHO is consistent with a model in which the EHO is an edge kink-peeling mode that is destabilized by shear in the edge toroidal rotation at an edge current density slightly below that on the standard ELM boundary. Peeling-ballooning stability calculations have been used as a guide in developing the best plasma shape. Experiments confirm the prediction that a high-triangularity, double-null plasma has the best stability against ELMs. Plasmas with the best stability require the least reduction in edge transport to achieve ELM-free operation.

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EX/8-5 · Heat Transport and Pedestal Structure of H-mode in the Variation of Current Density Profiles in JT-60U

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Abstract: H-modes operated at higher l_i with the current ramp down have shown higher energy confinement with higher density. The H-factor evaluated for the core plasma depends strongly on l_i with the relation of $H_{89core} \sim l_i^{0.74}$ for the case without sawtooth activities. Center peaked profiles of electron density and electron temperature are obtained in high l_i H-modes. While the peripheral current density profiles are largely modified by the current ramp, the pedestal pressure is not significantly changed. Thus, the increased density in high l_i H-modes is attributed to the peaked density profile with no explicit difference in pedestal profiles. It was found for the first time that the electron heat diffusivity is reduced at the centre in high l_i case, resulting in the centre peaked T_e profile while the T_i profiles are approximately stiff.

EX/9-1 · Deuterium Inventory in Tore Supra : Reconciling Particle Balance and Post Mortem Analysis

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Abstract: Fuel retention in the vessel walls of tokamaks is a major concern for next step devices, as the tritium inventory is limited for safety reasons. With its ability to perform discharges on relevant durations with actively cooled components, Tore Supra offers a unique opportunity to address these issues in true steady state from the plasma wall interactions point of view. This paper reports the recent progress achieved in this field, in particular to understand the discrepancy between particle balance and post mortem analysis, the two methods commonly used to assess fuel retention, and to identify the main retention mechanisms at stake. In order to do so, a dedicated experimental campaign was performed, followed by the dismantling of selected plasma facing components to extract samples and an extensive analysis phase in the frame of the EU Plasma Wall Interaction Task Force. The aim of the campaign was to load the vessel walls with a significant additional deuterium (D) inventory. Repetitive long pulses (2 minutes) were performed, allowing to run the equivalent of 1 year of operation within 2 weeks. No sign of loss of density control or saturation of wall retention was observed, although 5 hours of plasma were carried out without any conditioning procedure. The retention rate is found to be constant throughout the campaign. A final wall inventory of $\sim$ 10 g of D was reached at the end of the campaign, corresponding to 52% of the injected gas. Plasma facing components, selected in the 3 zones of interest (erosion, thin deposits and thick deposits), were then dismantled for analysis. Thermodesorption was carried out to determine the global D content of the samples. A preliminary extrapolation from these measurements would lead to $\sim$ 40% of the inventory deduced from particle balance already found, in better agreement than previous results. In addition, Nuclear Reaction Analysis and Secondary Ion Mass Spectrometry were performed to determine the D profile as well as the impurity concentration within the samples. In all cases, results seem to indicate a significant penetration of D inside the carbon fiber composite (CFC), well beyond the implantation depth. Sophisticated analysis tools, such as a NRA microbeam, are then used to assess the D distribution at the micron scale, showing that porosities between the CFC fibers and matrix could play an important role.

EX/9-2 · Non-Boronized Operation of ASDEX Upgrade with Full-Tungsten Plasma Facing Components

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Abstract: The paper reports about plasma operation and H-mode performance in ASDEX Upgrade with 100% tungsten plasma facing components (PFCs). The conversion into a fully tungsten coated tokamak was finished prior to the 2007 experimental campaign, which was carried out without boronisation. Soon after the first successful discharge a milestone pulse was achieved with a full performance H-mode at $\beta_N = 2$, 70% of the Greenwald density and confinement factor $H_{98} = 1$. The carbon content of the ASDEX Upgrade plasma was reduced less than originally expected to values around 0.5%. Remaining sources, with unknown relative contributions, are arc traces cutting through the 4 micron W coating at the inner divertor baffle, photo-desorption from remote surfaces, and the release of carbon from uncoated tile sides. Residual gas analysis suggests that C erosion by O forming CO and CO_2 also contributes to the carbon content in in the unboronised device. Nevertheless, strongly reduced C deposition in remote areas was observed. The recycling nature of carbon due to chemical erosion is supposed to explain the resilience of the core C concentration against the reduction of the primary sources. The H-mode operational space turns out to be limited towards lower density by central tungsten accumulation. Medium density H-mode operation is possible with central ECRH heating, while the beneficial effect of central ICRH suffers from increased tungsten sources due to rectified sheath formation. The limitation of H-mode working space can be understood by the central power balance. As long as enough central power flow, Pheat-Prad, is available, central ion transport is anomalous. For lower heat fluxes, ion transport gets neoclassical, and the Ware pinch leads to a peaking of the central density profile. This process then self-amplifies due to increased central W radiative losses. Higher density operation is advantageous due to the lower central W content caused by the combined effects of lower sputtering yields and more frequent W expulsion by ELMs.

EX/9-3 · Dust Studies in DIII-D and TEXTOR

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Abstract: Studies of naturally occurring and artificially introduced carbon dust are conducted in DIII-D and TEXTOR. Dust production and accumulation impose safety and operational concerns for ITER by contributing to tritium inventory rise and leading to radiological and explosion hazards. In DIII-D, dust does not present operational concerns except immediately after entry vents. In the first $2-3$ plasma discharges after an entry vent cameras detect thousands of dust particles per discharge. After a few days of operations (~ 70 discharges) dust levels are reduced to a few observed events per discharge. Energetic plasma disruptions produce significant amounts of dust. However, dust production by disruptions alone is insufficient to account for the estimated in-vessel dust inventory in DIII-D. Submicron sized dust is routinely observed using Mie scattering from a Nd:Yag laser. The source is strongly correlated with the presence of Type I ELMs. Migration of pre-characterized carbon dust is studied in DIII-D and TEXTOR by injecting micron-size dust in plasma discharges. In DIII-D, a sample holder filled with ~ 30 mg of dust is introduced in the lower divertor and exposed to high-power lower single-null ELMing H-mode discharges with strike points swept across the divertor floor. After a brief exposure (~ 0.1 s) at the outer strike point, part of the dust is injected into the plasma, raising the core carbon density by a factor of $2-3$ and resulting in a twofold increase of the radiated power. Individual dust particles are observed moving at velocities of $10-100$ m/s, predominantly in the toroidal direction, consistent with the drag force from the deuteron flow and in agreement with modelling by the 3D DustT code. In TEXTOR, instrumented dust holders with $1-45$ mg of dust are exposed in the scrape-off layer $0-2$ cm radially outside of the last closed flux surface in neutral beam heated discharges (NBI power of 1.4 MW). Dust is launched either in the beginning of the discharge or at the initiation of NBI. Preliminary analysis of the video sequences show the launch of dust perpendicular to the toroidal magnetic field presumably because of $E \times B$ drift.

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EX/9-4 · Study on Impurity Screening in Edge Ergodic Layer of Large Helical Device

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Abstract: Impurity screening in ergodic layer of the Large Helical Device (LHD) has been demonstrated with the measurement of carbon line radiation. When the screening occurs in the ergodic layer, the carbon ions with higher charge states decrease compared to ones with lower charge states. Since the radiation intensity from each charge state is not so sensitive to the edge Te unless it has a peculiar profile, the change in such carbon density behaviors is found to appear in the radiation intensity. i.e., radiation intensity from CIII and CIV monotonically increase with increasing density, while those of CV and CVI stay almost constant, resulting from screening. The result indicates a great advantage of heliotron configuration with the ergodic layer, i.e., intrinsic prevention of impurity contamination to core plasma. Utilizing 3D edge transport modelling, it is identified that the screening is caused by the friction force exerted on the impurity with background plasma flow. The effect is switched on when the friction force dominates over the thermal force with increasing density. With higher ergodization of edge magnetic field, the screening becomes stronger due to the larger radial impurity out-flux induced by the braiding field lines. Classical force balance analysis shows that this effect is insensitive to charge states (Z). It is then expected to be applicable also for heavy impurities. It should be noted, that the impurity screening by the friction force is usually limited only to near-divertor region, and thus there is no effective screening against the impurity entering scrape-off layer around mid-plane in tokamak X-point divertor configuration. On the other hand, the ergodic layer can provide impurity screening effect in all poloidal direction due to the braiding magnetic field lines caused by island chains.

EX/10-1 · Dependence of the L- to H-mode Power Threshold on Toroidal Rotation and the Link to Edge Turbulence Dynamics

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Abstract: Injected power required to induce a transition from L-mode to H-mode plasmas is found to depend strongly on the injected neutral beam torque and consequent plasma toroidal rotation. Edge turbulence and flows, measured near the outboard midplane of the plasma ($0.85 < r/a < 1.0$) on DIII-D with the high-sensitivity 2D beam emission spectroscopy (BES) system, likewise vary with rotation and suggest a causative connection. L-H power threshold in plasmas with the ion ∇B drift away from the X-point decreases from $4 - 6$ MW with co-current beam injection, to $2 - 3$ MW near zero net injected torque, and to < 2 MW with counter injection. Plasmas with the ion ∇B drift towards the X-point exhibit a qualitatively similar though less pronounced power threshold dependence on rotation. Preliminary analysis of similar ECH-heated plasmas (with ion ∇B drift towards the X-point) indicates that the L-H power threshold with zero-torque electron heating is similar to or less than that of comparable balanced-torque NBI-heated discharges. 2D edge turbulence measurements with BES show an increasing poloidal flow shear as the L-H transition is approached in all conditions. As toroidal rotation is varied from co-current to balanced in L-mode plasmas, the edge turbulence changes from a uni-modal character to a bi-modal structure, with the appearance of a low-frequency ($f = 10 - 50$ kHz) mode propagating in the electron diamagnetic direction, similar to what is observed as the ion ∇B drift is directed towards the X-point in co-rotating plasmas. At low rotation, the poloidal turbulence flow near the edge reverses direction prior to the L-H transition, generating a significant poloidal flow shear that exceeds the measured turbulence decorrelation rate and perhaps facilitates the L-H transition. No such reversal is observed in high rotation plasmas. The high frequency poloidal turbulence velocity spectrum exhibits a transition from a Geodesic Acoustic Mode zonal flow to a higher-power, lower frequency zero-mean-frequency zonal flow as rotation varies from co-current to balanced during a torque scan at constant injected neutral beam power, perhaps facilitating the L-H transition. This reduced power threshold at lower toroidal rotation may benefit inherently low-rotation plasmas such as ITER.

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EX/10-2Ra · Turbulent Fluctuations with the Electro Gyro-scale in the National Spherical Torus Experiment

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Abstract: Various theories and numerical simulations support the conjecture that the ubiquitous problem of anomalous electron transport in tokamaks may arise from an electron gyro-scale turbulence driven by the Electron Temperature Gradient (ETG) mode. To check whether such a turbulence is present in NSTX plasmas, measurements of turbulent fluctuations were performed with coherent scattering of electromagnetic waves using a novel scattering geometry with a radial resolution of ± 2.5 cm. Measurements in plasmas heated by high harmonic fast waves (HHFW) confirm the existence of turbulent fluctuations with the electron gyro-scale in the range of $k \perp r_s = 8 - 16$. This seems to suggest that neither the Ion Temperature Gradient mode nor the Trapped Electron Mode is the source of measured fluctuations. Agreement with numerical results from the linear gyrokinetic GS2 code supports the conjecture that the observed turbulence is caused by the ETG mode.

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EX/10-2Rb · Evolution of ETG-mode Scale Turbulence Spectra and Anomalous Electron Transport in Dynamic Experiments at FT-2 Tokamak

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Abstract: Small scale drift wave turbulence is discussed nowadays as a possible candidate for explanation of the anomalous electron energy transport in tokamak plasmas. In the present paper we report results of systematic investigations of the small scale turbulence performed in dynamic (fast (20 MA/s) current ramp up and lower hybrid (LH) heating) experiments at FT-2 tokamak ($R = 55$ cm; $a = 7.9$ cm; $B_t = 2.2$ T). Both frequency and wave number spectra are measured with correlative enhanced scattering (CES) diagnostics. It is found that during the dynamic discharge the turbulence component identified with the dissipative TEM mode possesses a wide q-spectrum which could be described by universal exponential dependence in the range of $3 - 4$ orders of amplitude characterized by two parameters - the turbulence level and scale length. Both parameters are found to decrease substantially when the shear of the poloidal plasma rotation estimated from Doppler frequency shift of the ES signal increases at plasma periphery. Simultaneously transition to the improved confinement resulting in suppression of anomalous electron transport is observed in the experiment. On contrary, the wave number spectrum of the component, identified as the ETG mode, looks very different from exponential. It is characterized by pronounced maximum at wave number corresponding to the largest ETG instability growth rate. Its behavior in the dynamic LH heating experiment is correlated not with the electron thermal conductivity, but with the ratio of electron temperature and density scale lengths which is quite natural near the ETG mode threshold.

EX/P3-1 · Suppression of Turbulent Transport in NSTX Internal Transport Barriers

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Abstract: The understanding of turbulent transport mechanisms and methods for their suppression is crucial for the performance of fusion devices. Electron transport is of particular importance for future burning plasma devices such as ITER where fast particles, both the energetic beam particles and fusion alphas, will primarily heat electrons. Direct and local suppression of electron turbulence via reversed (negative) magnetic shear has recently been observed in reversed shear discharges with internal transport barriers (ITBs) in both the electron thermal, ion thermal, as well as the momentum channels on the National Spherical Torus Experiment (NSTX). Variations in time resolved magnetic shear profiles made using Motional Stark Effect (MSE) internal magnetic pitch angles [1] have been observed to correlate with variations in local fluctuations observed on a high-k microwave scattering diagnostic [2, 3] measuring at electron turbulence wavenumbers. It is observed that the high-k fluctuation amplitude is greater during periods where negative magnetic shear is absent than in periods with strong reversed shear. These observations are consistent with previous non-linear gyrokinetic simulation results from Jenko and Dorland [4] showing reduced transport due to ETG turbulence as a direct result of magnetic shear. These results provide strong evidence that electron transport is suppressed by negative magnetic shear rather than E„eB shear, and that high-k fluctuations are correlated with enhanced electron thermal transport, highlighting the need for current profile control in future burning plasma devices.

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[1] Levinton, F. et. al., Phys Plas. 14, 056119 (2007) [2] Mazzucato, E., Phys. Plasmas, 10, 753 (2003) [3] Smith, D.R., Rev. Sci. Inst., 75, 3840 (2004) [4] Jenko, F. and Dorland, W., Phys. Rev. Lett. 89 225001 (2002)

EX/P3-2 · Characteristics of Internal Transport Barrier under Reactor Relevant Condition in JT-60U Weak Shear Plasmas

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Abstract: Characteristics of internal transport barrier (ITB) have been investigated under reactor relevant condition with edge particle fueling, electron heating and low toroidal rotation in JT-60U weak shear plasmas. The ion temperature (T_i) in the central region decreased both with edge particle fueling by pellet injection or supersonic molecular beam injection (SMBI) and with electron cyclotron heating (ECH). The T_i decrease with edge particle fueling was larger inside the ITB than that outside the ITB, which can be described by cold pulse propagation using the ion thermal diffusivity (χ_i) estimated from power balance analysis. By optimizing the injection frequency and the penetration depth, high confinement was sustained at high density with shallow pellet injections. The T_i ITB significantly degraded when the electron temperature profile shows stiffness against ECH. The value of χ_i in the ITB region increased with the electron thermal diffusivity (χ_e), indicating existence of clear relation between ion and electron thermal transport. The density fluctuation level in the frequency range of ITG measured in the ITB region did not increase with ECH. Thus, longer correlation length could lead enhanced transport rather than increase in fluctuation level. It is important for sustainment of high T_i during electron heating to break this T_e profile stiffness by producing the strong T_e ITB. The increases of χ_i and χ_e can not be explained by destabilization effect of fluctuations due to reduction of sheared flow induced by the change in the toroidal rotation (V_t) toward lower rotation during ECH.

EX/P3-3 · Correlation Between the Edge and The Internal Transport Barriers in JT-60U

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Abstract: For understanding of the physics processes determining the radial profiles of the plasma pressure in the advanced tokamak plasmas, correlation between the edge and the internal transport barriers (ETB and ITB) has been studied. (1) We found that the edge pedestal β_p, β_p-ped, increases almost linearly with the total β_p over a wide range of the plasma current for the type I ELMing H-mode, and the dependence becomes stronger with increasing triangularity. This dependence is not due to the profile stiffness. However, with increasing the stored energy inside ITB (W-ITB), the total thermal stored energy (W-th) increases and then the pedestal stored energy (W-ped) increases. (2) With increasing W-ped, the ELM penetration depth expands more inward and finally reaches the ITB-foot radius. At this situation, the ITB radius cannot move outward and the ITB strength becomes weak. Then the fractions of W-ITB and W-ped to W-th become almost constant. We also found that the type I ELM expels/decreases edge toroidal momentum larger than ion thermal energy. The ELM penetration radius for toroidal rotation is deeper than that for ion temperature, and exceeds the ITB radius. This may cause degradation of ITB.

EX/P3-4 · Pedestal Dynamics in ELMy H-mode Plasmas in JET

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Abstract: Periodic Edge Localised Modes (ELMs) in H-mode plasmas are accompanied by an erosion of the edge pedestal due to both conductive and convective processes. The extent of the collapse varies with ELM size and determines the energy and power load to plasma facing components. In advanced tokamak (AT) scenarios Type I ELMs pose an additional problem in that the inward propagating cold pulse following an ELM crash has the potential to affect internal transport barriers and hence can deteriorate core confinement. ELM mitigation techniques are therefore required. High temporally and spatially resolved electron temperature and density (T_e and n_e) profile measurements at JET reveal the dynamics of T_e and n_e profiles in the Type I ELM cycle. First during a highly dynamic period of < 0.1 ms after the collapse radial profiles of filamentary structures are observed. They are seen to leave the pedestal area, and enter the scrape-off layer (SOL). In the next 2 ms, during the so-called relaxation phase the T_e and n_e geometry evolve differently; The temperature collapse is solely downwards and inside the separatrix, whereas the inward density profile collapse provokes a rise in the density just outside the separatrix. After ~ 2 ms the density in the SOL has disappeared. For standard H-mode conditions it has been observed that the area where the profiles of T_e and n_e are affected by the ELM, is limited to $r/a > 0.85$. However, in JET plasmas relevant for ITER steady state operation, with $q_{95} = 5$ and optimised magnetic shear, ELMs are found to affect a large region of $r/a > 0.5$, and can affect ITBs. Impurity seeding experiments have demonstrated that these ELMs can be mitigated and the ELM affected area can be reduced. ELM mitigation experiments have also been performed using the error field correction coils in JET. Changes in pedestal structure are observed, with a wider pre-ELM p(e) pedestal during the mitigated compared to the unmitigated phase. The ELM affected area is reduced from $r/a = 0.85$ to $r/a > 0.95$ for the mitigated ELMs.

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EX/P3-5 · Progress Towards a Predictive Model for Pedestal Height in DIII-D

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Abstract: A new theoretical model predicts the observed pedestal pressure height for a range of conditions in DIII-D. This model uses the peeling-ballooning MHD model as a constraint on the total pedestal pressure gradient and the empirical scaling for the pedestal pressure width: p_{wid} proportional to $\sqrt{\beta_{pp}}$, where p_{wid} is measured in poloidal flux and β_{pp} is the total β-poloidal on top of the pedestal. This empirical scaling has been observed in scans of heating power, plasma shape, and plasma current in DIII-D and is obtained from pedestal measurements just before the onset of an edge localized mode (ELM). Pedestal studies in DIII-D have also provided tests of several theoretical pedestal models. Some of these tests are enabled by the observation that the electron pedestal pressure width and electron pedestal pressure height increase together during the pedestal buildup after an L-H transition and during pedestal recovery after an ELM. The pedestal density $n_{e,wid}$ and the pedestal density $n_{e,ped}$ also increase in time during these ELM-free periods. The simultaneous increase of $n_{e,wid}$ and $n_{e,ped}$ contradicts an analytic neutrals model, which predicts that $n_{e,wid}$ decreases with increasing $n_{e,ped}$. These results indicate that transport effects, perhaps an inward particle pinch, must be invoked to explain the density pedestal expansion. However, the increase of the pedestal barrier width in time is qualitatively predicted by time-dependent theoretical models, based on $E{\times}B$ shear suppression of turbulence in the pedestal, where the pressure gradient is the dominant term that balances the radial electric field. The $E{\times}B$ shear hypothesis has also been tested with initial comparisons of the TGLF transport model against experimental data. The results show that the experimental $E{\times}B$ shearing rate is of about the right magnitude to suppress the long wavelength modes at the edge. In summary, scaling studies provide good prospects that the pedestal pressure height can be predicted in future DIII-D experiments and detailed studies of pedestal evolution and structure provide new tests of pedestal models.

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EX/P3-6 · Global Plasma Oscillations in Electron Internal Transport Barriers in TCV

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Abstract: Global plasma oscillations have been observed in the Tokamak à Configuration Variable (TCV; R/a = 0.88 m/0.25 m, B_t < 1.54 T) in reversed magnetic shear discharges (fully non-inductive driven) with Electron Internal Transport Barriers (eITBs). These slow oscillations involve the electron temperature, density and the plasma current. The triggering mechanism is the periodic destabilization and stabilization of a magnetic island near the foot of the barrier. This regime is similar to the O-regime first observed on Tore Supra, but the link to the MHD and the global nature are specific of TCV [1,2]. The regime observed on TCV relies on the large pressure gradients in a region of low magnetic shear, typical in eITB plasmas. Depending on the proximity to the marginal infernal stability limit [3], the phenomenology of the mode can be more ideal (minor or major disruptions) or resistive. Regardless of the character, the result is detrimental to the confinement and leads to a lowering of the electron pressure and thereby a decrease of the bootstrap current. These fully-non-inductive discharges have large bootstrap fractions, therefore the total plasma current is affected. Due to the proximity to the marginal stability limit, the modes can be stabilized by the changes induced by the MHD in the pressure gradient and local q profile. This results in a recovery of good confinement properties with the re-attainment of the internal transport barrier, and the cycle can begin again. The oscillations can be controlled by modifying the current or the pressure profiles, either with Ohmic current density perturbation or by means of RF heating and localized current drive.

[1] V.S. Udintsev et al, Fusion Science and Technology 52, vol 2 (2007) [2] G. Turri et al, to be published in Plasma Phys. Control. Fusion, 2008 [3] O. Sauter et al, Phys. Rev. Lett. 94, 105002 (2005)

EX/P3-7 · Pedestal Studies at ASDEX Upgrade

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Abstract: Properties of the edge transport barrier (ETB) and its breakdown during ELMs are studied at ASDEX Upgrade in order to a) explore the nature of the L- to H-mode transition, b) evaluate if ion heat transport can be described within the neoclassical framework and c) to provide significant experimental data for ELMs in various scenarios. While the ratio of gradient lengths of density and temperature for electrons (η_e) was shown to be around 2 in the ETB region, η_i is found to be around 1 in ELMy H-modes with ELM frequencies smaller than 100 Hz. SOLPS modelling suggests that ion heat transport can be described neoclassically in the ETB region of such discharges. A comparison of transport coefficients derived from SOLPS modelling of experimental midplane profiles of L-mode and H-mode plasmas very close to the transition reveals that even before the transition to H-mode ion heat transport is close to neoclassical values. For these investigations a large effort was put into a series of improvements to existing pedestal diagnostics as well as the implementation of two new diagnostic systems for the measurement of edge T_i profiles from charge exchange spectroscopy and for the determination of radial electric fields from passive He II emission. The application of integrated data analysis in a probabilistic framework contributes substantially to the quality of E_r and n_e profiles. Especially the improved temporal resolution allows to study the development of the radial electric field in L- to H-mode transitions: while the width of the E_r dip does not change, a sudden increase of the well very close to the separatrix is seen which is in agreement with neoclassical expectations.

EX/P3-8 · Link of Internal Transport Barriers and the Low Rational q Surface on HL-2A

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Abstract: Reduced core transport or ITB has been observed with off-axis deposited ECRH around rational surfaces ($q = 1.5$ or 2 surface). This is inferred from the experimental observations that the central soft x-ray remains constant in the timescale of energy confinement time after ECRH switch-off while the off-axis intensity decrease concurrently. Besides the similarities with the experiments on other tokamaks, the results in HL-2A sometime show an obvious increase in the central soft x-ray intensity, indicating a much steeper temperature profile. It was found that such phenomenon is closely linked to the change in the central magnetic shear causing by the off–axis ECRH heating. A scan of ECRH power deposition along plasma minor radius also shows that there exist several electron temperature plateaus between low q rational surfaces, suggesting that electron heat transport is strongly correlated with a sequence of narrow transport barriers around rational q surfaces.

EX/P3-9 · Improved Confinement with Internal Electron Temperature Barriers in RFX-mod

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Abstract: In RFX-mod, improved axysimmetry of the magnetic boundary with rotating tearing modes has been obtained by correcting the aliasing of the sideband harmonics generated by the discrete actuators of the real time feedback control scheme(Clean Mode Control, CMC). The improved boundary avoids the highly localized power loads that at high current would otherwise represent an operational limit; this enabled to reach currents up to 1.5 MA, the highest ever reached in a RFP. During high current discharges spontaneous quasi-single helicity (QSH) regimes take place, during which magnetic dynamics is dominated by the innermost resonant mode: the level of magnetic chaos is reduced and an internal magnetic field configuration close to a pure helix is accessed. Inside the helical structure energy confinement is enhanced and electron temperatures of the order of 1 keV are measured. Transient strong QSH are reproducibly induced by oscillating poloidal current driven operation (OPCD), also at lower currents ($I_p > 500$ kA). During the OPCD the edge toroidal field is modulated in order to transiently induce a current profile modification in the plasma, concentrating the toroidal flux in the core. As an effect, secondary modes decrease while the resonant one increases providing enhanced confinement properties. The electron temperature shows steep gradients, which identify an internal transport barrier. The measured electron temperature peak results nearly symmetric and involves a large fraction of the plasma cross section, corresponding to an improvement of the global electron energy confinement up to 50%. Perturbative experiments (pellets and impurity laser blow off injections) have been performed to study particle confinement inside and outside the thermal island. The plasma energy and particle transport properties are compared with the topological properties of the magnetic field both in standard, or multiple helicity (MH), and QSH cases and compared with numerical simulation result. In conclusion, thanks to improved feed-back control and to advanced operation techniques RFX-mod displays clear and robust transitions to QSH regimes. The synergic interplay between the growth of the dominant mode and the reduction of the secondary modes provides enhanced confinement properties with steep gradients in the electron temperature profiles.

EX/P3-10 · Counter-NBI Assisted LH Transition in Low Density Plasmas in the TUMAN-3M

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Abstract: An increase in the LH transition power threshold towards low density has been observed in many experiments. In the TUMAN-3M the H-mode operational domain has a low density boundary of $(1.2 - 1.4) \times 10^{19}$ m^{-3}. No transitions have been observed at densities below the boundary in typical scenarios: ohmic and co-current NBI heating schemes. In the peculiar cases of electrode and pellet assisted H-modes, the transition have been found at densities down to $(0.6 - 1.0) \times 10^{19}$ m^{-3}. According to the theory developed in [1] in the electrode and pellet assisted H-modes an artificial radial electric field could help transition at lower densities. In the recent experiments on NB injection in the counter-current direction the H-mode transition at target density as low as 0.5×10^{19} m^{-3} has been observed. Low input power, P-inp $<$ 240 kW ($\sim$ 1.3 P-ohm), in these experiments should be noted. No transition is possible in the co-current NBI heating scheme at the above density with P-inp up to 500 kW. The paper presents results of the experiments and transport modelling allowing formulation of the hypothesis predicting generation of a negative (inward directed) radial electric field E_r which could help LH transition at low densities. Proposed model conjectures the development of the E_r and toroidal rotation in the presence of the return radial current balancing radial current caused by the large ion orbit losses in counter-NBI scheme. Direct measurements of the plasma potential by HIBP diagnostic confirming negative E_r appearance in counter-NBI scenario will be presented in the paper. Using Doppler spectroscopy an increase in the toroidal rotation V_{tor} of 30 km/s has been observed after application of the counter-NBI. The measured V_{tor} is in good agreement with the above model predictions. The above experimental observations allow concluding validity of the proposed model of the low density LH transition assisted by counter-NB injection. The mechanism could play a role in facilitation of the LH transition in larger devices in scenarios with essential losses of fast charged particles.

[1] L.G. Askinazi, et al, Phys.Fluids B, v.5 (1993), 2420

EX/P3-11 · Simultaneous Realization of High Density Edge Transport Barrier and Improved L-mode on CHS

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Abstract: An edge transport barrier (ETB) formation and an improved L-mode (IL mode) have been simultaneously realized in high density region ($\overline{n_e} \sim 1.2 \times 10^{20}$ m^{-3}) on the Compact Helical System (CHS). The plasma is produced by two co-NBIs (total power is 1.6 MW) injection to a target plasma that is produced by a 54.5 GHz gyrotron. The IL mode of the CHS is initiated by shutting off fueling after a strong gas-puffing. After the IL mode start, the plasma radiation continues to decrease resulting from the reduction of the peripheral neutral particle density. The electron temperature in the peripheral region is raised up resulting from the reduction of the neutral particles causing the charge exchange loss and the peripheral confinement is improved due to a radial electric field formation caused by an ion root (High T_i mode). Because the power threshold decreases due to the density reduction, the ETB is formed in the edge region ($\rho(r/a) \sim 0.9$) during the IL mode, and the density reduction in the edge region is suppressed by the barrier formation. As a result of the continuous increasing of the temperature by the IL mode, the stored energy during the combined mode increased up to the maximum stored energy ($W_p \sim 9.4$ kJ) of the CHS was recorded. The plasma pressure in the peripheral region ($\rho(r/a) > 0.6$) increases up to three times larger than that of the L-mode, and the large peripheral plasma pressure gradient is formed and the pedestal structure appears. The fluctuation measurement with the YAG phase contrast interferometer is carried out. When the ETB is formed during the IL mode, the fluctuation sharply drop to the same level of the ETB without the IL mode, though the density during the IL mode is larger than that of the first ETB phase. Therefore, the anomalous transport is considerably suppressed in the high density region by the combined mode. The H_α signal behavior shows that the neutral particle density of the ETB phase during the IL mode is lower than that of the first ETB phase. It seems that the confinement improvement by the combined mode in the high density relates to the neutral particle reduction in the peripheral region.

EX/P4-1 · Hydrocarbon Characteristics in Fusion Edge Plasmas from Electron-Molecule and Ion-Surface Collision Experiments

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Abstract: The compatibility of reactor grade plasmas with plasma facing materials at the first wall is one of the present challenges in fusion research. In order to understand and elucidate the role of radiative and collisional processes in the edge plasma region, it is essential to have available detailed and quantitative knowledge on the corresponding elementary reactions proceeding in the volume before and at the wall. Surface processes involving hydrocarbons and their ions, which are probable vacuum contaminants in plasma devices, and the role of hydrocarbon chemistry and transport in divertor plasmas are some of the key components in modelling predictions for ITER based on atomic and molecular data. Improved data sets for hydrocarbons and detailed and accurate knowledge of the cross sections of the relevant plasma chemical volume and wall processes are needed. In this context we have carried out: electron (proton and helium) impact excitation/ionization reactions with possible plasma edge atoms, molecules and ions; determination of relevant differential, partial and total cross-sections and reaction rate coefficients; investigation of the temperature dependence of ionization energies and ionization cross sections and electron attachment cross sections; and calculations of inelastic interactions between electrons and atoms, molecules and molecular ions. The reported work supports the provision of essential data commonly needed for diagnostics and modelling across the fusion programme.

EX/P4-2 · Operational Limits during High Power Long Pulses in Tore Supra

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Abstract: The Tore Supra tokamak has a unique long pulse capability in an actively cooled machine, with injected power provided exclusively by radiofrequency (RF) waves. Due to its good diagnostic coverage, in particular the infrared (IR) imaging system, Tore Supra is a highly relevant device for studying issues related to power handling of RF antennas and plasma-wall interaction in true steady state conditions. The last two years experimental campaigns have been particularly intense in terms of injected power, energy and accumulated plasma time (10 hours plasma time in both 2006 and 2007, with 65 GJ injected RF energy in 2006 and 40 GJ in 2007). Large experience has been gained in the understanding of localised heat loads due to RF sheath effects and interaction by fast particles. However, an increasing operational difficulty has emerged, limiting the high power and long pulse performance. The analysis undertaken suggest that this limitation could be linked to the growth and flaking of re-deposited carbon layers on the main plasma facing component. This subject is presented in this paper, together with issues related to RF coupling and antenna heat loads.

EX/P4-3 · Dynamic and Static Deuterium Inventory in ASDEX Upgrade with Tungsten First Wall

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Abstract: To investigate the retained D in a device different techniques are used. For the sample technique tiles are removed after a campaign and investigated using surface analysis techniques. For gas balances the amount of gas retained during a shot is measured as the difference between the puffed and pumped gas. This allows distinguishing different scenarios and wall conditions. From probe investigation, it is known that the majority of the retained deuterium is chemical bound as a:CH layers, which exist only for C first wall. AUG made a stepwise transition to full W PFCs, so carbon still remained in the vessel due to deposited layers from previous campaigns. For C PFCs the whole divertor shows carbon deposition. The majority is found at the inner divertor with 5 mg/s deposition rate. During the transition to the full W PFC deposition at the outer divertor vanishes. For full tungsten, almost no deposition is found at the divertor with the exception of the private flux region. The D inventory is mostly found in C and B dominated layers, formed by deposits remaining from earlier campaigns. For extrapolation to ITER we have to investigate the gas balance during a high density discharge in detail. Most of the puffed gas was retained during the plasma build up. For a carbon device retention of 70 mg D is needed to saturate the wall and reach steady state conditions. Due to the limited accuracy of dynamic gas balance a retention rate of $9 \pm 12\%$ of the puffed gas is observed during this phase. After the wall is saturated no additional D is retained. Comparing high density discharges with C and W wall material, first investigations show a higher dynamic retention for W materials. For highest puffing rates, no retention of D is observed after wall saturation. It seems that higher gas puff rate is needed in W to reach wall saturation, in comparison with C wall. D implantation was measured at the outer divertor strike point area. Only 50 mg was found for the outer divertor VPS coatings, whereas for polycrystalline Langmuir probes the inventory is less by a factor of 10.

EX/P4-4 · Issues Associated with Codepostion of Deuterium with ITER Materials

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Abstract: Codeposition of fuel species with the various plasma-facing materials present in ITER is expected to be the primary source of tritium accumulation within the vacuum vessel. Codeposition with the Be and C (if C is present) is expected to dominate tritium retention. While codeposition with C is fairly well understood, the scatter in the database of codepostion with Be is large and leads to uncertain predictions of the accumulation rate of tritium in Be codeposits. Understanding the dependence of the D/Be ratio in codeposits on the experimental parameters, such as the energy of impinging D atoms, the depostion rate of Be and the surface temperature on which the codeposit forms, leads to a reduction in the uncertainty of predictions for the expected retention in the Be codeposits. Recent data from the PISCES-B device verifies the expectation that the rate of codepostion of fuel with W is small. Regardless of the level of retention expected in ITER, ultimately the accumulation rate within the vacuum vessel will be determined to a large extent by the release behavior of fuel species from codeposited layers. The thermal release behavior of the fuel species is critical to determining the necessity of designing additional removal techniques and predicting how frequently such techniques will need to be employed in ITER. The thermal release behavior of deuterium from codeposits of each of the three materials is compared and the efficiency of removal due to the ITER bake temperature and duration is calculated.

EX/P4-5 · Detection of Dust Particles in FTU

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Abstract: Recent experiments were dedicated to dust detection during plasma discharges in FTU by utilising two Langmuir probes separated in the poloidal direction and located at the equatorial plane very close to the wall. The analysis of the fluctuations of the ion saturation current collected by the two probe tips allows to identify uncorrelated, large amplitude spikes which can be ascribed to dust impact ionization events. These are due to micrometer size dust grains impinging on the probes at velocity of the order of 10 km/s. Scanning electron microscope analysis revealed presence of impact craters of diameters in the range from several to 100 micrometers. The number of the craters found is consistent with the number of the ion saturation current spikes interpreted in terms of dust impact ionization events. A characterization of the dust component present in the FTU vacuum chamber after disruptions was also attempted by laser light scattering. Vaporization of the dust particles implies that only a rough estimate of the particle size (0.1 micrometers) and density (10 particles per cc) could be derived from the elastic scattering of the laser light. Preliminary results indicate that particle size and density evaluation can be improved considering the broad band emission.

EX/P4-6 · Status and Perspective of the Liquid Material Experiments in FTU and ISTTOK

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Abstract: The main goal of the experiment was to demonstrate the capability of the Capillary Porous System (CPS), firstly tested on T-11M tokamak, to confine liquid lithium avoiding sudden release, as consequence of $J \times B$ electromagnetic force on the surface. These latter should be balanced by the surface tension forces associated to the liquid lithium in capillary channels. Very encouraging results have been obtained on FTU with Li. A new plasma regime is obtained that is characterized by spontaneous peaking of the electron density profile with line average values of the density near or beyond the Greenwald limit. The higher density shots show an enhancement of energy confinement time over ITER97 L-mode scaling (H97) of up to a factor 1.2. A detailed transport analysis with the JETTO code on the database of lithized discharges will be shown and compared with the available data on transport relevant to plasma discharges with boronized and metallic walls conditions. Although the lithium liquid limiter is 2 cm away from the last closed magnetic surface and FTU is a fully metallic device with a TZM toroidal limiter, the only impurity detected by the survey spectrometer is lithium in all electron density range from $(0.15 - 3) \times 10^{20}$ m^{-3}. In addition, the application of additional power (LH and ECH) on strongly lithized plasmas at relatively low density $(0.5 - 0.8) \times 10^{20}$ m^{-3} has shown that the access to internal transport barrier (ITB) can be remarkably simplified. The reasons for this easier access could be related to the increased current drive efficiency in plasmas with lower Z_{eff} and/or to the different edge conditions that could reduce the local turbulence. From a technological point of view, the FTU liquid lithium limiter has withstand heat load up to 5 MW/m^2 without any surface damage and major plasma contamination. The results will be discussed in detail together with theoretical explanation and plasma edge modelling interpretation. Technical improvements of liquid lithium limiter are in progress: a new single element actively cooled and lithium refilled able to withstand heat loads up to 10 MW/m^2 for 4 s is being designed. Following the results of the plasma-liquid metal experiment in ISTTOK we are planning the installation of a liquid gallium jet system in FTU, in collaboration with the Portuguese and Latvian Associations.

EX/P4-7 · Characterization of Heated Dust Particles Using Infrared and Dynamic Images in LHD

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Abstract: Dynamic behavior of dust particles has been investigated in the Large Helical Device (LHD). A high-speed infrared camera viewing the inboard side from an outer port provides information about the movement of dust particles as well as their radiated thermal intensities. Imaging measurements in the infrared range have an advantage over visible light imaging because they can detect invisible dust particles which do not heavily interact with the main plasma. These heated dust particles are only observed during main discharges and directions of the moving dust particles are not uniform. A position of dust with respect to the plasma can be determined by using the reflected image of the heated dust particle on the first wall in this experiment. Using this method, a tracing velocity with three-dimensional position is determined. In this paper, moving dust phenomena, direction patterns, velocities and infrared intensities are reported to contribute a theoretical model of accelerating dust mechanisms.

EX/P4-8 · Hydrogen Concentration of Co-deposited Carbon Films Produced in the Vicinity of Local Island Divertor in Large Helical Device

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Abstract: In ITER, one of the major concerns is tritium retention in carbon film/flake from the point of view of safety. The tritium retention can be predicted by the data on hydrogen or deuterium retention.The deuterium concentration of carbon film/flake produced in JET showed a relatively high deuterium concentration, with a D/C ratio of 0.4. The co-deposited carbon films produced at the wall facing to plasma in deuterium arc discharge showed a deuterium concentration, D/C = 0.2 for the ITER wall condition. However, it is pointed out that the hydrogen or deuterium retention in carbon films produced at shadow regions (walls not facing plasma) in fusion devices may become higher than the existing value, D or H/C = 0.4. In the present study, the hydrogen concentration of the co-deposited carbon film produced in LHD was investigated. In the Local Island Divertor configuration, a divertor head made of carbon fiber composite receives high heat and particle fluxes, and the eroded carbon co-deposits with hydrogen on the wall of duct. Four sets of material probes made of Si were installed in the vicinity of the LID head. One set of probes was placed inside the pumping duct and faced to the head. The other set of probes has a shallow line of sight to the head. After the campaign, the probes were extracted and their surface morphology, depth profiles of atomic composition, crystal structure and hydrogen retention were investigated. The hydrogen concentration of the co-deposited carbon film produced at the shadow wall with a relatively low temperature was remarkably high, $2-3$ times higher than the existing value. Such the high concentration with the atomic ratio of $0.55-1.25$ has not been observed so far. The high hydrogen concentration is owing to the deposition of hydrocarbons. The crystal structure analyzed by Raman spectroscopy was very unique, a polymer structure, which is consistent with the high hydrogen concentration. Similar co-deposited carbon films with a high hydrogen concentration can be produced in tokamak devices with a carbon plasma facing material, so that the present result is important to evaluate an in-vessel tritium inventory in ITER and demonstration reactors.

EX/P4-9 · Plasma Performance Improvement with Lithium-Coated Plasma-Facing Components in NSTX

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Abstract: Lithium as a plasma-facing material has many attractive features, including a reduction in the recycling of hydrogenic species and the potential for withstanding high heat and neutron fluxes in fusion reactors. Recent NSTX experiments have shown, for the first time, significant and recurring benefits of lithium coatings on plasma-facing components (PFC's) to the performance of divertor plasmas in both L- and H-mode confinement regimes heated by high-power neutral beams. They included decreases in the plasma density and inductive flux consumption, and increases in the electron temperature, ion temperature, energy confinement time, and DD neutron rate. Extended periods of MHD quiescence were also achieved, and measurements of the visible emission from the lower divertor showed a reduction in the deuterium, carbon, and oxygen line emission. Other salient results with lithium evaporation included a broadening of the electron temperature profile, and changes in edge density gradients that benefited electron Bernstein wave coupling. There was also a reduction in ELM frequency and amplitude, followed by a period of complete ELM suppression. In general, it was observed that both the best and the average confinement occurred after lithium deposition and that the increase in WMHD occurs mostly through an increase in We. In addition, a liquid lithium divertor (LLD) is being installed on NSTX this year. As the first fully-toroidal liquid metal divertor target, experiments with the LLD can inform on the behavior of metallic ITER PFC's should they liquefy during high-power divertor tokamak operations. The NSTX lithium coating and LLD experiments are important near-term steps in demonstrating the potential of liquid lithium as a solution to the first-wall problem for both magnetic and inertial fusion reactors.

EX/P4-10 · Plasma-Wall Interaction Study in the Open Divertor of Globus-M Spherical Tokamak

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Abstract: Globus-M is the first Russian spherical tokamak, which was built at the A.F. Ioffe Institute in 1999. Currently about 90% of in vessel area, which is faced to plasma, is covered by protection tiles. Tiles are constructed from RGTi graphite, doped by 2 at.% Ti and 0.3-0.7 at.% Si. Gradual increase in plasma parameters was recorded as the wall area protected by tiles constantly grew up. The plasma-facing surface including graphite armor was covered by a-B/C:H layers periodically. Globus-M is usually operating in the density range $n \sim (3 - 10) \times 10^{19}$ m^{-3} with plasma currents $0.2 - 0.25$ MA and toroidal magnetic field of 0.4 T. The main working gas is deuterium. NB injection and IC resonance heating at hydrogen minority fundamental frequency are used as auxiliary heating methods. Specific power deposition is high up to value of several MW/m. Power density deposited at the first wall is also high. This is due to "spherical" geometry of plasma column in which the first wall area is small with respect to plasma volume and plasma is close to the first wall position. The "focusing" of power fluxes along separatrix strike points could increase power density up to 10 MW/m^2 at the divertor target. RGTi diverter tiles analysis was performed after irradiation by plasma during big number of shots (10000 shots in average). Composition and morphology of the surface layers were examined by different diagnostic tools (electron probe microanalysis, scanning electron microscope, Rutherford backscattering, nuclear resonance reactions, thermal desorption spectroscopy). The most of tiles were covered with deposited mixed layers. Deposits exist even in high flux regions (separatrix strike points). The mixed layers are composed from the elements used in the vessel construction and in conditioning technology processes. The concentration and absorbed deuterium depth profiles in tiles are analyzed and conditions of deuterium desorption are studied. Important result is that deuterium is absorbed only inside the mixed layers and its concentration is vanished in the bulk of the tiles. Conditions of deuterium desorption are analyzed and merits of T_i doped graphite are discussed.

EX/P4-11 · Overview of Recent ISTTOK Results

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Abstract: ISTTOK is a large aspect ratio circular cross-section tokamak ($R = 46$ cm, $a = 8.5$ cm, $B_T = 0.5$ T, $I_p = 4 - 7$ kA), with a poloidal graphite limITER. ISTTOK has been very important for the creation and consolidation of the Portuguese fusion research team, being its main objectives the formation of students in fusion plasma physics and technologies and the development of new diagnostic techniques and instrumentation systems. In this contribution the work developed recently on ISTTOK will be reviewed with emphasis on the following topics: (i) Study of fusion relevant materials as liquid metals and nano-structured materials; (ii) Control and data acquisition that consisted mainly in the upgrade of the data acquisition system and the real-time plasma position control; (iii) Development and upgrade of diagnostics; (iv) Plasma physics studies that has been based mainly in the characterization of the edge transport using different types of probes and the study of the MHD activity; and (v) The achievement of regular AC discharges with 250 ms, extending the plasma duration for almost one order of magnitude. Joint Experiments in the framework of the IAEA Coordinated Research Project (CRP) on "Joint Research Using Small Tokamaks" have also been carried out on ISTTOK in October 2007 with the participation of 24 scientists from 13 countries. The ISTTOK achievements demonstrate that small tokamaks can play an important role in the fusion plasma physics community as a result of their flexibility, high availability and good opportunity for the development of sophisticated diagnostics and technology tools.

EX/P4-12 · Active Particle Control Experiments and Critical Flux Discriminating between the Wall Retention and Release in the CPD Spherical Tokamak

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Abstract: Active recycling control experiments using a rotating tungsten limiter coated with lithium are performed in a small ST device CPD (Compact Plasma wall interaction experimental Device). A clear reduction in hydrogen recycling by factor of ~ 3 is found both near the limiter and centre stack region. In order to understand "retention and release" mechanisms of the wall pressure difference before and after the RF plasma is investigated without external pumping. It is found that a critical particle flux exists to discriminate between them. A large fraction $\sim 80\%$) of particles is found to be retained in the wall beyond it. The role of low energy neutrals is considered. Both low recycling and wall retention are necessary to increase RF driven current.

EX/P4-13 · Study of Dust Morphology, Composition and Surface Growth under ITER-relevant Energy Load in Plasma Gun QSPA-facility

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Abstract: The experimental date on structure, chemical and X-ray analysis of re-deposited material from eroded CFC, W and Al/Mg material under ITER ELMs and disruptions heat loads are presented. Experiments were realized in QSPA plasma facility. CFC and W macrobrush targets were designed and manufactured in EU. As analogue of Be-like material the Al/Mg plate were used. Each macrobrush targets were exposed to a large number of repetitive pulse (from 10 up to 1000) of QSPA plasma gun with heat loads $0.5-2.0$ MJ/m^2 with 0.5 msec time duration. The experiments was directed to collect all eroded material, to have the dates on angular distribution of evaporated material and dust particles spread and to determine correlation of the deposit characteristics with the dominant mechanism for target erosion - evaporation, melt layer losses with splashed droplets, brittle destruction and flakes crumble. The principal result of impact type I type ELMs-like plasma on material was in formation the films, consisted of the dust particles with spherical shapes, typical dimensions of $0.02-2.0$ μm and fractal surface structures ("cauliflower"). The particles with solid state structure have bigger dimensions (up to 20 μm). 1000 shot tungsten macrobrush erosion experiment was carry out for testing the dust particles size growth. The results of dust particles growth on heated collectors are comparing for various temperature. The particles size distributions are presented. In dust particles the Auger spectrometer/SIMS analysis found out many of elements from which the vacuum chamber consist. X-ray crystal analysis show that tungsten re-eroded in the form of tungsten-carbide (nearby 70%). This effect confirms by the small surface electroconductivity. The analysis of small angle scattering gives "halo" picture, which indicate of $10-30$ nm clusters. Sorption surfaces (SSA) for tungsten dust particles are presented. The results of cauliflower particles growth simulations shows two mechanism of clusters agglomeration - particles compression on collector - CCA model, which depends from surface tensions, and growth of particles in vapour as whole, in accordance with DLA model.

EX/P4-14 · Experiments with Lithium Gettering of the T-10 Tokamak.

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Abstract: The first experiments with gettering of the T-10 chamber with lithium were carried out. The experiments were aimed to prove the lithium abilities to reduce the recycling level of the working gas recycling and to reduce impurities level in T-10 configuration with graphite limiters. The main goals of the T-10 experiments were to extend the operational space of discharge parameters in conditions of the clean plasma with low recycling and to test compatibility of the Li-gettering with the proper work of the primary vacuum windows of the T-10 ECRH system. The lithium element was manufactured by NPO "Krasnaia Zvezda" on the base of the kapillar porous system, which have been already successfully used in experiments in T-11M and FTU. The lithium system was placed in the same port as the graphite rail and circular limiters. The element was introduced to the port by means of the special transport vacuum system. The element had the inner heater, which can heat it up to the 5500C. The gettering were made by evaporation of the lithium during 0.5 – 2 hours. Such technique was sufficient to reduce density decay time from about 1.5 s to 0.5 s. The significant decrease of the recycling was sustained during 10 discharges. The decrease of the SXR emission in a factor of 3, the central radiative losses in 1.7 and the total radiative losses 1.2 were observed. This corresponds to the decrease of the high Z in a factor of 1.9 and oxygen in 7, while the main carbon impurity decrease up to 30% and Z_{eff} from 1.4 to 1.25. Thus the experiment showed significant decrease of the impurities and recycling. At the same time the influence of the lithium gettering on the window transmission and stable work of the full gyrotron complex were not observed.

EX/P4-15 · Overview of Experimental Studies on IR-T1 Tokamak

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Abstract: An overview of experimental studies on IR-T1 tokamak is presented. Several issues of plasma displacement measurement are investigated. An analytic solution of the Biot–Savart law, which is used to calculate magnetic fields created by toroidal plasma current, is presented. Results of calculations are compared with the experimental data obtained in no-plasma shots with a toroidal current-carrying coil positioned inside the vessel to simulate the plasma movements. The results show a good linear behavior of plasma position measurements. An array of Mirnov coils employed for measurement of plasma position, too. The results show that Mirnov array can not be used for this measurements without concerning high field side errors. The helical perturbation of the shape of the plasma column due to MHD modes has been studied and typical results are presented. The mode number and frequency analysis has shown that the plasma column rotates as non-rigid body. A relative correction for island rotation frequency has been suggested. A resonant external magnetic helical field (RHF) applied to IR-T1 tokamak plasma. The aim of these experiments was to understand the effect of RHF on light impurities radiation and suppressing major disruptions. Measurements results of visible line emissions of OII, CIII impurities and Halpha radiation with and without RHF ($L = 2$) show that the addition of a relatively small amount of resonant magnetic helical field ($L = 2$ & $L = 3$) could be effective for improving the quality of the discharge by reducing of light impurities radiation and suppressing major disruptions. Also, the results show that by using RHF, Z_{eff} of the plasma could be decreased. In other experiments, Studies of plasma interaction with titanium coated Ferritic Steel have been performed on IR -T1 tokamak. Depth of impurity penetration and retention, and the surface roughness were measured using surface analysis methods. The results show that titanium acts as an effective getter for oxygen but its fast erosion poses a problem. A change in roughness with respect to position of samples has been observed.

EX/P4-16 · Localized Tungsten Deposition in Divertor Region in JT-60U

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Abstract: Deposition profiles of tungsten released from the outer divertor were studied in JT-60U. A neutron activation method together with XPS and EDX was used for the first time to accurately measure deposited tungsten on the graphite tiles in tokamak devices. Surface density of tungsten in the thick carbon deposition layer can be measured by this method. In the experimental campaigns 2003-2004 (1^{st}) and 2005-2006 (2^{nd}) in JT-60U, 12 tungsten-coated CFC tiles (10 tiles for the 2^{nd} campaign) were installed in the outer divertor (section P-8). This tile array covered about 1/21 toroidal length. In most discharges, W-tiles were faced with the outer SOL plasma. Although some of discharges (2-3% of total number) had outer strike points on the W-tiles, most of tungsten erosion took place without strike points on the W-tiles. It was observed that eroded tungsten was mainly deposited on the inner divertor (around inner strike points) and on the outer wing of the dome. Tungsten deposition on the dome tiles was found only near the top surface (within the depth of a few μm), while tungsten on the inner divertor tiles was codeposited with carbon to the depth up to about 60 μm. Toroidal distribution of the W deposition was significantly localized near the tungsten released position both on the outer wing and on the inner divertor. Toroidal localization of tungsten deposition was more pronounced on the outer wing than the inner divertor. These data suggested that inward drift in the divertor region played a significant role in tungsten transport in JT-60U. In addition, asymmetry of tungsten toroidal distribution on the outer wing suggested the effect of plasma flow along magnetic fields on the tungsten transport.

EX/P4-17 · The Influence of Filaments on Scrape-Off Layer Transport

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Abstract: The tokamak scrape-off-layer (SOL) is a primary component of the interface between the core plasma and the material surfaces of the vacuum vessel. As such, an accurate model of SOL processes is required for reliable prediction of the boundary plasma environment in next-step devices such as ITER, in particular the distributions of particle and heat fluxes exhausted to the plasma facing components, and the transport back to the confined plasma of the associated neutral and impurity influxes. However, observations on several devices indicate that plasma intermittency in the SOL is significantly more prevalent than previously thought, prompting a re-evaluation of the standard steady-state approximation for the boundary plasma and raising the question of whether a new model paradigm must be developed based on transient plasma states, one that explicitly includes filamentary transport. Distribution functions for L-mode, ELM and inter-ELM filaments on the Mega-Ampere Spherical Tokamak (MAST) have been measured using fast cameras, reciprocating probe data and time-resolved Thomson scattering measurements. A 4D (x,y,z,t, guiding centre only for kinetic components) dedicated interpretive code for the boundary plasma has been developed, based on extensions to the OSM-EIRENE package. Initial simulation results for filament evolution are presented, with filament properties prescribed from the measured data. The model captures the essential features of intermittent transport and can therefore be used to quantitatively assess the validity of the steady-state SOL approximation.

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EX/P4-18 · Statistical Analysis of Fluctuation Characteristics at High- and Low-Field Sides in L-mode SOL Plasmas of JT-60U

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Abstract: Intermittent convective plasma transport, so-called "plasma blobs" has been observed in LFS SOL of several tokamak devices, which is thought to play a key role for cross-field transport. On the other hand, there has been little investigation on electrostatic fluctuation in HFS SOL. Then, it becomes one of the most important issues to reveal an influence of plasma blobs on cross-field transport in the HFS SOL. Furthermore, the detailed analysis of the fluctuation property is required to understand the generation mechanism of plasma blobs and to predict the cross-field transport, including non-locality of transport in the edge plasma turbulence. Detailed comparison between fluctuation characteristics at high-field-side (HFS) and low-field-side (LFS) scrape off layers (SOLs) has been made, for the first time, in the L mode plasma of JT-60U tokamak using reciprocating Langumuir probes. Statistical analysis based on probability distribution function (PDF) was employed to describe intermittent (non-diffusion) transport in SOL plasma fluctuations. It was found that the positive bursty events appeared most frequently at LFS midplane associated with blobby plasma transport, then the PDF is strongly skewed positively, while the PDF in HFS SOL is close to Gaussian distribution. Conditional averaging analysis of the positive bursty events at LFS midplane indicates the intermittent feature with a rapid increase and a slow decay is similar to that of plasma blobs theoretically predicted. Statistical self-similarity was also investigated with Fourier power spectrum, and statistics of waiting-time and duration-time of the fluctuation. It was found, for the first time, that clear statistical self-similarity was observed both at HFS and LFS SOLs, showing fractal property of the fluctuation. The scaling exponent of duration time statistics at HFS-SOL is larger than that at LFS-SOL, and both scaling exponents disagree with the predictions for the self-organized criticality (SOC) paradigm.

EX/P4-19 · Effect of Cross-field Drifts and Core Rotation on Flows in the Main Scrape-Off Layer of DIII-D L-mode Plasmas

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Abstract: Deuteron flows measured in the scrape-off layer (SOL) measured in the region vertically opposite the divertor of low-confinement (L-mode) plasmas in DIII-D show a marked dependence on the direction of the ion ∇B direction with little sensitivity on core plasma rotation. In contrast, the parallel flow of low charge state carbon ions in the SOL was measured to be independent of both parameters, and thus appear to be de-coupled from the deuteron flow. Improved understanding of the SOL flows is required to reliably predict impurity transport in the SOL and pedestal regions, and material migration and fuel retention in future fusion devices, such as ITER. In single-null magnetic configurations, as the ion ∇B drift direction is changed, the deuteron flow parallel to the magnetic field $\mathbf{B}$ toward the inner plate reduces in magnitude from $M = 0.5$ to $M \sim 0$, while the magnitude of parallel carbon flow toward in the inner plate remains approximately the same. The poloidal flow of singly ionized carbon, on the other hand, increases roughly three-fold as the ion ∇B drift direction is reversed. These measurements suggests that the dynamics governing the deuteron flow in the SOL is a complex interaction of radial transport, and $E{\times}B$ and $B \times \nabla B$ drifts in the SOL, and that the carbon SOL flow is not simply determined by entrainment in the deuteron flow. In L-mode plasmas with the ion ∇B direction away from the divertor x-point, reversing the core plasma rotation does not affect the flow of deuterons and low-charge state carbon ions in the SOL. Scrape-off layer transport simulations with the fluid edge code UEDGE for the L-mode plasma in the ion ∇B toward the divertor do not predict the measured deuteron flow at the crown. Low charge state carbon ions at the crown are predicted to flow in the direction of the outer divertor, opposite to the observation, while higher charge state carbon ions, carrying the bulk of the total carbon flow, are calculated to flow toward the high-field divertor.

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EX/P4-20 · SOL Transport in TCV

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Abstract: Radial profiles of parallel SOL ion flows at several poloidal locations in the outer midplane vicinity have been measured in SNL and SNU equilibria. Matched ohmic density scans for both toroidal field directions yield co-current directed flows which decrease as density increases and reverse direction with field inversion. Excellent agreement is found between experiment and a simple analytic expression for the neoclassical Pfirsch-Schlüter ion flow. At any given density, the mean value of the flows obtained above or below the plasma midplane for both field directions reveals a field independent component. This offset increases with density, is directed away from the midplane and is absent on the midplane, indicating a ballooning type origin. It can be clearly identified with the time average of transient over-pressure generated in field aligned filaments driven by interchange motions. These "blobs" increase in number with density and are observed more frequently with decreasing plasma current. This is consistent with an increase in fluctuations and turbulence driven radial transport with increasing collisionality and is evidently related to the blob formation process. During ELM activity filaments are observed in IR images of the outer divertor target, consistent with energy release in the outer midplane region from a number of discrete toroidal locations. They are also seen by Langmuir probes in the SOL and at the outboard midplane walls. For Type III ELMs, radial velocities are in the range ~ 1 km/s. On average, between $5-8$ substructures are detected at the wall in each ELM (Type III and Type I ELMs) and the fluctuating parallel particle flux statistics during the filaments appear very similar to those of the inter-ELM phases. Tomographic reconstructions of the radiation distribution during ELMs on the ~ 5 micros timescale clearly show that a disturbance occurs in the X-point region before any particle flux reaches first wall surfaces. Most of the ELM energy reaching the targets does so in the separatrix vicinity, with the integral energy deposited up to the peak energy deposition normalised to total deposited energy at the outer target in the range $0.2-0.3$ for Type III ELMs and $0.3-0.4$ for Type I ELMs. modelling of the ELM parallel transport is progressing favourably, both using the 2D edge code package SOLPS5 and the 1D particle-in-cell kinetic code BIT1.

EX/P4-21 · The Effect of Magnetic Balance and Particle Drifts on Radiating Divertor Behavior in DIII-D

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Abstract: The success of the "puff-and-pump" radiating divertor approach to heat flux reduction at the divertor targets while maintaining high performance H-mode tokamak plasmas depends sensitively on the divertor magnetic geometry and the ion ∇B drift. In the puff-and-pump scenario [1] used here, Argon (Ar) was injected near the outer divertor target, while plasma flows into the inner and outer divertors were enhanced by a combination of particle pumping near divertor targets and D_2 gas puffing upstream of the divertor targets. The key to the success of this approach is in preventing the injected Ar from escaping the divertor and contaminating the main plasma. Under similar conditions, Ar accumulated at a $2-3$ times faster rate in the main plasma of magnetically balanced double-null (DN) configurations than in magnetically unbalanced DN shapes (where $\mid dRsep \mid$ was typically 1.2 cm). For balanced DNs, stronger increases in Ar concentration were observed in the divertor opposite to the ion ∇B drift direction, regardless of the divertor into which the Ar was initially injected. For unbalanced DNs, there was significantly higher divertor enrichment and lower Ar leakage out of the divertor when the ion ∇B drift was directed away from the dominant X-point. That configuration provides the best prospect for coupling a radiating divertor approach to a high performance H-mode plasma. Exhaust-to-core argon concentration ratios ≈ 35 were achieved under this scenario, along with good H-mode conditions: $\tau_E/\tau_{89P} = 2$, $\beta_N = 2$, $n_e/n_{eG} = 0.6$, $P_{RAD}/P_{INJ} \approx 0.6$, and $Z_{eff} \approx 2$ [2]. Related experiments using ELMing H-mode discharges compared the plasma behavior in the open lower divertor to that of the more closed upper divertor and demonstrated that control of D_2 and Ar inventory was more sensitive to the ion ∇B drift direction than to the relative closure of the divertor. The roles of particle drifts in the divertor(s) and scrape-off layer are discussed in the context of UEDGE [3] analysis.

[1] M.R. Wade, et al., Nucl. Fusion **38** (1998) 1839. [2] T.W. Petrie, et al., J. Nucl. Mater. **363-365** (2007) 416. [3] T.D. Rognlien, et al., Plasma Phys. **34** (1994) 362.

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EX/P4-22 · Divertor Heat Flux Mitigation in High-Performance H-mode Plasmas in the National Spherical Torus Experiment

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Abstract: Experiments conducted in high-performance $4-6$ MW NBI-heated H-mode plasmas with a high flux expansion radiative divertor in NSTX demonstrate that significant divertor peak heat flux reduction and access to detachment may be facilitated naturally in a highly-shaped spherical torus (ST) configuration. Radiative (detached) divertors and optimized divertor geometry are considered candidate techniques for steady-state mitigation of divertor heat flux and erosion of divertor material in ITER and the ST-based Component Test Facility concept and the proposed National High-power Advanced-Torus eXperiment. Improved plasma performance approaching the performance level of CTF with high β $15-25\%$, a high bootstrap current fraction $45-50\%$, longer plasma pulses, and an H-mode regime with smaller ELMs has been achieved in lower single null NSTX plasmas with higher end elongation $2.1-2.4$ and triangularity $0.5-0.7$. Because of the high poloidal magnetic flux expansion factor $(12-23)$ and higher SOL area expansion, the divertor peak heat flux is $20-40\%$ lower than in similar plasmas with lower-end shaping parameters. Access to detachment was demonstrated using additional deuterium injection and divertor radiation from intrinsic carbon. A partially detached divertor (PDD) phase was induced at several SOL power levels while good core confinement and pedestal characteristics were maintained. Measured properties of the PDD regime indicated similar trends with large aspect ratio tokamak experiments: a $30-60\%$ increase in divertor plasma radiation, a peak heat flux reduction up to 60%, measured in a ~ 0.1 m radial zone, a $30-80\%$ increase in neutral compression, and a significant volume recombination rate increase. A five-region SOL heat conduction model with constant heat and particle sources and sinks predicts that large radiated power and momentum loss fractions are required to achieve detachment in the NSTX range of parallel SOL heat flux $25-60$ MW/m^2 and connection lengths $6-10$ m. Because of a large SOL magnetic shear, the parallel connection length rapidly decreases in the outer SOL thereby radially limiting the detachment zone, whereas the plasma volume available for the radiated power and ion momentum loss is maximized in the region of longer connection length.

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EX/P4-23 · Experimental Investigation of Turbulence at the Transition from Closed to Open Field Lines

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Abstract: Investigations of turbulent transport at the transition from closed to open field lines are summarised. The data are from the toroidal low-temperature plasma of the torsatron TJ-K and the high-temperature plasma of ASDEX Upgrade. A close comparison of TJ-K data with results from the gyro-fluid code GEM shows a great structural similarity in all aspects analysed. This indicates the dominance of drift-wave turbulence. Data from both devices indicate a flow-shear layer at the separatrix incorporating pinch phenomena and the origin of SOL transport intermittency. The role of Reynolds stress in the generation of shear flows is studied on TJ-K with a novel poloidal array equipped with 128 Langmuir probes.

EX/P4-24 · Advanced Divertor Scenario in the Large Helical Device

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Abstract: The modification of the open helical divertor (HD) to the closed HD in the Large Helical Device (LHD) heliotron has been investigated using fully three-dimensional neutral transport code EIRENE, to accomplish the active particle control to improve plasma confinement and to sustain high performance long pulse discharges. For the first step to the full closed HD, the torus-inboard side divertor is planned to be modified first in the LHD. The exploring of the structure has been conducted using EIRENE code by trial and error. The code was examined in advance by comparing to spectroscopic observations, and the results of calculation agreed well with the observations. Results of the exploring show that proper rearrangement of divertor plates and additional components, such as dome structure make the neutral particles to be compressed in the divertor region, and effective divertor pumping to be possible. Based on the simulation and experimental results, design and installation of closed HD is programmed in LHD.

EX/P4-25 · Impurity Accumulation in the Main Plasma and Radiation Processes in the Divetor Plasma of JT-60U

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Abstract: Two studies are included: tungsten accumulation in the main plasma and radiation processes of carbon ions in the detached plasma of JT-60U. It has been found that tungsten accumulation becomes more significant with increasing plasma rotation velocity against the plasma current direction. The accumulation level does not depend on the tungsten generation flux, suggesting the transport plays a dominant role to determine the accumulation level. In the detached divertor plasma, C^{3+}, which contributes 60% of the total radiation, comes from the main plasma via volume recombination of C^{4+}, and from the divertor via ionization of C^{2+}, comparably. Hence CD_4 seeding would increase the C^{3+} ion through the ionization of C^{2+}, leading to an increase of the radiation power. However the increase of the radiation may be limited because the C^{3+} source from the main plasma is difficult to be controlled.

EX/P4-26 · On Impurity Handling in High Performance Stellarator/Heliotron Plasmas

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Abstract: In stellarators and Heliotrons the plasma confining magnetic fields are exclusively produced by external coil systems without induced net currents. Therefore, they offer the intrinsic potential for continuous plasma operation. The large helical device (LHD) and Wendelstein 7-X (W7-X), the latter presently being under construction, already comply with the technical requirements of long pulse operation and focus upon the establishment of high performance long-pulse plasmas. A crucial point is the avoidance of impurity accumulation which can lead to early pulse termination by radiation collapse. Predictions by theory for non-axisymmetric configurations underline the importance of this issue because screening of impurities by ion temperature gradients is not expected in standard ion root plasmas. Trends for increasing impurity confinement with rising densities were indeed found in TJ-II and W7-AS leading to accumulation phenomena at higher densities in CHS, W7-AS and LHD. In W7-AS, the rise of impurity concentration could be stopped by the onset of drag forces in the high density and low temperature scrape-off-layer which flush out impurities and reduce the net impurity influx into the core. Additionally, the impurity core confinement is reduced (HDH-mode). High performance discharges with steep density edge gradients and low edge density do not benefit from this mechanism. Therefore, the utilization of ELMs, which seem to have a similar effect on impurity screening as known from tokamaks, might be a possible way for plasma purification. Degradation of impurity confinement and avoidance of accumulation by increased ECRH heating power was demonstrated at W7-AS as an additional tool against impurity accumulation, but it is not clear whether this effect can be extrapolated to other heating methods. Nevertheless, the exploration of such purification mechanisms is a demanding issue for steady-state operation.

EX/P4-27 · Development in DIII-D of High β Discharges Appropriate for Steady-State Tokamak Operation with Burning Plasmas

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Abstract: A research focus at DIII-D is development of methods for reliable production of high β discharges suitable for high fusion gain power plant operation with 100% of the plasma current generated noninductively ($f_{NI} = 1$). Included in this work is validation of the physics basis necessary to confidently design next step tokamaks for steady-state operation. Recent work has encompassed a complete discharge scenario, from feedback control of the q profile evolution during the initial, low beta, discharge formation phase to the study of the optimum discharge shape and electron cyclotron current drive (ECCD) deposition profile for stationary operation of the high beta, $f_{NI} = 1$ phase of the discharge. Stationary discharges with $\beta_N = 3.4$ that project to $Q = 5$ in ITER have been demonstrated for a resistive time. Normalized confinement (H_{89}) was found to increase approximately 20% when the position of the X point was changed with respect to the ∇B drift direction, and the maximum β_N stable to tearing modes was increased when ECCD was deposited with a broad profile. Transport code models of the current profile evolution during discharge formation have been validated against the experiment. modelling of the high-performance phase is being used for analysis of the current contributions from neutral beam current drive, ECCD, bootstrap current and ohmic current drive and to design experiments with additional heating and current drive power. In addition, experiments to increase β_N to 4-5 with profiles suitable for steady-state operation have begun, motivated by the requirement for high power density and neutron fluence in a demonstration power plant. Two approaches capable of reaching $\beta_N = 5$ have been identified through modelling and tested experimentally. In a wall stabilized scenario with $q_{min} > 2$, $\beta_N = 4$ is produced and in a high internal inductance scenario, which maximizes the ideal no-wall stability limit, $\beta_N > 4.5$ has been reached.

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EX/P5-1 · Observations of Spontaneous Toroidal Flow on LHD

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Abstract: Spontaneous toroidal flows driven by the radial electric field and ion temperature gradient are studied in the Large Helical Device (LHD). Radial electric field can be controlled by changing collisionality as predicted by neoclassical transport, independently of the NBI direction, in helical system. The radial electric field at the plasma edge is changed its sign from negative to positive by reducing the electron density in the NBI sustained plasma. It is observed that the magnitude of the toroidal flow velocity in the counter-direction increases when the radial electric field changes from negative to positive at the plasma edge. The dependence of the toroidal flow velocity on the radial electric field at plasma edge, where the helical ripple is strong, shows that positive radial-electric-field drives toroidal flow into counter-direction at the plasma edge in the helical system. In contrast, the positive radial-electric-field drives toroidal flow in the co-direction is observed at the plasma core where the modulation of the magnetic field due to the helical ripple is comparable to that due to toroidal effect in the LHD. The difference in the direction of the spontaneous flow between core and edge is considered to be due to the difference in the ratio of the helical ripple to the toroidal ripple. It has been observed that profiles of both the ion temperature and the toroidal flow velocity are changed when the heating power of the plasma is changed. The line averaged electron density is kept constant during the discharge and there is no significant change of radial electric field. The profiles shows that the toroidal flow in the co-direction is increased associated with the increasing of ion temperature gradient. The toroidal flow driven by the ion temperature gradient is clearly observed in the large helical device.

EX/P5-2 · Intrinsic Rotation in H-Mode Pedestal in DIII-D

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Abstract: The boundary condition for the intrinsic rotation measured in DIII-D H-mode plasmas is consistent with a simple model based on thermal ion orbit loss. This toroidal rotation in DIII-D is measured to be nonzero, in the direction of the plasma current, co-I_p, near and at the last closed flux surface (LCFS) on the outboard midplane in H-mode discharges without toroidal acceleration due to neutral beam injection (NBI). The boundary condition for intrinsic rotation is presently an unknown, both experimentally and theoretically. Without knowing this boundary value, no theory of intrinsic rotation will be able to predict an absolute intrinsic rotation profile for a burning plasma, in particular ITER. These DIII-D measurements are focused upon establishing and understanding this boundary condition. We find that the pedestal region boundary intrinsic velocity is roughly proportional to the ion temperature measured at the same location. This is the case for velocity measurements of both the minority impurity constituent C^{6+} in bulk D^+ discharges, and also the bulk ion in He^{++} H-mode discharges. A model for this edge velocity based upon thermal ion orbit loss from the edge reproduces this $V_\phi \sim T_i$ result, where V_ϕ is the toroidal velocity and T_i the ion temperature. The counter-I_p directed thermal ions can be lost from the pedestal region leaving a hole in velocity space with net co-I_p momentum. In this model we consider only the mechanical momentum of the lost ions, assuming a steady state has been reached and thus a steady state return current established to balance the lost thermal ion electric current. The radial electric field in the pedestal region is measured and found to be only a first order correction to the lost orbit boundary in phase space at the measured T_i values. This effect is only operative near the LCFS for thermal ions. Some other mechanism, such as a momentum pinch due to turbulent or higher order neoclassical effects is still required to establish a rising toroidal momentum profile going inward from the pedestal region boundary value.

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EX/P5-3 · Momentum Transport from Tearing Instability

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Abstract: In the MST reversed field pinch (RFP), the plasma spontaneously rotates toroidally and poloidally. The rotation speed varies with radius, peaking at the plasma centre. The radial profile of the plasma rotation undergoes a rapid flattening during reconnection events. The RFP experiences periodic reconnection events (sawtooth crashes), associated with bursts of multiple tearing modes. The rapid flattening of the rotation profile represents a transport of angular momentum in the radial direction; it occurs much faster than can be explained by classical, collisional diffusion. This effect could also be active in tokamaks under conditions of strong MHD activity, thereby affecting mode rotation. We report here both detailed experimental and theoretical studies that put forth a picture of momentum transport from tearing instability. Experimentally, we report (1) spectroscopic measurements of momentum transport, (2) core measurements of Maxwell stress with laser Faraday rotation, and (3) edge probe measurements of both the Maxwell and Reynolds stresses. These measurements reveal a somewhat surprising picture in which the Maxwell and Reynolds stresses are each individually much larger than needed to account for the measured change in momentum. However, in the edge they are opposite in direction and nearly equal in magnitude, so that their difference is comparable to the inertial term. Theoretically, we report (1) quasilinear, analytic calculation of the tearing mode stresses for a flowing plasmas and (2) full MHD computation of the stresses from nonlinearly coupled tearing modes, and their effect on the flow profile. The combined experimental and theoretical studies establish that tearing modes transport momentum, and that the transport is enhanced through nonlinear mode coupling.

EX/P5-4 · Counter-current Rotation and ITB Formation in Alcator C-Mod LHCD Plasmas

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Abstract: Tokamak plasmas with internal transport barriers (ITBs) are attractive for their potential high fusion performance and large bootstrap current. Most ITBs are generated with external rotation and/or current drive applied during the plasma current ramp. Of particular interest are ITBs formed during the current flat top and under reactor relevant conditions, with equilibrated electrons and ions, and without external momentum input. Following application of LHCD during the current flat top, the toroidal rotation in Alcator C-Mod L- and H-mode plasmas is found to increment in the counter-current direction in conjunction with a decrease in the plasma internal. Along with drops in l_i and the core V_{Tor}, there are increases in the electron density and central ion and electron temperatures. The mechanism giving rise to counter-current rotation is unknown, but it is certainly not due to energetic electron loss, which would induce co-current rotation. During this evolution, transport barriers form in the density, momentum and energy channels. These ITBs develop over a time scale similar to the current relaxation time but slow compared to the energy and momentum confinement times. For most conditions, the ITB foot is located near $r/a \sim 0.4$ for electron density and temperature, and for ion (impurity) rotation, temperature and density. These discharges exhibit sawtooth oscillations throughout, with an inversion radius well inside the ITB foot. The role of magnetic shear in the ITB formation is under investigation, in addition to techniques for manipulating the foot location and increasing the strength of the barrier. The magnitude of the changes in the central rotation velocity and the internal inductance is correlated and found to increase with increasing LHCD power and decreasing electron density. The maximum effect is found with a phasing of 60° ($n_\parallel = 1.6$), with a smaller magnitude at $120^\circ (n_\parallel = 3.1)$, and no effect for negative or heating phasing. These results are consistent with the current drive efficiency scaling as $P_{LH}/(n_e n_\parallel^2)$. Regardless of the plasma parameters and $n_\parallel$, there is a strong correlation between the rotation velocity and l_i changes, possibly providing a clue for the underlying mechanism. Since the magnitude of the ITB scales with the input LHCD power, the future prospects for this technique look promising.

EX/P5-5 · Measurement of Temporal Evolution of Plasma Rotation in the TCABR Tokamak

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Abstract: In toroidal plasma configurations there is a great interest in the damping of poloidal and toroidal plasma rotation. However the diagnostic techniques used to measure the temporal profiles (charge-exchange spectroscopy and multichannel diode array detector) do not have sufficient time resolution and in general are not available in small laboratories. Other methods, such as Mach probes are also frequently used, although the latter technique is related to edge plasma rotation measurements. A new method for measuring the temporal evolution of plasma rotation in tokamaks is used in this work. It is based on the ratio of spectral line intensities of an impurity detected with two different photomultipliers installed at the exit slits of a monochromator. The light from the plasma is collected and transmitted to the entrance slit of the monochromator through an optical fiber. Inside the monochromator, using a semi-transparent mirror, the light is divided into two parts and directed to the photomultipliers. Therefore the photomultiplier at the exit slit 1 detects the left part of spectral line profile while the slit 2 detects the right part. When the plasma begins to move, the centre of the spectral line will move changing the ratio, which is directly correlated to the plasma rotation velocity. Preliminary results of the temporal profile of the poloidal rotation velocity of CIII (464.74 nm) at $r = 16$ cm in TCABR tokamak obtained using this set up show good agreement, within experimental uncertainty, with previous results [1].

[1] J. H. F. Severo, et al - 2003 - Nuclear Fusion 43 1047.

EX/P5-6 · Experimental Investigation of Particle Pinch Associated with Turbulence in LHD Heliotron and JT-60U Tokamak Plasmas

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Abstract: Comparative studies were carried out to elucidate the most essential parameters for control of density profiles in LHD heliotron and JT60U tokamak plasmas. Clear differences and similarities of characteristics of density profiles have been found between the two plasmas in low collisionality regime. Density peaking increased clearly with decreasing collisionality in JT60U, while it increased moderately at an inwardly shifted configuration and decreased with decreasing collisionality at an outwardly shifted configuration in LHD. Particle sources from walls decreased exponentially and did not affect the core density profiles inside the last closed flux surface in both devices. In JT60U, the central particle source was changed by a factor three by the combination of neutral beam injection and electron cyclotron heating powers. However, the density peaking factor did not change. In LHD, central particle fueling was increased by a factor eight with an increase of neutral beam injection powers, resulting in the change of density profiles from peaked to hollow although the beam fueling supplied particles more to the core than to the edge. Carbon impurity profiles did not change in both devices, indicating that impurity accumulation did not affect density profiles. These observations suggest the changes of density profiles to be not due to the difference of particle fueling, but due to the difference of transport in both devices. In JT60U, neoclassical effects were negligible for peaked density profile. In LHD, the peaked and hollow density profiles were observed in the reduced neoclassical transport configuration. Turbulence driven anomalous transport is believed to play an important role in both devices. The turbulence was measured in LHD, and clear differences were found for peaked and hollow density profiles not in fluctuation amplitude but in fluctuation spatial structure.

EX/P5-7 · Investigation of Compact Toroid Penetration for Fueling Spherical Tokamak Plasmas on CPD

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Abstract: In previous Compact Toroid (CT) injection experiments on several tokamaks, although CT fueling had been successfully demonstrated, the CT fueling process has been not clear yet. We have thus conducted CT injection into simple toroidal or vertical vacuum magnetic fields to investigate quantitatively dynamics of CT plasmoid in the penetration process on a spherical tokamak (ST) device. Understanding the process allows us to address appropriately one of the critical issues for practical application of CT injection on reactor-grade tokamaks. In the experiment, the CT shift amount of about 0.26 m in a vertical magnetic field has been observed by using a fast camera. In addition to toroidal magnetic field, vertical one appears to affect CT trajectory in not conventional tokamak but ST devices operated at rather low toroidal fields. We have also observed the CT kinetic energy loss larger than the energy required for penetration into a toroidal magnetic field. Observation of CT attacks on the target plate with an IR camera has indicated that CT shifts 39 mm at the toroidal field of 261 G and the input energy due to CT impact in vacuum without magnetic fields is estimated to be 44 J, which is less than 10% of the initial CT kinetic energy. These suggest that unexpected adverse effects occur in CT penetration process.

EX/P5-8 · Central Fueling of Globus-M Plasma with the Help of Coaxial Plasma Gun.

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Abstract: New experimental results of the plasma jet injection into Globus-M are presented. The improved jet parameters generated by the two stage plasma gun are: density $> 2 \times 10^{22}$ m^{-3}, total number of accelerated particles $(1-5) \times 10^{19}$ and the flow velocity >200 km/s. The source produces stable gas and plasma flow for many shots (>1000 pulses) due to modification of the gas and plasma generating stages. Kinetic energy of the jet was increased by the higher discharge power at limited storage capacitor energy (2 kJ). Calorimeter registered > 200 J of the jet flux. Behaviour of the jets in the vacuum magnetic field and plasma of Globus-M was investigated. Video and streak cameras registered that the jet consists of different components with separate velocities from 100 to 200 km/s. Deeper penetration into the tokamak plasma in comparison with previous experiments was recorded. A fast density increase in a shot was recorded with the microwave interferometer and Thomson scattering. Temporal evolution of the electron temperature and density profiles obtained by means of multi-pulse Thomson scattering diagnostics during the first millisecond after the injection is analysed in the report. The measurements show that already in 100 μs after the start of the plasma jet injection into the target plasma with the current of 0.2 MA and density of $(2-6) \times 10^{19}$ m^{-3} the plasma density at the magnetic axis increases and the temperature drops, at that, changes in the central region are stronger than at the periphery. This fact seems to testify that the most part of the injected particles is stopped just in the central region of the plasma column. The experimental results are compared with computations.

EX/P5-9 · International Stellarator/Heliotron Database Activities on High-β Confinement and Operational Boundaries

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Abstract: The worldwide research program in stellarators and helical systems aims at developing an economically attractive fusion energy source, which is inherently adapted for disruption-free steady-state operation. In this paper, high-β data from the advanced stellarator Wendelstein 7-AS (W7-AS) and the heliotron type Large Helical Device (LHD) are emphasised, and their relevance for the International Stellarator/Heliotron Confinement and Profile Databases (ISHCDB/ISHPDB) is discussed. A particular issue concerns the dependence of the global confinement and local transport on the plasma β. Since most of the important magnetic configuration and equilibrium parameters can change with β, the standard ISS95 or ISS04 scaling laws for the global confinement time and the choice of configuration parameters have to be reviewed. In LHD, the observed deterioration of the confinement towards high-β can be traced back to an increased turbulent transport induced by changes of the magnetic configuration, whereas in W7-AS such a clear trend is not observed. Nevertheless, effects of the plasma β on the confinement are significant as concluded from comparing low and high β data subsets in the database. For this purpose a probabilistic model comparison approach is used, which is based on fundamental models that include or neglect magnetic (and collisional) effects, respectively. The different properties with regard to confinement, equilibrium and stability in stellarators and helical systems result in extended operational limits compared with tokamak devices. The plasma β has been raised to reactor relevant values of about 5% (volume averaged) in LHD without provoking deleterious MHD instabilities. The effects determining the maximum achievable β in experiment appear to be associated with an excessive Shafranov shift and/or the density limit. Actually, the density limit in stellarators is set by the available heating power. Therefore, very high densities can be realised with sufficient power, so that a favourable high density, low temperature path to an ignited reactor plasma seems feasible. Efforts are made to extend the the high-β subset in the present ISHCDB/ISHPDB databases by including more specific configuration parameters, local (profile) data and MHD data characterising the onset and magnitudes of pressure driven MHD modes.

EX/P5-10 · Radial Interaction in Dynamic Heat Transport of LHD Plasmas

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Abstract: The radial interaction of heat transport is clarified by cross-correlation between electron heat flux and electron temperature gradient in LHD. Correlations between the heat transport dynamics and the radial structures of turbulence are also investigated. The existence of a turbulence modulator is discerned by using the conventional reflectometry signals. The amplitude of high frequency (> 100 kHz) density fluctuations is modulated with low frequency (< 2 kHz) in edge-core coupled plasmas.

EX/P5-11 · Impact of Magnetic Shear Modification on Confinement and Turbulent Fluctuations in LHD Plasmas

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Abstract: For the comprehensive understandings of transport phenomena in toroidal confinement systems and improvement of the predictive capability for burning plasmas, the impact of magnetic shear has been extensively investigated in the Large Helical Device for comparison with tokamaks. In order to extract the contribution of the magnetic shear, a series of dedicated experiments has been designed and performed, where inherently weak negative shear is largely modified solely by the beam driven current at nearly constant densities and heating powers. The tangential NB was switched from co- to ctr-direction and vice versa at various densities and ion to electron temperature ratios. In addition, modulated ECH was applied for the perturbative transport analysis. For discharges where the dominant rational surfaces reside in the low magnetic shear region, local flattening of the temperature profile was observed due to the vigorous MHD activities accompanying the island formation, following further reduction of the magnetic shear to the negative values. As for the MHD stable plasmas with the magnetic well structure expanding over half the minor radius, on the other hand, it has been found that the responses of temperature gradient and thermal diffusivity in the core region are quite subtle under substantial modification of the magnetic shear. Indeed, not only the kinetic profiles but also the characteristic density and temperature gradient lengths as well as the ion and electron temperatures were amazingly sustained within a few percent across the beam switching. Consequently, the pronounced effect of magnetic shear, which has been hitherto considered to be ubiquitous and strongly impacts the core transport in tokamaks, was not quite obvious. According to the gradual changes in the magnetic shear, the broadband component of the turbulent fluctuations in the core evolves as a whole at a similar rate to the current diffusion time, following the conventional paradigm. However, the vigorous dynamics in the detailed spectrum structure have also been documented. In addition, noticeable changes in the thermal diffusivity evaluated using the perturbative technique have occasionally been observed under the active magnetic shear modification, both of which respond in much faster time scale than the characteristic time for either the magnetic diffusion time or the profile evolution.

EX/P5-12 · Density Collapse in Improved Confinement Mode on Tohoku University Heliac

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Abstract: In Tohoku University Heliac (TU-Heliac), a small helical device, the electrode biasing experiments have been carried on to trigger and achieve an improved confinement mode using the hot cathode. And we tried to clarify the effects of the ion viscosity maxima on the transition to the improved confinement mode [1] and the effect of magnetic Fourier components on the neo-classical viscosity by the externally controlled $J \times B$ driving force for a poloidal rotation [2]. Here J and B are the electrode current and the magnetic field. The ion viscosities were estimated in various magnetic configurations [3]. In this improved confinement mode we observed the density collapse, which occurred periodically around the magnetic axis and accompanied with the bursting high frequency fluctuations. The density profiles showed the steep gradient around the core plasma region before the collapse. The steep density profile collapsed with the bursting fluctuation ($100 < f < 400$ kHz), which had $m = 2$ poloidal mode number and the frequency agreed well with the $E \times B$ plasma rotating frequency calculated from $f = mE/(2\pi < r > B)$. Here E, $< r >$ and m are the radial electric field, the averaged minor radius and the integer that corresponds to the poloidal mode number. The high frequency fluctuations in connection with the density collapse strongly depended on the plasma poloidal rotation driven by the hot cathode biasing. After the collapse the steep gradient in the density profile disappeared and the density outside the core region increased in the level consistent with the decrease of the core region. The collapse did not seem to degrade the overall performance in the improved mode. The appearance of bursting high frequency fluctuations and the relation between the density collapse and the bursting fluctuation in a biased plasma were observed for the first time.

[1] S. Kitajima et al., Nucl. Fusion 46, 200 (2006). [2] H. Takahashi et al., J. of Plasma and Fusion Research SERIES 6, 366 (2003). [3] S. Kitajima et al., Nucl. Fusion, 48, 035002 (2008).

EX/P5-13 · Effect of Bumpy Magnetic Field on Energy Confinement in NBI Plasmas of Heliotron J

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Abstract: This paper describes the effect of the bumpy magnetic field (bumpiness) on the energy confinement in NBI plasmas of Heliotron J. Control of bumpiness is key issue for the collisionless particle transport in helical-axis heliotron configuration since bumpiness has a role to reduce the grad-B drift by aligning the mod B_{min} contour with the flux surface. The experiments are carried out using three configurations with different bumpiness of 0.01 (low), 0.06 (medium) and 0.15 (high). In the high and medium bumpiness configurations in the counter-NBI plasmas at the averaged electron density of 2×10^{19} m^{-3}, relatively higher stored energy is observed than that in the low bumpiness case. The enhancement factor of energy confinement time to the international stellarator scaling law ISS95 (H_{ISS95}) is 1.8 and 1.7 in the high and medium bumpiness configurations, respectively, which is higher than that for the low bumpiness case ($H_{ISS95} \sim 1.4$). The improvement in the energy confinement in the two configurations can be explained qualitatively in terms of the increase both in ion and electron temperatures. The bulk ion temperature deduced by charge-exchange (CX) neutral particle analysis increases with bumpiness. The electron cyclotron emission measurement indicates increase in the electron temperature both in high and medium bumpiness configurations. From the study for the energy confinement in ECH plasmas of Heliotron J, it has been found that H_{ISS95} around 2 has been obtained in the medium bumpiness case and it has been about $1.4 - 1.6$ for the high and low bumpiness configurations. The difference in the energy confinement between the NBI and ECH plasmas is interpreted by the increase in the temperature due to the improvement in the energetic ion confinement by bumpiness. It has been found that the 1/e decay time of the high energy CX flux (≥ 18 keV) after the turning-off of NBI increased with increasing bumpiness. These results suggest that the control of the bumpiness is effective not only in the energetic ion but also in the global energy confinement in the NBI plasmas of Heliotron J.

EX/P5-14 · Fast Transport Transitions in High Shear L-2M Stellarator: Role of Moderate-Order Rational Magnetic Surfaces

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Abstract: Fast transport transitions in high shear L-2M stellarator under zero net current mode of operation are analyzed. The characteristic feature of the experiment is that plasma pressure is small enough ($\beta < 0.2\%$ where β is the volume averaged ratio of the plasma pressure to the pressure of magnetic field). Therefore, with the exception of central region magnetic surfaces are weakly disturbed by plasma pressure effects and the only source for plasma instabilities is the thermal energy of plasma. In much detail we discuss recently found [1] fast transport transitions to the regime with improved confinement (with duration < 200 ms) having specific sandwich structure of the plasma edge. It is shown that transport events are triggered by local disturbances of plasma parameters in the vicinity of moderate-order rational magnetic surfaces caused by plasma instabilities. As plasmas are stable in the framework of ideal MHD special analysis is performed within two-fluid magnetohydrodynamic and it is shown that found instabilities can be trigger for transport transitions. For reference we use conventional transport transition having no sandwich structure that happens at the initial stage of the discharge. The availability of several transport states with different behavior of plasma edge turbulence gave a chance to draw definite conclusion on electromagnetic effects role in plasma edge turbulence. It is shown that for low frequency part (important for turbulent transport) electromagnetic effects govern non adiabatic response of electrons. For high-frequency part non adiabatic response behavior is governed by several different reasons.

[1] Shchepetov S.V. et al. (2008) Plasma Phys. Control. Fusion, v. 50, 045001

EX/P5-15 · Physics Insight and Performance Benefit in MHD and Energy Transport from Plasma Shaping Experiments in the TCV Tokamak

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Abstract: The Tokamak à Configuration Variable TCV is unique in the world fusion program in having the flexibility to produce plasmas with extreme shaping of the plasma cross section, including elongations k up to 2.8, positive and negative triangularities $-0.8 \leq 0.9$, limited/diverted configurations, and with up to 4.5 MW localized EC heating. The characterization of the effects of shaping is crucial for the design of future devices beyond ITER, and is used in TCV as a tool to test transport and stability theory. This paper enables a comprehensive and integrated view of the various specific effects of plasma shaping observed. Confinement has been found to depend strongly on triangularity in electron cyclotron (EC) heated L-mode plasmas. For otherwise identical discharges e.g., the heat flux necessary to sustain the same profiles and stored energy in a discharge with $d = -0.4$ is half of that required for a discharge with $d = +0.4$. This measured change in confinement directly arises from a strong dependence of the electron energy diffusivity on triangularity and on collisionality. Gyrokinetic simulations predict the Trapped Electron Modes turbulence (TEM) to be dominant in these low collisionality and hot electron plasmas, and the TEM induced transport is shown to strongly depend on triangularity and collisionality. The non-linear heat flux simulation results are in notable agreement with observation. Plasma shaping also modifies MHD and is found essential in stabilizing modes and preventing disruptions in low-q operation, e.g., in the current ramp-up, essential for ITER high-current scenarios. High k is destabilizing to the internal kink (frequent sawteeth), and can even suppress sawteeth at very high elongation, of interest to avoid neoclassical tearing modes (NTM) seeding. In contrast, high positive and negative d is stabilizing, in agreement with ideal MHD stability predictions. Experiments with negative triangularity H-mode plasmas will be started, a domain where no experience exists so far. Initial ideal stability calculations with an X-point on the low field side (LFS) indicate a lower pedestal height attainable in negative triangularity configurations, which can potentially lead to ELM regimes different from the positive triangularity case. Similarly, electron internal transport barriers (eITB) studies will be extended to negative triangularity.

EX/P5-16 · Study of Globus-M low Aspect Ratio Plasma in Improved Confinement Regime.

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Abstract: Results on ion heating in experiments with neutral beam injection ($23-25$ keV, 0.8 MW) and ion cyclotron heating at the fundamental harmonic of light atomic mass (hydrogen) minority in deuterium plasma ($7.5-10.0$ MHz, $100-300$ kW) in the Globus-M spherical tokamak are presented. Ion temperature reached 0.75 keV value during NB injector heating pulse and exceeded electron temperature 1.5 times. Numerical simulation of fast ion trajectories is performed in the magnetic field geometry reconstructed from magnetic measurements with the help of the EFIT code. Experimental spectra of slowing down fast ions are measured with tangential and perpendicular NPAs and compared to classical Coulomb theory. Reasons limiting the ion heating efficiency are discussed. Reliable H-mode transition was achieved in Globus-M tokamak after the alteration of the toroidal magnetic field direction. The ion drift now is directed towards the lower X-point, which exists inside the vessel even in a limiter configuration. The most stable H mode operation was achieved in NBI experiments. Also L-H transition was observed in OH regime and at ICR heating in limiter and divertor configurations. The H-mode transition was achieved under conditions when the plasma boundary was very close to the vessel wall, and the safety factor had extremely low value ~ 2.7 for spherical tokamaks. By means of multi-channel multi-pulse TS diagnostic we observed twofold density increase near the plasma boundary right after the L-H transition and a formation of a steep density gradient. Similar effect is observed in electron pressure profiles. Flat density distribution in the plasma bulk is sustained during the all H-mode period. Modeling of transport processes in Globus-M in the L and H-modes is performed with the help of the ASTRA code. The calculated energy confinement time is consistent with the ITER scaling for L-modes. Neoclassical electric field and its shear near the separatrix are of the same order as those in the typical L-mode shots in ASDEX-Upgrade and MAST. It is shown that in the H-mode with NBI auxiliary heating the ion heat conductivity remains neoclassical and no barrier is formed for the ion temperature. The particle diffusion coefficient is reduced down to the values $(0.2-0.5)$ m^2/s inside the transport barrier near the separatrix.

EX/P5-17 · Confinement and Fueling in MAST

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Abstract: The dependencies of energy confinement on the main engineering parameters have been investigated in H-mode by expanding operational space towards higher plasma currents (up to 1.2 MA) and heating powers (up to 3.5 MW). Multivariate fits show that the dependence of global energy confinement time on plasma current is weaker than in tokamaks with conventional aspect ratio while the toroidal field dependence is stronger. The I_p and B_t dependencies have also been analysed by single parameter scans using TRANSP together with high spatial resolution electron temperature diagnostics. This analysis shows that the favourable I_p and B_t scalings come from the outer zone of the plasma ($r/a > 0.7$), i.e., the zone affected by fueling. Fueling and particle confinement have been studied using high field side pellets with relatively small penetration depth. Pellet deposition is captured by images in visible bremsstrahlung, fast cameras and pellet-triggered high resolution Thomson scattering. The pellet deposition profiles can so far only be explained by invoking the ∇B drift of the pellet ablatant. The pellet creates a distinct zone with sharp positive density gradient and doubled temperature gradient. Simulations using the linear GS2 code and the global nonlinear CUTIE code show that these changes could change the character of micro-turbulence and thus modify the role of the pedestal as a boundary for core transport. The complex post–pellet dynamics of the density profile is characterised by the pellet retention time determined from 200 Hz Thomson scattering. The pellet retention time decreases rapidly for shallower pellet deposition, $r_{pel} \to a$, and for the ITER-like case, $r_{pel} \sim 0.8a$, the pellet retention time is about 20% of the energy confinement time. For ITER this would extrapolate to a pellet particle throughput of about 70% of the ITER design steady-state value. For a Component Test Facility based on the spherical tokamak predicted particle throughput is about 1/4 of that in ITER.

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EX/P5-18 · Tokamak Plasma Self-Organization and Possibility to Have the Peaked Density Profile in ITER

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Abstract: The extrapolation from to-day tokamaks to ITER dimension hitherto remain to be very far. We can't do such extrapolation with the help of scaling built without deep understanding of the physical processes. The main process in the turbulent plasma is its self-organization. Transport coefficients may be changed in plasma in a wide range by changing of the instabilities level. This permit plasma to build the most stable under given external impacts energy distribution profile, i.e., plasma pressure self-consistent profile. This process cannot be realized only in the regions, where the turbulent transport value is not enough for the p(r) regulation. The enhanced grad p appears here. We call these regions "transport barriers". The plasma pressure self-consistent profile and transport barriers are two main factors, which determine the plasma pressure profile. At T-10 tokamak the phenomenon of plasma pressure self-consistent profile was investigated and it was shown that normalized plasma pressure does not depend on any plasma parameters if we normalize the minor radius to the radius $r_c = (IR/kB)^{1/2}$, $\rho = r/r_c$, k is plasma elongation coefficient, I plasma current, R major radius and B magnetic field strength. In this case $p(\rho)/p(0)$ practically the same for T-10 and JET if in given shot there was not pronounced ITB. The process of $p(\rho)$ restoration very fast and links to plasma minor radius equillibrium. Now we can predict the self-consistent pressure profile for ITER, which can be changed by ITB formation only. As we expect in ITER the peaked power deposition profile, and so the peaked temperature profile, we can conclude that density profile will be flat even in the low-collisionality case.

EX/P5-19 · Simultaneous Analysis of Ion and Electron Heat Transport by Power Modulation in JET

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Abstract: This paper presents a simultaneous investigation of ion and electron heat transport carried out in the JET tokamak by means of heating power modulation methods. It is widely recognized that the dynamical response of a system to a known external perturbation provides a rich information on the underlying physics. This approach has been very successfully applied to investigate electron heat transport in fusion plasmas, evidencing in particular the existence of a threshold in normalized electron temperature gradient as well as the increase of transport above it. These experimental results agree well with theoretical expectations provided by turbulent transport calculations. However, the dynamical method has not been used, so far, to investigate ion heat transport. This is the main aim of the present work, for which JET provides a unique possibility. We describe experiments in which the perturbative method has been applied simultaneously to the ion and electron heat channels using power modulation of Ion Cyclotron Resonance Heating in the 3He minority scheme. For the electrons, the results agree with those obtained previously in JET and other tokamaks. For the ions, the rate of increase of transport, also known as stiffness factor, could be determined. It is, as expected, larger than that of the electrons by a factor of 2 to 3 and rather close to that predicted by theory, as underlined by a comparison with gyro-kinetic calculations. Results from transport modelling taking into account these experimental values for threshold and stiffness reproduce simultaneously and with accuracy the time-averaged and modulated data for both ion and electron channels. The different contributions of the ICRH heating processes, provided by the PION code, are used in the modelling. Essential here is that the modulation data clearly indicate that the time constants for the different processes as yielded by PION are correct. Therefore, these investigations yield comprehensive and coherent results for both ion and electron heat transport, together with an assessment of the heating mechanisms in the 3He ICRH scheme.

EX/P5-20 · Density profile Behavior in JET H-mode plasmas

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Abstract: As observed on many different tokamak devices, density profiles in H-mode plasmas are not necessarily flat. In many cases observed density gradients are considerably high and cannot be explained only by core particle sources and neoclassical pinch effects. Recent research done on JET, AUG and C-MOD machines suggests a peaked density profile for the ITER baseline scenario plasma conditions, which should greatly affect its expected performance – namely increased fusion power gain and non-inductive bootstrap current generation. In this work we've studied the density profile behavior in JET H-mode in various plasma conditions. A comprehensive database was created on the basis of recently done experiments (2006-2007) which allowed us to investigate the dependencies of density peaking on various plasma parameters, such as electron and ion temperatures, the q profile and core particle sources. General conclusions of previous studies on this topic were confirmed: effective collisionality seems to be the most important value determining the density profile shape while all other parameters have much less impact or no effect at all. Along with data analysis, series of quasilinear gyrokinetic simulations using GS2 code were done in order to investigate the effect of ITG turbulence on the density profile formation. More than 250 simulations were performed for the JET relevant plasma parameters with different collisionalities, density gradients and T_e/T_i ratio. Calculated particle fluxes were normalized to ion heat fluxes (assuming that ion heat transport is driven dominantly by ITG turbulence) and compared with the actual JET experimental values. These simple calculations show a good qualitative agreement with the experiments making the anomalous fluxes generated by ITG turbulence a promising candidate for explanation of density profile behavior observed in JET H-mode plasmas.

EX/P5-21 · Particle Transport and Electron Density Relaxation Due to Stochastic Magnetic Fields in the MST Reversed Field Pinch

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Abstract: Particle transport resulting from stochastic magnetic fields has long been important in fusion research. Stochastic magnetic fields can arise from global tearing instabilities that often underlie the sawtooth oscillation and lead to density relaxation. Conversely, stochastic fields have also been externally imposed (resonant magnetic perturbation) to mitigate the effect of edge localized modes (ELMs) by locally enhancing edge transport in Tokamaks. Basic understanding of stochastic field driven transport processes is thus of great interest and possibly critical to ELM control in ITER. Progress to date is largely limited by the lack of measurements of the magnetic fluctuation-induced particle flux in hot plasmas. In this paper, we report the first measurements of stochastic magnetic field driven particle flux in the core of the MST reversed field pinch (RFP), particularly during the sawtooth crash when the stochastic field resulting from tearing reconnection is strongest. Direct measurements of the magnetic fluctuation-induced particle flux are made using a newly developed differential interferometer in combination with a fast Faraday rotation system. Measurements show that convective electron particle flux from stochastic magnetic field can account for the equilibrium density change at the magnetic axis. The electron particle flux primarily arises from electron density fluctuations while ion particle flux is due to ion velocity fluctuations.

EX/P5-22 · High Density Physics in Reversed Field Pinches : Comparison with Tokamaks and Stellarators

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Abstract: Reversed Field Pinches (RFPs) share with Tokamaks and Stellarators the experimental evidence of an upper limit for the maximum value of the electron density at which each device can operate. Above a certain density level, well described by the Greenwald law for Tokamaks and RFPs, a radiative collapse with strong plasma cooling is observed, predominantly due to processes occurring at the plasma boundary. In the RFX-mod RFP close to the density limit a radiating belt, poloidally symmetric and toroidally localized develops in the region where the plasma is shrunk as an effect of the $m = 0$ tearing modes. The phenomenology recalls that of MARFES or plasma detachment, however it has to be mentioned that, differently from Tokamaks, the appearance of the radiating belt is associated to a soft landing of the plasma discharge. This paper reports a complete experimental picture of the RFX-mod plasmas close to the density limit, including density and radiation profiles, plasma flow and turbulence and the results from the spectroscopic measurements. The data analysis indicates that particles, mainly produced where the plasma-wall interaction is stronger due to the phase locking of the $m = 1$ modes, are then toroidally conveyed towards the region of maximum shrinking of the plasma column where, if the transport is low enough, they accumulate. The interpretation is related to the topology of MHD $m = 0$ and $m = 1$ modes: the detailed reconstruction of the magnetic topology as obtained from magnetic field ray tracing and guiding centre codes shows that the highly radiative region corresponds to the presence of peripheral $m = 0$ magnetic islands well detached from the wall. The indication emerging from the paper is that in RFPs a reduction of the $m = 0$ activity can be a way to overcome the high density limit.

EX/P5-23 · Non-local Transport Phenomenon with SMBI on HL-2A

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Abstract: The non-local transport phenomena induced by supersonic molecular beam injection (SMBI) has been firstly observed in HL-2A tokomak in both the low density discharges and the high density discharges. For low density (less than 1.5×10^{19} m^{-3}, besides the similarities with the phenomena induced by other methods in various Tokamaks, the results on HL-2A show some experimental progress in this effect: the duration of the effect induced by SMBI in HL-2A could be prolonged by changing the condition of SMBs injection; both the bolometer radiation and the H_a emission decrease when the non-local effect appears in low density. Repetitive non-local effect induced by modulated SMBs allows Fourier transformation of the temperature perturbation, yielding detailed investigation of the pulse propagation. The investigation indicates that, although the fast core T_e response to edge cooling in high density is opposite to that in low density, the nonlocality of electron heat transport appears in both the two conditions. Analytic results also suggest a common underlying physical mechanism between the "non-local" transport phenomena in the two different conditions

EX/P5-24 · Observation of a Natural Particle Transport Barrier in HL-2A tokamak

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Abstract: In this paper we report the observation of a natural or spontaneous generated internal particle transport barrier (pITB) in purely ohmically heated plasmas in the HL-2A tokamak without any external input momentum or core particle source. The pITB have been observed when the electron density exceeds a threshold, which is 2.2×10^{19} m^{-3}. It was found that the threshold becomes lower during ECRH heating, which is about 1.8×10^{19} m^{-3} with about 200 kW ECRH power injection. This phenomenon is perfectly reproducible and identified by profile analysis, perturbative transport studies with SMBI modulation, and the Doppler reflectometry, which measures the perpendicular rotation velocity of the turbulence. The barrier is generally located around $r/a = 0.6 - 0.7$ with a width of $1 - 2$ cm both in purely ohmic and ECRH discharge. By analysing the propagation of a particle wave generated by SMBI modulation across the barrier, the particle diffusivity and the convective velocity have been separately determined. The experimental results have been compared to an analytical model for particle transport to quantitatively characterize this barrier. The diffusivity is rather well-like than step-like and lower in the zone of barrier than outside. The convection is inward outside of the barrier, while the convection is outward inside the barrier. The measurement of the plasma rotation velocity by Doppler reflectometry has shown the drastic change in the rotation velocity, which is likely due to the steepness of the density gradient in the barrier. In the paper, the explanation for the convective velocity found with SMBI modulation has also been given. The change of sign for the convective velocity has been explained by the turbulence TEM/ITG system. The density threshold may be correlated to the TEM/ITG transition via the collisionality. The mechanism leading to the pITB formation remains unclear. Some possibilities of the mechanism have been discussed and need further experiments to be confirmed.

EX/P5-25 · High β Plasma in Improved Confinement Regime on the TPE-RX Reversed-Field Pinch Plasmas

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Abstract: A very high poloidal β was achieved by the Pulsed Poloidal Current Drive (PPCD) in Toroidal Pinch Experiment-RX (TPE-RX, was operated to March 2007 from April 1998). The plasma electron density and temperature increased, hence, the poloidal β was improved to 30% from 5% during PPCD (poloidal β is almost equal to total β in the reversed-field pinch). D_α emission from recycling deuterium by the plasma-wall interaction decreased during the PPCD and the ratio of the total number of particles to the D_α emission increased ten-fold at the end of the PPCD with a resulting improvement of the particle confinement time. Fast magnetic fluctuation levels in the edge region of TPE-RX were measured using a newly developed complex edge probe installed inside a vacuum vessel and sensitive to fast magnetic fluctuation. During the PPCD operation, the magnetic fluctuation associated with the dynamo effect is reduced, and improved confinement is reached, which results in this high β value. Unique NBI system suitable for the RFP machine was developed to control plasma parameters. In the case of high power neutral beam injection to the improved confinement plasma which was attained by PPCD operation with an ice pellet injection, the meaningful increments in the soft X-ray signal and the plasma current were found.

EX/P5-26 · Transport Mechanisms in the outer Region of RFX-mod

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Abstract: The transport processes taking place in the outer region of fusion-relevant magnetically confined plasmas are generally believed to be driven by the fluctuating electric field associated to plasma microturbulence. The edge plasma of RFX-mod, a large reversed field pinch (RFP) toroidal device, shares many peculiar characteristics of other configurations. The edge turbulence exhibits an intermittent behaviour, associated with blobs, coherent eddies and structures emerging form the turbulent background. Despite the crucial role in the improved confinement regimes of the interaction between the generation/dissipation of the flow and the creation/destruction of turbulent structures, its description is far from being completely understood. RFX-mod is equipped with a full set of edge diagnostic systems with sub-microsecond time resolution: the Gas Puffing Imaging diagnostic which gives great details of the structures and of their dynamics; the Integrated System of Internal Sensors, a distributed system of in-vessel magnetic and electrostatic probes, which can follow the structures in their motion and relate them to large scale instabilities; an insertable "U-probe" which simultaneously measures local magnetic field, current density, plasma pressure, $E\times B$ flow and turbulent particle flux; a "Gundestrup probe" which provides direct measurements of both parallel and perpendicular flow. Very recently new Thomson scattering and thermal Helium beam spectroscopy diagnostics for measurement of edge electron density and temperature profiles have been put into operation. The paper presents the new insights achieved starting from the obtained measurements: the properties of the structures in terms of pressure, flow and current density; an estimation of the diffusivity associated to the coherent structures, found to decrease with increasing $n/n_{Greenwald}$ between $0.1 < n/n_G < 0.35$ with a tendency to saturate at higher values; the analysis of turbulent momentum fluxes, described in terms of the Maxwell and Reynolds stress terms of momentum transport equation, where the Reynolds stress has been found to dominate the process over the Maxwell stress for at least a factor five.

EX/P5-27 · Advances in Plasma Heating and Confinement in the GOL-3 Multiple Mirror Trap

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Abstract: The paper reviews recent experimental results from multi mirror trap GOL-3. In present configuration of the GOL-3 a deuterium plasma with density of $10^{20} - 10^{22}$ m^{-3} is confined in a 12-meter-long solenoid, which comprises 52 corrugation cells with mirror ratio $B_{max}/B_{min} = 4.8/3.2$ T. The plasma in the solenoid is heated up to ∼2 keV temperature by a high power relativistic electron beam (∼0.8 MeV, ∼20 kA, ∼12 μs, ∼120 kJ) injected through one of the ends. During the upgrade of the relativistic diode assembly of the U-2 generator the beam duration was increased up to $11 - 12\mu$s. Parameters of electron distribution function during the beam injection and shortly after it stops were studied with the Thomson scattering system wich covers $0-20$ keV energy range. It was found that the electron distribution function is rather complex. In particular, preliminary data indicate that energetic electron tail exist till the end of the heating pulse. Phenomenon of oscillation of flux of D-D neutrons from near-the-end cells of the trap was found in previous experiments. Current understanding of this effect is that the plasma motion along the corrugated magnetic field due to axial pressure gradient leads to electrostatic instability of bounce oscillations of slightly trapped ions. Excitation of these oscillations leads to improvement of the axial plasma confinement in the multi mirror system. A special experiment with a change in magnetic configuration of the trap was done to validate the theory predictions. A high-mirror-ratio section was formed at first 2 meters of the solenoid with 5 cells at $B_{max}/B_{min} = 6.0/2.2$ T and 44 cm length each. Corresponding change in dynamics of neutron emission was detected. First results of the experiments with injection of sub-MW NBI are presented in the paper. In general, current GOL-3 parameters demonstrate good prospects of a multi mirror trap as a fusion reactor.

EX/P5-28 · High-β (Hot Electron) Plasma in Ring Trap 1 (RT-1)

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Abstract: The Ring Trap 1 (RT-1) experiment has succeeded to produce a magnetosphere- like high-β (hot electron) plasma in a dipole magnetic field. Magnetic levitation of the super-conducting magnet, freeing it from mechanical supports, brought about significant improvement of plasma parameters (density, temperature, confinement time, etc.). The maximum (local) β exceeds 0.1, which is primarily contributed by energetic (>1 keV) component of electrons ($n \sim 10^{16}$ m^{-3}) produced by ECR.

EX/P5-29 · Footprint of the Magnetic Configuration in ECH Plasmas of the TJ-II Flexible Heliac

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Abstract: The configurational flexibility of the TJ-II Heliac has been upgraded with the commissioning of a mode of operation that allows changing the magnetic configuration dynamically: the currents feeding the different coil sets can be ramped during the discharge, which allows, for instance, moving up or down the offset of the rotational transform profile. In these experiments the ohmic transformer is also activated so as to counteract the inevitably induced currents. This capability has been used to investigate the effect of low order rational values of the rotational transform, $\iota/2\pi$, in transport magnitudes, like effective diffusivities, without altering considerably the magnetic shear. The experiments in plasmas created and sustained with Electron Cyclotron Resonance Heating show, in agreement with previous experience from the TJ-II, that such low order rational values of $\iota/2\pi$ are locally coincident with lower, rather than higher as conventionally expected, effective diffusivities.

EX/P5-30 · Multi-scale Physics during Shear Flow Development in the TJ-II Stellarator

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Abstract: Sheared flows can be easily driven and damped at the plasma edge of the TJ-II stellarator by changing the plasma density [1] or during biasing experiments [2], which makes TJ-II an ideal plasma physics laboratory to unravel the physics of shear flow development and momentum transport in fusion plasmas. The main results reported in this paper are: 1) The discovery of long-range correlations in potential fluctuations that are amplified during the development of radial electric fields and transitions to improved confinement regimes. These experimental findings suggest the importance of long-range correlations during the development of edge shear flows. Comparative studies stellarator - tokamak (during L-H transition) should be stimulated to provide a critical test for the L-H transition physics mechanisms. 2) The first experimental evidence of the dual role of electric fields as a stabilizing mechanism of plasma turbulence and as an agent affecting the parallel momentum balance via turbulence modification. Then, turbulence driven momentum flux should be taken into account when considering the parallel momentum balance equation, particularly in high electric field shear regimes. Experiments were carried in ECRH plasmas in the TJ-II stellarator. Different edge plasma parameters were simultaneously characterized in two different toroidal positions using two similar multi-Langmuir probes installed on fast reciprocating drives and fast cameras during spontaneous and biasing induced transitions to improved confinement regimes.

[1] M. A. Pedrosa et al., Plasma Phys. Control. Fusion 49 (2007) B303 [2] C. Hidalgo et al., Plasma Phys. and Control. Fusion 46 (2004) 287

EX/P5-31 · Characterization of the Perpendicular Rotation Velocity of the Turbulence by Reflectometry in the Stellarator TJ-II

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Abstract: In TJ-II ECH plasmas, the perpendicular rotation velocity of the turbulence can be either positive or negative (ion or electron diamagnetic direction) and it is dominated by the radial electric field. Parametric studies performed in plasmas confined in different magnetic configurations and heated with different ECH power levels point to plasma collisionality as the parameter that controls the inversion in the perpendicular rotation velocity of the turbulence. Dedicated experiments have been performed to trigger the perpendicular rotation inversion either by changing the plasma collisionality (modulating the plasma density or the ECH power) or by plasma biasing. The inversion in the perpendicular rotation velocity and its radial propagation are characterized by using a two-channel fast frequency hopping reflectometer. Changes in the direction of the perpendicular rotation velocity as fast as $20 - 40$ ms can be resolved with good spatial resolution. Density modulation is used as an external knob to trigger the inversion of the perpendicular rotation of the turbulence. At the critical density, the perpendicular rotation velocity changes at the plasma edge region; a closer look using high temporal resolution reflectometry measurements reveals fast quasi-periodic changes in the propagation direction. No radial propagation of these fast changes in the rotation direction is observed. ECH power modulation experiments allow the study of the plasma rotation response time. The perpendicular rotation velocity reverses following the ECH modulation on a fast time scale and in a wide plasma region, from the plasma edge up to $r/a \approx 0.6$. Further inside the plasma, the velocity is modulated as well, but no rotation inversion is observed. Changes in the radial electric field at the plasma edge can be also induced by means of electrode biasing. The modification in the rotation velocity is more pronounced as the bias voltage increases; while the radial extension of the plasma affected by the biasing and the radial propagation of the rotation velocity changes depend on the plasma density. Finally, changes in the radial correlation characteristics of the plasma turbulence have been measured associated with changes in the perpendicular rotation velocity of the turbulence.

EX/P5-32 · Three-dimension Spectral Characteristics of Low-frequency Zonal Flow in the Edge Plasma of HL-2A Tokamak

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Abstract: The low-frequency Zonal Flow (LFZF), a potential structure which is symmetrical in both the poloidal and toroidal directions, spreads outward with small radial wave vector and coexists with the geodesic acoustic mode (GAM) is observed using several Langmuir probe arrays in the edge plasma of the HL-2A tokamak. It is identified as a low–frequency coherent mode peaking near zero frequency and broadening several kilohertz on the self power density spectrum (PSD) of the floating potential fluctuations. The estimated three-dimensional wavenumber-frequency spectral characteristics are almost consistent with the theoretical predications and simulation results: $k_\theta = k_\phi \simeq 0$, $k_r\rho_i \simeq 0.011$ and with the full width at half maximum $\delta(k_r\rho_i) \simeq 0.42$. The envelope analysis approach reveals that the envelopes of the radial electric field fluctuations is coherent with the LFZF, as similar as the GAM, just with a smaller coherent coefficient because the GAM is dominant in the edge plasma, which suggests that the ambient turbulence is modulated by the LFZF in the same way as the GAM.

EX/P5-33 · Density Fluctuation of GAM Zonal Flows on the HL-2A Tokamak

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Abstract: Although Zonal Flow (ZF) characteristics have been reviewed theoretically and experimentally due to its suppression effect on turbulent transport, there are some unclear issues necessary to be elucidated. Especially, there is little understanding for the mode structures of zero frequency ZF and the density fluctuation of the Geodesic Acoustic Mode (GAM) ZF, on which the recent experimental study in HL-2A is focused. The poloidal and toroidal mode numbers of the density fluctuation for the GAMZF are identified in HL-2A for the first time. Recently, a novel design combined rake probes with three-step Langmuir probes is used to measure the GAM density fluctuation in HL-2A. Each tip as single probe measures the saturation ion current (I_s). The relative density fluctuation is replaced by the relative I_s fluctuation because the relative temperature fluctuation contributes less than 10% of the I_s fluctuation. The zonal flow on density fluctuation in shot 8847 has the GAM frequency $f = 9.8$ kHz, full width (∼3.9 kHz) at half magnitude (FWHM) and lifetime ∼0.3 ms. The relative I_s fluctuation within the FWHM at the GAM frequency is 4.1%, corresponding to the relative density fluctuation, which is much higher than the value predicted by the ZF theory (0.03%). The phase difference of the signals between tips is used to estimate the poloidal or toroidal mode number, respectively. The poloidal mode number at the GAM frequency is $m = 1.3$ in shot 8847, while the corresponding toroidal mode number is $n = 0.04$. Both are in agreement with the theoretical prediction mode numbers ($m = 1/n = 0$). Statistic analysis results in different discharges also indicate that the poloidal mode numbers approach unit. The radial wave vector at the GAM frequency is estimated as 0.9 cm^{-1} in a similar discharge. The squared auto-bicoherence of the ion saturation current fluctuation has a spectral peak at the GAM frequency, indicating that the GAM density fluctuation may be driven by nonlinear three-wave coupling of ambient turbulence. In addition, the poloidal and toroidal mode numbers ($m = 0/n = 0$) of the GAM potential fluctuation predicted by the ZF theory are simultaneously confirmed as well within experimental error bars ($m < 0.3$, $n < 0.1$).

EX/P5-34 · Two Distinct Regimes of Turbulence in HL-2A Tokamak Plasmas

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Abstract: The distinct characteristics of low frequency fluctuations (termed quasimode (QM) tentatively here) of several tens kilohertz and the higher frequency ambient turbulence (HFAT) of 100 kHz or higher are measured with high spatial and temporal resolution Langmuir probe arrays at the edge of the HL-2A tokamak plasmas. Three dimensional wave number spectra and dispersion relations are investigated and compared between the QM and HFAT. Significant differences, revealing the existence of the two distinguishable regimes, are observed for the first time. The poloidal and toroidal correlation lengths of the fluctuations in the QM regime are one order of magnitude longer than those in the HFAT regime. The timescale ratio of the QM and HFAT is of the same order. The nonlinear three wave coupling between the QM and HFAT turbulence is identified with bi-coherence analysis to be a plausible generation mechanism for the former. The phase shift between the QM and the envelopes of the HFAT is around 180^o , indicating possible regulating effects of the former on the latter. Possible correlation of the results with the theory and simulation predictions on QM is discussed as well.

EX/P5-35 · First Observation of Reduced Core Electron Temperature Fluctuations and Intermediate Wavenumber Density Fluctuations in H- and QH-mode Plasmas

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Abstract: We report the first observation of reduced core electron temperature and intermediate wavenumber density fluctuations in H and quiescent QH-mode plasmas in the DIII-D tokamak. These new measurements allow a more complete picture to be drawn of anomalous electron heat transport and the physics of the L-H transition. L-mode electron temperature fluctuation levels ($0.5\% \leq \tilde{T}_e/T_e \leq 2\%$ for $k_\theta \rho_s \leq 0.3$ as measured by correlation ECE (CECE) radiometry) are observed to decrease by at least a factor of four in H and QH-mode regimes ($r/a = 0.7$). In addition, recent Doppler backscattering data indicate that intermediate wavenumber density fluctuations ($1 \leq k_\theta \rho_s \leq 2$) are also reduced substantially ($\sim 30\%$) in these regimes for $r/a = 0.7$. A dramatic reduction within 100 ms of the L-H transition is seen at the top of the pedestal ($r/a \leq 0.9$). Backscattering and FIR scattering data confirm reductions of $\tilde{n}/n$ in this intermediate-k (TEM) range in the core region ($r/a \leq 0.65$). The measurements reported here also show that during L-mode operation low-k relative temperature fluctuation levels are similar in magnitude to low-k density fluctuations levels. In the absence of phase measurements between the various fluctuating quantities it must be concluded that $\tilde{T}_e$ and $\tilde{n}_e$ fluctuations are equally likely to contribute to the observed anomalous electron heat transport. The observed $\tilde{T}_e$ reduction at the transition to H-mode suggests that temperature fluctuations potentially play a major role in the improved confinement properties ($\chi_e^{QH}/\chi_e^L < 0.25$) observed during H and QH-mode operation. Using linear stability calculations with the TGLF code, the observed low k_θ temperature fluctuations ($k_\theta \rho_s \leq 0.6$) are attributed to ion temperature gradient (ITG) modes stabilized by $E{\times}B$ shear at the L-H-mode transition. The observed reduction of core electron temperature fluctuations and intermediate-k density fluctuations during H-and QH-mode are fundamental new observations that provide a fresh perspective on the physics of electron heat transport in L- and H-mode operation in tokamak plasmas.

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EX/P5-36 · The Study of the Statistical Properties of Electric Potential Oscillations in the T-10 Tokamak

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Abstract: Highlights in fusion plasma performance have been obtained under conditions where $E{\times}B$ shear stabilization mechanisms are likely play a key role. The oscillating shear flows, zonal flows and Geodesic Acoustic Modes (GAMs) are widely considered now as a turbulence stabilization mechanism. The study of the properties of plasma electric potential oscillations may explain the turbulence - zonal flow interplay. Low-frequency ($f < 50$ kHz) electric potential oscillations were investigated on the T-10 tokamak using Heavy Ion Beam Probe (HIBP), Correlation Reflectometry (CR) and Langmuir probes (LP). Regimes with Ohmic heating and with on- and off-axis ECRH were studied ($B = 2.2 - 2.5$ T, $I_p = 180 - 330$ kA, $n_e = (1.3 - 2.5) \times 10^{19}$ m^{-3}. It was shown that GAMs might have a complex structure, not similar to conventional periodical oscillations with a single frequency. GAMs have an intermittent character presenting the stochastic sequence of the wave packages with a "lifetime" in a range of $0.5 - 2$ ms. GAMs are more pronounced in ECRH plasmas. At the outer one third of the plasma column GAMs may be characterized not only by single, but also by a couple of closed but separated frequencies in a narrow interval $22 - 27$ kHz. Each frequency peak is characterized by high potential/density correlation (coh > 0.6) and constant cross-phase, $-\pi/2$ for the main GAM peak, similar to observed in JIPP T-IIU, and $+\pi/2$ for the satellite. The low-frequency MHD mode $m = 3$ is pronounced at 7 kHz. The long-term correlation between potential (HIBP) and density (CR) oscillations were studied by the CR antenna shifted toroidally by one quarter of the torus similar to the duo-HIBP experiments in CHS. The result of the correlation measurements of the two diagnostics shows the high coherency (coh > 0.4) for GAMs and low-frequency MHD modes, suggesting the global character of the modes. The MHD mode $m = 3$ is also pronounced in core plasma potential and density (HIBP) and in floating potential at the plasma edge (LP) by quasicoherent oscillations with $f = 7$ kHz with high coherency between plasma potential (HIBP) and floating potential (LP). It also exhibits the stochastic character with $\sim 1 - 2$ ms time-scale. Sawtooth oscillations presents the common mechanism modulating potential oscillations for both GAM and MHD mode $m = 3$.

EX/P5-37 · Spatial Structure of Density Fluctuations and Geodesic Acoustic Mode in T-10 Tokamak.

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Abstract: Investigation of turbulence properties could reveals the nature of anomalous transport in tokamak. This paper presents results of systematic investigations of poloidal and toroidal structure of small-scale density fluctuations as well as the properties of Geodesic Acoustic Modes (GAM) in T-10 tokamak. It was found that total density fluctuations amplitude is strongly non uniform in poloidal direction. Turbulence amplitude had a maximum at low magnetic field side and minimum at high magnetic field side in accordance with theory predictions. The asymmetry of turbulence amplitude increased with the rise of total heating power. Poloidal dependence of the amplitude of different spectra components are showed in this paper. Improved reflectometry system allowed to made long distance toroidal correlations without parasitic cross-talk between different antennae arrays. It was found that quasi-coherent oscillations had a toroidal correlation length about 12.5 m whereas the length of stochastic low frequency fluctuations was 2.5 m. It was found some evidence of oblique propagation of the quasi-coherent oscillation at small angle (about 0.1 degree) with respect to magnetic field line. Experiments with joint turbulence observations by different diagnostics were made in T-10 tokamak. Strong correlation at GAM frequency was found between fluctuations of plasma potential, measured by heavy ion beam probe, and density fluctuations, measured by correlation reflectometry. GAM amplitude in density fluctuations spectra was high at the top antennae positions and negligibly small at equatorial plane in accordance with theory predictions. It was reported previously that density perturbations by GAM exist at local electron densities lower than 2×10^{19} m^{-3} in Ohmic discharges. Recent experiments with ECR heating have shown presence of GAM in density fluctuation spectra at density about $(3 - 3.5) \times 10^{19}$ m^{-3}. Effect of the GAM on the turbulence amplitude at LFS and HFS is also discussed.

EX/P5-38 · Turbulence and Geodesic Acoustic Mode Behavioural Studies in ASDEX Upgrade Using Doppler Reflectometry

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Abstract: The interaction of turbulence with stationary and oscillating zonal flows is important for the behaviour of turbulent transport in magnetically confined plasmas. On ASDEX Upgrade Doppler reflectometry has been developed for direct measurement of E_r profiles, its fluctuations and density turbulence properties such as radial correlation lengths L_r and perpendicular k-spectra. Around the H-mode pedestal the turbulence amplitude at high k together with L_r are reduced, coincident with enhanced E_r shearing. L_r also falls with increasing plasma triangularity, consistent with improved global confinement. In both L and H-mode the k-spectrum become narrower towards the core region. Coherent E_r fluctuations with geodesic acoustic mode (GAM) behaviour are observed in the ohmic and L-mode plasma edge. GAMs are not detected deep in the core or in H-mode. The mode frequency scales as sound speed over major radius but with an inverse dependence on plasma elongation. The GAM amplitude also decreases inversely with elongation, but linear with q (consistent with collisionless damping) and the temperature gradient (drive) up to the L to H-mode transition.

EX/P5-39 · Effect of ECRH Regime on Characteristics of Short-Wave Turbulence in Plasma of the L-2M Stellarator

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Abstract: The present paper reports on studies of short-wave turbulence in plasma of L-2M stellarator under markedly different conditions: with doubling the ECR heating power (100 and 200 kW) and with restricting the plasma radius by a sector limITER. The role of such short-wave turbulence in anomalous transport can appear important for conditions of a thermonuclear reactor. Experiments were carried out in a basic magnetic configuration of the L-2M stellarator during ECRH at the second harmonic of the electron gyrofrequency (75.3 GHz) at average electron densities of $1.5 - 1.7 \times 10^{13}/\mathrm{cm}^{-3}$. The energy lifetime was ~3.5 ms at $P_0 = 100$ kW and dropped to about 2 ms at a $P_0 = 200$ kW. When the limiter was introduced inside the plasma to a depth of 2 cm from the last closed flux surface, τ_e decreased by a factor of $1.3 - 1.4$. Plasma density fluctuations were measured from the scattering spectra of gyrotron radiation at the second harmonic of operating frequency ($\sim 150\ GHz$) [1]. A quasioptical receiving system allowed measurements of scattered radiation from plasma regions $r/a \leq 0.6$ at scattering angles $\pi/4 \leq \theta \leq \pi/2 (24cm^{-1} \leq k_\perp \leq 44cm^{-1})$. The short-wave turbulence was studied for two radial positions of the scattering region: $r/a = 0.3 - 0.4$ and $r/a = 0.5 - 0.6$. Short-wave turbulence related connected with ETG instability in core plasma of L-2M stellarator was investigated by the method of collective scattering of gyrotron radiation on the second harmonic of operating frequency. It is shown, that short-wave turbulence is close to the type of strong plasma turbulence. It is shown, that decreasing of energy lifetime power at doubling of the power of ECR heating correlates with growth of energy of short-wave noise. It is established, that cooling of plasma near last close surface by means of a sector limiter causes decreasing of falling energy lifetime and growth of short-wave turbulence in core plasma. Thus, it is experimentally established, that change of energy lifetime in L-2M stellarator correlates with the level of short-wave turbulence.

[1] G.M. Batanov, L.V. Kolik, M.I. Petelin et al. Plasma Physics Reports. 2003. 29. 1019.

EX/P5-40 · Multi-scale Nonlinear Turbulence Dynamics Studies

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Abstract: Critical gradient transport dynamics, breaking of gyro-Bohm transport scaling, transport bifurcation, and transport barrier formation are thought to be both generated and regulated by the nonlinear interaction between gradient-driven turbulence and larger meso-scaled structures such as zonal flows and streamers. Direct study of the underlying physics requires simultaneous spatiotemporal turbulence measurements that are only possible in smaller experiments; several such results are reported here. A dual turbulent cascade, wherein pressure fluctuations are nonlinearly transferred to large wavenumber while the fluctuating kinetic energy is transferred to small wavenumber, is found in collisional plasmas similar to those found in the edge region of fusion devices. The nonlinear kinetic energy transfer is manifested in configuration space by the formation of an azimuthally symmetric radially sheared flow sustained against damping by the turbulent Reynolds stress. The sheared zonal flow, turbulence amplitude and turbulence Reynolds stresses evolve on slow time scales indicating the existence of a complex dynamical system formed by the coupling of the shear flow dynamics and damping to the turbulence. The transfer of internal energy is manifested in configuration space by the shearing apart of turbulent structures immersed in the zonal flow – the essential process that leads to transport barrier formation and shear decorrelation. This process has been directly visualized by a fast imaging diagnostic. These results directly confirm zonal flow formation from turbulence, regulation of the turbulence by the zonal flow, and introduction of non-diffusive transport events, providing insight into the origins of critical gradient transport dynamics, gyro-Bohm breaking and transport barrier formation in magnetic fusion confinement devices.

EX/P5-41 · Turbulence and Transport in Simple Magnetized Toroidal Plasmas

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Abstract: Progress in understanding turbulence and related cross-field transport of fusion relevance is achieved in the simple magnetized plasmas of the TORPEX toroidal device, which provide a test bed for code benchmarking and theory validation. In TORPEX (R = 1 m, a = 0.2 m), a small vertical magnetic field B_z <4 mT is superposed on a toroidal magnetic field B_t < 100 mT to form open helicoidal magnetic field lines, which, similarly to Scrape-Off Layers of fusion devices, feature ∇B, and magnetic field curvature. Plasmas of different gases are produced by microwaves (2.45 GHz, P < 10 kW) with typical electron temperatures ∼5-15 eV and densities ranging from $\sim 3 \times 10^{16}$ m^{-3} for hydrogen to $\sim 3 \times 10^{17}$ m^{-3} for neon. Drift and interchange instabilities are observed and characterized in terms of dispersion relation, driving mechanisms, and non-linear development into turbulence and blobs. Density fluctuations associated with drift-interchange waves exhibit universal statistical properties. Their probability density functions (PDFs) are best fitted by a β distribution, with a unique quadratic polynomial relation linking their skewness and kurtosis. At large vertical fields, PDFs with positive skewness are associated with blobs, which form from radially elongated structures that are sheared off by the $E \times B$ flow. These structures are in turn generated by interchange waves that increase in amplitude and extend radially following an increase of the radial pressure gradient. Using an 86 electrostatic probe array, we investigate the scaling of the cross-field speed with blob size, connection length and gas pressure for different gases and compare the results with semi-analytical blob models. The transport of heat, particles and momentum associated with both the interchange wave and the blobs is quantified. The cross-field transport caused directly by instabilities constitutes a different mechanism from that associated with the blobs. We present also fluid simulations, which, together with an analytical theory, predict the existence of an improved confinement regime with some common features to tokamak observations. The accessibility of this regime on TORPEX is explored by varying a large number of experimental parameters.

EX/P5-42 · Comparison of Turbulence Measurement Results on the Wendelstein 7-AS Stellarator and the TEXTOR Tokamak

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Abstract: A considerable amount of evidence has been collected over the past decade which supports the view that heat and particle transport across the magnetic field in magnetic confinement fusion devices is caused by microturbulence of the plasma. The aim of this paper is to compare properties of density fluctuations measured on the Wendelstein 7-AS stellarator and the TEXTOR tokamak using the same diagnostic technique, notably Beam Emission Spectroscopy with the accelerated Li-beam diagnostic. The detailed comparison is made possible by the recent installation of a Li-beam diagnostic on TEXTOR optimised for turbulence measurements while the characterisation of density fluctuations in Wendelstein 7-AS has been previously published. Fluctuations in the Scrape-off Layer of TEXTOR and Wendelstein 7-AS appear to have quite similar characteristics: broadband with $10 - 100$ ms correlation time and $10 - 50\%$ RMS amplitude. In the core plasma the temporal and spatial structure of turbulence is also similar, although in the case of the older stellarator measurements their observation was marginal: about 10 ms correlation time and $< 1\%$ relative RMS amplitude. The spatial correlation is in the cm range in agreement with other core plasma measurements. The only missing element between the stellarator and tokamak observations is the appearance of transient profile flattening seen in the stellarator data which could not be found so far in TEXTOR. These phenomena were speculated to be caused by sudden poloidally localised transport events, for which streamers are a likely candidate. The paper will discuss properties of turbulence in Wendelstein 7-AS and TEXTOR measurements in detail.

EX/P5-43 · Steady-State Confinement of Anisotropic Hot Ion Plasma in the Gas Dynamic Trap

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Abstract: The paper summarizes recent results obtained in the gas dynamic trap experiment. Gas dynamic trap (GDT) is a mirror device where Maxwellian plasma component and fast anisotropic ions are confined in the axially symmetric magnetic field. The GDT NB heating system is capable of providing six focused $20-25$ keV hydrogen or deuterium beams. The total beam power delivered to the central cell plasma reached 4 MW in the 5 ms pulse. Two additional focused 25 keV beams provide up to 1.2 MW in the compact mirror cell attached to the GDT central cell at the end Comparison of the experimental data on global energy and particle balance with the results of the Monte-Carlo modeling of the plasma equilibrium parameters indicates that the two-component plasma in GDT reaches steady-state within 5ms shot. The characteristic plasma lifetimes are $4-5$ times shorter than the shot duration. In these experiments peripheral gas-puff near the end mirror is used to maintain the radial profile of main plasma during the NBI pulse. The peak density of anisotropic ions with the mean energy of 10 keV exceeded 4×10^{19} m^{-3} near their turning points, that is close to main plasma density in the reported experiments. Accordingly, in these experiments the maximal beta value was increased from 0.4, which was reported previously, to about 0.6. Electron temperature of plasma was also significantly increased from 100 eV up to 200 eV. The stability against MHD interchange modes was established using the set of biased radial limiters and sectioned end wall. The new results of experiments with the compact mirror cell attached to the GDT central cell are be also presented in the paper. In particular, micro-instability limits in extremely anisotropic hot ion plasmoid were studied..

EX/P6-1 · Filament Dynamics and Transport in the Scrape-Off-Layer of ASDEX Upgrade

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Abstract: Measurements at different experiments have shown that a few 10% of the energy loss from the confined plasma during an ELM cycle is deposited to non-divertor components. Usually, these components are not designed to receive high heat loads. The measurement and the extrapolation of these non-divertor loads to a next step fusion reactor are essential for the design of in-vessel components. The energy released from the pedestal plasma during an ELM is concentrated in field aligned helical structures, so called filaments, distributed around the torus with a quasi mode number in the order of 10. These filaments are rotating and moving radially. Theoretical modelling of the filament dynamics relies on different assumptions and results in contradictory predictions for the velocity and size scaling of such filaments. This paper reports on measurements of velocity, radial extent, and electron density of filaments in the far SOL of ASDEX Upgrade, as well as on the energy content of such filaments by a new filament probe. The probe carries a set of radially and poloidally separated Langmuir pins around a magnetic pick-up coil and can be moved towards the plasma by a magnetic drive. Radial and poloidal velocity of filaments are calculated from time delay measurements together with the amplitude and width of the corresponding ion saturation current signal. 466 filaments have been analysed. Several trends have been derived from a series of scatter plots. Most important, it has been shown that filaments with large radial extent move faster than smaller ones, and that filaments with higher density move faster than filaments with less density. From a comparison of filament size and density, we could show that the filaments lose density by parallel losses and broaden radially with time. In AUG, the filaments would be depleted after a distance of about 12 cm from the separatrix. The radial velocity of filaments does not seem to vary with time or distance from the separatrix, i.e., no acceleration could be identified from the data. The modelling of the magnetic signal requires the assumption of a rotating mode structure inside the separatrix otherwise the required currents are not consistent with the measured ion saturation currents. From the comparable decay length of heat load and particle flux a simple one point model for the ion energy and heat load is derived.

EX/P6-2 · Divertor Heat Loads in RMP ELM Controlled H-mode Plasmas on DIII-D

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Abstract: On DIII-D successful edge localized modes (ELMs) suppression with resonant magnetic perturbation (RMP) is always accompanied by splitting of the separatrix observable in the particle flux and under some conditions in the divertor heat flux patterns. In this work, we study the effects of $n = 3$ magnetic perturbation on the heat flux deposition patterns (including ELMs) at different levels of electron pedestal collisionalities ($\nu^* < 0.2$ and $\nu^* \sim 1$) and triangularity. High collisionality results are compared to TEXTOR findings with Dynamic Ergodic Divertor at similar collisionalities ($\nu^* \sim 0.9$). It has been found that the heat transport shows different reactions to the applied RMP depending on the plasma triangularity and collisionality of the plasma boundary. In the ITER-like discharges ELMs produce several helical striations on the target with width of the deposition pattern increasing linearly with the ELM size. Application of the RMP causes remaining ELMs (before they are completely suppressed) to lock to the externally induced magnetic structure, which holds true also for ELM-like bursts caused by D2 pellets.

EX/P6-3 · ELM Power Loadings and Control on MAST Using Resonant Magnetic Perturbations

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Abstract: Although it is generally agreed that the filament structures observed during ELMs are responsible for transporting energy and particles to the wall it is less clear what role they play in transporting particles and heat to the target near to the strike point. Studies on MAST have been performed to investigate if the filaments could be playing a role in both processes and what determines the fraction of ELM energy transported to the wall. Information on the temporal evolution and spatial structure of the target power distributions in the near SOL, have been used to produce a Monte Carlo model based on ion parallel transport. The consequences of such a model for the power loads to the divertor targets and first wall on ITER will be discussed. All current models, this one included, indicate that in order to ensure the lifetime of the divertor targets on ITER a mechanism to decrease the amount of energy released by an ELM, or to eliminate ELMs altogether is required. One such amelioration mechanism relies on perturbing the magnetic field in the edge plasma region, enhancing the transport of particles and keeping the edge pressure gradient below the critical value that would trigger an ELM. MAST is equipped with a set of ex-vessel error field correction coils, similar to the ones used successfully on JET for ELM mitigation experiments. Experiments on MAST, using an $n = 1$ configuration of the coils were performed, however, it was difficult to find an operational window where sufficient current could be applied without either causing a back transition H-L or locked mode. Experiments were also performed using an $n = 2$ configuration. In low collisionality discharges the ELM frequency was approximately doubled with the maximum available coil current of 15 kA$\times$turns. In these discharges the density was reduced by $\sim$10% while the electron temperature remained unaltered. Calculations of the perturbation to the plasma for 15 kA.t in these coils using the ERGOS code (vacuum magnetic modelling) predict a broad ergodised layer and complete ELM suppression may therefore have been expected. The difference between on and off-axis calls will be investigated further following the installation on MAST of a set of 12 in-vessel coils (6 upper and 6 lower), similar in layout to those used in DIII-D and experiments are planned to start in May 2008.

EX/P6-4 · Comparison of Small ELM Characteristics and Regimes in Alcator C-Mod, MAST, and NSTX

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Abstract: We report on a set of ITPA-endorsed experiments among the Alcator C-Mod, MAST and NSTX devices to compare the characteristics of small edge-localized modes (ELMs) and the conditions in which they can be obtained, with the goal of assessing their extrapolability to ITER. The urgency for development of small and no ELM scenarios has increased with the recent revision of the allowable ELM size in ITER to 1 MJ, representing about 0.3% of the 350 MJ plasma stored energy. In Alcator C-Mod, the Enhanced D_α (EDA) regime has been shown to have individual small ELMs at sufficiently high pedestal temperature and/or pedestal b. In MAST small ELMs have been observed in specific discharge scenarios and shapes. Finally in NSTX a small ELM regime, termed Type V ELMs, has been shown to have a wide operating window with unique ELM structure. A second small ELM regime was also found in NSTX. One straightforward conclusion from these studies is that small ELMs can indeed have different toroidal mode structure and operational windows, even within a single device. Thus there may be multiple scenarios by which small ELMs could be achieved naturally in ITER. A common observation from all of the devices is an apparent bped threshold for small ELM access, although the threshold value varied widely between the devices. Stability analysis of these ELMs is complicated by the fact that the profile relaxation following small ELMs is quite subtle. The status of simulations of these results with M3D will be presented.

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EX/P6-5 · Analysis of Pedestal Characteristics in JT-60U H-mode Plasmas Based on Monte-Carlo Neutral Transport Simulation

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Abstract: This paper describes the result of pedestal characteristics in H-mode plasmas of JT-60U based on neutral transport simulation using the DEGAS Monte-Carlo code. A three-dimensional (3-D) mesh structure including isolated dome, baffle and divertor plates is precisely implemented and the simulation space is extended from SOL to core region. The present mesh model for DEGAS simulation has been developed based on the previous one for the DEGAS2 code and a consistency with the mesh model of the 2-D fluid UEDGE code has been ensured. In this simulation, therefore, background plasma parameters and the intensity of particle flux on the divertor plates are determined from the UEDGE code. The dependence of plasma density on the neutral penetration and ionization area was investigated in two cases of H-mode discharge with gas puff and without gas puff. The simulation result showed that the increase of edge pedestal density causes a noticeable decrease of 1/e scale length in the neutral penetration. Consequently the neutral penetration decreases with the increase of the density-pedestal width, which leads to the localization of ionization area near edge transport barrier (ETB) region. The detail of 3-D structure of neutral transport is also being investigated from the viewpoint of the local effect of the gas puff.

EX/P6-6 · Non-inductive Plasma Current Start-up by EC and RF Power in the TST-2 Spherical Tokamak

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Abstract: Key issues in spherical tokamak (ST) research are plasma current start-up and formation of the ST configuration without the use of a central solenoid (ohmic coil). Successful start-up, ST formation and sustainment have been achieved by injecting RF power (usually in the EC frequency range) to a configuration with a toroidal field and a weak vertical field. However, the current generation mechanism is still not clearly understood. Non-inductive plasma current start-up experiments were performed in the TST-2 spherical tokamak using EC (2.45 GHz, up to 5 kW) and RF (21 MHz, up to 30 kW) power. In order to determine the mechanism experimentally, dependences on such parameters as EC wave polarization and vertical field configuration are studied. It was found that the sustained current after a current jump is proportional to the vertical field strength, but dependences on other parameters are very weak. On the other hand, the initial current ramp-up rate depends on several parameters such as the filling pressure, the injected EC power, the vertical field strength, the resonance position and the mirror ratio, and a scaling law was obtained for the first time. Low frequency RF power was injected to the low current EC plasma. A current jump was induced and ST configuration was sustained after the EC power was terminated. The sustained currents were similar to those sustained by the EC power. These results demonstrate that a low frequency RF power can be used as an important and flexible tool for starting up ST plasmas.

EX/P6-7 · Experimental and Computational Studies of MHD Relaxation Generated by Coaxial Helicity Injection in the HIST Spherical Torus Plasmas

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Abstract: Coaxial Helicity Injection (CHI) experiments in the Helicity Injected Spherical Torus (HIST) device have produced discharges with peak plasma currents up to 150 kA and successfully sustained the plasma current for much longer than a resistive decay time. The flow measurements using a Mach probe have shown that the intermittent generation of the flow is correlated to the fluctuation seen on the electron density and current signals. The repetitive merging of the magnetized plasma jet ejected from the helicity injector may be responsible for the current sustainment in these discharges. This CHI current drive mechanism is supported by the 3D-MHD numerical simulation. The acceleration of ion flow up to about 50 km/s has been observed during the transition phase from the spherical torus (ST) to the flipped-ST relaxed state. We have found that the current self-reversal process involves the non-linear growth of the $n = 1$ kink instability of the central column and the relevant magnetic reconnection flow.

EX/P6-8 · Non-Solenoidal Formation of Spherical Torus by ECH/ECCD in LATE

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Abstract: Start-up and formation of spherical torus (ST) without central solenoid (CS) is a key issue to realize compact tokamak reactors. Electron cyclotron heating and current drive (ECH/ECCD) is potentially attractive since both the breakdown and current start-up can be realized by the injection of microwave beam from a small launcher remote from the plasma. The LATE device has been exploring its feasibility. LATE has no CS but is equipped with two microwave sources at 2.45 GHz (30 kW, 2 sec) and 5 GHz (200 kW, 0.07 sec). They are injected obliquely to the toroidal field Bt for ECH/ECCD by electron Bernstein (EB) waves mode-converted via O-X-B process. Previously, the 3^{rd} ECR layer was located near the center of the vacuum vessel and the plasma current Ip was saturated at 15 kA as the vertical field B_v was increased. By increasing B_t as well as the microwave power so that the 2^{nd} ECR layer locates near the vessel center, I_p increases without saturation and reaches 20 kA at a large ramp-up rate up to 300 kA/s, suggesting improvement of wave coupling to the current-carrying fast electron tail. The plasma current is carried by energetic tail electrons. A measure of the tail momentum range C_f is obtained from magnetics. Initially, C_f increases as Ip increases, but becomes steady when $I_p > 15$ kA. This means that while the tail momentum range is limited by the orbit loss due to the outer shift of electron orbit from the flux surfaces, the shift becomes small at $I_p > 15$ kA and the tail velocity distribution is rather determined by the wave N// spectrum. During the fast I_p ramp at ~ 300 kA/s, the loop voltage at the plasma core is negative (-0.06 V). The tail energy range estimated from C_f at $I_p = 20$ kA is ~ 500 keV, which exceeds the runaway critical energy ~ 2 keV for the reverse voltage. The production of such high energy tail at the energy range far beyond the critical energy may be realized by the forward driving force on tail electrons via EC absorption of high N// EB waves. The detail of the velocity distribution of the fast electron tail is investigated by X-ray pulse height analysis for the 2.45 GHz 2 sec discharges with $I_p = 5.9$ kA. Comprehensive analysis of X-ray energy spectra detected at both the forward and backward line-of-sights shows that a uni-directional electron tail up to 250 keV is developed and carries the current.

EX/P6-9 · Physics Study of EC-Excited Current Generation via Current Jump in CPD

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Abstract: Non-inductive current generation mechanism in electron-cyclotron (EC) heated toroidal plasmas under steady toroidal and vertical fields are investigated in CPD (Compact Plasma-wall-interaction experimental Device). CPD is a small spherical tokamak (ST) device whose diameter and the height are both 1.2 m. As a result of the current generation and increase closed flux surface configurations are formed from the externally applied open field structures. Such change of the field topology is accompanied by the spontaneous rapid current increase called “current jump”. Three topics related to the current jump phenomena are described, 1) threshold power dependence on injected EC-wave mode, 2) topological change in the density profiles as well as the magnetic field configurations at the current jump, and 3) change in the characteristics of density and magnetic fluctuations at the current jump.

EX/P6-10 · Solenoid-free Plasma Start-up in NSTX Using Transient CHI

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Abstract: Experiments in NSTX have now unambiguously demonstrated the coupling of toroidal plasmas produced by the technique of Coaxial Helicity Injection (CHI) to inductive sustainment and ramp-up of the toroidal plasma current. This is an important step because an alternate method for plasma startup is essential for developing a fusion reactor based on the spherical torus concept. Elimination of the central solenoid would also allow greater flexibility in the choice of the aspect ratio in tokamak designs now being considered. The CHI technique has previously been studied in smaller experiments, such as the HIT-II device at the University of Washington. As demonstrated on the HIT-II machine, the method is compatible with tokamak and ST designs that use a superconducting poloidal coil system, as rapidly changing coil currents are not required for generating the initial closed flux equilibrium. CHI is implemented in NSTX by driving current along field lines that connect the inner and outer lower divertor plates. To understand and assess the full potential of the transient CHI method to NSTX and to future machines we have used the Tokamak Simulation Code (TSC) to simulate NSTX CHI discharges. This work is strongly linked to analysis of experimental data. In the simulations the plasma evolves in much the same way as observed experimentally for a similar current discharge. TSC, which is a time-dependent, free-boundary, predictive equilibrium and transport code, uses as input the NSTX vessel geometry and external circuit parameters. Generation of closed flux in TSC is as a result of an effective toroidal loop voltage induced by the CHI ejected poloidal flux that decreases as the injector current is reduced to zero. The code has been able to show consistency with earlier theory for the scaling of CHI produced current with the injector and toroidal fluxes. These results in conjunction with experimental work on HIT-II and NSTX, two machines with different sizes, suggest that the amount of injector current required to pull the injector flux into the vessel increases with the injector flux but decreases with the toroidal field indicating that the scaling to future machines with stronger toroidal field is favorable.

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EX/P6-11 · Current Profile Modification Influence on MHD and Non-Solenoidal Plasma Startup in the Pegasus Toroidal Experiment

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Abstract: The Pegasus Toroidal Experiment operates at A < 1.3 and has achieved record high normalized plasma current for a tokamak, $I_N > 12$. Operation at high I_N allows access to high β_t and requires $j(r)$ modification to avoid deleterious MHD. Very broad, stable current profiles are obtained when washer-stack current sources (plasma guns) are used to initiate non-inductive discharges via DC helicity injection. This startup technique is scalable and requires no modification of the vacuum vessel. Equilibrium reconstructions of gun discharges show high edge current ($l_i = 0.2$) and elevated q ($q_{min} > 6$), which allow access to the high I_N regime. Plasma gun discharges relax into a tokamak-like configuration with toroidally-averaged closed flux surfaces, large $n = 1$ activity and toroidal current amplification up to 3 times the vacuum windup. Maximum I_p is determined by helicity balance and up to 50 kA of toroidal current has been generated with this technique. Nonlinear 3D simulation with NIMROD shows that gun injection excites a line-tied kink that produces amplification of poloidal flux beyond the vacuum wind-up. Experimental evidence of flux amplification includes: reversal of the edge poloidal magnetic flux; increase of the toroidal plasma current over that of the vacuum geometric windup; plasma position subject to radial force balance; and persistence of the plasma current after gun shut-off. Coupling gun discharges to other current drive is straightforward. Gun-only plasmas which reach a maximum plasma current of 20 kA have been coupled to Ohmic drive applied at the time of the plasma gun turn-off and ramped up to 80 kA with $1 - 2$ V of loop voltage totaling < 10 mVs of Ohmic flux. Early TSC results qualitatively agree with experimental observations of gun plasma dynamics. Very low TF and high edge currents in Pegasus allow study of peeling-ballooning modes, even in the absence of transition to H-mode. Ideal MHD stability calculations indicate that Pegasus is susceptible to peeling mode instability at low beta. The visible onset of edge filaments believed to be caused by peeling modes coincides with magnetic oscillations in the $20 - 100$ kHz range with low to intermediate toroidal mode numbers. The amplitude of these oscillations falls off quickly with increasing radius. I_p ramp-down decreases the edge $j_\parallel$ and coincides with the suppression of these modes.

EX/P6-12 · Electron Cyclotron Resonance Heating Assisted Plasma Startup in the Tore Supra tokamak

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Abstract: In ITER and future fusion reactors, plasma startup will have to be achieved with a low in-vessel toroidal electric field ($\leq$0.3 V/m in ITER) due to the use of superconducting coils and the presence of thick toroidally continued in-vessel structures. The benefit of using additional power to assist plasma startup has been demonstrated and studied in several devices, however some uncertainties remains on the requirement on the ECRH pulse characteristics for ITER. This paper reports on further results obtained on ECRH assisted startup on Tore Supra ($R = 2.4$ m) at the fundamental EC resonance as it is foreseen to be used for ITER start-up. Tore Supra ECRH system is based on two 118 GHz gyrotrons ($\sim$ 600 kW). The power is injected into the plasma as Gaussian beams by an antenna located on the low field side which, using actively cooled mirrors inside the vacuum vessel, allows control of both poloidal and toroidal injection angles. Pre-ionization was obtained with one gyrotron in less than 1 ms at the fundamental resonance location. Successful plasma startups were obtained with EC assistance down to 0.15 V/m at 5 mPa deuterium filling pressure ($\sim$ 350 kW). The average stray magnetic stray field ($B_\perp$) in the vacuum vessel at the breakdown time is around 7 mT, due to the pre-magnetisation coils currents and the presence of the iron core. It was possible to initiate the discharge at up to 25 mPa deuterium pressure with 160 kW of ECRH (5 mPa maximum without). This significantly enhances the start-up reliability and limits the runaway electrons generation. After a disruption, the use of two gyrotrons ($\sim$ 600 kW) during 70 ms helped to survive the impurity burnthrough phase reducing the number of failed attempts compared to the case with lower power. The location of the resonance was varied from $R \sim 2.2$ m, near the center of the torus toward the inner wall, $R \sim 1.9$ m, and startup was achieved up to mid-radius, $R \sim 2$ m. Besides, the toroidal and poloidal injection angles were both varied, respectively from -30° (co-current) to 0° (perpendicular) and from -15° (aiming at the center of the torus) to $+2^\circ$ without any observed significant difference. A simple model is used to extrapolate these results to ITER. It shows that, assuming similar impurity content, 1 MW of ECRH power at fundamental resonance could be enough to reliably start-up ITER plasmas.

EX/P6-13 · Electron Bernstein Wave Heating via the Slow X-B Mode Conversion Process with Direct Launching from the High Field Side in LHD

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Abstract: Electron cyclotron heating (ECH) by electron Bernstein wave (EBW) excited by direct launching of the extraordinary (X-) mode from the high field side has been demonstrated for the first time in helical systems. In the large helical device (LHD), the X-mode can approach the upper hybrid resonance (UHR) layer directly through the ECR layer without installation of any additional mirror by oblique launching. Power absorption in the inner region occurred apart from the ECR layer with direct X-mode launching from the outer high field side of the ECR layer. It suggests EBW is excited and absorbed in the Doppler shifted ECR layer.

EX/P6-14 · Profile Control by Local ECRH in LHD

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Abstract: The specific and the important feature of electron cyclotron resonance heating (ECRH) in magnetic fusion experiments is the well defined deposition profile and its controllability. This feature is realized and utilized for the electron temperature profile control, investigating the transport properties and current profile control in LHD. Temporally and spatially high resolution diagnostics, motional Stark effect (MSE), heavy ion beam probe (HIBP) as well as Thomson scattering, electron cyclotron emission (ECE) are applied to such locally heated or current driven plasma in LHD. The relations between the ECRH induced flux or current and the formation of the potential and density or induced current profile structure are clearly shown experimentally for the first time.

EX/P6-15 · Effect of Magnetic Field Ripple on ECCD in Heliotron J

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Abstract: Electron cyclotron current drive (ECCD) experiment has been made in Heliotron J. A wide configuration scan shows that the EC driven current is strongly dependent on the magnetic ripple structure where the EC power is deposited. As the EC power is deposited on the deeper ripple bottom, the EC driven current flows in the direction determined by the Ohkawa effect, and the high energy tail is suppressed, indicating that the EC driven current is closely related to the population of trapped electrons. The dimensionless ECCD efficiency, $\zeta = e^3 n_e I_{EC} R/\epsilon_0^2 P_{EC} T_e$, is estimated by scanning the single pass EC absorbed power.

EX/P6-16 · Real Time Control of Plasmas and ECRH Systems on TCV

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Abstract: Developments in the real time control hardware on TCV paired with the flexibility of plasma shaping and ECRH actuators are opening up many opportunities to develop real time experiments, algorithms and methods for fusion applications. This paper presents an overview of recent real time control experiments on TCV, developments in the control hardware, algorithms and plans for the future. Recent experiments using the multi-launcher ECRH system have demonstrated the ability to control the plasma current in fully non-inductive (ECCD) plasma discharges using the ECRH power as the actuator and a PID controller. The ability to control the plasma elongation in real time was also demonstrated: under a constant magnetic quadrupole shaping field, varying the plasma current profile leads to a change in the plasma elongation. By heating off-axis with the ECRH system, the current profile is broadened leading to an elongation of the plasma. A flux observer was used to estimate the real time elongation, generate an elongation error signal and actuate the ECRH power to control the elongation. As the original TCV control system provided only linear feedback control of the coil currents (ie PID), a high performance multi-DSP controller, developed in collaboration with Association Euratom/IST has been installed. This firstly provides a more powerful, non-linear, procedural control ability and secondly introduces more channels for controlling the ECRH systems. The latest results from this new controller will be presented in the paper. A further multi-channel, real time controller is being developed to enable real time reconstructions, eg equilibrium, soft x-ray etc to be run. This will produce observables for processing by the DSP system, or directly control the actuators as necessary.

EX/P6-17 · Investigation of Electron Bernstein Wave (EBW) Coupling and its Critical Dependence on EBW Collisional Loss in High-Beta, H-Mode ST Plasmas

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Abstract: In conventional aspect-ratio tokamaks, electron cyclotron (EC) waves can provide local heating and current drive. However, the high-beta spherical tokamak (ST) plasma conditions cut off propagation of EC waves. The electron Bernstein wave (EBW) does not experience a density cutoff and is strongly absorbed and emitted at the EC harmonics, allowing EBWCD to be used in STs for current profile control to increase global stability, as well as possible NTM suppression. A critical challenge is to establish and maintain efficient EBW coupling in these high-beta, H-mode ST plasmas. EBW emission (EBE) diagnostics and EBE modelling have been employed on NSTX to study oblique EBW to O-mode (B-X-O) coupling and propagation. Initial EBE measurements in H-mode plasmas exhibited strong emission before the L-H transition, but the emission rapidly decayed after the transition. EBE simulations show that collisional damping of the EBW prior to mode conversion (MC) can significantly reduce the measured EBE for $T_e < 20$ eV, explaining the observations. Lithium evaporation was used to reduce EBW collisional damping near the MC layer. The evaporation rate was increased from 0 to 19 mg/min to reduce the electron density and increase the electron temperature outside the last closed flux surface (LCFS) in an H-mode plasma. With edge conditioning, an increase in T_e near the fundamental MC layer from 10 eV (no Li) to 20 eV (with Li) was observed. As a result the measured B-X-O transmission efficiency increased from $< 10\%$ (no Li) to 60% (with Li), consistent with EBE simulations. This work has significant implications for future ST devices that use EBWCD, such as an ST-CTF, if the O-X-B MC layer is shifted outside the plasma to a region where $T_e < 20$ eV. For these conditions, O-X-B heating and CD may become inefficient due to significant EBW collisional losses. This work has demonstrated that using edge conditioning to raise T_e outside the LCFS can significantly reduce these losses.

EX/P6-18 · Fundamental Investigation of Electron Bernstein Wave Heating and Current Drive at the WEGA Stellarator

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Abstract: Overdense plasma heating with electron Bernstein waves (EBWs) is of large importance for fusion, since ultra high density operation is a promising fusion scenario for stellarators and heliotrons. The EBW physics is also important for spherical tokamaks (ST). Here the plasma core is inaccessible for standard ECRH with electromagnetic waves. The EBW driven current could help to stabilize MHD-modes and to achieve steady state operation at high beta. The typical EC-frequency range in STs is $10 - 30$ GHz. This paper reports about EBW heating and current drive experiments at frequencies of 2.45 GHz and 28 GHz at the WEGA stellarator. WEGA is a classical five period $l = 2$ stellarator with a major radius of 0.72 m and an aspect ratio of 7. For the 2.45 GHz the mode conversion could be investigated in detail by hf-probes. The results were consistent with full wave simulations. The propagation of the EBWs was calculated by a 3D- ray-tracing code for 2.45 GHz and 28 GHz. The local resonant power deposition was measured by power modulation technique. Both the predicted EBW driven toroidal current and the current density profile could be clearly detected experimentally.

EX/P6-19 · Lower Hybrid and Electron Bernstein Wave Current Drive Experiments in MST

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Abstract: Inductive current profile modification in MST has been successful in reducing fluctuations and transport but is transient and radially non-localized. Current profile control with rf waves offers steady and more precise control. Studies of lower hybrid (LH) wave and electron Bernstein wave (EBW) injection are underway. This first application of LH waves to the high dielectric RFP presents challenges in rf physics, e.g., limited wave accessibility. The novel interdigital line antenna, chosen because of stringent vacuum vessel constraints, operates at 800 MHz and $n_{\parallel} \sim 7.5$ parameters chosen to drive current in the edge ($r/a \sim 0.8$) with strong single-pass absorption. Extensive antenna loading studies have been performed to validate the design up to the present source power limit of 225 kW with up to 125 kW being coupled to the plasma. Hard-x-ray emission with energies as high as 50 keV has been observed. The emission is spatially localized to the antenna location with a toroidal spread of about 60 degrees. This interesting toroidal localization of the emission that occurs in the dominantly poloidal magnetic field of the RFP could result from the formation of a localized current structure. Presently, a 250 kW system designed to heat electrons and drive current via the electron Bernstein wave is in operation on the MST reversed field pinch. The antenna is a grill of four half-height S-band waveguides with each arm powered by a separate, phase controlled traveling wave tube amplifier at 3.6 GHz. The X-mode polarization is being used to launch electromagnetic waves that mode convert to EBWs in the edge plasma. Coupling to the plasma (as measured by the reflected power ratio) is dependent on the relative phasing between adjacent waveguides. The total reflected power can be maintained near the 10% level. The antenna face is outfitted with a pair of triple Langmuir probes to measure local electron density; the density gradient at the upper hybrid resonance (typically within $1-2$ cm of the antenna) strongly influences coupling efficiency. The x-ray spectrum ($5-200$ keV) is monitored as a way to detect modification to the electron distribution as full transmitter power is approached. Recent experiments have noted toroidally localized soft-x-ray emission.

EX/P6-20 · Time Evolution of the Bootstrap Current Profile in LHD Plasmas

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Abstract: Since the rotational transform profile, which has important roles on the MHD stability and transport, is sensitive to the net plasma current, it is indispensable to estimate the toroidal current profile quantitatively. It is predicted theoretically for LHD plasmas that the configuration dependence of the bootstrap current is strong in the outward shifted plasmas and it affects the MHD equilibrium limit. In the previous studies, total amount of the bootstrap current was estimated experimentally in the LHD plasmas and compared with that obtained from the theoretical calculation. But there had previously been no experimental estimation of the radial profile of the bootstrap current. The Motional Stark Effect (MSE) measurement is now equipped in the LHD, and the rotational transform profile can be measured. Moreover, a numerical simulation code, TASK/EI+BSC, is available to calculate the time evolution of the bootstrap current profile consistently with the three-dimensional MHD equilibrium. In this paper, we compare the rotational transform profile obtained by the MSE measurement and that calculated by the numerical simulation of the toroidal current including the non-inductive bootstrap current and the inductive current, and the consistent result can be obtained.

EX/P6-21 · Lower Hybrid Heating and Current Drive on the Alcator C-Mod Tokamak

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Abstract: Lower Hybrid Current Drive (LHCD) is an attractive scheme because of its inherent high efficiency and its ability to drive current well off-axis in tokamak discharges. On the Alcator C-Mod tokamak, LHCD is being used to modify the current profile with the aim of obtaining advanced tokamak (AT) performance in plasmas with parameters (plasma density, magnetic field) similar to those that would be required on ITER. To date, power levels in excess of 1 MW at a frequency of 4.6 GHz have been coupled into a variety of plasmas. Experiments have established that LHCD on C-Mod behaves globally as predicted by theory. Bulk current drive efficiencies, $n_{20} I_{lh} R/P_{lh}$ ~0.25, inferred from magnetics and MSE are in line with theory. The entire data set of C-Mod LHCD discharges is found to behave consistent with Karney-Fisch theory including the finite electric field. Quantitative comparisons between local measurements, MSE, ECE and hard x-ray bremsstrahlung, and theory/simulation have been performed. Measurements and modelling using the GENRAY, TORIC-LH CQL3D and TSC-LSC codes have demonstrated the off-axis localization of the current drive, its magnitude and location dependence on the launched $n_{\|}$ spectrum, and the use of LHCD during the current ramp to save volt-seconds and delay the peaking of the current profile. Broadening of the x-ray emission profile during ICRF heating indicates that the current drive location can be controlled by the electron temperature, as expected. In addition, an alteration in the plasma toroidal rotation profile during LHCD has been observed with a significant rotation in the counter current direction being generated accompanied by peaking of the density and temperature profiles on a current diffusion time scale inside of the half radius where the LH absorption is taking place. Power modulation experiments have demonstrated that fast electron diffusion is not playing a role in the broader than expected x-ray emission profiles while comparisons between ray tracing and full wave deposition codes indicate that wave diffraction may explain the observations. LHCD in H-mode plasmas has been demonstrated with efficiency consistent with modelling.

EX/P6-22 · Effect of Gas Injection during LHCD Experiments in ITER-Relevant Coupling Conditions

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Abstract: LHCD experiments were carried out on the JET tokamak in ITER-relevant conditions which include: low ($\delta \sim 0.22$) and high ($\delta \sim 0.45$) triangularity ELMy plasmas, high safety factor ($q_{95} = 4.0-6.8$) and gaps between the separatrix and the antenna varying between 0.08 m and 0.16 m. Good coupling conditions, throughout the entire antenna, are found when the power injection is assisted by local D2 gas feed. The beneficial modification of the SOL in the flux tubes passing in front of the antenna is documented by measurements of a reciprocating Langmuir probe. This allowed to couple up to 3.1 MW, with a gap of 0.15 m, corresponding to a power density of 15 MW/m^2, but 22 MW/m^2 on half of the antenna which is close to the power density required at 3.7 GHz for ITER. modelling of the modification of the SOL by LH power absorption was performed with the EDGE2D fluid code. Flat j_{sat}/density profiles similar to experimental ones are obtained when a small fraction of the LH power is supposed to be absorbed in a ~ 20 mm thick layer in front of the antenna. The parallel heat flux ($F_{\|}$) on a plasma-facing component due to LH power dissipation in the SOL is computed from infra-red data. The strong dependence of $F_{\|}$ with the injected LH power gives further evidence of the LH-induced density modification. It is concluded that with optimized gas injection $F_{\|}$ in H mode should not exceed 5 MW/m^2 at maximum power density (25 MW/m^2). The effect of large gap and gas injection on LH current drive efficiency was studied in L-mode discharges for which the real-time control was used in order to keep

EX/P6-23 · Ion Cyclotron Antenna Impurity Production and Real Time Matching in Alcator C-Mod

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Abstract: Ion cyclotron range of frequency (ICRF) heating is expected to provide auxiliary heating for ITER and future fusion reactors where high Z metallic plasma facing components are being considered. In C-Mod, high performance plasmas require impurity control, and reliable RF requires minimizing the reflected power throughout a discharge. To control Mo impurities boronization is frequently applied where the lifetime is proportional to number of RF joules injected. We infer from post campaign surface analysis and experiments that important ICRF-derived Mo sources are linked to the active antenna and the source location can be distant from the antenna. Furthermore, the erosion rate of boron films by ICRF-derived processes is estimated to be $15 - 20$ nm/s. Assuming a similar net erosion rate for beryllium in ITER, the 1 cm beryllium thickness is eroded in ~ 1000 discharges (400 second discharges), a shortened lifetime. Emissive probe measurements of the local plasma potential at the plasma limiter confirm the presence of an enhanced plasma voltage when ICRF power is applied and the probe is magnetically linked to an active antenna. Measurements showed that the plasma potential voltage scaled with the square root of the RF power for L-mode, was enhanced in H-mode, and was present with both insulating and conducting limiter tiles. While the L-mode power dependence was expected, the increase in plasma potentials voltages with H-mode was significantly larger than expected. Furthermore, insulating tiles were expected to eliminate the enhanced potential; however, and it persisted. To transfer power to the plasma throughout a discharge, we have deployed a real-time matching system to minimize the VSWR. This system consists of two fast ferrite tuners (FFT) arranged in a double stub and triple stub configurations. We have recently demonstrated FFT usage at high power (1.5 MW coupled) while low VSWR was maintained over a wide range of plasma conditions. Furthermore, operation with ELMs shows that system matching was maintained throughout the ELM with low reflection coefficient. Results from experiments investigating impurity sources and real time matching in a wide range of plasma conditions will be presented.

EX/P6-24 · Experimental Study of Fast Wave Absorption Mechanisms in DIII-D in the Presence of Energetic Ions

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Abstract: Experiments on DIII-D have shown that fast Alfvén waves (FWs) can be damped on core electrons with minimal damping on fast ions at high harmonics, in contrast to early code predictions of strong fast ion absorption; these differences may result from known edge losses not yet included in the modelling. Fast wave current drive (FWCD) is an important option for central current drive in ITER. Since ITER and any future fusion reactor will have large fast ion populations (alphas and perhaps injected ion species), a validated quantitative model of the interaction between fast ions and FWs is needed to assess the application of FWs for central heating and current drive. DIII-D experiments have studied interaction between injected 80 keV deuterons and FWs at 60 MHz (up to 1.4 MW), 90 MHz (up to 2 MW), and at 116 MHz (up to 1.6 MW), at ion cyclotron harmonic numbers between 4 and 8. While strong absorption of FWs at the 4th harmonic of injected deuterons has been clearly observed, ion absorption under the same conditions at 6th and 8th harmonics is seen to be weak. Data on fast wave absorption on the injected beam is obtained from confinement analysis and equilibrium reconstruction, from fusion neutron rate measurements, and from direct measurements of the fast ion population by D_{alpha} spectroscopy. The experimental results also clearly indicate the importance of edge losses in cases in which the central absorption mechanisms are weak, so that quantitative modelling must take the edge losses into account. Such modelling has been carried out with combinations of ray-tracing or full-wave approaches for the wave fields in the plasma and with Monte Carlo or bounce-averaged Fokker-Planck solvers to compute the self-consistent distribution functions. Edge loss models are being incorporated into both field-solving approaches. Core absorption mechanisms will be much stronger in ITER, so that edge losses should be much less significant than in DIII-D, but the competition between core damping mechanisms will be similar. Hence, establishment of a quantitative model of the partitioning of FW power among competing absorption processes is important in both cases.

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EX/P6-25 · Spectral Effects on Fast Wave Core Heating and Current Drive

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Abstract: RF heating and current drive is an essential component in most magnetic fusion devices, including ITER. Recent results obtained with high harmonic fast wave (HHFW) heating and current drive (CD) on NSTX strongly support the hypothesis that the onset of perpendicular fast wave propagation right at or very near the launcher is a primary cause for reduced core heating efficiency at long wavelengths [1], observed in ICRF heating experiments in numerous tokamaks. A dramatic increase in core heating efficiency was achieved in NSTX L-mode helium majority plasmas when the onset for perpendicular wave propagation was moved away from the antenna and nearby vessel structures. These results indicate that careful tailoring of the edge density profiles in ITER should be considered to limit RF power losses to the antenna and plasma facing materials. The location of HHFW CD measured by MSE [2], obtained with reduced edge losses, is in reasonable agreement with predictions from AORSA [3] and TORIC [4] full wave simulations. This paper presents detailed experimental observations and compares them to predictions with a range of advanced state-of-the-art rf simulations.

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[1] J. Hosea et al, to be published in Phys. Plasmas May 2008. [2] F. M. Levinton et al, Phys. Plasmas 14, 056119 (2007). [3] E. F. Jaeger et al, Phys. Plasmas 8, 1573 (2001). [4] M. Brambilla, PPCF 44, 2423 (2002).

EX/P6-26 · Current Profile Control Studies on MAST

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Abstract: The high neoclassical resistivity observed in spherical tokamaks (STs), and consequent faster current penetration rate, results in rapid approach to low values of core safety factor. This provides specific challenges for initiating $q > 1$ regimes, desirable for long pulse or steady-state spherical devices. The extent to which the q-profile may be modified during the current ramp-up phase has been studied experimentally. Three different parameters were investigated separately: the current and density ramp rates and the neutral beam heating (NBH) start-time. The TRANSP code was used to model the current penetration in these experiments. MHD signatures of Alfvén cascades and sawteeth were used to benchmark the modelling results. The most significant result was seen when early NBH was applied which caused reversed magnetic shear with the most pronounced effect seen with the fast current ramp-rate. TRANSP simulations show that the NBH peaks the electron temperature profile, consequently lowering the current diffusion rate into the core, and causing the current to "pile-up" off-axis. To achieve a steady state scenario in STs with $q > 1$ both high efficiency CD and current profile control is required which can potentially be provided by off axis Neutral Beam Current Drive (NBCD). In recent MAST experiments, off-axis heating and NBCD has been studied in vertically displaced Single Null Divertor (SND) plasmas. Experiments have benefited from the ongoing NBI upgrade to JET-style PINI sources allowing the studies to be extended to higher power and duration (up to 3.8 MW, 0.5 s). Results indicate that broadening the fast ion deposition profile by off axis NB injection helps to avoid harmful plasma instabilities and significantly extends the operational window of MAST. Long pulse (> 0.65 s) H-mode plasmas are achieved with plasma duration limited only by present machine and NBI engineering limits. In order to match the experimentally observed neutron rate and stored energy a low level of anomalous fast ion diffusion ($D_b \sim 0.5$ m^{2}s^{-1}) is required. The introduction of the fast ion diffusion broadens the NBCD profile and degrades the relative contribution of NB driven current from $\sim$40% to $\sim$30%. Direct measurements of the current profile are highly desirable; a multi-chord MSE diagnostic is currently being installed on MAST to confirm the off axis location of the NBI driven current.

EX/P6-27 · Parametric Decay Instability during High Harmonic Fast Wave Heating Experiments on the TST-2 Spherical Tokamak

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Abstract: Spherical Tokamaks (STs) are attractive because of their superior stability at high beta. The high harmonic fast wave (HHFW) is a promising candidate for heating ST plasmas, because it has good accessibility to the core even in high β plasmas with high dielectric constants, and can be absorbed efficiently by electrons. In HHFW experiments on NSTX and TST-2, degradation of heating efficiency was observed when parametric decay instability (PDI) occurred. Suppression of PDI is required to make HHFW a reliable heating and current drive tool. In order to understand PDI, measurements were made using electrostatic and electromagnetic probes, reflectometry, and fast optical diagnostics, and compared with the results of a full-wave calculation. The results of these measurements are consistent with the HHFW pump wave decaying into the ion Bernstein wave (IBW) and the ion-cyclotron quasi-mode (ICQM). The IBW power varies approximately quadratically with the local pump wave power, which becomes smaller as absorption of the pump wave by the plasma increases. In addition, a new decay mode involving a sub-harmonic of the ion-cyclotron frequency was discovered.

EX/P6-28 · Velocity Distribution of Fast Ions Generated by ICRF Heating in Heliotron J

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Abstract: Fast ion velocity distribution is investigated using ICRF minority heating in Heliotron J with special emphasis on the effect of the toroidal ripple of magnetic field strength. The effect of the magnetic configuration on the fast ion confinement is one of the most important issues in helical devices. Here, the pitch angle dependence of energy spectra for high bumpy case is measured for the first time and compared with the medium bumpy case, then, the fast ions up to 34 keV are observed during ICRF heating in Heliotron J. The configurations used in this study are as follows; the bumpiness B_{43}/B_{00}, where B_{04} is the bumpy component and B_{00} is the averaged magnetic field strength) are 0.15 (high) and 0.06 (medium) at the normalized radius of 0.67. The configuration of $B_{04}/B_{00} = 0.06$ corresponds to the standard configuration in Heliotron J. In both cases, the higher energy flux is measured near 120° in pitch angle although the ions are considered to be accelerated in the perpendicular direction by ICRF heating. To understand experimental results, Monte Carlo calculation is performed. The numerical model consists of orbit tracing, Coulomb collisions and acceleration by the ICRF heating. Minority protons are regarded as test particles and the heating is simulated by the velocity kick in the perpendicular direction in velocity space when ions cross the cyclotron layer. The calculation results using Monte Carlo method represents that the accelerated ion distribution has its peak in the range between 20° and 30° from the perpendicular direction. This result is considered to be caused mainly due to the existence of the loss region around the perpendicular direction.

EX/P6-29 · Progress towards RF Heated Steady-State Plasma Operations on LHD by Employing ICRF Heating Methods and Improved Divertor Plates

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Abstract: The steady-state plasma discharge experiment was started at the third campaign of 1999 on Large Helical Device (LHD). Many experimental data have been obtained up to the 9th campaign of 2006 before the last IAEA conference. The long pulse plasma operation can be easily carried out on the current-less LHD using RF power, i.e., ion cyclotron range of frequency (ICRF) heating of 2 MW and electron cyclotron heating (ECH) of a few hundred kW. The world record was achieved in the duration time and the input heating energy (a product of plasma duration time and heating power) employing the magnetic axis sweeping to disperse the plasma heat load. Progress towards the steady-state plasma discharge data after the last IAEA conference is reported. It was found that the pulse-length so far achieved was decreased with the RF heating power. However it exceeded half hour and reached one hour when the RF heating power was smaller than the critical value, i.e., 0.7 MW. A empirical relation between the pulse-length and the RF heating power can be introduced using the temperature increase in divertor plates. The plasma was collapsed by the metal impurity penetrating at the end of the long-pulse plasma discharge. This phenomenon is closely related to the temperature on divertor plates. After the last conference were replaces to ones with more improved heat-removal capability some divertor plates, in which the temperature increase was observed to be higher. The plasma pulse-length was extended beyond the empirical relation: The maximum pulse length was achieved at 800 s in the plasma with 1 MW RF heating. Fe-pellets coated by C were injected to examine how much size of Fe/C sphere collapsed plasmas. It was found that the Fe sphere with diameter of 230 micron meter was a critical size for that. We found many thin flakes consisting of Fe and C layers with thickness of a few micron-meter on the divertor plates. When the flake of a few mm^2 penetrates to the RF heated plasma, it is thought that the plasma should be collapsed.

EX/P6-30 · High Harmonic Fast Wave Heating and Beam Driven Ion Cyclotron Emission Behavior in the LHD

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Abstract: The electron heating with higher harmonic fast wave (HHFW) has been considered to be one of the useful approaches to heat electrons in the high density and high β regime, and initial direct electron heating with weak single-pass damping is achieved by the use of HHFW in the large helical device (LHD). ECH is worked as more effective than bulk electron heating and a substantial electron heating is observed in the central region of the plasma due to the synergy effect of ECH and HHFW. The heating efficiency of HHFW estimated by energy balance equations at break-in-slope using plasma stored energy which is measured by diamagnetic current. In these discharges, the heating efficiency of HHFW increases up to 0.55 as the electron density increases due to multi-pass damping. On the other hand, ion cyclotron emission (ICE), which produces fast wave of ion cyclotron range of frequencies (ICRF) and damping mechanism is similar to HHFW with weak single-pass damping, is observed in high ion temperature discharges. By connecting the ICRF heating antenna to the detector instead of the ICRF power supply, ICE is clearly detected as magnetic fluctuations at the resonance frequencies during high ion temperature discharge. The ICE intensity signal rapidly increases associated with the increase of central ion temperature which is measured with charge exchange spectroscopy. The ICE intensity has strong correlation to an ion temperature and it is interesting that the harmonic ICE signals strongly grow up rather than fundamental ICE signal. Since the radiation pattern in the direction of magnetic field is different in each ICE harmonic numbers for the ICE calculation in tokamak, this experimental result may indicate that the excited toroidal location of ICE is changed as ion temperature is increased in the LHD.

EX/P6-31 · ICRF Antenna Operation with Full W-wall in ASDEX Upgrade

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Abstract: For ICRF (Ion Cyclotron Range of Frequencies) system to remain a good candidate for heating and current drive system in a fusion reactor, it is important to make the use of ICRF power more compatible with high-Z plasma facing components (PFCs). ASDEX Upgrade (AUG) with its W first wall and ICRF system allows to study ways to do this. A significant improvement of the ICRF operation with W wall can be achieved by forcing low temperature conditions at the plasma edge. This results in lower effective sputtering rates at PFCs. An increase of the clearance between the antenna and the plasma as well as an additional gas puff is used to decrease the edge temperature. These conditions are very similar to those ITER ICRF antenna will operate at. However, an additional improvement is required for further reduction of the high-Z impurity sputtering during ICRF in the present and the future devices. The progress in theoretical modelling of ICRF antenna near-fields shows that the RF voltages along the magnetic field lines may to a large extent originate not from antenna straps and their RF flux, but rather from parasitic RF currents on the antenna box. Experimental results in AUG corroborating this picture are described. For future antenna design, the calculations show that a reduction of the antenna box contribution can be achieved for an antenna with a large number of toroidally distributed straps and with an extension of the antenna box parallel to the magnetic field.

EX/P6-32 · Suprathermal Electron Beams and Large Sheath Potentials Generated by RF-antennae in the Scrape-off Layer of Tore Supra

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Abstract: In Tore Supra detailed radial-poloidal mapping of antenna-SOL interaction zones are made in a single shot using a retarding field analyzer (RFA) or Mach probe. Up to 15 reciprocations at a rate of 1 Hz have been made in a single full power discharge. Real time feedback on edge safety factor is used to vary the magnetic connection between the probe and the antennae. The probe position itself is controlled by feedback on fast magnetic reconstruction to guarantee safe, reliable, and reproducible operation. A retarding field analyzer was used during LH current drive experiments to provide direct measurements of the particle and power fluxes of suprathermal electrons emanating from the region in front of the LH grill. When one of the active wave-guide rows is magnetically connected to the RFA, a strong particle flux due to suprathermal electrons is observed. A fraction of the electrons have energies greater than 1000 eV. Theory predicts that waves with high refractive index should be totally absorbed within at most 5 mm from the grill, and that the electron current should be stationary in time. In reality, electron current is measured at least $4-5$ cm in front of the grill, and it exhibits a highly intermittent temporal evolution with a characteristic burst rate in the 10 kHz range. A Mach probe was used to map the SOL plasma parameters on flux tubes connected to both the standard and the new ITER-like ICRH antennae. A 1 cm layer of large positive floating potential (about one order of magnitude higher than the local electron temperature) is observed on flux tubes that graze the leading edges of the lateral protection limiters. Preliminary measurements of the true sheath potential using the RFA show that it is slightly more positive than the floating potential by around $2-3$ times the local electron temperature. Large floating potentials are also observed on Langmuir probes at specific toroidal positions on the high-field side of the bottom toroidal limiter, in good correlation with the magnetic connection map. The presence of long, positively biased flux tubes causes $E\times B$ convection that transports plasma radially and dumps it on the antenna structure, leading to strong heating and impurity release. The measured floating potential depends little on antenna power or phasing, but decreases strongly as the edge density is increased.

EX/P6-33 · Heating Optimization Studies at JET in Support of ITER

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Abstract: Due to the large gap ($\sim$ 20 cm) between the plasma separatrix and the antenna straps, it will be challenging to couple the foreseen 20 MW ICRF power to ITER plasmas. A method to enhance coupling is needed for ITER, even in the case of an ELM resilient ICRF antenna system, as also in that case the coupling in between ELMs will be low. Applying D_2 gas puffing in the plasma edge in H-Mode plasmas we found: (i) increase in coupling by a factor of $\sim$ 6 compared to similar plasmas without gas puff up to $\sim$19 cm distance between plasma and straps, with up to 8 MW of ICRF power coupled so far under such conditions; (ii) the influence of ELMs on the antenna coupling is reduced with increasing ROG. Applying D NBI on top of D majority ICRF heating at the fundamental cyclotron resonance frequency increased the neutron rate $\sim$ 1000 times, turning a very poor heating scheme into a more viable one. Not only the lower collisionality of the pre-heated plasma but mostly the Doppler-shifted ion-cyclotron resonance absorption of the fast beam ions is responsible for this. Simulations using a combined full wave solver / quasi-linear Fokker-Planck code (taking into account the influence of non-Maxwellian distributions of the ICRF heated particles and the injected beam ions in the wave equation) show that maximum ICRF power absorption was found at $\sim$ 0.5 m off-axis, to the high field side. modelling confirms that the fast D particles, which represent only $5-10\%$ of the total plasma particles, are responsible for the absorption of as much as $30-40\%$ of the ICRF power. These results show that absorption of the ICRF waves can take place even in scenarios (e.g second harmonic T heating in ITER) where the D cyclotron layer is near the plasma edge or outside the plasma. Anomalous behaviour of the beam ions when using off-axis NBI was reported earlier on ASDEX Upgrade, suggesting the possibility of a reduction of the current drive capabilities of the NBI system on ITER. A detailed analysis of JET experiments which used injection of short on- and off-axis tritium beam blips ($\sim$ 300 ms), have led to the following conclusions: (i) at low to intermediate plasma densities, the behaviour of fast particles is consistent with neoclassical theory; (ii) at high densities, fast particle losses are anomalous. These results show that for high density plasmas on ITER anomalous losses for fast ions could occur.

EX/P8-1 · Oscillatory Zonal Flows Driven by Interaction between Energetic Ions and Fishbone-like Instability in CHS

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Abstract: The turbulence-driven zonal flows are one of the most important subjects for realizing a nuclear fusion device. Besides, the behavior of energetic particles is another important issue since it is anticipated that the fusion-generated alpha-particles in a burning plasma should drive MHD instabilities, such as TAEs, Energetic Particle Mode EPMs, and interact the instabilities to lose themselves out from the plasma and to degrade the plasma performance. On the other hand, it has been clarified in the H-mode studies that the energetic particle loss could be the major cause to drive the sheared radial electric field to stabilize the plasma turbulence and create the transport barriers. The internal structural measurements of electrostatic potential and density, using twin HIBPs, have been performed in CHS to elucidate nonlinear evolution of fishbone-like instability (the mode numbers are $m/n = 2/1$), called here CHS fishbone that is MHD cyclic oscillation (with a period of $\sim$3 ms) driven by the interaction with energetic ions. This paper presents the finding of a new kind of zonal flows developing during a cycle of the CHS fishbone by the interaction between the energetic ions and MHD instabilities in the electrostatic potential measurements using twin HIBPs. The measurements also allow us to infer the degree of the distortion of the magnetic field flux surface from the $m = 2$ perturbation of potential, the change of the density gradient that could reflect the energetic ion fraction, and their mutual relationship during a cycle of the CHS fishbone. The observation also provides a quite reasonable view of the nonlinear evolution scenario of beam-driven mode, i.e., the local increase in the energetic ion gradient grows MHD instabilities to lose the energetic ions. Then the decreasing energetic ion gradient may stabilize the instability. The finding of the new kind of zonal flows should be emphasized since the resultant shear of electric field in a future burning plasma could reach a sufficiently large level to reduce the turbulence transport. This leads us to a simple expectation that the alpha-particle loss induced by the magnetic field distortion during alpha-particle driven MHD instabilities in a burning plasma may cause a sudden change of the plasma structure by resultant structured electric field.

EX/P8-2 · Study of Ion Cyclotron Emissions due to DD Fusion Product Ions on JT-60U

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Abstract: Ion cyclotron emissions (ICEs) due to deuterium fusion product (FP) ^{3}He and T-ions on JT-60U are described. The electromagnetic waves are measured by using ion cyclotron range of frequency antennas. The excitation of slow Alfvén waves is elucidated for the first time as the mechanism for ICE due to T-ions, ICE(T). The excitation of magneto-acoustic waves for ICEs due to ^{3}He-ions, ICE(^{3}He), is consistent with those in the previous JET and TFTR experiments. The toroidal wave numbers are evaluated from the phase differences between two antenna elements arrayed in the toroidal direction. ICE(T) with lower frequency has larger wave numbers than ICE(^{3}He). The density dependence of the excitation of ICE(^{3}He) is measured. The precise behaviors of ICEs due to FP ions on JT-60U are studied.

EX/P8-3 · Radial Profile and Confinement of Energetic Particles during NBI and ICRF Heating in LHD

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Abstract: The energetic ion distributions and confinements are studied in the LHD plasma applying a newly developed radial profile measurement system FICXS (Fast Ion Charge Exchange Spectroscopy) based on the Doppler shifted Balmer-alpha (H_α) emission from the energetic ions charge-exchanged with injected neutral beams. A clear difference is found in the H_α emission spectrum changing the magnetic configurations from the neoclassical optimized ($R_{ax} = 3.53$ m) to the classical heliotron ($R_{ax} > 3.75$ m) configuration. The radial profiles of energetic ion distributions during ICRF heating are also investigated by the pellet charge exchange (PCX) measurement. These results are compared with the GNET simulations, in which the drift kinetic equation is solved in 5-D phase space, and show relatively good agreements. These are the first experimental studies of the radial profiles of energetic ions that make clear the energetic ion distributions and, also, the effect of the magnetic configuration optimization on the energetic ion confinement in helical systems.

EX/P8-4 · Alfvén Eigenmodes and Geodesic Acoustic Modes Driven by Energetic Ions in an LHD Plasma with Non-monotonic Rotational Transform Profile

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Abstract: The reversed shear Alfvén eigenmode (RSAE) or Alfvén cascade (AC) attracts much attention, because it is excited in a reversed magnetic shear (RS-) tokamak plasma where an internal transport barrier (ITB) is easily formed. The RSAE frequency is swept upward from a certain minimum frequency when the minimum of the safety factor q_{min} passes integer values (3, 2 and so on). In the Large Helical Device (LHD), production of a RS-helical plasma with non-monotonic rotational transform profile was attempted by using off-axis counter neutral beam current drive. In the configuration of $R_{ax} = 3.75$ m (R_{ax}: the magnetic axis position in the vacuum field), more than 100 kA was driven by counter NB of ~ 4 MW at the toroidal field $B_t = 1.3$ T. The line averaged electron density was kept relatively low (less than 1.0×10^{19} m^{-3}). In thus produced plasma, realization of non-monotonic rotational transform profile was confirmed by a motional stark effect spectrometer. In the plasma, $n = 1$ RSAE (n: toroidal mode number) and the geodesic acoustic mode (GAM) with $n = 0$ were clearly excited by energetic ions. The frequency of $n = 1$ RSAE first chirps downward and then chirps upward from the minimum ($\sim$ 18 kHz) which is close to the GAM frequency estimated including beam ion pressure. The observed frequency of the fundamental RSAE with largest amplitude is explained very well by shear Alfvén spectra calculated from the experimentally data. This fact also indicates the realization of RS-configuration in LHD. Other $n = 1$ chirping modes are excited through nonlinear coupling between the fundamental RSAE and GAM. These modes are also detected by a microwave interferometer, having a lot of satellites of the RSAE and GAM extended up to 200 kHz. Electron cyclotron emission (ECE) polychromator has clarified that these RSAE and GAM are localized at $r/a \sim 0.4$ to 0.6 near the minimum of the rotational transform ($\iota_{min} = 1/q_{max}$). Large plasma potential fluctuations of GAM up to ~ 0.6 kV have been detected by a heavy ion beam probe. When the ι_{min} passes 1/3, the central ion temperature T_{io} measured by soft X-ray crystal spectrometer starts to increase continuously for ~ 0.3 s much longer than the global energy confinement time, while the central electron temperature has a small drop. This T_{io}-rise may link to an ITB formation.

EX/P8-5 · Fast-ion Transport during Repetitive Burst Phenomena of Toroidal Alfvén Eigenmodes in the Large Helical Device

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Abstract: Alfvén instabilities induced fast-ion losses have been directly observed for the first time by a newly developed scintillator lost ion probe (SLIP) in the Large Helical Device (LHD). The SLIP can measure the pitch angle and gyro radius of escaped fast ions toward loss region. Neutral beam driven Alfvén Eigenmodes (AEs) are excited under the reactor relevant conditions: the ratio of fast ion and Alfvén velocities is more than $0.3 \sim 4.0$. The β value for fast ions is considered roughly to be $\sim 10\%$. During repetitive Toroidal Alfvén Eigenmode (TAE) bursts, synchronized fast ion losses are observed by SLIP. From the orbit calculation the measured fast ion with pitch angle of 130° and beam energy of 150 keV surely pass through the locations of TAE gaps. The orbit analysis found that the observed fast ions interact strongly with the excited TAEs. This result becomes the first experimental evidence of radial transport of fast ions predicted theoretically during TAE activities. In addition, from the correlation between stored energy degradation and fast-ion loss rate, it is found that fast-ion losses induced by TAE activities with low toroidal mode numbers categorize two phenomena with and without fast- ion loss enhancements, which indicate the fast-ion redistribution and loss.

EX/P8-6 · Characterization of Stable and Unstable Alfvén Eigenmodes in Alcator C-Mod

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Abstract: Experiments designed to characterize both stable and unstable Alfvén eigenmodes are a key part of the Alcator C-Mod physics program. Stable intermediate toroidal mode number Alfvén eigenmodes, which are expected to be the most unstable in ITER, are excited with a set of active MHD antennas and their damping rates are measured as a function of plasma parameters. This is part of an international effort within the ITPA, which includes similar experiments on JET, JT-60U, and MAST. Alcator C-Mod provides results at the same toroidal field and density as ITER in the presence of an ICRF generated fast ion tail, while JET and JT-60U provide results in large high temperature plasmas and MAST will provide results on aspect ratio and beta. Unstable toroidal Alfvén eigenmodes (TAEs) are also produced with fast ions generated by hydrogen minority ICRF heating and with fast electrons generated with Lower Hybrid Current Drive (LHCD). The fast ion driven TAEs are observed to be more unstable in steady H-mode discharges than in lower density L-mode discharges with the same ICRF power. The stability properties have been analyzed with the NOVA-K code. The fast electron driven Alfvén eigenmodes with LHCD have only been observed very early in the current rise phase of the discharge at very low densities. These modes, nonetheless, provide a measure of the early evolution of the resonant q and help to constrain the fast electron energy distribution in comparison with hard x ray energy measurements.

EX/P8-7 · Fast Particle Instabilities in MAST

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Abstract: In MAST the population of fast ions created by the neutral beam injection system is super-Alfvénic and both the normalised fast ion pressure and the ratio of fast ion energy content to thermal energy content are in excess of the values expected in DEMO, thus allowing interpolation of the parameter space. MAST has recently installed a 12-coil TAE excitation system capable of measuring the spectra of stable modes in the system. An analysis of the measured plasma response to perturbations applied using a subset of up to three coils has shown modes in the toroidal and ellipticity induced Alfvén eigenmode frequency ranges to have damping rates comparable to the values found on other devices such as JET. Analysis tools have been developed to interpret the orthogonal magnetic field measurements to identify the polarisation of the perturbations observed. This has led to modes observed at around the on-axis cyclotron frequency being identified as compressional Alfvén eigenmodes. Their toroidal mode numbers, n, have been measured to lie in the range $4 < |n| < 10$ and they have been simultaneously observed to propagate both co and counter to the plasma current. These modes usually appear separated in discrete frequency groupings, with the frequency splitting occurring on up to three frequency scales. Nonlinear frequency sweeping of the modes has also been observed in accord with hole-clump formation. 1-D and 2-D models have been developed to further investigate these modes. They indicate that the modes are localised at approximately half the minor radius and why negative toroidal mode numbers can be driven by the beam ion distribution via the Doppler-shifted cyclotron resonance. Recent results on MAST in neutral beam heated plasmas have also observed rapid downwards sweeping $n = 1$ chirping modes which start at ∼100 kHz and gradually decrease in frequency to around ∼20 kHz before smoothly transitioning into a long-lived mode which appears to be stationary in the plasma rest frame. An analysis of the soft X-ray data indicates that these modes have an internal kink-like mode structure. Charge exchange data indicates a monotonic decrease in plasma rotation over the duration of the mode, whilst bolometer data suggests that these modes are responsible for the loss of fast ions from the plasma.

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EX/P8-8 · Recent Progress in Fast-Ion Physics on JET

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Abstract: In a burning plasma there will be several different groups of fast ions: alpha-particles, deuterium NBI and ICRH-accelerated ions in the MeV range. It is therefore important to measure and understand the temporal and spatial evolution of the separate fast ion populations. In this regard, JET is exceptionally well equipped for such studies using gamma-ray diagnostics, NPA and two new lost ion diagnostics: a thin foil Faraday Cup array and a Scintillator Probe. This paper presents recent results on fast-ion studies in JET. A synergy of the unique diagnostic set was utilised in JET for studies of the response of fast ions to MHD modes, toroidal field (TF) ripple and fast ^{3}He ions behaviour in shear-reversed plasmas. The temporal evolution of pitch-angles and gyro-radii of lost ions during a sawtooth crash was studied. Gamma-spectrometry indicated the presence of fast protons and deuterons accelerated by ICRH. During the crash, the majority of the loss was close to the trapped-passing boundary. After the crash, a relaxation to the initial pre-crash state was observed. The temporal evolution of the fast ions during sawtooth crashes was further investigated using the 2D gamma-camera in JET discharges with ^{3}He minority ICRH. Tomographic reconstruction of the gamma-emission produced by the ^{3}He distribution shows a significant re-distribution of the fast ions with a cooling down effect and a change in the topology of the fast ion orbits from potato to banana type. A study of various plasma scenarios had shown that a significant redistribution of the fast ions happens during the change in q-profile from strongly shear-reversed to monotonic. It was found that significant changes in the losses of ICRH accelerated protons are associated with confinement transitions in plasmas. After an L-H transition, an abrupt decrease in the ICRH proton losses was observed. In plasmas with an internal transport barrier, the loss of ICRH-accelerated ions increases by an order of magnitude as the barrier forms. Further results concerning fast ion losses were obtained during the JET experiments in which the magnitude of the TF ripple was varied. The ripple losses of fusion products in high triangularity plasmas were determined to be different than the classic prompt losses, and in good agreement with the ASCOT code modelling.

EX/P8-9 · Observation of Fast Particles Modes in Tore-Supra

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Abstract: MHD modes are known to be destabilised by energetic particles (EP). This is a potential threat for plasma performances and particularly critical in burning plasmas. On Tore-Supra, suprathermal electrons are generated by Lower Hybrid (LH) and fast ions by Ion Cyclotron Resonance Heating (ICRH). Different EP-MHD modes can be excited from few kHz up to 200 kHz. In LHCD plasmas, electron fishbones have been observed below 20 kHz with ECE and reflectometry diagnostics. Two classes of modes have been identified. The first class is characterized by a frequency range $8 - 10$ kHz, consistent with an estimation of the precession frequency for barely trapped fast electrons. Hence these modes have been identified as precessional electron fishbones. The second category of modes, whose frequency is around $2-4$ kHz, is likely a diamagnetic electron fishbone since the frequency follows the thermal electron temperature and not the suprathermal energy. In presence of precessional fishbones the electron temperature exhibits an oscillating behavior ($\Delta T_e/T_e \sim 1$) which seems strongly related to the discontinuous evolution of the mode frequency. For large enough LH power, diamagnetic electron fishbones take over and the electron temperature oscillation disappears. In this last regime, when increasing LH power, we observe a frequency rise in agreement with the theory. In ICRH plasmas, modes in the frequency range $30-70$ kHz are observed with reflectometry, ECE correlation and soft X-ray diagnostics. This frequency range corresponds to the acoustic frequency, a domain where both Geodesic Acoustic Modes (GAMs) and BAEs are expected. However density perturbations associated with GAMs are predicted to vanish in the equatorial plane, whereas such perturbations are clearly measured by reflectometry. Hence this mode is rather identified as a beta Alfvén Eigenmode (BAEs). Using a variational formalism and a gyrokinetic MHD model we have derived the BAE dispersion relation. The excitation threshold results from a balance between the fast ion drive and Landau damping by thermal ions. This threshold has been quantitatively calculated using a fast ion distribution computed with the PION code. It is found that the experimental mode frequency follows the BAE dispersion relation. Also the excitation threshold is found to be in reasonable agreement with the experimental observations.

EX/P8-10 · TF Ripple Induced Stochastic Diffusion of Energetic Particles in Advanced Tokamak Configurations on HL-2A

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Abstract: A NBI heated discharge of optimized configuration is modeled with TRANSP under the HL-2A condition. The target plasma ($I_p = 220$ kA, $B_t = 2.0$T, and line averaged $n_e = 1.0 \times 10^{19}$ m^{-3}) is heated by NBI during $0.5 - 1.8$ s. 1.5 MW LH wave power with a nearly symmetric power spectrum is injected into the NBI heated plasma during $t = 1.0 - 1.5$ s to control the current profile and elevate the electron temperature. Due to the off-axis driven current and electron heating, an optimized q-profile, of which the magnetic shear is weak in the central region and negative in the mid-plasma region ($x \sim 0.5 - 0.65$), is formed. Accordingly an ITB is developed because of the shear reversal. The ripple loss of the enrgetic partiles is studied by using the Golaston-Whiet-Boozer threshold for stochastic ripple diffusion. The ripple loss fraction of the captured NBI power is rather large. It is more than 20% in the RS phase ($t = 1.0-1.5$ s). To show the effect of q-profile on the ripple loss, a conventional tokamak discharge is established with the same NBI injected into a target plasma of I_p=400 kA, B_t=2.43 T, in which ECRH is used at $t = 1.0-1.5$ s to keep T_e being the similar value as in the RS discharge above. In the conventional tokamak discharge a monotonic q-profile is formed, and the ripple loss fraction of the captured NBI power in roughly an order of magnitude less than that in the RS discharge. We evaluate the ripple diffusion threshold for the energetic ions of initial velocity to depict the stochastic ripple loss region in the plasma cross-section. For the RS configuration, the stochastic ripple loss domain occupies more than half of the plasma cross-section on the low field side excluding some small islands. For the non-RS configuration, however, the ripple loss domain is only a small area located on the outside plasma boundary. The factor of the ripple loss fraction of particles against that of energy, KNE, is much higher than unity showing that most of the particles are lost during slowing down. KNE is dependent on the energy of injected particles, the injecting angle of neutral beam line, and electron temperature of the bulk plasma. With NBI of lower particle energy, the ripple loss fraction of the captured NBI power is larger than that for the case of higher NBI energy because the lost particles are richer in ones of relatively higher energy in the case of lower NBI energy.

EX/P8-11 · Destabilization of the Internal Kink Mode by Energetic Electrons on the HL-2A Tokamak

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Abstract: Strong burst of the internal kink mode excited by energetic electrons have been observed on the HL-2A tokomak. The mode has obvious features of the fishbone-like structure, which characterized by the frequency chirping, and amplitude bursting. The fishbone-like structures are located at the $q = 1$ flux surface. The frequency is between 4 and 8 kHz, and the modes with poloidal/toroidal mode number ($m/n = 1/1$) are characterized by periodic recurrence in the scale time about $2 - 4$ ms. The mode is also proved to relate with non- Maxwell distribution of the electrons obviously. It is found that an energetic electron beam, which departs Maxwell velocity distribution, interacts with internal kink mode. The energy distribution of the electrons is measured by a hard X-ray detector (CdTe) with the pulse height analysis (PHA) mode. As long as the energetic electrons with energy of $35-70$ keV come into being, the fishbone-like instabilities are driven during Electron Cyclotron Resonance (ECR) heating. The fishbone-like instabilities can be excited on the HFS ECR heating, and it has been found for the first time that the modes also present themselves LFS ECR heating. The modes propagate toroidally parallel to the precession velocity of deeply trapped ions which is in the same direction as the plasma current and poloidally parallel to the electron diamagnetic drift velocity. When the average line density exceeds 4.0 (units in 10^{13} cm^{-3}) or the ECRH power is larger than 900 kW, the fishbone-like instabilities don't occur on HL-2A. In order to further assess the identification of this instabilty with fishbone mode, the resonance condition of wave-parcticle has been adopted. Comparing with experimental results, the calculation analysis show that the mode frequency is close to the precession frequency of barely the trapped electrons and the barely circulating electrons when the magnetic shear is very weak or negative. The fishbone instabilities are supposed to be excited by the barely circulating electrons via precessional resonance with off-axis ECRH on the LFS. However, on the HFS, the modes are likely to be driven jointly by the barely trapped electrons and the barely circulating electrons.

EX/P8-12 · Search for Fishbone-like Internal Kink Instabilities Driven by Suprathermal Electrons in FTU

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Abstract: Fishbone-like internal kink instabilities driven by electrons are observed in FTU during lower hybrid (LH) current drive experiments. By a general theoretical framework, this phenomenon is interpreted as effect of the level of LH power input, which controls the transition from nearly steady state to bursting electron fishbone oscillations. The bursting regime corresponds to a continuum resonant mode well above marginal stability, and associated with significant fast electron redistribution that is analogous to fast ion losses expected when ion fishbones are excited. This phenomenon is relevant to burning plasmas, as the bounce averaged dynamics of fast electrons depends on energy (not mass), so that the effects associated with low frequency magnetohydrodinamic (MHD) modes can serve simulating and analyzing analogous charged fusion product effects. Combining theory, modelling and the flexible experimental tools available in FTU, such as the lithizated vessel, LHCD and ECRH, optimal excitation conditions for the fishbone-like instability, expected to occur for $q_{min} \geq 1$, are being investigated via sensitivity studies on LH power, q-profile conditions, as well as perpendicular and parallel electron energy, obtained by changing the ratio of ECRH to LH power.

EX/P9-1 · Disruptions Mitigation in ASDEX Upgrade with the In-vessel Fast Valve.

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Abstract: According to our present knowledge, a disruption in ITER is going to convert a large fraction of the plasma current into a beam of highly energetic runaway electrons. In order to prevent the generation of the runaway electrons, the electron density must be increased up to 5×10^{22} m^{-3}, which is a factor of 500 larger than the nominal one. Such density is difficult to reach, it has a strong impact on the design of the pumping system and on the operation of the tokamak. Therefore a significant effort must be addressed in understanding which processes control the fueling efficiency in order to maximize it. This work illustrates the progress made at ASDEX Upgrade in (1) developing and installing a valve close to the plasma with a relatively large fueling efficiency and (2) understanding the experimental results. The experimental data collected up to now indicate that the fueling efficiency ranges between 25 and 40% for the in-vessel valve and is independent of the type of gas injected (experiments with helium have not been done yet). In addition it does not depend clearly on the quantity of gas injected and is not a strong function of plasma energy. The code package SOLPS is used to model gas injection in an ASDEX Upgrade plasma; the experimental measurements are directly compared with the code prediction.

EX/P9-2 · Study on Stabilization of Tearing Mode with ECRH and its Resultant Transport Properties on HL-2A Tokamak

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Abstract: The important role of the plasma current profile in influencing plasma performance is becoming increasingly recognized, and HL-2A uses ECRH/ECCD to provide active control of the local current profile to control MHD activity. Stabilization of $m/n = 2/1$ tearing modes has been achieved by depositing ECRH/ECCD power off-axis in the vicinity of the $q = 2$ surface, resulting in a dramatical increase in plasma density and stored energy. In particular, it is evidenced as a new finding that the suppression of $m/n = 2/1$ tearing modes can be sustained by modulated ECRH but with low frequency(about 10 Hz). Furthermore,concurrently with the suppression the discharge is characterized by a continuous confinement improvement, i.e., the plasma density, temperature, stored energy increase steadily. This provides a low cost, effective means of controlling $m/n = 2/1$ tearing modes and extending the beneficial effects of MHD suppression on particle and energy confinement.

EX/P9-3 · Investigations of Magnetohydrodynamic (MHD) Instabilities in the STOR-M Tokamak

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Abstract: MHD modes in the STOR-M tokamak ($R/a = 0.46/0.12$, $B < 1$ T, $I < 50$ kA) have been measured using 32 discrete external Mirnov coils. The data have been analyzed using (a) conventional Fast Fourier Transform (FFT) cross-correlation analysis, (b) direct Fourier components analysis, and (c) Singular Value Decomposition (SVD). During the nominal Ohmic discharge phase, the dominant poloidal mode numbers are 2, 3, and 4 and toroidal mode number $n = 1$. The oscillation frequencies are in the range $20 - 30$ kHz. During the improved confinement phase triggered by compact torus injection, the Mirnov oscillations are strongly suppressed to a level lower than noise. The MHD activities reappear BEFORE the back transition from the H-mode-like to L-mode discharge phase. The reappearance of the MHD activities is led by an m=1 gong mode burst which is followed by a propagating $m = 2$ mode.

EX/P9-4 · Ion and Electron Heating Characteristics of Magnetic Reconnection in TS-3 and UTST Merging Startup Experiments

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Abstract: Cause and mechanism for magnetic reconnection heating were directly measured for the first time in TS-3 spherical tokamak (ST) merging experiment by use of new two dimensional ion and electron measurements. Two ST plasmas were merged together in the axial direction for high-power reconnection heating/ startup without center-solenoid (CS) coil. A new finding is that ions and electrons were heated in the down-stream area and in the current sheet, respectively. The 2-D line-integrated data of spectral lines were measured by polychromators with ICCD cameras, and were transformed into local ion temperature data using the tomography reconstruction. The electron temperature was measured by a scanning electrostatic probe array. While the electron temperature outside the sheet was uniformly $5 - 6$ eV, it clearly peaked around the current sheet. While electrons were quickly hearted inside current sheet by its ohmic heating power, ions were heated in the downstream area by shock or viscosity damping of the reconnection outflow. The ion heating power ~ 4 MW was an order of magnitude larger than the electron heating power ~ 0.2 MW. This heating mechanism is consistent with the squared B scaling (B: poloidal field) of reconnection heating energy, because the outflow speed is equal to the Alfvén speed in the Sweet-Parker model. Based on these successful results, a new ST device: UTST ($R \sim 0.4$ m) started the initial operation with plasma current $50 - 100$ kA. The UTST device has all PF coils outside of the vacuum vessel to demonstrate (1) double-null startup of STs without CS coil, (2) their high-power reconnection heating for high-beta ST formation and (3) their sustainment by advanced RF and NBI techniques.

EX/P9-5 · Comprehensive Control of Resistive Wall Modes in DIII-D Advanced Tokamak Plasmas

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Abstract: Comprehensive control of the resistive wall mode (RWM) with plasma rotation and magnetic feedback is needed for robust operation above the no-wall β limit. Although the existence of high β operation above the no-wall β limit was successfully demonstrated at low plasma rotation [1,2], the robustness of steady-state operation is not guaranteed. Above the no-wall stability limit MHD activity such as edge localized modes (ELMs) often cause a sudden increase in amplitude of an $n = 1$ RWM to $3 - 5$ Gauss within $10 - 100$ ms (i.e, on the quasi-MHD time scale). Such perturbations have the potential to develop into a large scale RWM followed by collapse of the rotation and beta. Thus, the amplitude of RWM excited by ELM events must be reduced by feedback before the next event in order to prevent a uncertain deterioration in the plasma performance. Feedback control with internal control coils and proportional gain only is sufficient to shorten the decay time of the perturbation from up to 5 ms to values of typically 1 ms, improving the stability. Perturbations in the ion temperature in the region outside the $q = 3$ surface are also substantially reduced by feedback. The feedback system is most effective at controlling the ELM-driven RWM when the fast feedback is combined with real-time error field compensation with a second set of coils using independently optimized feedback setting. The possibility of a non-rigid plasma response during feedback has been studied using the NMA stability code. Without adequate feedback design, the mode helicity can switch direction from positive to negative as β increases toward the ideal-wall limit. The calculations predict that an optimized coil geometry can greatly improve feedback performance at high β. In ITER, these experimental results support the combination of port-plug coils along with the external error field correction coils for achieving robust operation above the no-wall limit, even in the absence of plasma rotation.

[1] H. Reimerdes, et al., Phys. Rev. Lett. **98**, 055001 (2007). [2] M. Takechi, et al., Phys. Rev. Lett. **93**, 055002 (2007).

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EX/P9-6 · Control of External Kink Modes Near the Ideal Wall Limit Using Kalman Filtering and Optimal Control Techniques

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Abstract: We report on the first experimental demonstration of feedback suppression of rotating external kink modes near the ideal wall limit in a tokamak using Kalman filtering to discriminate $n = 1$ kink mode amplitude from background MHD noise. Kalman filtering was observed to suppress the kink over a broad range of feedback phase angles between the sensed mode and applied control field. This suppression is accomplished with no excitation of higher frequencies under feedback as was observed in previous experiments using simple lead-lag loop compensation. In order to achieve the highest plasma pressure limits in ITER, resistive wall kink mode stabilization is also required. A novel resistive wall mode Kalman filter and feedback controller designed using model reduction and optimal control theory and employing only proportional gain have been designed for ITER allowing operation up to 86% of the ideal wall limit using the present design external control coils. We find an order of magnitude reduction of the required control coil current and voltage in the presence of white noise from the no-wall limit to the optimal feedback system performance limit as compared with a traditional, classical controller.

EX/P9-7 · Reversed-field Pinch Contributions to Resistive Wall Mode Physics and Control

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Abstract: Optimal feedback control of resistive-wall modes (RWM) is of common interest for toroidal fusion concepts that use conducting walls for stabilization of ideal MHD modes. From the RWM control point of view, the RFP situation is in many respects similar to the advanced tokamak situation in the presence of very low plasma rotation, where the most effective stabilizing mechanism is the feedback action of a set of active coils. Results from EXTRAP T2R (Sweden) and RFX-mod (Italy) RFP experiments have shown that full feedback control of multiple RWMs is possible and their deleterious effects can be completely suppressed. Both for the RFP and tokamak configurations, it is important to optimize the RWM control systems for future implementation. Important aspects of optimization are avoidance of the harmful effects of sideband modes (aliasing) in the control spectrum, fast transient response time, effective mode identification and tracking capability, minimized power requirements and robust controller stability. The paper describes collaborative work carried out on the two machines. Experimental studies of RWM control physics and progress in development of optimal control schemes are presented. In addition, simulation modelling of the functioning control systems is presented. An important general conclusion of the RWM physics experiments is that linear MHD stability theory for perturbation mode dynamics is adequate for use in simulation modelling of control systems. For implementation of RWM control systems in future experiments, both for the RFP and tokamak configurations, it is important to optimize the RWM control system. In order to optimize the control systems it is now important to introduce advanced control theory. The work involves development of a MIMO plant model by coupling of SISO dynamics with consideration of the bias in sensor signals introduced by aliasing. These simulations describe very well the operation of the controller including the controller stability. The model of the full control system has been used to optimize transient response and power requirements by appropriate selection of the PID gain matrix.

EX/P9-8 · Active Control of Resistive Kink-Tearing Modes in RFX-mod

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Abstract: The presence of error fields and the growth of some MHD instabilities have negative influences both on plasma core transport and plasma-wall interactions in fusion devices. The RFX-mod reversed field pinch (RFP) experiment is well equipped to study this issue: its magnetic boundary is, in fact, composed by a thin shell and a set of 48(toroidal) $\times$ 4(poloidal) actively controlled saddle coils. In the last two years a systematic error due to the aliasing of the sideband harmonics generated by the saddle coils has been identified and the control system modified in order to perform the correction in real-time (sideband cleaning). Moreover, the control algorithms operate on the measurements performed by a set of "virtual sensors" located exactly at the plasma edge. We present here results obtained with a control algorithm that aims at zeroing Fourier harmonics, named "Mode control". Real-time sideband-cleaned measurements are the feedback variables for this algorithm. It is found that, by properly selecting different gains for different resistive-kink tearing modes, wall unlocking and partial phase unlocking can be obtained. Therefore the Mode Control scheme, applied to sideband cleaned measurements reduces effectively the plasma column deformation due to tearing modes. The flat top averaged maximum non-axisymmetric displacement can be as low as 1 cm. Consequently, localized plasma wall interactions decrease and are spread in different locations during the discharge. Moreover, the systematic reduction of the deformation of the plasma column allows reliable operations at plasma currents up to 1.5 MA with reproducible electron density behaviour and frequent onset of quasi single helicity spectra.

EX/P9-9 · MHD Modes Associated to Hollow Current Density Profile Configuration: Experiment and Modelling

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Abstract: Hollow current density profiles often exist in Tokamak discharges, either transiently in the current ramp up, or in stationary phases where their main interest is to be generally associated with improved core confinement. Such a configuration is particularly relevant for Steady-State non-inductive scenarios where the self-driven bootstrap current dominates, as envisaged for ITER, or where external current drive peaks off-axis, as in Tore Supra. The reversed safety factor profile then obtained is moreover generally considered to reduce turbulent transport and form Internal Transport Barrier. However, MHD limits associated to such a configuration can be detrimental for the energy confinement, or even for the plasma control, due to dual-resonant surfaces. We show experimental analysis of MHD activity from Tore Supra experiments, as well as non-linear numerical simulations that probe our understanding of the underlying processes. The MHD behaviour at higher, ITER relevant normalized pressure is also discussed. The Double-Tearing Mode dynamics in Tore Supra discharges is well recovered by non-linear simulations, with different regimes corresponding to different width of the reconnected region. As q_{min} gets around $3/2$, the $n = 1$ mode is often absent. A conventional MHD model without heat transport can recover this result, which is consistent with the large stabilizing contribution from toroidal curvature at low resistivity. However, taking into account heat transport removes this dependence on resistivity, which also cancels in the Drift-Tearing model. Preliminary non-linear simulations suggest that the $n = 1$ mode could saturate at reduced amplitude when two-fluid effects are taken into account. Several observations show that the Double-Tearing Mode remains a severe MHD limit in high performance discharges, resulting in large performance degradation similar to the MHD regime seen on Tore Supra. The non-linear behaviour of the infernal mode has been studied on the basis of a JET-like equilibrium. We find that the redistribution of the pressure due to the MHD mode can trigger a secondary instability, namely a $m = 1$ internal kink. This type of cascade process represents the main concern for evaluating the impact of initially localized modes, with potential triggering of Neoclassical Tearing Modes or pressure driven modes.

EX/P9-10 · The Role of Stochastization in Fast MHD Phenomena on ASDEX Upgrade

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Abstract: Studies of fast MHD events in ASDEX-U suggest that stochastization plays an important role in these processes. In spite of the short time duration and small region of stochastization, it can lead to very strong changes in plasma confinement. Main reason for such influence is a strong mixing of the magnetic field lines which destroy magnetic surfaces and strongly increase radial transport. In this paper three such phenomena are discussed: frequently interrupted regime of neoclassical tearing mode (FIR-NTM), minor disruption due to interaction of the (2,1) and (3,1) tearing modes and sawtooth crash. The role of stochastization of magnetic field lines is analyzed by applying the mapping technique to trace the field lines of toroidally confined plasma. The sawtooth crash dynamics is also studied by means of spectral analysis and reconstruction of phase trajectories using delay coordinates which are standard techniques for stochastic systems and have been employed for identification of the transition to chaos in different physical systems.

EX/P9-11 · Configurational Effects on Stability and Confinement in the H-1NF Heliac

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Abstract: H-1 is a helical axis stellarator of major radius $R = 1$ m, and average minor radius $< r >\sim 0.15 - 0.2$ m. Its flexible heliac coil set permits access to a wide range of magnetic configurations. RF-generated plasma shows a very complex dependence on configuration of both the electron density and the nature of fluctuations. Magnetic fluctuations range from highly coherent, often multi-frequency in sequence or simultaneously, to approaching broad-band ($\delta f/f \sim 0.02 - 0.5$), in the range $1 - 200$ kHz. Application of datamining to a wide range of configurations classifies these fluctuations and extracts poloidal and toroidal mode numbers, revealing that a significant class of fluctuations exhibit scaling which is Alfvénic with electron density (within a constant factor) and shear Alfvénic in rotational transform. An array of optical and interferometric diagnostics is combined with the magnetic probe arrays to provide information on internal structure of the MHD modes, and associated 3D effects. The configurational dependence is closely related to the presence of low order rational surfaces. The plasma density falls to very low values at these rationals prompting the question: how much is this due to the effect of magnetic islands? Magnetic islands are expected to be relatively less important in a moderate shear device such as the heliac, than in conventional stellarators. Results from a uniquely accurate magnetic field mapping system, combined with a comprehensive model of the vacuum magnetic field in H-1 show that magnetic islands do not dominate the configuration for the low order resonances in the configuration scan. More detailed studies of plasma in configurations with magnetic islands show some surprising albeit small improvements in confinement and rapid changes in radial electric field. The results to be presented will demonstrate how H-1's unique combination of flexibility and variety of advanced diagnostics can contribute to two key issues for future fusion devices: Alfvénic activity, and the effects of magnetic islands.

EX/P9-12 · Overview of Tearing Mode Physics and Control Program at TEXTOR

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Abstract: This paper presents the work on Tearing Modes (TMs) carried out in TEXTOR. The dynamic ergodic divertor reproducibly produces TMs. These are manipulated by the localized heating and current drive from the 1 MW, 140 GHz ECRH system, and diagnosed with the 2D Electron Cyclotron Emission (ECE)-Imaging diagnostic. Heating is shown to be the dominant mechanism for TM suppression in TEXTOR. This is explained by modeling. In ITER, the TM stabilizing effects of heating and direct current drive are of the same order of magnitude. 2D heat pulse propagation studies show that inside islands with a flat temperature profile, the electron heat transport is strongly reduced with expect to the ambient plasma. Power balance analysis of heated islands shows that the electron transport is similar to the ambient plasma. These results are consistent with critical gradient driven heat- transport within the island. A 6 channel ECE diagnostic has been integrated in the transmission line of the ECRH. This links the observed TMs to the ECRH power deposition radius. TMs and sawteeth can be appreciated from the data during ECRH. Under some conditions, a scattering is observed showing a clear correlation with MHD events. Model-based strategies for control of the electro-mechanical launcher have been developed. This involves input-output models derived from experimental analysis of the launcher dynamics.

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*** Fusion Technology and Power Plant Design***

FT/1-1 · Overview of Recent Commissioning Results of KSTAR

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Abstract: KSTAR is an advanced tokamak with fully superconducting coils for the steady state plasma research. The detailed engineering design and infrastructure setup for R&D have been completed in 2001. In March 2008, after the construction period of seven years, the KSTAR assembly has been completed, by connecting the cryogenic transfer lines between the tokamak and the cryogenic distribution system. The commissioning on KSTAR has been progressed from April to July, through these four steps: vacuum, cryogenic cool down, superconducting magnet test, and plasma start-up. KSTAR has successfully passed the vacuum and cool down commissioning at the first trial. All of the subsystems have been tested and showed to satisfy the design requirements. All of the magnets were cooled down to 4.5 K stably and then charged successfully without any serious faults. The first ohmic plasma was achieved on June 10th. The breakdown was successfully achieved at $B_t = 1.5$ T at 1.8 m. After several tens of breakdown shots, the first plasma of more than 100 kA has been successfully achieved on June 13th. At the plasma start-up stage, it has been verified that the key issues for the breakdown and the current ramp-up are: the null field, breakdown electric field, toroidal field, gas pressure, blip duration, ECH pulse length and impurity control. This first plasma commissioning has demonstrated that KSTAR has been successfully constructed and is ready for operation. All of the commissioning progress, including various problems and interesting test results, are summarized in this paper. Furthermore, details on the individual subsystem commissioning results will be presented in the KSTAR-related paper at this conference.

FT/1-2 · The IFMIF/EVEDA Project: Outcome of the First Engineering Studies

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Abstract: IFMIF is an accelerator driven neutron source consisting of two deuteron accelerators bringing two intense beams carrying each of them 125 mA to an energy of 40 MeV. These beams interact with a 25 mm-thick liquid lithium flow circulating at a speed of nominally 15 m/s and generate an intense flux of neutrons whose energy is centered at 14 MeV, the energy of D-T fusion reactions. The accelerator main subsystems are the following: (i) The injector, creating the deuteron beam by means of electron cyclotron resonance and at an energy of 100 keV; (ii) The RadioFrequency Quadrupole (RFQ), bunching and accelerating the beam up to 5 MeV; (iii) The Matching Section, optimizing the beam before its entrance into (iv) The Drift Tube Linac, with a Half Wave Resonator superconducting structure, being currently assessed (the first section up to an energy of 9 MeV will be tested during EVEDA); (v) The High Energy Beam Transport Line, transporting and shaping the beam to its required characteristics at the entrance of the Lithium Target (a rectangle of 20×5 cm^2 with a flat energy profile of 40 MeV). (vi) Diagnostics and auxiliaries complete the system. The rapid flow of lithium and the highly activated environment pose many challenging issues: (i) Hydrodynamics of the flow (a laminar flow is required with a maximum ripple of ± 1 mm); (ii) Erosion and corrosion of the loop; (iii) Purification and monitoring of impurities (hydrogen, nitrogen, oxygen and carbon) at only a few ppm, because of the point above; (iv) Safety for the installation and the personnel; (v) Diagnostics implementation in a highly radioactive environment; (vi) Rapid change of the backplate, subject to neutrons flux up to 60 dpa per year. Finally, the Test Facilities, which will host the samples to be characterized, also raise severe issues: (i) Resistance to the intense flux of neutrons; (ii) Stable operating conditions for all samples (temperature, stress, neutron flux, etc.); (iii) Replacement and Remote Handling of all constituents; (iv) Cooling of the Test Cell. The main first activities were mainly centered on the preparatory work for the prototypes, with detailed thermal, thermo-mechanical, neutronics, etc. calculations and the first engineering work. First technological demonstrations were performed (RFQ brazing, backplate welding, erosion and corrosion measurements, etc.). The paper will review this work and present a synthesis of all activities conducted so far.

FT/1-3 · Fusion Technology Development for DEMO in the Broader Approach Activities

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Abstract: As a part of the Broader Approach activities from 2007 to 2016, research and developments on blanket related materials and tritium technology have been initiated toward DEMO by EU and Japan. According to the common interest of EU and Japan toward DEMO, R&Ds on reduced activation ferritic martensitic (RAFM) steels as a DEMO blanket structural material, silicon carbide (SiC/SiC) composites, advanced tritium breeders and neutron multiplier for DEMO blankets, and tritium technology will be implemented as a part of the DEMO Design and R&D coordination Center activity of BA. As a preparatory work, 5-tone heat of F82H was done recently. It was found that Electro Slag Remelting was required as for secondary melting process in order to control unexpected impurities. Double notch tensile test to evaluate failure behavior was performed for NITE-SiC/SiC composites with different size and different notch depth. It was found that neither significant notch sensitivity nor specimen size effect was observed in proportional limit tensile stress and fracture strength, implying the importance of stress (or strain) criterion for the failure evaluation of composites. On the neutron multiplier, reactivity of Be-Ti and Be-V alloys for F82H was investigated. The reactivity of Be-Ti and Be-V alloys was much smaller than that of beryllium meal. As a preliminary work on the recycle of Li_2TiO_3 pebbles, solving technique for Li_2TiO_3 pebbles was investigated. It was found that those pebbles were solved with H_2O_2 without acid. This technique has large advantage from the safety point of view compared with conventional solving technique using acid.

FT/1-4 · The Construction of the Wendelstein 7-X Stellarator

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Abstract: This paper reviews the recent progress in the construction and assembly of the large, superconducting stellarator device Wendelstein 7-X in Greifswald, Germany. Technical solutions are discussed as well as the impact on the later physics program. Also the "lessons learned" during the long construction phase are discussed. Last not least, the role of Wendelstein 7-X in the international stellarator program and its relevance for ITER are outlined. Wendelstein 7-X is a large (30 m^{-3} plasma volume) stellarator with superconducting modular magnetic field coils (3 T induction on axis). The design of Wendelstein 7-X is based on the quasi-isodynamic concept. The mission of W7-X is to demonstrate the basic reactor suitability of this concept under steady-state operation conditions. The project has experienced a number of technical set-backs, in particular in the area of superconducting coils and the mechanical structure of the device. Both problem areas are now solved and the manufacturing of component progresses well. In particular, all magnetic field coils are manufactured and more than half have already passed the cold test. The project now orients towards device assembly. The device consists of five almost identical modules forming the torus. Each two half magnet modules are assembled in parallel and joined after completion. This important step was recently accomplished by the project in schedule. A large variety of assembly technologies have been developed and successfully qualified. They will be reviewed in the present contribution. The first step in the future operation of Wendelstein 7-X is to develop integrated discharge scenarios with competitive performance parameters in terms of equilibrium, stability and transport. Reference discharge scenarios will be developed with a simplified (inertially cooled) robust test-divertor. After this first (short) operation phase, the pressure-cooled system of in-vessel components is completed and the best-suited discharge scenarios are extended to steady-state (30 min) at full 10 MW ECRH heating power. Local current drive by ECR waves will be used to control the edge magnetic field structure, required for proper divertor operation. A main task is the full integration of discharge scenarios, plasma heating, control and plasma-wall interaction to achieve reactor-relevant plasmas under steady-state conditions.

FT/1-5 · The Fusion Advanced Studies Torus (FAST): A Proposal for an ITER Satellite Facility in Support of the Development of Fusion Energy

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Abstract: The successful development of ITER and DEMO scenarios requires a preparatory activity on devices smaller than ITER, with sufficient flexibility and capable of investigating the peculiar physics of burning plasma conditions. The aim of the Fusion Advanced Studies Torus (FAST) proposal is showing that preparation of ITER scenarios and development of new expertise for DEMO design and R&D can be effectively implemented on a new facility that: a) will work with Deuterium plasmas, avoiding the problems associated with the use of Tritium, and will investigate non linear dynamics that are relevant for the understanding of alpha particle behaviors in burning plasmas by using fast ions accelerated by heating and current drive systems; b) will work in a dimensionless parameter range close to that of ITER; c) will test technical innovative solutions for the first wall/divertor directly relevant for ITER and DEMO, such as full-tungsten plasma facing components and advanced liquid metal divertor target; d) will exploit advanced regimes with long pulse duration with respect to the current diffusion time; e) will provide a test bed for ITER and DEMO diagnostics; all fast particles expected in FAST plasmas (p, D, ^{3}He) are assumed to be diagnosed; as in ITER, diagnostics of fast particles require intensive R&D, that can be carried out on FAST with reduced costs and development time; f) will provide an ideal framework for model and numerical code benchmarks, verification and validation in ITER and DEMO relevant plasma conditions. The scientific rationale of the FAST conceptual design will be discussed. The choice of high equilibrium magnetic field $B = 7.5$ T, the consequent possibility of operating routinely at high plasma current $I_p = 6.5$ MA, high plasma density and moderate temperature, while maintaining a significant fusion performance at $Q > 1$ together with the possibility of extended pulse operations (up to 80 resistive times) will elucidate the capability for FAST of reaching its objectives in a flexible and cost-effective way.

FT/2-1 · Reflectivity Reduction of Retro-Reflector Installed in LHD due to Plasma Surface Interaction

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Abstract: Optical reflectivity of the retro-reflector installed in LHD as the first mirror was reduced seriously by plasma wall interaction. In order to understand the mechanism of the reflectivity reduction, optical and material properties of the mirror surfaces have been examined extensively. It was found that the deposited impurity layers caused the serious reduction of the reflectivity. Formation of iron oxide, bulges structure and He bubbles are the major factors for the reflectivity reduction in the wide wave length range. Based on these results, a new concept of the reduction-free retro-reflector is proposed.

FT/3-1Ra · Demonstration of 1 MW Quasi-CW Operation of 170 GHz Gyrotron and Progress of EC Technology for ITER

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Abstract: In last two years, significant progress has been obtained in the development of electron cyclotron heating and current drive (EC H&CD) technology for ITER. On the 170 GHz gyrotron, 1.02 MW power generation was attained at 800 s pulse duration with the efficiency of 55% at the first time in the world, which satisfy the ITER criteria of 1 MW, 500 s, 50%, respectively. The total output microwave energy from the gyrotron is 144 GJ without a trouble. A ~ 0.97 MW, quasi-CW power transmission was proved using ITER relevant waveguide system. The basic design has completed for the equatorial launcher, and mock-up of key components such as a movable mirror were fabricated. These give a clear prospect for the accomplishment of the EC H&CD system on ITER.

FT/3-1Rb · Status of the Development in Russia of a Megawatt 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. During last years several new gyrotrons were designed and tested at IAP/GYCOM. Main development efforts were spent for development 170 GHz/1 MW/50%/CW gyrotron for ITER and multi-frequency gyrotrons. The industrial gyrotron prototype for ITER operates at very high order mode $TE_{25.10}$ which allows efficient cooling of the cavity walls. The calculations show the possibility of 1 MW microwave generation in the cavity in CW regime. Potential depression at the collector provides power load on the collector surface essentially (up to two times) lower than without electron energy recovery. A new efficient mode converter is used in the gyrotron. The gyrotropn is equipped with a CVD diamond output window. The tests were performed at the test stand at Kurchatov Institute. The following gyrotron output parameters (power/pulse duration) were demonstrated so far: 1 MW/116 sec and 0.85 MW/203 sec. The very efficient operation of the cavity cooling system demonstrated in our megawatt gyrotrons opens the way to increase the generated power by the gyrotron with traditional cylindrical cavity of slightly increased size. Such kind of project is under IAP/GYCOM development. First experiment is carried out with a short-pulse (0.1 sec) mock-up of 170 GHz gyrotron with 1.5 MW output power. In the first tests of the gyrotron the power 1.44 MW was achieved with efficiency of 41%. A gyrotron capable to operate at several frequencies is very attractive for plasma experiments. The use of step-tunable gyrotrons can greatly enhance flexibility and performance of ECRH/ECCD systems due to larger accessible radial range, possible replacement of steerable antennas, higher CD efficiency for NTM stabilization. Even two-frequency gyrotrons can bring real improvement

FT/3-1Rc · Experimental Investigations on the Pre-Prototype of the 170 GHz, 2 MW Coaxial Cavity Gyrotron for ITER

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Abstract: The development of a 2 MW, CW, 170 GHz coaxial cavity gyrotron at Forschungszentrum Karlsruhe (FZK) is in progress in cooperation with several European research institutions. The gyrotron is foreseen as a source for heating, current drive and stabilization of plasmas in the International Thermonuclear Experimental Reactor (ITER). Tests with a first industrial prototype started recently at CRPP Lausanne. The design of critical components, especially the electron gun, cavity and quasi-optical (q.o.) RF-output system has been verified at FZK in tests with a short pulse gyrotron, the so called pre-prototype. When starting the experiments the gyrotron operation was strongly limited due to the excitation of parasitic LF-oscillations with very high amplitude mainly around 265 MHz. The oscillations have been successfully suppressed due to small modification of the geometry of the gyrotron components. In recent experiments another parasitic oscillation around 160 GHz appears simultaneously with the desired gyrotron working mode. This parasitic oscillation belongs to a multi-moding scenario observed at operating parameters near of the nominal values. In the region with multi mode operation a reduction of the generated power in comparison with calculations, has been observed. The reason for this behavior is currently under investigation. The performance of the q.o. RF-output system was experimentally tested at the low power. The measured pattern at the window position has shown some discrepancies to the calculated one. The Gaussian content of the RF-beam at the window position evaluated from the experimental values was only 77%, whereas from design calculations a value of 86% is expected. The recent activities towards the improvement of the gyrotron operation will be presented and discussed.

FT/4-1 · Plasma Surface Interaction Issues of an All-metal ITER

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Abstract: We assess and review key plasma surface interaction issues of an all-metal plasma facing system (PFC) for ITER, in particular an all-tungsten coated divertor and a beryllium or tungsten coated first wall. Such a system eliminates problems with carbon erosion and T/C codeposition, and for an all-tungsten system would better extrapolate to post-ITER devices. The key issues are mixed-material sputtering/transport and formation of mixed surface layers, tritium implantation and codeposition, core plasma contamination, divertor response to ELM's, helium implantation effects, and overall surface thermo-mechanical performance. Code package OMEGA (UEDGE/DEGAS/REDEP/TRIM-SP etc.) is used to compute first wall and divertor sputtering material transport and redeposition, in an ITER full power D-T plasma edge regime with convective and conductive transport. The HEIGHTS code package analyzes divertor response to disruptions and ELM's. Code results together with PISCES, C-MOD and other tokamak data on sputtering, tritium codeposition, and mixed-material properties, are used to assess PFC performance. We conclude that an all-metal system is likely much better than a system using carbon, but critical issues remain in areas of plasma transient induced surface erosion (of all materials), tungsten surface integrity with Be/W alloy formation and energetic He bombardment, and tritium codeposition in sputtered/redeposited beryllium. Steps are suggested to ameliorate problems and reduce uncertainties.

FT/4-2Ra · Development of Wall Conditioning and Tritium Removal Techniques in TEXTOR for ITER and Future Fusion Devices

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Abstract: ITER needs the availability of wall conditioning techniques for routine operation and to control the in vessel long term tritium retention. Methods have to be developed which can be used in the presence of the toroidal magnetic field. For this purpose, Ion Cyclotron Wall Conditioning (ICWC) have been optimised for wall cleaning and fuel removal in TEXTOR. The ICWC plasmas have been produced from two ICRF antennas in a single-pulse mode in He and He/H_2, gas-mixtures in a pressure range of $(2-4)\times 10^{-2}$ Pa and a coupled RF power $10-135$ kW. To optimise the cleaning efficiency, a proper He/H_2 gas mixture and the overlay of an additional steady state or modulated vertical magnetic field $(0-0.04$ T) are important. The observed effects are compared with the predictions from 1-D RF and 0-D transport modelling which give the basis for scaling up the ICWC conditioning discharge parameters to ITER and other devices. He, H_2, oxygen, nitrogen, ammonium (NH_3) and mixtures of it were investigated to improve the removal efficiency for retained fuel and redeposited carbon layers. ICWC in (He+O_2)-mixture in the pulse mode (5 s) showed a complete consumption of the injected molecular oxygen with a significant part that increases with operation time converted to CO and CO_2 . The upper limit of the CO and CO_2 production is given by the maximal neutral pressure of CO and CO_2 permitted for non-arcing antenna operation, showing that high pumping speeds are favourable for optimisation of the removal efficiency. Very recently N_2 and NH_3 have been used and compared with oxygen with the analysis ongoing. However, first data show also a complete consumption of injected neutral N_2 and N from NH_3, but with no corresponding conversion to other neutral reaction products, indicating a strong pumping of N by the walls. This contribution gives an overview of the results and discuss the prospects and further R&D needs for routine application in ITER and future devices.

FT/4-2Rb · Hydrogenic Retention of High-Z Refractory Metals Exposed to ITER Divertor Relevant Plasma Conditions

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Abstract: Fuel retention in the wall is a serious issue for ITER both in terms of tritium inventory safety limit and efficiency of the reactor. The prediction of the hydrogenic retention characteristics of the tungsten (W) divertor in ITER is difficult due to the lack of experimental data for high-Z materials at the high plasma fluxes and fluences expected in ITER. This lack of data and understanding must be resolved if the plasma-surface interactions in the ITER divertor are to be understood. With the low hydrogenic solubility of high-Z materials, refratory metals in particular, exposures to high plasma flux densities can cause a super-saturation of the high-Z lattice that results in the formation of hydrogenic trap sites within the lattice. This process has been proposed as an explanation for hydrogenic retention results in laboratory experiments but it is not clear how this process extrapolates to very high fluxes achieved in linear devices and tokamaks. Some issues and open questions with extrapolations of hydrogenic retention in high-Z materials to higher flux densities are 1) trap production as a nonlinear function of plasma flux density; 2) impurities impacting the surface, or residing on the surface, and the role they play in trap formation; 3) inherently different hydrogenic retention for different high-Z materials; or some combination of the above. The Pilot-PSI experiment can reproduce the plasma conditions expected at the ITER divertor strikepoints (1×10^{24} D/m^{2}s, ∼2 eV, ∼5 MW/m^2). Using these plasma parameters, tungsten and molybdenum targets are exposed to deuterium plasmas in Pilot-PSI. The retention will be measured as a function of plasma fluence using ex-situ nuclear reaction analysis and thermal desorption spectroscopy. The purpose of this study is to determine how high-Z materials behave under high plasma flux exposures and to determine the role of target composition in the hydrogenic retention characteristics for high-Z materials. Initial nuclear reaction results for tungsten indicate very low retention rates ($\sim 1 \times 10^{-6}$ incident D ions retained) but no saturation up to incident fluences of $\sim 1 \times 10^{26}$ D/m^2.

FT/4-3Ra · Irradiation Effects on Reduced Activation Ferritic/Martensitic Steels -Mechanical Properties and modelling-

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Abstract: Although reduced activation ferritic/martensitic steels (e.g., Japanese F82H) are leading candidate structural materials for DEMO blanket and ITER-TBM, irradiation by 14 MeV fusion neutrons produces severe damage in the steels to cause large changes of their properties. Results recently obtained by irradiation experiments about the life limiting properties (e.g., ductile to brittle transition temperatures and reduction of ductility) are briefly introduced. Also, Models of macroscopic behaviors, such as post irradiation tensile and fatigue properties are introduced, as well as the simulation methods of microstructural development under irradiation at relatively low temperatures. The macroscopic models have been obtained by the detailed analysis of strain distribution in the tensile specimen during post irradiation testing and that of the softening behavior during cyclic loading on irradiated fatigue specimen. These macroscopic models and the simulation methodology are essential for design criteria development for in-vessel components.

FT/4-3Rb · Compatibility of Reduced Activation Ferritic/Martensitic Steels with Liquid Breeders

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Abstract: Liquid Li, molten salt Flibe and Li-Pb are candidate tritium breeding materials for advanced liquid breeder blankets of fusion reactors. Although each liquid breeder has pros and cons, compatibility with the structural materials is the common key issue. The Fe-Cr-W based Reduced Activation Ferritic/Martensitic Steels (RAFM) are regarded as the leading candidate for near and middle-term blankets of fusion reactors. Although some data are available on the compatibility of RAFM with Li-Pb, the data on liquid Li and Flibe are quite limited. The purpose of this paper is to characterize and access the compatibility of RAFM with the liquid breeders. Specimens used in this study are Fe-9Cr-2W based RAFM (JLF-1) normalized at 1323 K for one hr and tempered at 1053 K for one hr. Static compatibility tests were carried out in which the specimens were immersed into liquid Li or Flibe in isothermal autoclaves. Also carried out were compatibility tests in flowing liquid Li by a thermal convection loop. After the exposure tests, weight change and hardness of the specimens were measured in addition to microstructural observations. In the case of liquid Li, the corrosion rate increased with temperature. The corrosion was almost one order larger for the loop tests than for the static tests. Transformation from martensitic to ferritic phase and the resulting softening were observed in near-surface area of Li-exposed specimens. The chemical analysis of the specimens showed that carbon was heavily depleted from the specimens in which the phase transformation was observed, implying that the change was induced by the loss of carbon. In the case of Flibe, the corrosion loss was much larger in Ni than in JLF-1 crucibles. Both fluorides and oxides were observed on the surfaces. The key corrosion process of Flibe is the competing process of fluoridation and oxidation. Possible mechanism of the enhanced corrosion in Ni crucible is electrochemical circuit effect. It was suggested that the corrosion loss rate of RAFM by liquid lithium and Flibe is comparable with or smaller than that by Li-Pb at identical temperature. However, effects of the flowing rate needs to be accessed for further comparison.

FT/4-4 · Structural Materials for Fusion Power Reactors - the RF R&D Activities in 2006-2008

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Abstract: The results of the RF investigations (2006-2008) of initial and radiation physical and mechanical and activation properties of low activation structural materials for ensuring the development of test blanket modules DEMO in ITER (ceramic and lithium self-cooled) and fusion power reactor are listed. Two types of structural materials of Russian technologies are considered: the RAFM 12%-Cr heat resistant steel RUSFER-EK-181 (ceramic blanket) and low activation vanadium alloy V-4Ti-4Cr (lithium self-cooled blanket). Theoretical, modelling, experimental and technological investigations were performed for further widening of temperature, mechanical and dose application windows of the materials. The results of the application of the impact method and the acoustic (ultrasonic) non-destructive method to research the low temperature embrittlement of materials (measuring of ductile-to-brittle transition temperature) are listed. Solid solution state and phase-structure state were studied by methods of X-ray structural and electron microscopy analysis in the specimens of V-4Ti-4Cr alloy after their deformation in the temperature range up to 800°C. The results of investigation of microstructure, phase-structure states and mechanical properties of the steel RUSFER-EK-181 after irradiation in reactor BOR-60 (irradiation temperature $320 - 330$°C, doses up to 15 dpa) are presented. The results of relaxation of microstructure and elastic properties of the steel RUSFER-EK-181 and V-4Ti-4Cr alloy by acoustic (ultrasonic) method during the irradiation by protons with the energy 10 MeV are listed. Molecular dynamics simulation of the collision cascades and their influence on the generation of radiation damage areas was carried out in iron and vanadium crystals. The formation energy, the relaxation volume, the dipole-force tensor, the self strain tensor and strain fields of interstitial dislocation loops have been calculated by molecular statics in bcc iron. As the aggregate result of investigations, the values of the functional properties of the steel RUSFER-EK-181 and V-4Ti-4Cr alloy for ensuring the development of test blanket modules and fusion power reactor are presented.

FT/4-5Ra · Development of the JET Ion Cyclotron Resonance Frequency Heating System in Support of ITER

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Abstract: Three new improvements have recently been implemented to the existing JET Ion Cyclotron Resonance Frequency antennae to (a) increase power density and (b) develop matching systems that are tolerant to rapid coupling variations during Edge Localised Modes (ELMs); both of which are key developments for the future design of the ITER ICRF antenna. Firstly, 3dB couplers have been fitted to two antennae in 2004/5. Initial results from these antennae have highlighted the need for parallel development of arc detection and ELM-tolerant systems. Secondly, a new ITER-like antenna (ILA) was installed during 2007 to couple an ITER-relevant power density at the required plasma/antenna spacing using a close-packed array of straps. ELM tolerance is incorporated using an internal (in-vacuum) conjugate-T junction with each strap fed through in-vessel matching capacitors from a common vacuum transmission line. The mechanical engineering design challenges posed by the ILA were significant, given the need for the antenna to: withstand high disruption forces; operate for long pulse length; and achieve a demanding positional accuracy on the in-vessel capacitor tuning system. In addition, it has proved necessary to develop both existing and novel arc detection systems. Prior to installation, the antenna has been high power tested on a test-bed. First results, as well as related simulations, have highlighted the challenges inherent in developing a matching system for an ICRF antenna with closely spaced straps. Thirdly, an externally-mounted conjugate-T (ECT) system has been installed on antennae C and D during the 2006/07 shutdown. The detailed engineering design features of all three developments; the results achieved to date; and the implications for the ITER antenna design will all be reported.

FT/4-5Rb · Validation of the Load-Resilient Ion Cyclotron Resonance Frequency Antenna Concept on Tore Supra Plasmas

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Abstract: In the framework of the ion cyclotron resonance frequency (ICRF) heating development at CEA Cadarache, a prototype antenna based on the load-resilient electrical layout foreseen for ITER has been built. This prototype was recently tested in Tore Supra. The ITER-like electrical scheme has been validated during fast perturbations at the edge plasma. Clear load resilience properties are reported. Main conclusions and consequences to be learned for the development of ITER antenna are discussed.

FT/P2-1 · Mechanical Properties of Reduced Activation Ferritic/Martensitic Steels after High Dose Neutron Irradiation

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Abstract: The Reduced Activation Ferritic/Martensitic (RAFM) steels and their Oxide Dispersion Strengthened (ODS) variants are considered as primary candidate structural materials for fusion reactor breeding blankets with operating temperatures between 250 and 550°C. The irradiation performance of the European reference steel EUROFER97 was thoroughly studied in various low (IRFUMA up to 2 dpa, SUMO, SIWAS up to 9 dpa), mid (WTZ RUS 01/577, SPICE up to 15 dpa) and high dose (ARBOR 1 up to 33 dpa) irradiation programmes carried out in a wide temperature window from 60 to 450 °C. Although, the irradiation damage resistance of EUROFER97 is superior to that of conventional ferritic martensitic steels, the low temperature ($< 350 - 400$°C) irradiation hardening, accompanied by embrittlement and reduced toughness and ductility, did not reach saturation up to 30 dpa and remain the limiting factors for material application. High dose, up to 70 dpa/330°C, mechanical properties of RAFM steels were studied in ARBOR 2 irradiation experiment performed in BOR 60 at SSC RF RIAR. The yield stress and the Ductile-to-Brittle-Transition Temperature (DBTT) of EUROFER97 indicate saturation behaviour of low temperature hardening and embrittlement. Saturating behaviour of these quantities with irradiation dose can be qualitatively understood within a Whapam and Makin model. The thermal recovery experiments were performed on selected specimens from ARBOR 2. Annealing of irradiated (70 dpa/330°C) EUROFER97 at 550°C for 1 h lead to substantial reduction of the yield stress, resulting in a residual hardening of 62 MPa. A recovery heat treatment at 550°C for 3 h lead to further reduction of the yield stress and a residual hardening of only 37 MPa was observed. Similar to tensile properties, the impact properties are also significantly improved compared to the as irradiated state in thermal recovery tests. After annealing of 70 dpa irradiated EUROFER97 at 550°C for 3 h a residual DBTT shift of 48°C was obtained. Taking into account a need for continuous development and characterisation of materials and welded technologies to be potentially used in ITER-TBM (EUROFER and EUROFER ODS steels) and DEMO (ferritic ODS steels, tungsten alloys) and in order to increase basic scientific knowledge and to develop a materials data base for DEMO design, needs for specific irradiation campaigns are discussed.

FT/P2-2 · Microstructure and Tensile Properties in Reduced Activation $8-9\%$ Cr Steels at Fusion Relevant He/dpa Ratios, dpa Rates and Irradiation Temperatures

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Abstract: Helium production by irradiation in reduced-activation martensitic/ferritic $8-10$ Cr-WTaV steels and their impact on mechanical properties are primary subjects of fusion materials research. In this connection, three heats were melted with different contents of B-nat (natural boron) and the isotope B-10 on the basis of the chemical composition of EUROFER97. The alloy EUROFER97 had the lowest content of B-nat <10 ppm. Two experimental heats were alloyed with boron, one with $\sim$ 82 ppm B-nat and the other with $\sim$ 83 ppm B-10, and a third heat with a very high concentration of $\sim$ 1160 ppm B-10. Tensile specimens were fabricated of the heats and irradiated. The thermal neutron irradiation was carried out in 771 full power days up to $\sim$ 16.3 dpa, and was set between 250 and 450°C. The post-irradiation tensile tests showed the typical strengthening in the temperature range of 250 and 350°C. The achieved He contents by B-10(n,α)Li-7 reaction were $<$ 10 appm, $\sim$ 80 appm, $\sim$ 415 appm, and $\sim$ 5800 appm. Effect of the dpa and irradiation temperature was explained by microstructural changes, such as dislocation loops, α '-precipitates, and He bubbles. The lower B-10 containing specimens broke all ductile, but with a loss in elongation. The specimens with the highest concentration of B-10 broke always brittle. The phenomena observed in the microstructures and fractures were correlated with the mechanical properties. The significant implications of these results are obvious. Even at higher He levels of 400 appm, despite moderate uniform elongations, good levels of tensile ductility have been found in the entire temperature range of irradiation hardening (< 400°C). In addition, the results supported the hypothesis that at least up to $\sim$ 400 appm He, doping with B-10 can be used to simulate He embrittlement effects, at least as long IFMIF with fusion relevant neutron spectra is not yet available.

FT/P2-3 · Optimization of the Chemical Composition and Manufacturing Route for ODS RAF Steels for Fusion Reactor Application

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Abstract: As the upper temperature for use of reduced activation ferritic/martensitic (RAFM) steels is presently limited by a drop in mechanical strength at about 825 K, Europe, Japan and the US are actively researching steels with high strength at higher operating temperatures, mainly using stable oxide dispersion. In addition, the numerous interfaces between matrix and oxide particles are expected to act as sinks for the irradiation-induced defects. Main R&D activities are aimed at finding a compromise between good tensile and creep strength and sufficient ductility, especially in terms of fracture toughness. Oxide Dispersion Strengthened (ODS) Reduced Activation Ferritic (RAF) steels appear as promising materials for application in fusion power reactors up to about 1025K. Six different ODS RAF steels, with the compositions of Fe-(12-14)Cr-2W-(0.1-0.3-0.5)Ti-0.3Y_2O_3 (in weight percent), were produced by powder metallurgy techniques, including mechanical alloying, canning and degassing of the milled powders, and compaction of the powders by hot isostatic pressing, using various devices and conditions. The materials have been characterized in terms of microstructure and mechanical properties. Results have been analysed in terms of optimal chemical composition and manufacturing conditions.

FT/P2-4 · Progress in Flibe Corrosion Study Toward Material Research Loop and Advanced Liquid Breeder Blanket

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Abstract: Liquid breeder blanket is promising for DEMO and commercial fusion reactors. Flibe molten salt (BeF_2 + LiF) is a very attractive liquid breeder material. Flibe blanket has been designed for Force-Free-like Helical Reactor (FFHR). One of the critical issues for Flibe blanket system is corrosion of the structural material, low activation ferritic steel, by HF (TF in tritium breeder). The reduction of HF by a REDOX (REDuction and OXidation control) reaction in Flibe has been demonstrated in Japan USA Program JUPITER-II. The next milestone to the blanket development is material corrosion research with non-isothermal convection loop (material research loop) to predict corrosion rate in the blanket condition. In the present study, development of purification technique for Flibe, its scale-up, and fundamental corrosion experiments have been performed toward construction of the material research loop and development of Flibe blanket. High purity Flibe is required for materials corrosion test, because HF, O and metal impurities heavily affect corrosion rate in Flibe. Purification was started with the volume of 50 g, and then it was increased to 150 g. In the present study, especially Fe impurity level was successfully reduced to 5-70 wppm, compared with a previous research (260 wppm). A blanket structural material (JLF-1 steel, Fe-9Cr-2W-0.1C) and a loop structural material (316L SS, Fe-18Cr-8Ni-2Mo) were exposed to the purified Flibe at $500-600^\circ$C for up to 2003 hr in static condition. The weight loss of JLF-1 at 550°C was equivalent to 0.83 micron/yr in corrosion rate. While, corrosion rate for 316L SS at 500 and 600°C was 3.3 and 5.4 micron/yr. The corrosion rates for both JLF-1 and 316L SS were considered as acceptable level for the structural materials. A material research loop was designed to investigate the effect of flow velocity ($3-10$ m/s) and temperature difference (100°C) in the blanket condition. From the above results, 316L SS is applicable to the structural material. 200 kg or more Flibe is required for the loop, and is feasible with the scale-up of the above-established process. Development of components of the loop, such as valve, flow meter and impurity censor, has been started. Feasibility of the material research loop was verified by the above results and other related R& D.

FT/P2-5 · R&Ds on Li_2TiO_3 Pebble Bed for Test Blanket Module in JAEA

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Abstract: In JAEA, a Test Blanket Module (TBM) with water-cooled solid breeder has been developing for in-situ experiments in ITER. In the present paper, we report recent R& D activities for the TBM. In order to analyze thermo-mechanical behavior in the blanket including the pebble bed, the average thermal expansion coefficient of the Li_2TiO_3 pebble bed has been experimentally obtained. The obtained data enable us to conduct thermo-mechanical calculation accurately. For the breeding material, the chemical stability of Li_2TiO_3 has been improved by Li_2O additives. This leads to high temperature use and to high efficiency in the energy conversion in DEMO reactors. In the field of the neutronics, the activation foil method has been proposed to verify nuclear properties of the TBM. For the water-cooled blanket, tritium permeation into the coolant water is the key issue. The chemical densified coating on the F82H steel has been well developed and high permeation reduction factor is obtained. In accordance with the current design of the TBM and the refined analysis of tritium behaviour, tritium recovery system has been modified. Based on these achievements, test program in the TBM is discussed.

FT/P2-6 · Achievements of the Water Cooled Solid Breeder Test Blanket Module of Japan to the Milestones for Installation in ITER

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Abstract: As the primary candidate of ITER Test Blanket Module (TBM) to be tested under the leadership of Japan, Water Cooled Solid Breeder (WCSB) TBM is being developped. Six TBMs will be tested in ITER simultaneously, under the leadership of different coutries. To ensure the installation of reliable TBMs, it is necessary to show feasibility on the TBM milestones for installation in ITER. This paper shows the recent achievements toward the milestones of ITER TBMs prior to the installation, that consist of design integration in ITER, module qualification and safety assessment. With respect to the design integration, it is necessary to show the consistency with ITER design on time with ITER design progress, targeting the detailed design final report in 2012. Structure design of the interfacing components between the WCSB TBM structure and the interfacing components (Common Frame and Backside Shielding) that are placed in a test port of ITER has been developed. The design work also consists of procedures of fabrication and replacement of TBM, the consistency with ITER port structure and TBM interface structure, and the layouts of the auxiliary systems of TBMs including the tritium extraction system and water cooling system. As for the module qualification, it is necessary to show fabrication capability and the integrity of prototypical size mockup in corresponding operation condition before the delivery of the TBM to ITER. A real scale first wall mock-up was successfully fabricated by using Hot Isostatic Pressing (HIP) method by structural material of reduced activation martensitic ferritic steel, F82H. High heat flux test with real cooling water condition is planned using this mock-up. Other essential R& Ds for the WCSB TBM also showed steady progress on investigation of mechanical behavior of breeder pebble beds, development of advanced breeder/multiplier pebble, neutron measurement technology for TBM and purge gas tritium recovery technology. As for safety milestones, preliminary safety report in 2008 and final safety assessment of individual TBMs starting from 2010 are critical. According to the requirements in the licensing process, the preliminary safety report consists of source term identification, Failure Mode and Effect Analysis (FMEA) and identification of Postulated Initiating Events (PIEs), and safety analysis.

FT/P2-7 · Preliminary Safety Analysis for the Chinese ITER Dual-Functional Lithium-Lead Test Blanket Module

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Abstract: Safety analysis is a part of the ITER Test Blanket Module (TBM) design process to ensure that the TBM does not adversely affect the safety of ITER. The TBM is subjected to special safety requirements and shall not compromise the safety objectives, principles, requirements and guidelines of ITER. To get the license for TBM as a whole with ITER, the relevant safety analysis is required for each TBM system proposed by each party. The safety analysis for the Chinese Dual-Functional Lithium-Lead Test Blanket Module (DFLL-TBM) has been performed based on the latest DFLL-TBM design. In this paper, the following safety considerations, such as source terms, operational releases, accident sequence analysis and waste assessment, have been analyzed. Source terms including tritium production, neutron activation products, energy and hydrogen, etc. have been first considered. Tritium release to the environment in normal operation and Occupational Radiation Exposure (ORE) assessment during the TBM maintenance procedure represent the important safety aspect, which should be less than the environmental criteria and dose limits. The accident sequences analysis covering all the accident scenarios envisaged in incidents and accidents have been completed to assess the ultimate safety margins of the DFLL-TBM. Both deterministic approach and complementary systematic approach starting with FMEA (Failure Mode and Effects Analysis) studies have been adopted in the accidental analysis. An assessment of waste arising during operations and decommissioning has been made to investigate the feasibility of recycling and clearance for the materials in the DFLL-TBM. The preliminary results show that the DFLL-TBM system at normal operating conditions and under accident scenarios don't add additional safety hazards to ITER machine and can meet the ITER safety requirement and additional safety requirements for TBM system. Keywords: DFLL-TBM; Safety analysis; Source terms; Accident sequence

FT/P2-8 · Thermal-hydraulic Safety Study of Chinese Helium Cooled Solid Breeder Test Blanket Module for ITER

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Abstract: Chinese Helium Cooled Solid Breeder Test Blanket Module (CH-HCSB TBM) is designed to be tested in ITER and its aim is to validate its feasibility for DEMO fusion reactor. The thermal-hydraulic safety analysis has to testify that the TBM and its Helium Cooling System (HCS) will not influence the safe operation of ITER under normal and accidental conditions. In order to simulate the transient accidents and temperature evolution, the TBM and HCS are modeled using system code RELAP5/MOD3 and helium non-condensable gas model in the RELAP5/MOD3 is used to model the coolant. The Steady-State and three kinds of important Loss of Coolant Accidents (LOCA) have been analyzed. The Steady-State results indicate that the designed TBM inlet/outlet temperatures are obtained to be about 300/500 degree centigrade the design values. The temperature of First Wall beryllium armor is limited to the acceptable range. This means that the TBM structure and neutronics designs can satisfy the demand of thermal hydraulic design. The Ex-Vessel LOCA will induce the melting of First Wall beryllium armor after about 80 seconds of the LOCA initiation and some controlling measures have to be taken before melting. The pressurization of Vacuum Vessel induced by In-Vessel LOCA is within the allowable value of ITER design and will have little damage to TBM system and ITER. The In-Box LOCA would lead to pressurization of the TBM box, including all pebble beds and purge gas pipes to the system pressure of 8 MPa in about 2 seconds. So there must have a pressure relief for the blanket box, and at the same time the fast isolation of the Tritium Extraction System (TES) from TBM has to be taken to keep the TES safety.

FT/P2-9 · Overview of Design and R&D of Solid Breeder TBM in China

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Abstract: Testing TBM (Test Blanket Modules) is one of important engineering test objectives in ITER project. China is performing the TBM design and R&D based on Chinese development strategy of fusion DEMO. Ceramic breeder with helium-cooled test blanket module concept for testing on ITER will be one of the basic options in China. The current progress and status on design and R&D of CH HC-SB (Chinese Helium-cooled Solid Breeder) TBM are introduced briefly. A modified design based on 2×6 sub-modules arrangement, and 3-D global neutronics calculation of HC-SB TBM and related ancillary sub-systems, test strategy on ITER, and relevant R&D activities are summarized. An international collaboration plan on R&D and construction of related facilities of TBM are proposed.

FT/P2-10 · Neutronics R&D Efforts in Support of the European Breeder Blanket Development Programme

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Abstract: The EU fusion technology programme considers two blanket development lines, the Helium-Cooled Pebble Bed (HCPB) blanket with Lithium ceramics pebbles as breeder material and beryllium pebbles as neutron multiplier, and the Helium-Cooled Lithium-Lead (HCLL) blanket with the Pb-Li eutectic alloy acting both as breeder and neutron multiplier. The long-term strategy aims at providing validated engineering designs of breeder blankets for a fusion power demonstration reactor (DEMO). As an important intermediate step, the breeder blankets need to be tested in a real fusion environment as provided by ITER. HCPB and HCLL Test Blanket Modules (TBM) have been accordingly designed for tests in dedicated ITER blanket ports. The nuclear design and performance of the breeder blanket modules rely on the results provided by neutronics design calculations. Validated computational tools and qualified nuclear data are required for high prediction accuracies including reliable uncertainty assessments. Complementary to the application of established standard tools and data for design analysis, a dedicated neutronics R&D effort is therefore conducted in the EU. This includes the development of dedicated computational tools, the generation of high quality nuclear data and their validation through integral experiments. The recent neutronic design efforts have been devoted to the European DEMO reactor study comprising (i) Monte Carlo based pre-analysis for the dimensioning of the shielding system, (ii) the generation of a generic CAD based Monte Carlo geometry model, and (iii) performance analysis for HCLL and HCPB based DEMO variants. The recent focus of the validation effort is on neutronics TBM mock-up experiments. The first experiment of this kind was performed on a TBM mock-up of the HCPB breeder blanket. The follow-up experiment on a neutronics HCLL TBM mock-up is currently under preparation. Computational pre-analysis were performed to optimise the design of the mock-up configuration and provide first uncertainty assessments. The MCSEN code has been extended to enable, in an efficient way, the calculation of sensitivities by the track length estimator approach. A first successful test application to a complex tokamak configuration was recently performed for the HCPB TBM in ITER.

FT/P2-11 · Study of Radiation-Damaged Fusion Materials under High-Power Plasma Stream

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Abstract: Plasma-facing materials (PFM's) of a fusion reactor will be affected by high heat flux, fast particles and 14 MeV-neutron irradiation. All these factors are crucial for the lifetime of the reactor components. Fast neutrons produce a high level of radiation damage in materials during long operation of the fusion reactor (estimated value is up to hundred dpa). At the same time, PFM's will suffer from erosion induced by the plasma. While important data on the plasma erosion have been collected for non-irradiated materials, it is difficult to qualify PFM's at present taking into account radiation damage effect. This paper is devoted to the experimental results on radiation damage effect on erosion of materials under plasma impact. To obtain a high level of radiation damage, we simulated neutron irradiation by fast ions with energies of $1-60$ MeV accelerated on the cyclotron at Kurchatov Institute. Using this method we can accumulate the radiation damage equivalent to fast neutron effect at the dose of up to 10^{26} neutron/m^2 in a few days operation of the cyclotron. Both carbon materials and tungsten were taken for the study as the targets: MPG-8 (Russian graphite), SEP NB-31 (ITER PFM candidate) and W (99.95% wt). Irradiation of these materials on the cyclotron has been performed with 5 MeV carbon ions (for carbon materials) and alpha particles 3.4 MeV for tungsten. The experiments have been performed on the materials having accumulated $0.1-10$ dpa of radiation damage. Plasma erosion was studied on the linear plasma simulator LENTA. Irradiated samples were exposed to steady-state deuterium plasma at 100 eV (deuterium ions) to dose of up to 10^{25} ion/m^2. The microstructure modification was observed and comparison was made of damaged and non-irradiated materials. Enhancement of the erosion process was detected for the radiation-damaged materials. New experimental approach developed in this work to explore the plasma-facing materials for accounting of neutron effect and the results obtained appear to be important for the further studies of the combined plasma and neutron effect on fusion PFM's .

FT/P2-12 · Studies and Developments of Tritium Recovery System for Blanket of Fusion Reactor in Japan Atomic Energy Agency

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Abstract: Tritium technology has reached the level where they allow us to design the main fuel cycle of ITER. On the other hand, for the blanket tritium recovery system, a series of fundamental studies have still been carried out even though the system is essential to realize the fusion reactor from the viewpoint of the fuel production. In the case of a water cooling solid breeder blanket, the blanket tritium recovery system will be composed of three processes: tritium recovery from the helium sweep gas as hydrogen, that as water vapor and tritium recovery from the coolant water. For these processes, the present authors have proposed a set of advanced systems, and have proved that the proposed systems would be feasible for a DEMO reactor. For tritium recovery as hydrogen, an electrochemical hydrogen pump with a ceramic proton conductor has been proposed. In this work, the correlation between the proton concentration in the ceramic and the hydrogen pressure in the gas phase has been investigated to describe the proton conductivity specifically. And then, it has been shown that the partial pressures of hydrogen and water vapor control the proton concentration in the ceramic as a key parameter of the system. A ceramic electrolysis cell has been proposed to process the tritiated water vapor. In this work, the present authors have developed a new electrode that contained the cerium oxide, and it has shown a fairly large current density. The performance of the ceramic electrolysis cell would be largely improved by using this electrode. For tritium recovery from the coolant water, the reduction of the processing water by tritium condensation is necessary. The present authors have studied about the fixed-bed adsorption process of synthesis zeolite, and the development of the adsorbent that can release water vapor easily is required. In this work, the effect of the silica/alumina ratio of the adsorbent on the separation factor was examined. NAY10.0 (faujasite-type, silica/alumina = 10.0) showed quite unique characteristics for the water adsorption and desorption. The system size is expected to decrease significantly by using NAY10.0.

FT/P2-13 · Assessment of Radiation Streaming Effect upon Radiation Loads to Inboard TF-coil of DEMO Utilizing HCLL Blanket

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Abstract: Neutron and photon calculations have been performed to assess the effect of radiation streaming through the gaps between blanket modules upon the radiation loads to the super-conducting TF-coil at the inboard side. These include the fast neutron fluence to the superconductor, the peak nuclear heating in the winding pack, the radiation damage to the copper stabilizer and the radiation dose absorbed by the Epoxy resin insulator. Three-dimensional Monte Carlo radiation transport code MCNP4C and a suitable 20 deg.-sector model of DEMO reactor, developed by FZK, have been employed in the analysis. The model is based on PPCS AB concept employing HCLL blanket. A 4000 MW fusion power and 40 fpy of plan operation have been assumed. The poloidal profiles of radiation loads in respect to the torus mid-plane have been investigated in several variants of the gap widths and compositions of additional shield plugging the gaps. It was found that the design limit of all the responses considered could only be met if the toroidal and poloidal gaps are plugged by LTS material, the later being made of borated steel, WC and water coolant.

FT/P2-14 · Chemical Forms of Tritium in Be Pebbles after Different Treatments

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Abstract: Beryllium in a form of pebbles is foreseen as neutron multiplier in the Test Blanket Module for ITER and also in DEMO fusion devices. In order to understand the features of tritium release from the pebbles under operational conditions, it is very important to know chemical forms of localized tritium as they have different mechanisms of diffusion. The paper focuses on the latest results achieved on analysis of chemical forms of tritium in neutron-irradiated beryllium pebbles (EXOTIC 8-3/13 and BERYLLIUM) before and after different conditions of treatment. The pebbles differ by method of production, by irradiation conditions and content of impurities, but they have the same grain size. An annealing of pebbles was performed in the special radiation thermo magnetic rig based on a linear electron accelerator. The chemical forms of tritium were analyzed using a lyomethod. The dissolution of samples was performed in a special system in the presence of only 2 M sulfuric acid and in the presence of 2 M sulfuric acid together with a scavenger of atomic tritium – 0.5 M sodium dichromate. The chemical forms of tritium in pebbles were investigated after they annealing at temperature (490-1023 K) in the presence of ionizing radiation (5 MeV fast electrons, the dose rate 14 MGy/h) and magnetic field (1.7-2.35 T) separately or simultaneously. In both the types of pebbles, the three forms of tritium (ionic, atomic and molecular) were determined. In the untreated EXOTIC 8-3/13 Be pebbles, tritium was in an ionic $10-15\%$, in an atomic $20-25\%$ and in a molecular form $60-70\%$. After the treatments at 553K, their abundances had the following values in an ionic form $11-25\%$, atomic $50-60\%$ and molecular $30-35\%$. After the treatments at 770K, they had the following values ionic form $10-14\%$, atomic $40-45\%$ and molecular $30-55\%$, but after annealing at 1023K only ionic form of tritium was detected. In untreated and treated pebbles of BERYLLIUM, the same chemical forms were determined, but their abundance ratios were different.

FT/P2-15 · Main Results and Prospects of Lithium Capillary-Porous System Investigation as Tokamak Plasma Facing Material

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Abstract: Concept of lithium fusion reactor has been offered in FEC-16 and lithium Capillary-Porous Systems (CPS) is the key element of its realization. Experiments in T-11M tokamak with lithium CPS based rail-type limiters have been started in 1998 for the purpose to prove the compatibility of lithium CPS with tokamak boundary plasma. Further stage was the beginning of liquid limiter tests in Italian tokamak FTU in 2005. The main reasons of this investigation were confirmation of CPS ability to lithium surface renewal and lithium confinement during normal plasma operation and in disruption, possibility for withstanding of high power flux on the limiter surface without damage in real tokamak condition with ohmic plasma heating mode and with additional heating. Litization effect on plasma discharge parameters of all metallic and carbon-free tokamak has been investigated. Experiments on inner-wall litization of graphite containing camera of T-10 tokamak with application of CPS based unit have been also started in 2006. All this experiments have allowed to solve the problems in MHD stability of liquid lithium film in tokamak conditions, study of lithium CPS compatibility with tokamak boundary plasma, demonstrate the positive lithium effect on plasma parameters and determine the next steps on further activity in this area. At the same time the promising results of experiments on the lithium CPS serviceability and possibility for withstanding of steady-state (up to 3 hours) high power flux ($1-10$ MW/m2) and plasma disruption effect has been demonstrated. The progress in development of lithium technology allows deciding the problems in the development of projects of Steady-State Operating Lithium Limiters (SLL) for FTU and T-15, lithium divertor for tokamak KTM. The first stage of this activity is development and experimental study at power flux up to 10 MW/m^2 of the single-element prototype of SLL/divertor with systems for surface temperature stabilization in the range of $350-550^o$ C and controllable lithium supply. SLL/divertor prototype structure, operating parameters and new tungsten-based CPS are presented. Safety analyses and critical aspects of prototype technology are considered. Investigations on structure material compatibility with lithium in DEMO-type tokamak conditions have allowed for proving the possibility of lithium application in fusion reactor.

FT/P2-16 · A–C:H Film Removal in H_2 and H_2/N_2O Glow and Afterglow Discharge

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Abstract: In ITER project CFC materials are proposed as divertor targets for exhaust D/T plasma. The main advantage of the CFC materials is an ability to take a high heat load and the serious shortcoming is chemical erosion susceptibility under interaction with hydrogen plasma when hydrocarbon molecules and radicals are formed. Carbon and hydrogen isotopes joint deposition may be a principal mechanism of "dead" tritium retention. Thereupon, it is necessary to develop both methods for chemical erosion decreasing of carbon materials and techniques for T-C film removing. The aim of the work is to develop an effective gas-discharge technology for removal of hydrocarbon deposits from remote parts of divertor plasma. The experiments have been shown that the main a-C:H film gasification product is methane in the afterglow region (hydrogen discharge). In film contact directly with the discharge plasma, C_2H_x and C_3H_x hydrocarbons appeared in the erosion products. The hard a-C:H film erosion rate in the afterglow region decreased exponentially with distance from the atomic hydrogen source. The disadvantage of the hydrogen discharge is low erosion yields at $320-420$ K ($Y \sim 0.001$ at.C/at.H) in the remote parts from divertor plasma. For efficiency enhancement of hydrocarbon film removal the H_2/N_2O mixture containing oxygen component has been proposed. In the indicated medium for removal of the hard a-C:H film (1 μm thick) located in the afterglow region, i.e., in the shadow of plasma, it needed about an hour at $420-500$K and N_2O partial pressures of $3-24$ Pa. The technique suggested is one of possible variants of an effective gas-discharge technology for hydrocarbon deposit removal from the remote parts of ITER divertor plasma. However, for using of plasma cleaning (on the base of H_2/N_2O mixture) of large machines from carbon films redeposited it is necessary to carry out a systematic investigation of possible side effects connected with oxidizing and oxihydrogenating of divertor material surface layers.

FT/P2-17 · Evaluation on Failure Resistance to Develop Design Basis for Quasi-Ductile Silicon Carbide Composites for Fusion Application

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Abstract: Silicon carbide composites (SiC/SiC) are promising candidate materials for fusion blanket with advanced features such as high thermal efficiency. Of many composite types, a nano-infiltration and transient-eutectic-phase sintered (NITE) SiC/SiC composites, as well as chemical-vapor-infiltration (CVI) SiC/SiC composites, is believed to be viable because of excellent baseline mechanical properties and proven radiation stability of microstructure and strength under certain irradiation conditions. With a completion of the "proof-of-principle" phase, the R&D on SiC/SiC composites is now shifting to the more pragmatic phase of material data-basing and development of design basis. This paper will provide a present status in development of design basis for SiC/SiC composites as a structural application. Composites exhibit "quasi-ductility" in fracture, which is totally different from the ductility of metals since this "quasi-ductility" occurs as a result of cumulative accumulation of irreversible permanent damages such as interface debonding, fiber pullouts with friction at the fiber/matrix interface, and fiber breaks. From this aspect, the damage accumulation behavior of SiC/SiC composites was first evaluated in this paper with a final goal to develop the design basis for generation of practical database with a direction to use this class of composites for structural application. Recent fatigue test result demonstrated that the latest NITE-SiC/SiC composites are more crack resistant compared with the conventional low-stiffness, porous SiC/SiC composites. The improved crack resistance of NITE-SiC/SiC composites thus results in better helium gas tightness. In contrast, detailed crack extension behavior of NITE-SiC/SiC composites was evaluated by the single-edge notched-bend technique. A developmental analytical model based on the non-linear fracture mechanics, which can separately discuss the effect of irreversible energies such as interfacial friction, thermal-residual strain energy and fiber breaks, provides a non-linear fracture toughness for NITE-SiC/SiC composites of ~ 4 kJ/m^2, which is an actual energy consumed during macro-crack extension. Besides it is worth noting that, with a fact of the notch-insensitivity and very minor size effect on the failure behavior, a stress criterion is suggested in failure of SiC/SiC composites.

FT/P2-18 · Development and Operation of an ITER relevant Inspection Robot

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Abstract: Robotics and remote operations are one of the main challenges for ITER and future fusion reactors; in particular, first wall inspection and treatment in between pulse could be required for better Tokamak exploitation and for safety purpose. Since 2002, CEA laboratories have started developments to demonstrate the feasibility and reliability of a light ITER relevant inspection robot. A multipurpose carrier was manufactured with the aim to be deployed in the plasma vessel of a tokamak without breaking the Ultra High Vacuum and temperature conditioning. The robot called Articulated Inspection Arm (AIA) is an 8 meter long multi link carrier with 5 modules of 160 mm diameter. With a payload of up to 10 kg, the poly-articulated arm total weight is about 150 kg and can be introduced through a remarkably small port of 250 mm diameter. A fatigue test at atmospheric pressure and under representative loading was conducted in 2005 to qualify the module concepts. Vacuum and temperature tests campaign on the first prototype module was also carried out in 2006 and 2007. Particularly, cycles of motion and temperature were performed during several weeks; 120°C during operating and 200°C for outgassing phase. The first diagnostic built to be plugged at the front head of the carrier is the viewing system able to make accurate visual inspection of Plasma Facing Components under darkness conditions. In September 2007, after an integration phase and command control devising, the robot with its vision diagnostic plugged in has realized its first complete deployments inside Tore Supra's plasma vessel under atmospheric conditions. To complete the demonstration, deployments and operations of the AIA robot will be performed in the plasma vessel of Tore Supra under ITER relevant conditions (10^{-6} Pa and 120°C) in the first semester of 2008. In parallel, new tools are being developed by CEA, which will be connected to the robot for tokamak in-vessel components diagnostic: a leak detection system and a laser compact system for carbon co-deposited layers characterizations or treatments. Besides bringing a relevant proof of principle, lessons learned on Tore Supra with such a multipurpose robotic device will give very helpful information for future ITER remote handling activities.

FT/P2-19 · Advancement of Nuclear Analysis Method for ITER

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Abstract: Advancement of nuclear analysis, which means improvement and validation of radiation transport codes and nuclear data libraries, is a key issue for ITER licensing. We have developed an automatic conversion system from CAD data to MCNP input data in order to easily make precise MCNP input data even for complicated ITER structures. We also performed a narrow slit streaming experiment for analysis validation on streaming effects at JAEA/FNS. We developed an automatic conversion system from three-dimensional CAD drawing data to the Monte Carlo code MCNP input data under the ITER/ITA Task. This system consists of a void creation program (CrtVoid) and a conversion program (GEOMIT) from CAD drawing data to MCNP input data. First CrtVoid creates void region data. CrtVoid automatically divides the overall region to many small cubes, and the void region data can be created in each cube. Next GEOMIT generates surface data for MCNP from CAD data including the void data generated with CrtVoid. These surface data are connected, and cell data for MCNP input data are generated. By using this system, geometry data in MCNP inputs are automatically produced, which leads drastic time- and labor-saving. We applied this system to ITER 40 deg. benchmark model and Japanese test blanket module for ITER. The calculation with the generated input showed adequate neutron flux and nuclear heating distributions. There are slits between the vacuum vessel port walls and the port plugs in ITER, providing possible radiation streaming paths. In order to verify the nuclear analysis for such slit streaming, we have carried out a narrow slit streaming experiment at the Fusion Neutronics Source (FNS) facility in JAEA under the ITER/ITA Task. The experimental assembly with a slit of 2 cm in width, 195 cm in depth and 3 cm offset at 56 cm depth from the surface was constructed with iron blocks, which was located 20 cm from the D-T neutron source. Fission rates of U^{238} and U^{235} were measured along the slit with micro-fission chambers. Fission rates were obtained up to 116 cm in depth successfully. This experiment was analyzed by using the MCNP-4C with the nuclear data library FENDL-2.1 and other libraries. The calculation results with MCNP-4C agreed well with the measured ones. It is confirmed that the present ITER nuclear analysis method can also give accurate results for such slit streaming.

FT/P2-20 · Technology Progress in Real-Time Control of MHD Modes at RFX-mod

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Abstract: The RFX-mod reverse field pinch (RFP) experiment applies active control on the MHD instabilities by acting on the magnetic field configuration at the boundary. These studies allow acquiring expertise on the control of MHD modes that is also relevant for high β discharges in tokamaks. To control the MHD activity, 192 saddle coils are installed to cover the whole torus surface. A large number of field and current measures are acquired and processed in real-time ($3 \times 192 = 576$ channels) [1]. Impressive results have been achieved by exploiting the real-time MHD mode control system, especially on MHD mode suppression at the boundary, toroidal loop voltage reduction, particle and energy time increase and duration of the plasma discharge [2]. The paper discusses the technological enhancement developed in the last two years. Two new techniques have been introduced recently. The first one is referred to as Clean Mode Control (CMC). It consists in the correction of the aliasing, due to the sidebands produced by the discrete grid of coils, which determines a systematic error on the Fourier analysis on the measurements [3]. As the CMC correction requires the simultaneous availability in the same real-time station of all acquired channel, a significant upgrade of the architecture of the real-time feedback system has been necessary. The second technique consists in the introduction of a Multiple Input Multiple Output (MIMO) approach to substitute the Single Input Single Output model used so far, whose dynamic behavior is not satisfactory. The MIMO approach relies on a full electromagnetic model of the coil and sensor system in the presence of passive structures. The MIMO controller has been developed following the concept of model decoupling [4]. To compute the new algorithms without exceeding the real-time constraints, faster CPUs have been introduced and the data throughput in the real-time network has been optimized.

[1] A. Luchetta et al., proc. 21th IAEA Fus. Energy Conf., Chengdu, China, 16-21 Oct. 2006 [2] S. Martini et al., Nucl. Fus. 47 783 (2007) [3] P. Zanca et al., Nucl. Fus. 47 1425 (2007) [4] A. Soppelsa et al., Fus. Eng. Des., to appear

FT/P2-21 · Definition of Workable Acceptance Criteria for the ITER Divertor Plasma Facing Components through Systematic Experimental Analysis

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Abstract: Plasma Facing Components (PFCs) which will be installed in the ITER Divertor require a heat flux removal capability in the range $5-20$ MW/m^2 in thermal steady-state conditions. Experience shows that the most critical part of a PFC is the armour to heat sink joint, therefore, maximum dimensions of an acceptable defect at the interfaces of bonding appeared to be a relevant criteria with regards to thermomechanical fatigue. A comprehensive study was thus launched in order to define acceptance criteria with regards to thermomechanical fatigue of the ITER Divertor PFCs. This study, which includes the manufacturing of samples relevant with the Divertor design with calibrated artificial defects, their non-destructive examination and their related fatigue testing is reported in this paper. As a first step, metallic samples (W armour) with calibrated defects were inspected with conventional Non Destructive Examination (NDE) technique whereas the composite ones (CFC armour) were inspected by transient infrared thermography NDE technique (so-called SATIR technique). Detection of the imperfections on CFC monoblock geometry was found reliable with a limited uncertainty. In particular it was found that defect with an extension above 30° could be detected at the interface CFC/copper, with a SATIR inspection on the 3 sides of each monoblock. Secondly, the samples were high heat flux tested following a program relevant with the lifetime of the Divertor PFC into the ITER machine. For each artificial defect, propagation or stability during the fatigue testing was assessed. It was observed that defects detectable with US NDE or SATIR techniques appeared to be below the threshold of propagation during fatigue experiments. This allows to be confident in the capability of commissioning components reaching the required design values of the ITER divertor PFCs.

FT/P2-22 · Development and Demonstration of Remote Experimental System with High Security in JT-60U

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Abstract: Remote experimental system with high network security has been developed in JT-60U. The remote experimental system is produced by personal authentication with a digital certificate and encryption of communication data to protect the JT-60U supervisory control system against illegal access. Remote experiment in JT-60U has been successfully demonstrated from Kyoto University (Japan) in 2006 and internationally from IPP Garching (Germany) in 2007. The validity of the system was obtained. Results are great advances towards remote experiments in ITER.

FT/P2-23 · Alternate Concepts for Generating High Speed DT Pellets for Fueling ITER

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Abstract: There is a growing concern that the velocity of the DT-ice pellets being proposed to fuel ITER will not have the velocity to effectively penetrate through the pedestal of the plasma edge. For ITER, the present pellet-fueling concept has DT-ice pellets traversing through a guide tube to reach the high field side (HFS) launch location. The forces on the ice pellet as it goes around bends causes the pellets to disintegrate at velocities above 300 m/s. Thus, in order to achieve deeper penetration the pellet velocity has to be increased. To achieve this, either the pellet needs to have its velocity increased after the pellet has passed the last bend, or the pellet has to be made stronger to survive the increased forces higher velocities create. Previously Parks and Perkins have proposed a novel concept to acceleration DT-ice pellets using microwave power from MW gyrotrons to develop high-pressure gas, by absorbing the microwave in a composite "pusher" medium attached to the backside of the pellet. This gas boost is created in the last meter of the guide tube after the last bend. The basic concept of the microwave-based accelerator was validated in the lab by measuring the losses in a section of fundamental waveguide loaded with paraffin and various mixtures of $1-2$ micron zinc powder. The length of the paraffin in the waveguide was adjusted to have all samples contain the same total amount of Zn powder. It was demonstrated that for constant number of particles in the waveguide the absorption is found to be constant. The next test is to actually heat a sample of naphthalene seeded with $1-2$ micron Zn powder. For this test a 500 kW pulse from a 110 GHz gyrotron will be pulsed into a naphthalene slug 4 mm diameter, 25 mm long for $1-2$ ms. The pressure and temperature of the generated gas will be measured and compared to theory. Results of these tests will be reported. Methods have also been evaluated to increase the strength of the ice pellets by either encapsulating the pellet inside a solid shell of either metal or plastic. Or, to stiffen the ice by integrating it into a plastic foam sphere. Analysis indicates a plastic shell with $< 5\%$ of the volume of the pellet will increase the strength of a DT Ice pellet by a factor of 100, tests will be performed to validate the modelling.

FT/P2-24 · Experimental Results of Series Gyrotrons for the Stellarator W7-X

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Abstract: A powerful ECRH system with 10 MW RF power at 140 GHz and CW operation is foreseen for the stellarator W7-X, being under construction at IPP Greifswald. The RF power will be delivered by 10 gyrotrons, each capable of 1 MW, CW. Nine gyrotrons are being manufactured by Thales Electron Devices (TED), Vélizy, France, one gyrotron was produced by CPI, Palo Alto, CA. Testing of the TED gyrotrons is performed at the test stand at FZK (pulse duration 3 min) and the final tests are performed at IPP (pulse length 30 min). Both, the first TED series gyrotron and the CPI gyrotron have passed the acceptance tests successfully. The acceptance tests of the TED series gyrotrons are ongoing. The RF output beam quality of all tubes tested so far is at a constant high level of about 97% in the Gaussian beam. In short pulse operation the gyrotrons have achieved the specified parameters. However, for long pulse operation the performance decreases due to the occurrence of parasitic oscillations which are assumed to be excited by the electron beam in the beam tunnel close to the cavity. Experimental results of this effect and possible modifications of the beam tunnel geometry will be discussed.

FT/P2-25 · Development of 1 MW Gyrotron and Progress of ECH System for the GAMMA 10 Tandem Mirror in Tsukuba

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Abstract: An ECH is a reliable tool for the plasma heating, current drive and plasma control in the magnetic confinement fusion devices. It is especially essential for the tandem mirror machines to get high confining potential and high electron temperature. Power scaling of the ion confining potential predicts that further higher potential formation would be expected corresponding to the plug ECH power increase. Along this line, GAMMA 10 has started the ECH system upgrade program in collaboration with JAEA and NIFS High power gyrotrons with TE4,2 cavity at 28 GHz and with TE18,6 cavity and a diamond window at 77 GHz have been developed for GAMMA 10 and LHD in the joint program of NIFS and University of Tsukuba. The maximum outputs of 570 kW at 28 GHz and 1.1 MW at 77 GHz were obtained corresponding to each design. The calculated power profile at the window and the experimentally obtained burn pattern of the 77 GHz gyrotron agree well. The operations of 460 kW for 5 s and 810 kW for 3.5 s were achieved in the first test gyrotron at 77 GHz, which is the first high power-long pulse result of 77 GHz tube. The experimental simulation of the effect of the stray RF in the 28 GHz tube indicates the stray RF is the one of the major causes limiting gyrotron performance. Installation of the polarizer in the transmission line enhanced the performance of the ECH system in GAMMA 10, that is the first result which clearly showed ∼100% X mode excitation is a key to design the efficient fundamental ECH system in mirror devices.

FT/P2-26 · Long Pulse/High Power ECRF System Development in JT-60U

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Abstract: In the recent gyrotron development in JT-60U ECRF system, 1.5 MW for 1 s of the world highest power oscillation (> 0.1 sec) has been achieved at 110 GHz. In addition to the carefully designed cavity and collector in view of thermal stress, an RF shield for the adjustment bellow, and a low-loss DC break enabled this achievement. Moreover as a development step to realize a reliable ECRF system in future fusion experiments, long pulse demonstration of 0.4 MW, 30 s injection to the plasma has been achieved. It has been confirmed that the temperature of cooled components are saturated and no evidence of damage were found in the waveguides and antenna without forced cooling. As a forced cooling antenna for longer pulse in future, an innovative antenna having relatively wide range of beam steering capability with linearly-moving-mirror concept has been designed. Beam profile and mechanical strength analysis shows the feasibility of the antenna.

FT/P2-27 · Development of Long Pulse Neutral Beam Injector on JT-60U for JT-60SA

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Abstract: Long pulse neutral beam injectors (NBIs) are being developed to realize 100 s D^0 injection at 34 MW in total for JT-60SA. Significant progress has been made in the development of the negative-ion-based NBI (N-NBI) where a 10 MW injection is required. In the N-NBI, a long pulse injection up to ∼ 30 s was achieved at 1 MW using one ion source. The time response of the neutron flux from the plasma indicates constant D^0 power without degradation. In the R&D's, a field shaping plate (FSP) for tuning the beamlet steering angle was newly designed and tested in the JT-60U negative ion source. The FSP succeeds in the reduction of the grounded grid power loading to 6% of the drain power in the acceleration power supply. This power loading is an allowable level for full D^- ion beam for JT-60SA. Moreover, the power loading of the electrons ejected from the JT-60U negative ion source into the beamline is measured to be as small as 2.6% of the drain power, and can be acceptable for JT-60SA.

FT/P2-28 · Development of an Energetic He^0 Beam Injector for Fusion Application

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Abstract: For application to α-particle measurement on ITER, a ground state He atomic beam produced through auto-detachment of a $1-2$ MeV He^- ion beam of 10 mA region is required. The He negative beam can only be produced from the He positive ion beam through two-step electron capture processes in an alkali vapour cell. (1) We have developed a full-size strongly focusing He positive ion source. The required beam current above 2 A was obtained at a beam energy of 20 keV, which is suitable for He negative ion production. The 1/e-holding beam profile half-width was about 15 mm at the beam waist, while the extraction area was 100 mm in diameter. A full-size He^+ ion source. (2) A test stand for ground-state He atomic beam production was constructed and an efficient conversion factor, higher than 1% from He^+ to He^- in an lithium cell was observed. Moreover, strong focusing properties of the He^- beam were confirmed. (3) Several methods to measure the meta-stable fractions were developed. The present research indicated that a confined alpha particle measurement system using an energetic He^0 beam produced from He^- and accelerated to 1˘2 MeV is promising. In addition, improvement of the S/N ratio on ITER can be accomplished by subtraction of neutron-induced background noise using beam modulation, or/and by additional shielding and bending of the secondary particles after stripping at the detectors.

FT/P2-29 · The Physics of Design and Operation of High Power Neutral Beam Injection Ducts – Extrapolating JET Experience to ITER Situations

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Abstract: Neutral beam injection systems have proven to be the single most effective form of heating for tokamak plasmas. Typical beam pulse lengths are of the order of ten seconds and the major limitation to increased pulse length in multi-megawatt beamlines is the effect of re-ionised neutral particles in the restricted drift space, or "duct", connecting the beamline to the tokamak vessel. These particles are deflected and frequently focused by the stray magnetic field of the tokamak and can produce significant power density on the walls of the duct. In JET the power density due to re-ionisation can reach ten megawatt per square metre and is the main limitation to beam pulse length. The effect of the re-ionised power is to cause local heating of the duct wall and evolution of gas trapped within the wall material. This raises the pressure in the duct, causing further re-ionisation of the beam and hence increased wall heating. Unchecked, this process can lead to complete re-ionisation of the beam and possible structural failure of the duct wall. A new model is presented that describes an effective source rate of excess gas evolved from the wall in terms of the surface temperature and area subjected to heating. This approach reduces the predicted dependency of duct pressure on beam flux relative to conventional models parametrised by an ion-induced desorption coefficient and is validated by comparison with measurements from the 80 keV and 130 keV JET beamlines over similar power ranges. In conjunction with a particle trajectory re-ionisation code to determine the size and power loading of the affected area, a self-consistent description of the duct pressure balance may be determined for a given heat-transfer characteristic at the wall. This can be directly applied to the design of systems for ITER such as the duct liner and the electrostatic residual ion dump panels. The time response of the duct pressure can be used to establish the mechanism by which gas is released. It is shown that only the percolation of occluded gas within the structure of the wall can account for the timescale over which the pressure is observed to rise and the quantity of gas released. These occlusions occur as a result of localised damage within the wall material and hence it follows that gas evolution will be a function of the ageing process of future systems.

FT/P2-30 · Focused Neutral Beams with Low Chaotic Divergence for Plasma Heating and Diagnostics in Magnetic Fusion Devices

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Abstract: A series of neutral beam injectors has been developed in the Budker Institute of Nuclear Physics for plasma heating and diagnostics in modern fusion devices. Ion optical system of these injectors is optimized to produce ion beams with low angular divergence. In order to provide beam focusing, the grids are formed to be spherical segments. Such geometrically focused neutral beams are particularly advantageous for plasma diagnostics when high spatial resolution is required. Another application of these beams is plasma heating in the machines with narrow ports through which only small size, high power density beams can be transported.

FT/P2-31 · Beam-plasma Interaction in the ITER NBI

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Abstract: In the neutraliser of the ITER Neutral Beam Injector, a beam of negative ions (D^-) of about 1 MeV passes through a structure filled with deuterium gas, where negative ions are mainly converted to fast atoms. Once that the beam is neutralized no further optical correction is possible, i.e., transport from the neutraliser to the confinement chamber is ballistic. Because of this, the transport through the neutraliser determines ultimately the geometrical properties of the neutral beam. The ionization of the buffer gas deuterium filling the neutraliser induced by the D beam creates a rarefied and low temperature plasma. This plasma can screen the electrostatic well of the D beam and, consequently, affect the properties of the extracted beam and the energy transport to the neutraliser walls. Moreover, the plasma can eventually escape from the neutraliser and move back in the accelerator, toward the accelerating grids and the negative ion source. OBI-2 code (Orsay Beam Injector – 2 dimensions) simulates the beam propagation and plasma formation through the neutraliser. Particle-particle and particle-wall collisions are treated using a Monte Carlo collision approach and the plasma is treated via Particle-in-Cell. The geometry of the neutraliser used in the present work has been chosen as cylindrical, which is not the real configuration of the neutraliser plates, but remains realistic in terms of the volume to surface aspect ratio. Hence, it is possible to follow the residual plasma formed between the symmetry axis and the wall but also forward and backward the beam axis Simulations show that the secondary plasma effectively screens the beam space charge preventing beam radial expansion due to Coulomb repulsion between beam ions. At the steady-state a quasi-neutral plasma built-up in the neutraliser, with a density ten times higher than the beam one and the beam is well focused. The plasma ions created in the neutralizer can create a backstream current impinging the accelerator grid, due to the imperfect compensation of the beam spatial charge between the accelerator and the neutralized, and the leakage field coming from the accelerator. On one hand, the presence of these ions enhances the beam focusing. On the other hand, plasma ions can cross the accelerator grids and be accelerated towards the negative ion source.

FT/P3-1 · Commissioning Results of the KSTAR Cryogenic System

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Abstract: The cryogenic system for the KSTAR superconducting magnet is under commissioning. It consists of cold box, distribution box (DB) and cryogenic transfer lines. The cold box and DB#1 provides 600 g/s supercritical helium to cool the superconducting (SC) magnet, their SC bus-lines, and the magnet support structures, and provide 17.4 g/s liquid helium to the current leads, and also supplies a cold helium flow to thermal shields. The cooling power of the cold box at 4.5 K equivalent is 9 kW which is extracted by 6 turbo-expanders. The DB#1 include 49 cryogenic valves, 2 supercritical helium circulators, 1 cold compressor, and 7 heat exchangers immersed in a 6 m^3 liquid helium storage. The main duties of the DB#2 are the relative distribution of the cryogenic helium among the cooling channel of each KSTAR cold component and the emergency release of over-pressurized helium during abnormal events such as quench of SC magnets. After individual commissioning, the system will be integrated and cooled down with KSTAR device. In this paper, the construction and commissioning results of the KSTAR cryogenic system will be introduced. In addition, we will present the cool-down results of the KSTAR device.

FT/P3-2 · Commissioning Results of the KSTAR Superconducting Magnet System for the First Plasma Operation

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Abstract: The KSTAR device is a tokamak with a fully superconducting (SC) magnet system, which enables an advanced quasi-steady-state operation. Initial operation of the KSTAR SC magnet system has been accomplished for the first plasma discharge operation. The SC magnet system consists of 16 toroidal field (TF) coils and 14 PF coils, magnetic structures, SC buslines, current leads, and monitoring and protection system. After the completion of the cool-down, magnet system investigation has been followed including the insulation test and interlock test between power supplies and control system. The measuring of SC coil parameters such as joint resistance, flow uniformity per channel, mechanical stress and displacement is followed according to the current excitation and discharge. The current waveforms for the loop voltage generation and field null formation have been simulated and are controlled by a plasma control system which has been developed in collaboration with General Atomic, USA. The magnetic field inside vacuum vessel has been measured before and after magnet cool-down to measure the magnetic field distortion due to the Inoloy908 material which is used as jacket material in the superconducting coils. The results of the KSTAR magnet system commissioning is meaningful for the KSTAR operation because only a prototype TF coil and a pair of central solenoid model coil have been tested before the system assembly. And it could be used as technical bases for the SC tokamak operation especially using the Nb_3Sn superconductor. This paper will describe overall commissioning progress and related operational characteristics of the KSTAR SC magnet system.

FT/P3-3 · Operation Results of KSTAR Integrated Control System for First Plasma

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Abstract: After the accomplishment of vacuum commissioning assisted by vacuum vessel baking and DC glow discharge, KSTAR is ready to cool-down in April 2008 in the result of all our strength on the construction of cryogenic system with 9 KW cooling power @ 4.5K. For the surveillance and operation of KSTAR, 13 local control systems including diagnostic DAQ systems and machine interlock system have been commissioned during the vacuum commissioning period, and then are under operation for the next cool-down mission. Also, the main control room with 22 operator's consoles and a large display wall is implemented for the remote operation and control. The KSTAR integrated control system has the feature, which is the introduction of EPICS (Experimental Physics and Industrial Control System) as the middleware to integrate whole heterogeneous control systems, the first attempt in the fusion environment. Therefore, it is highly demanded to verify the adaptability of EPICS and the performance our control system because the operation of Tokamak is characterized in the both of the continuous plant operation and the pulse-related experiment. Before the first plasma experiment, we are focusing on the optimization of the data management system not to lose any data via EPICS channel access and archive engine, and development of the plant operational sequence supported by the timing system. In parallel with the preparation of cool-down, we are conducting the commissioning of the magnet power supply systems; one TF magnet power supply and 7 PF magnet power supplies. Each power supply is tested individually and simultaneously by the plasma control system in conjunction with the central control system and the machine interlock system in accordance to the operation sequence and the feedback control scenario. This paper will describe the features and development experience of KSTAR control system with EPICS middleware, and the operation results of the first plasma experiment by using the integrated control system and the plasma control system. In addition, future expansion plan of KSTAR control system for long pulse operation will be presented.

FT/P3-4 · KSTAR Assembly and Vacuum Commissioning for the 1st Plasma

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Abstract: The KSTAR (Korea Superconducting Tokamak Advanced Research) tokamak was at last completed in the site assembly by end of May 2007. During the almost 40-months assembly period, several key technologies were developed to assemble and integrate a fully superconducting magnet tokamak. As will be described in this paper, we have met several mistakes, faults and troubles that mainly stemmed from the characteristics of the superconducting tokamak, but all the problems were successfully solved and the assembly was satisfactorily finished. The vacuum commissioning for both the vacuum vessel and the cryostat followed the assembly finish to confirm the vacuum and helium leak tightness of the machine. The commissioning results showed that vacuum condition is good enough to start cool-down and 1st plasma. Wall conditioning and gas puffing experiments were also performed as a final step of the vacuum commissioning. As a result, the KSTAR vacuum commissioning was very successful and satisfactory for the integrated machine commissioning and 1st plasma.

FT/P3-5 · Load Assembly Design of the FAST Machine

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Abstract: The preliminary design of the FAST (Fusion Advanced Studies Torus) Load Assembly is presented. FAST is a compact and cost effective machine operating with plasma current I_p from 3 MA, in the long pulse advanced scenario, up to 7.5 MA, in standard H-Mode. FAST, has a pulsed, resistive copper magnets which are at cryogenic temperatures and adiabatically heated during the plasma pulse. The cooling is guaranteed by Helium gas allowing the extension of the flat-top to about 3 min. The magnet dimensions has been determined by the cooling requirements. TF magnet ripple has limitated with optimized ferromagnetic inserts. The Vacuum Vessel is adequate to accommodate the whole heating system as well as to withstand the electromagnetic loads produced by plasma disruption. The first wall and the divertor are actively cooled by pressurized water. Liquid Lithium as divertor target will be tested.

FT/P3-6 · Status of the COMPASS Tokamak Re-installation in IPP ASCR

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Abstract: The COMPASS-D tokamak, originally operated by UKAEA at Culham, UK, is being reinstalled at the Institute of Plasma Physics (IPP) AS CR. The COMPASS device was designed as a flexible tokamak in the 1980s mainly to explore the MHD physics. Its operation (with D-shaped vessel) began at the Culham Laboratory of the Association EURATOM/UKAEA in 1992. This tokamak (COMPASS since rei-nstalation in IPP Prague) tokamak will exhibit the following unique features after putting in operation on IPP Prague. It will be the smallest tokamak with a clear H-mode and ITER- relevant geometry. ITER-relevant plasma conditions will be achieved by installation of two neutral beam injection systems (2×300 kW), enabling co- and counter- injections. Re-deployment of the existing LH system (400 kW) is also envisaged. A comprehensive set of diagnostics focused mainly on the edge plasma will be installed. The scientific programme will benefit from these unique features of COMPASS and focuses mainly on scientific project highly relevant to ITER - Edge plasma physics (H-mode studies, pedestal physics, L-H transition etc.) Presently, COMPASS has been dismantled, transported to IPP Prague, placed in to new torus hall and is connected to the new energetic system, which is under commissioning now as well as other systems like vacuum, cooling, feedback and several basic diagnostics. The first plasma is expected at the end of the 2008 year.

FT/P3-7 · KTM Tokamak Project. Present and Future Activity

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Abstract: Specialized experimental complex on the basis of spherical tokamak for material testing (KTM) is created by Kazakhstani and Russian Federation organizations in National Nuclear Center of Kazakhstan. Tokamak KTM is intended for studies in the field of fusion material science and plasma-surface interactions. In 2007 the construction of tokamak KTM was completed and the facility was delivered to Kurchatov. The following activities were carried out: test assemblage of the facility at the manufacturing plant (Efremov Institute), production tests of the KTM facility in accordance with the programs and techniques for testing the vacuum chamber, electro-magnetic system, transport slice device and movable divertor device. 2008 activities are aimed at completion of the tokamak construction, mounting of the KTM systems at the site, off-line tests and preparations for physical start-up. Start-up of the complex is scheduled for 2009. The paper describes main parameters of spherical tokamak KTM, physical start-up program, goals of plasma-physical and material studies, main of which are study of plasma interaction with materials' surfaces, study of physics of tokamaks with aspect ratio $A = 2$ under ohmic and RF-heating, study of physics of divertor area and divertor, study of first wall materials and divertor armor under stationary mode and plasma disruption mode.

FT/P3-8 · Design and Construction Solutions in the Accurate Realization of NCSX Magnetic Fields

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Abstract: The goal of NCSX is to provide the understanding necessary to develop an attractive, steady state, disruption free compact stellarator-based reactor design. A fundamental requirement that NCSX must satisfy in order to ensure that islands and resonant field perturbations are maintained at acceptable levels is that the toroidal flux in island regions due to fabrication errors, magnetic materials, and eddy currents shall not exceed 10% of the total toroidal flux in the plasma (including compensation). This paper describes the recently revised designs of the critical interfaces between the (18) modular coil winding forms, the construction solutions developed to meet assembly tolerances, and the recently revised trim coil system that provides compensation for as built conditions. The revised interface designs utilize two types of friction shims clamped between the modular coil winding form flanges by high-strength studs. This configuration is used in all outer interface regions. The shims are custom made to the thicknesses required for accurate coil positioning. Besides ensuring against slippage during operation, the shims provide electrical insulation for eddy current control. The inner legs require special designs due to space limitations. The interfaces between the winding forms within each of the 3 field periods utilize low-distortion welded interfaces. This partially insulated configuration increases the time constant of eddy currents within a field period by $\sim 28\%$, but still permits adequate field penetration. The 3 interfaces between field periods are fully electrically insulated. At these locations, assembly access and space allows 12 bolts and shims to be added to each inner leg in the upper and lower regions. The small remaining regions at the mid-plane utilize compression shims to react centering forces. A 16 coil/field period trim coil set located around the periphery of the modular coils provides compensation for field errors due to modular coil alignment errors, coil leads, building steel, EM deflections, and eddy currents. Calculations based on projected field errors indicate that the trim coils can reduce the total flux trapped within islands to $\sim 5\%$ (vs. the 10% objective) with $\sim 100\%$ margin on the ampere-turns.

FT/P3-9 · Design Issues on Compact Low Aspect Ratio DEMO Reactor, SlimCS

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Abstract: SlimCS is a fusion DEMO concept featuring compactness (a major radius of 5.5 m) and low aspect ratio ($A = 2.6$). Key design issues on the DEMO reactor are presented. Recent magneto-hydrodynamic stability analysis validated the accessibility to a designed β_N (= 4.3) for bootstrap-dominated plasma. Neutronics analysis found that in-situ control of the tritium-breeding ratio would be likely to work for water-cooled solid breeding blanket using dilute borated water. Sector maintenance scheme has been under study on the basis on the extrapolation of existing technologies. These ideas will provide an important step in the conceptual design of SlimCS.

FT/P3-10 · Divertor Design Study on Compact DEMO Reactor for Handling of Huge Exhaust Power

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Abstract: Conceptual designing of the divertor system is required for DEMO reactor next to ITER for the handling of huge exhaust power. For the divertor design study using SOLDOR/NEUT2D code in the compact DEMO reactor, Slim CS, we find out a prospective divertor design to hold the detached divertor condition within allowable heat load for the first time. Although the heating power on Slim CS increases ~ 6 times larger than that for ITER, the tolerable heat load on divertor target is restricted to 10 MW/m^2, which is less by half than that for ITER. Assuming the heating power of 500 MW and ion out flux of $\sim 1 \times 10^{23}$ s^{-1} with a low-aspect ratio plasmas and vertical divertor target, the peak heat load is estimated to be 50 MW/m^2 on the outer target with attached condition. This value exceeds significantly the allowable limit of 10 MW/m^2. By installing the V-shaped corner in bottom of the outer divertor target and introducing argon impurity, the detached condition with higher particle recycling and radiation loss conditions is obtained, and the peak heat load is successfully reduced to 7.2 MW/m^2.

FT/P3-11 · FAST Plasma Scenarios and Equilibrium Configurations

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Abstract: The Fusion Advanced Studies Torus (FAST) conceptual study has been proposed as possible European ITER Satellite facility with the aim of preparing ITER operation scenarios and helping DEMO design and R&D [1, 2]. Insights into ITER regimes of operation in Deuterium plasmas can be obtained from investigations of non linear dynamics that are relevant for the understanding of alpha particle behaviours in burning plasmas by using fast ions accelerated by heating and current drive systems. FAST equilibrium configurations have been designed in order to reproduce those of ITER with scaled plasma current, but still suitable to fulfil plasma conditions for studying burning plasma physics issues in an integrated framework. The coupled heating power in all cases is assumed 30 MW provided by the ICRH system ($60-80$ MHz). However, for the long pulse AT scenario, 6 MW of Lower Hybrid (3.7 or 5 GHz) have been included to actively control the current profile, whereas 4 MW of Electron Cyclotron Resonant Heating (170 GHz – B_T=6 T) provide enough power for MHD control. In the reference H-mode scenario FAST preserves (with respect to ITER) fast ion induced as well as turbulence fluctuation spectra, thus, addressing the issue of micro- to meso-scale physics cross-scale couplings. The hybrid scenario corresponds to an equivalent Q of about 1, considering an enhanced confinement factor of 1.3*H98. Meanwhile, β_N=2 and $n/n_{GW} = 0.8$. In the AT scenario, the non-inductive/inductive current ratio is 60% and the pulse length is 25 times the resistive time. Predictive simulations of the scenarios described above have been performed by means of JETTO code, using a semi-empircal mixed Bohm/gyro-Bohm transport model. Plasma position and shape control studies will also be presented for the reference scenario.

[1]Romanelli F. et al. 2004 Fus. Sc. Technol. 45 483 ; [2]Pizzuto A. et al. 2008 “The Fusion Advanced Studies Torus (FAST): a proposal for an ITER satellite facility in support of the development of fusion energy”. Presented at the 22nd IAEA FEC, 13-18 October 2008, Geneva, Switzerland

FT/P3-12 · An Experiment to Tame the Plasma Material Interface

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Abstract: The plasma-material interface is the untamed frontier of fusion science. Approaches to heat flux handling and tritium retention that may work for ITER do not generally extrapolate to Demo, and certain ITER parameters, such as first-wall temperature and loss power / major radius do not approach those of Demo. Thus research will be required both in ITER and in parallel with ITER to develop viable approaches for Demo. A series of key questions needs to be resolved, and this provides the basis for a draft set of required research capabilities: (i) Input power / major radius $\sim$ 50 MW/m; Input power / surface area $\sim$ 1 MW/m^2 . (ii) Heating power / H-mode threshold power > 5, close to $n = n_g$. (iii) Flexible poloidal field system capable of wide variation in flux expansion. (iv) Non-axisymmetric coils to produce stellarator-like edge field structure. (v) Replaceable first wall and divertor, solid and liquid. (vi) High temperature $\sim$ 600°C first wall operational capability. (vii) Pulse length $\sim$ 200˘1000 sec (viii) Excellent access for surface and plasma diagnostics. (ix) A range of heating and current drive systems. (x) Extensive deuterium and trace tritium operational capability. A candidate configuration for such a facility has been identified using water-cooled demountable copper coils for flexibility and accessibility. At $R \sim 1$ m, $a \sim 0.55$ m, $B_t \sim 2$ T, $I_p \sim 3.5$ MA, this device, called the National High-Power Advanced Torus Experiment (NHTX), would provide a cost-effective platform, in conjunction with an enhanced program in plasma facing surface science and component technology, for developing the materials and techniques to tame the plasma material interface. Its results would be relevant to a tokamak, spherical torus or stellarator Demo.

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FT/P3-13 · Compact Stellarator Power Plants – Prospects, Technical Issue, and R&D Directions

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Abstract: Results of a detailed and integrated study of compact stellarator configurations as fusion power plant, ARIES-CS, are reported in this paper. Trade-offs among physics and engineering constraints are highlighted; key design features and analysis are described; and the major R&D issues are discussed. The first major goal of the ARIES-CS research was to investigatewhether stellarator power plants can be made to be similar in size to advanced tokamak variants. We focused our analysis on quasi-axisymmetric (QAS) configurations as they are able to operate at a low plasma aspect ratio ($\sim$ 4-5). Our efforts to reduce α-particle loss rate led to new criteria for optimizing QAS configuration. We also developed two new classes of QAS configuration in which strict adherence to linear, ideal MHD stability constraints were relaxed as recently achieved experimental values for β are higher than those predicted from linear stability theory. We also developed a non-uniform blanket and a WC-shield, optimized to provide shielding comparable to a regular breeding module but with a much reduced ($\sim$ 30%) radial thickness . The total tritium breeding including all modules is $\sim$ 1.1. The second major goal was to understand and quantify, as much as possible, the impact of complex shape and geometry of fusion core components. It became evident early on that the 3-D shape of the plasma and the coil (and the components between them) necessitates 3-D analysis of various components – typical correlations and insight developed for axisymmetric fusion devices are not appropriate for stellarator geometry. As such, we directly used 3-D CAD models in many of our analysis. Moreover, we found that the results are quite sensitive to the details of 3-D shape of components and slight variations can result in substantial changes. Second, we have found that engineering configuration as well as assembly/maintenance procedures are key elements in optimizing a compact stellarator – in some cases, these issues determine the choice of technologies. Examples include the selected port-based maintenance scheme which requires a compatible internal design of the fusion core and led to the choice of a ferritic-steel, dual-coolant blanket; and the irregular shape of the superconducting coil that necessitates development of inorganic insulators for high-field magnets.

FT/P3-14 · Effects of Physics Conservatism and Aspect Ratio on Remote Handling for Compact Component Test Facilities (CTFs)

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Abstract: High maintainability was identified in a recent FESAC panel report as a key gap in the knowledge base required for a fusion Demo. Remote handling designs for high maintainability of CTF as functions of physics conservatism (to approach disruption-free operations and avoid complex in-vessel plasma control components), aspect ratio, and device size was studied for the first time. It is shown that remote handling is feasible for compact, low aspect ratio CTF designs that allow modularity for all activated fusion core components, while assuming conservative physics to avoid complex plasma control systems. Such a design providing a neutron wall loading of 1 MW per meter squared is characterized by a minimized major radius near 1.2 m, aspect ratio near 1.5, elongation of 3, β normal near 3.8, and cylindrical safety factor near 3.7. The latter two values are conservative compared to the ideal limits of stability, thereby encouraging the possibility of highly desirable disruption-free operation. As aspect ratio increases with increasing inboard gap to the toroidal field coil, and size increases with increasing physics conservatism (such as by lowering β and increasing safety factor), so increases the complexity of and challenges to remote handling systems. Under constant physics conservatism, engineering assumptions, and outboard mid-plane test blanket areas, increasing the inboard gap from 10 cm to 55 cm is found to double the estimated minimum R to 2.4 m and raise the aspect ratio to above 3. This leads to tripling of the weight of the modularized center leg to 500 tons, requiring a higher level of weight lifting technology. Higher aspect ratio is found to also increase the complexity of and reduce the accessibility to mid-plane test module port assemblies, and may require a qualitatively different remote handling approach from that anticipated for low aspect ratio. Details of the tradeoffs under common plasma and engineering assumptions between physics conservatism to approach disruption-free operation, size, aspect ratio, and practicality of remote handling will be reported at the meeting.

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FT/P3-15 · Integrated Modelling of DEMO Advanced Scenarios with the CRONOS Suite of Codes

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Abstract: Simulations of the DEMO advanced scenarios are performed by means of the CRONOS suite of codes together with the theory-based transport model GLF23. These simulations include essential reactor physics ingredients such as i) proper external heating power deposition calculation; ii) time evolution of the various current sources (bootstrap, rf-driven, NBI-driven); iii) precise calculation of radiation losses, in particular the synchrotron loss, which is expected to be significant in DEMO, not only as a global loss, but also as a mechanism for substantial electron energy redistribution. The main challenges of these simulations are related to the simultaneous constraints the large fraction of non-inductive current required (even for pulsed operation), the large fusion gain (required to minimise the recirculated power) and the detailed current density alignment, needed to satisfy global stability requirements of the discharge. The analysis of these physics key issues made with CRONOS shows the intrinsic difficulty of having mid-size steady-state fusion reactors: the control of the q profile is essential for a steady-state machine (which must be done with precise current drive systems, usually radio frequency heating and current drive with low efficiency), high current drive levels are desirable to get zero loop voltage (which in turn means that NBI systems are essential for this purpose) and high bootstrap current fractions are needed to avoid excessive input power for generating current drive. These features can only be attained by assuming strong physics performance: high density peaking, high pedestal height, Greenwald limit fractions much higher than 1 or internal transport barriers lasting for very long times.

FT/P3-16 · Status of Development of the EU He-cooled Divertor for DEMO

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Abstract: A He-cooled divertor concept for DEMO has been pursued at Forschungszentrum Karlsruhe within the framework of the EU power plant conceptual study since 2002. The design goal is to achieve a DEMO-relevant heat flux of 10 MW/m^2 at least. The major R&D areas are design, analysis, fabrication technology, and experimental design verification. A modular design is preferred for thermal stress reduction. The current reference concept HEMJ (He-cooled modular divertor with jet cooling) relies on impingement cooling with high pressure helium (10 MPa, 600°C). It employs small tiles made of tungsten, which are brazed to a thimble made of tungsten alloy. The W finger units are connected to the main structure of ODS Eurofer steel by means of a copper casting with mechanical interlock. In cooperation with the Efremov Institute a combined helium loop & Electron beam facility (60 kW, 27 keV) was built in St. Petersburg, Russia, for experimental verification of the design. Technological studies were performed on manufacturing of the W finger mock-ups. The results of high heat flux tests till now confirm the divertor performance required. The helium-cooled divertor concept was demonstrated to be feasible. The knowledge gained from these experiments and some aspects on the design improvement are discussed in this contribution.

FT/P3-17 · Concept of Magnet Systems for LHD-type Reactor.

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Abstract: Heliotron reactors have attractive features for fusion power plants, such as no need for current drive and a wide space between the helical coils for the maintenance of in-vessel components. Their main disadvantage was considered necessarily the large size of their magnet systems. According to the recent reactor studies based on the experimental results in the Large Helical Device (LHD), the major radius of plasma of 14 to 17 m with a central toroidal field of 6 to 4 T is needed to attain the self-ignition condition with a blanket space thicker than 1.1 m. The magnetic stored energy is estimated at 120 to 140 GJ. Although both the major radius and the magnetic energy are about three times as large as ITER, the maximum magnetic field and mechanical stress can be comparable. In the preliminary structural analysis, the maximum stress intensity including the peak stress is less than 1000 MPa that is allowed for strengthened stainless steel. Although the length of the helical coil is longer than 150 m that is about five times as long as the ITER TF coil, cable-in-conduit conductors can be adopted with a parallel winding method of five-in-hand. The concept of the parallel winding is proposed. Consequently, the magnet systems for helical reactors can be realized with small extension of the ITER technology.

FT/P3-18 · Design Windows and Cost Analysis on Helical Reactor

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Abstract: Based on the recent experiment results of LHD and the technology-cost basis of magnets developed for ITER construction, the design window analysis has been carried out. We found that the LHD-type helical reactors have the technically and economically attractive design windows, where the major radius is increased as large as for the sufficient blanket space, but the magnetic stored energy is decreased to reasonable level because of lower magnetic field with the convenient physics basis of H factor near 1.1 and β value of 5%. For searching design windows of helical reactors and for discussing their potential as power plants, we have developed a mass-cost estimating model linked with system design code (HeliCos). The major relationships between plasma parameters and reactor parameters are identified with a power balance equation using the ISS04 energy confinement scaling, and with the LHD similar shape of plasma and helical coils depending on the coil pitch parameter gamma (1.15 $\sim$ 1.25). The design spaces of helical reactors are limited with the lower boundary of the major radius R_p given by the required blanket space for tritium breading, the minimum $B(0)$ for the required H factor, and the upper boundary of magnetic stored energy for avoiding the difficulty of manufacturing. With increasing R_p and gamma, the $B(0)$ decreases in the same β and blanket space conditions, that is why the magnet cost, in proportion to the magnetic stored energy, decreases even if in the larger coil size. The sensitivity analysis regarding current density, plasma profile and density limit are carried out. For estimating the magnet operation cost the inherent characteristics of magnet systems such as long lifetime and easy maintenance are considered. In regard to blanket the periodic replacement is necessary, therefore the availability factors are estimated in changing with neutron wall loads. The COE of helical reactors depending on R_p and gamma show the bottom as the result of the trade-off between the magnet cost and blanket cost. The estimation results on the COE of helical reactors suggest us that the technically and economically attractive design windows exist in the rather wide area of the large R_p ($\sim$15 m), medium γ ($\sim$ 1.20) and β values ($\sim$ 5%), and the reasonable magnetic stored energy ($\sim$ 130 GJ).

FT/P3-19 · The Low Temperature and High Density Ignition in the Helical Reactor FFHR2m Based on LHD Experiments

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Abstract: Super high-density up to 1×10^{21} m^{-3} has been obtained in large helical device (LHD) pellet injection experiments. However, an operation in the low temperature and high-density ignition regime usually suffers from the thermal instability. New control algorithm using the proportional-integral-derivative (PID) control on fueling has been proposed to overcome the thermal instability in FFHR2m. Although the large parameter variation would lose its control in the unstable ignition regime, we have demonstrated that the pellet injection can control the ignition access and steady state operation despite of its density variations. We finally discovered that major plasma parameters for FFHR2m are close to plasma parameters independently achieved in LHD experiments.

FT/P3-20 · Multi-functional Compact Tokamak Reactor Concept

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Abstract: An important milestone on the Fast Track path to Fusion Power is to demonstrate reliable commercial application of Fusion as soon as possible. One of the main challenges on the way to produce a GW-level of electrical power is the lack of fully validated suitable materials for the first wall. A new approach has been recently considered: to demonstrate power production in a compact reactor with low fusion output and modest first wall neutron loading, which will allow using available materials currently capable of being licensed for nuclear installations. The use of the small fusion power output of the pilot plant can be maximized either by energy multiplication methods in hybrid blankets or in applications such as production of fuel for nuclear industry (fuel breeding). Although demonstration of the electricity generation will be beneficial, the goal of a compact pilot plant is to demonstrate production, not to sell electricity. At this stage, Fusion will contribute to electricity generation by supporting nuclear industry mainly with production of fuel. The efficiency of the fissile fuel production in a fusion breeding device can be much higher than that in a fission breeder and a complementary fusion-fission power plant can be a good solution to resolve a problem of lack of cheap fission fuel needed for the fast expanding nuclear industry. This multi-functional compact reactor will also contribute to the mainstream GW Fusion power concept (ITER-IFMIF-CTF-DEMO) by providing data on burning plasma, test of diagnostics, remote handling, blanket design and operation, reactor integration etc. The analysis of the performance is based on the solution of the global power balance using empirical scaling laws considering requirements for the minimum necessary fusion power (which is determined by the optimized efficiency of the blanket design), positive power gain and constraints on the wall load. In addition, ASTRA and DINA simulations have been performed for the range of the design parameters, both for ITER and recently proposed advanced ST transport models. These studies show that a spherical tokamak with increased toroidal field, achieved by use of commercially available high temperature superconductors, is the most efficient candidate for proposed applications. In this paper, the main physics and technological challenges of the multi-functional compact reactor will

FT/P3-21 · The Fusion-fission Hybrid Reactor for Energy Production – A Practical Path to Fusion Application

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Abstract: Although the recent experiments and associated theoretical studies of fusion energy development have demonstrated the feasibility of fusion power, it is commonly realized that it needs hard work before pure fusion energy could be commercially and economically utilized. Some countries are speeding up the development of their fission industry, however, this change will face the shortage of fissile fuel and the trouble of safety and radioactive wastes. The fusion-fission hybrid systems have the potential attractiveness of good safety performances and plenty of fuel but easing the requirement of fusion plasma technology. Many studies have been performed to evaluate the feasibility and attractiveness of the hybrid systems. The main purpose of this contribution is to propose a practical path to the early fusion application for energy production in a sub-critical reactor based on the progress review on fusion-fission hybrid system studies in the past half centuries all over the world and the evaluation of the innovative fusion-fission hybrid reactor concept for energy production. The innovative fusion-driven subcritical system for energy multiplier (named FDS-EM) can generate about a net electricity power of ~ 1 GW with self-sustaining tritium cycle, which is driven by a tokamak core of the fusion power of ~ 50 MW, the power gain of 2, the major radius of 3 meters, the minor radius of 1 meter. The tokamak can be designed to achieve the energy multiplication factor of $50 - 100$ based on relatively easy-achieved plasma parameters extrapolated from the operation of the EAST tokamak considering the progress in recent experiments and associated theoretical studies of magnetic confinement fusion plasma all over the world. The subcritical blanket can be designed based the well-developed technology of pressurized water reactor, refering to the progress in studies of blanket concepts optimization. The neutron wall loading of ~ 0.3 MW/m^2 and the surface heat flux of ~ 0.1 MW/m^2 are considered in the design for lowering the requirements for the first wall materials. A preliminary conceptual design is presented with the analysis of energy balance, fuel cycle and electricity economy. The key issues of the concept design and system development are specified. A roadmap of design and construction is proposed.

FT/P3-22 · Conceptual Analysis of a Tokamak-Reactor with Lithium Dust Jet

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Abstract: The steady state operation of tokamak reactors requires re-radiating a substantial part of the fusion energy dissipated in plasma to make the heat loads onto the first wall more uniform and to reduce the erosion of divertor plates. In this paper a quantitative conceptual analysis of the reactor parameters with lithium dust jet injection is presented. The basic goal of this study is searching conditions when the major part of the fusion power is radiated by boundary plasma with injected lithium while the reactor core plasma remains clean enough with effective charge below 1.7 and low radiating. The interest to the lithium jet technique is stimulated by necessity to provide steady state wall conditioning and to reduce the total amount of lithium inside the vacuum vessel to a few kilogram level, which better corresponds to reactor safety requirements than thick lithium wall. Main conclusions of the study are as follows: (i) In accordance with the analysis presented the lithium jet injection technique might have good perspectives for tokamak-reactor performance. For ITER conditions the mantle and SOL might re-radiate more than 100 MW under lithium injection at ~0.15 g/s even in coronal approximation. For DEMO the coronal radiation of lithium changes insignificantly and becomes insufficient for the heat flows control. (ii) Stimulating the lithium recycling in the SOL volume by charge exchange and recombination processes may improve the situation for DEMO. (iii) An acceptable level of lithium in the SOL and core plasma at Li injection rate comparable with that of fuel was obtained in assumption of lower recycling coefficient for Li compared to the fuel gases. This assumption and a radiation Li level (corona approximation) should be checked experimentally, which is planned on T-10 tokamak. (iv) The analytic solutions for the density profiles of different components demonstrate a variety of shapes and significance of simultaneous accounting the terms responsible for particle diffusion, anomalous pinch and sources. (v) First experiments with Li jet on T-10 have been performed at 100 mg/s injection rate and have demonstrated opportunities of the technology for emergency shutdown.

FT/P3-23 · Relevant Developments for the Ignitor Program and Burning Plasma Regimes of Special Interest

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Abstract: Significant developments in the Ignitor program concerning both physics and technology are reported. These include i) the confirmation of the validity of the path to ignition based on the high density regimes ($n_0 \sim 10^{21}$ m^{-3}) following relevant experiments by LHD (Japan); ii) the adoption of intermediate temperature superconducting material (MgB_2) for the largest poloidal field coils and the relevant R&D effort; iii) the identification and the analysis of the optimal double-X point configurations with plasma currents $I_p \sim 9 - 10$ MA that permit achieving ignition in the H-regime; iv) the completed analysis of the ICRH strategy, optimizing the time necessary to reach ignition; v) the completion of the detailed engineering design of all the components of the machine core (e.g., toroidal and poloidal field magnets, supporting mechanical structures, plasma chamber, first wall system, remote handling system, cryogenic cooling system, tritium system); vi) the construction and initial tests of the first injector capable of launching pellets with almost 4 km/s as required by the high plasma densities and temperatures at which ignition can be achieved in Ignitor; vii) the development of new diagnostic systems that can complement the electromagnetic diagnostics to provide the input for the control system of both the position and the shape of the plasma column; viii) the development of prototype coils of the electromagnetic diagnostics with ceramic insulators that are resilient to neutron and gamma radiation; ix) the initial conceptual studies of power producing reactors based on the "reactor physics" that Ignitor is designed to investigate.

FT/P3-24 · Risk Analysis and Optimization of Safe Operation for Advanced Superconducting Tokamak

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Abstract: 1.Risk analysis: Any SC tokamak must have a cryostat. Its mechanical configuration is more complicated and risks of manufacture, operation is higher than the conventional tokamak. (i) The cryostat could be second discharge chamber if vacuum is poor and the voltage on PF coils is too high. (ii) All PF coils will suffer high voltage and AC loses by fast flux changes. (iii) Leakages inside cryostat will increase heat load and decrease quality of insulation, which can cause discharges in cryostat and then serious damages. (iv) The internal leakage is difficult to be found and the leaked gas is helium which is difficult to move out by any ways. (v) The sub-cryostat of current leads is the most danger place where highest voltage has. (vi) Quench protection of PF requires decoupling each other carefully. Otherwise mistakes can be easy happened. 2.EAST project [1]: EAST is an Experimental Advanced Superconducting Tokamak. Its parameters are $B_t = 3.5$ T, $R_0 = 1.70$ m, $I_p = 1$ MA, $a = 0.4$ m, $(b/a) =\sim 2$ with the flexibility of double and single null divertor. The maximum pulse will be 1000 seconds 3.Successful manufacture of EAST [1]: The commissions were success and all design parameters have been achieved under safe operation of all sub-systems. It means quality control during manufacture are success. The test items, the results during manufacture, the model coil tests, the cold magnets tests, the tests during final assembly, the two steps of engineering commissioning will be given in detail in the paper. 4.Successful operation of EAST [2]: The first plasma, the both single and double null divertor with $K \sim 1.7 - 2$ have been achieved safely. The safety of operations is obtained by 1) breakdown voltage should be as low as possible; 2)vacuum chamber was designed as the protection system for fast disruption by coupling with I_p and longer time constant $(10 - 20$ ms). 3) vacuum of cryostat is always require to below 5×10^{-5} Pa, 4) the value of integrated $(dB/dt)^2$ with time for any discharges was controlled to be as lower as possible to avoid high AC loses which can cause quench. More detail experimental results on safe operation of EAST will be provided in this paper.

[1] EAST Team, 'Design of the EAST (HT-7U)' October 2003, Hefei, China; [2] Y. X. Wan et al "First Engineering Commissioning of EAST Tokamak", Plasma Science and Technology Vol.8, No 3, May 2006

FT/P3-25 · Physical Design of MW-class Steady-state Spherical Tokamak, QUEST

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Abstract: QUEST ($R = 0.68$ m, $a = 0.4$ m) focuses on the steady state operation of the spherical tokamak (ST) by controlled PWI and electron Bernstain wave (EBW) current drive (CD). The situation of heat and particle handling is challenging, therefore W and high temperature wall is adopted. A new type antenna for QUEST has been fabricated to excite EBW effectively.

FT/P3-26 · 3-D Study of PFC and Dust Activation in ITER

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Abstract: Relating to the values characterizing the Plasma Facing Components (PFC) residual activity in ITER given in the appropriate documents earlier, a more careful 3-D activation analysis has been performed to specify those values taking into account such features of the ITER operation and irradiation conditions as: a positional relationship of the W, C and Be-coverage of the divertor and the first wall leading to a neutron spectrum variation; effects of the pulsed operation and long term irradiation scenario on the PFC activity and afterheat; contribution of the most important impurities (as N, Al, Co, Zr, Nb, Ag, U) in the candidate industrially produced PFC materials into the final activation characteristics of the PFC and dust. Using these data, proper calculation tools and the basic 3-D model of ITER, the radionuclide production rates and residual activity, contact, ingestion and inhalation doses and decay heat have been specified more precisely as a function of the ITER operation time and cooling time. A specific role and features of the long-lived radioactivity produced in the ITER PFC is clarified in the paper. The estimated values may be used to characterize the dust mixture radioactivity in different periods of the reactor life time: during operation, at the divertor maintenance and at the final waste disposal.

IC

*** Innovative Concepts***

IC/P4-1 · Hot Steady-State FRCs and the Field Reversed Mirror Concept

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Abstract: The temperatures in steady-state FRCs created and sustained by Rotating Magnetic Fields (RMF) have been increased from ~ 25 eV to ~ 250 eV in the upgraded Translation, Confinement, Sustainment (TCSU) experiment. Simple scaling laws show that modest increases in size and power could lead to keV temperatures and $50 - 100$ mWb flux levels, ideal parameters for Tangential Neutral Beam Injection (TNBI). These impressive results could lead to the long held dream of the Field Reversed Mirror reactor.

IC/P4-2 · Formation and Sustainment of Field-reversed Configuration by Rotating Magnetic Field with Spatial High-harmonic Components

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Abstract: Rotating magnetic field (RMF) is one of the most successful current-drive methods for high-beta field-reversed configuration (FRC) plasmas. A novel RMF method using spatial high-harmonic components has been proposed and experimentally investigated in the FRC injection experiment (FIX) in order to resolve the problem that the dipole RMF might open up the closed surface of the FRC and connect the bulk plasma to the vessel wall. The RMF generated by the antennas located inside a conducting vacuum vessel contains large spatial high-harmonic components, whose rotation frequency is slower and sometimes in the reverse direction compared to the fundamental component of the RMF. Since the electrons in the bulk FRC plasma are expected to rotate synchronously with the fundamental component of the RMF, the high-harmonic components will be absolutely excluded at the plasma edge, resulting in a generation of effective magnetic pressure near the separatrix, which helps to keep the separatrix away from the vessel wall. The proposed high-harmonic RMF method will be helpful in reducing the particle loss and thermal load when applied to the fusion core plasma. Experimental results show that the FRC plasma sustained by the high-harmonic RMF has steep density gradient at $r \sim 0.18 - 0.25$ m provided by magnetic pressure of excluded high-harmonic components of the RMF, which successfully keeps the separatrix away from the chamber wall at $r = 0.4$ m. The time-independent FRC equilibrium is also observed to show azimuthal deformation as predicted from the one-dimensional model of the RMF-FRC equilibrium, nevertheless, the experimental results rarely show severe degradation of the plasma including rotational instability. The azimuthal deformation of the FRC plasma sustained by the high-harmonic RMF might essentially eliminate the destructive modes caused by the rotation of the plasma column due to electron-ion collisions or some other reasons.

IC/P4-3 · Simulation Studies of Field-Reversed Configurations with Rotating Magnetic Field Current Drive

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Abstract: First results of 3D kinetic simulations of rotating magnetic field (RMF) current drive in field-reversed configuration (FRC) are presented. Self-consistent hybrid simulations have been performed using the HYM code for even- and odd-parity RMF and different FRC parameters. Simulations show that the RMF pushes the plasma radially inward, resulting in a reduced plasma density outside separatrix. Lower plasma density and larger RMF amplitudes result in faster RMF field penetration, in agreement with previous two-fluid studies. Numerical study of the effects of the applied RMF field on particle confinement shows that for relatively large ion Larmor radius and small RMF frequency, there is little difference between even- and odd-parity RMFs in terms of the ion losses. The rate of particle losses increases in larger FRCs, and with larger RMF amplitudes. In contrast, high-frequency RMF can reduce ion losses provided that the RMF is of even-parity. The improved particle confinement is related to ponderomotive forces due to the rapidly oscillating, inhomogeneous electromagnetic field. It is also found that high-frequency, odd-parity RMF forces particles away from the midplane toward the FRC ends.

IC/P4-4 · Current Drive and Heating in a D$-^3$He FRC Reactor, Relaxation in a Flux Core Spheromak and Oscillating Field Current Drive

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Abstract: The possibility of sustaining the current and heating the plasma with the fusion born protons in a D$-^3$He FRC ractor is investigated. A Monte Carlo code that follows the exact particle trajectories, including particle drag and pitch angle scattering, is imployed. It is found that a significant current and plasma heating can be obtained, depending on the shape of the plasma equilibrium. The same code was also used to study NBI current drive and heating. It is shown that, due to the large poloidal field gradient, a significant fraction of the ionized particles rotate in the opposite sense to the plasma current thus reducing the beam driven current. The spontaneous formation of a Flux-Core Spheromak (FCS) configuration from an unstable screw-pinch is studied by means of 3-D numerical simulations. The non-linear, zero β MHD system of equations is solved using the VAC code which includes several shock-capturing Godunov type base schemes and algorithms to maintain the $\nabla B = 0$ constraint. The column kinks and the instability grows with an $n = 1$ dominant mode until it saturates. Due to the relaxation, closed flux surfaces are formed and poloidal flux amplification occurs. Finally axisymetry is recovered and a FCS configuration is obtained which resistively decays. The effect of finite plasma length and the radial component of Ohm's law on RMF current drive in FRCs is studied. Considering large FRC elongation and assuming that the electrostatic potential is a surface function the problem reduces to the solution of two coupled non linear differential equations. Additional constraints are obtained by requesting that the radial component of the current density and the poloidal flux vanish at the boundary. Both dipole and quadrupole RMFs are considered and compared. Previous results on the use of oscillating helical magnetic fields to sustain the poloidal and toroidal currents in a RFP are extended to a lower resistivity regime. It is shown that the current drive efficiency continues to improve as the resistivity decreases.

IC/P4-5 · Spheromak Formation by Steady Inductive Helicity Injection

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Abstract: Since the last IAEA meeting, new understanding and parameters have been achieved on the Helicity Injected Torus with Steady Inductive helicity injection current drive (HIT-SI) experiment. The experiment has a bowtie shaped spheromak confinement region with two helicity injectors. The inductive injectors are 180 degree segments of a small, oval-crossection RFP. The Taylor-state model is shown to agree with HIT-SI surface and internal magnetic profile measurements. Helicity balance predicts the peak magnitude of toroidal spheromak current and the threshold for spheromak formation. The model also accurately predicts the division of the applied loop voltage between the injector and spheromak regions. Spheromaks with currents up to 29 kA have been produced.

IC/P4-6 · Performance Projections for the Lithium Tokamak eXperiment (LTX)

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Abstract: Use of a large-area liquid lithium limiter in the CDX-U tokamak produced the largest enhancements in ohmic tokamak confinement ever observed. Numerical simulations of CDX-U low recycling discharges have now been performed with the ASTRA code, utilizing a model with neoclassical ion transport and boundary conditions suitable to a nonrecycling wall, with fueling via edge gas puffing. This transport model has successfully reproduced the experimental values of the energy confinement (5˘6 msec), loop voltage (< 0.5 V), and density for a typical CDX-U lithium discharge. The model has also been used to project the performance of the new Lithium Tokamak eXperiment (LTX), with fueling either by edge gas puffing, or with core fueling via neutral beam injection (NBI). Core fueling with NBI in LTX, with a low recycling wall of liquid lithium, is predicted to result in core electron and ion temperatures of 1˘2 keV, and energy confinement times in excess of 50 msec. In addition, DEGAS 2 modelling has now been performed for low recycling CDX-U discharges. This is the first use of DEGAS 2 for a tokamak with liquid lithium plasma-facing components. The final value of the global recycling coefficient derived from DEGAS 2 is higher than earlier estimates indicated, and lies in the range of ∼75%. Finally, we will present the results of separate experiments which are designed to test the power handling capability of lithium-filled divertor targets. The motivation for this work was an earlier finding that shallow pools of liquid lithium could easily withstand spot power densities of up to 60 MW/m^2, during electron beam evaporation experiments on CDX-U. Similar results were obtained with solid surface overlaying a liquid lithium-filled target, although the solid surface was in this case composed of oxides of lithium. Here we reproduce the latter experiments, using a thin walled (0.125 mm) lithium-filled container, e-beam heated from the side. Results with various target geometries will be presented.

IC/P4-7 · The Super X Divertor (SXD) and High Power Density Experiment (HPDX)

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Abstract: The SuperX divertor (SXD), a radically new magnetic geometry that isolates the divertor from the main plasma, can solve the severe heat exhaust problem that will be faced by reactors as well as next generation high power density machines. The divertor plates are placed at the largest possible R, consistent with fitting inside existing TF coils, shielding, etc. Due to its larger R, the SXD increases the plasma-wetted area by a factor of $2-3$ over the best that can be obtained by any flux expansion method (plate near main X-point, extreme plate tilting, X-divertor, snowflake divertor, etc.); all of the latter, because of the engineering constraints of minimum 1degree between the field line and divertor plate, can raise the wetted area above ITER-like standard divertors by no more than a factor of ~ 2. The SXD, also, enables increases in the line length $5-10$ times greater than any other geometry. The SXD engineered extra gains seem essential for AT reactors; it reduces steady state and peak heat fluxes from ELMs and disruptions, shields the divertor from halo currents and neutrons, and enables operation at much lower core radiation and edge density – thus reducing disruption probability. The SXD uses axisymmetric PF coils with currents comparable to the standard divertor configuration. The SXD enabled reduction in core radiation requirements allows crucial core plasma optimizations needed for high fusion power density. A high power density experiment (HPDX) is proposed to attempt the challenging task of obtaining $\beta > 3$ times, and power density ~ 10 times that of ITER (with $\beta_N \sim 3-4.5$ and $f_{bs} = .4-.7$ as demonstrated in DIIID), simultaneously with appropriate power exhaust using SXD. As a prelude to an AT reactor, HPDX could run in the advanced tokamak (AT) mode with $R = 2.2$ m, $A = 2.5$, elongation $\kappa = 2.4-2.7$, produce 300-400 MW of fusion power in short periods of DT operation (like JET), and reach $1 < Q_{XT} < 2$ (= fusion power / total electrical input - JET and NIF achieve $Q_{XT} \sim 1/10$). The challenging κ is allowed by the lowish A, and gives higher β for fixed β_N and f_{bs}. HPDX-SXD could demonstrate integrated performance - simultaneous high β, good confinement, stability (including thermal stability for a self-heated plasma), and proper heat exhaust with only reactor-pertinent methods.

IC/P4-8 · A Snowflake Divertor: A Possible Way of Improving the Power Handling in Future Fusion Facilities

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Abstract: Handling high power loads on plasma facing components is one of the critical issues in developing an economically-competitive fusion reactor based on tokamak. In this study, we provide a detailed analysis of a relatively unexplored approach to this problem based on the use of divertors with the poloidal magnetic field structure closely approaching a second-order null. We demonstrate that this geometry opens up new possibilities for radiative divertors, has favorable effect on the convective transport, and provides an additional control over the ELM activity. In the ideal case where the null is exactly second order, the separatrix near the null acquires a characteristic hexagonal shape reminiscent of a snowflake, whence the name of this configuration. It can be created by a simple set of divertor coils situated outside the toroidal field coils (in particular, for ITER geometry). The snowflake geometry is consistent with the geometry of existing tokamaks with not-too-tight aspect ratio, such as DIII-D. It is also compatible with more tight aspect ratio tokamaks, including spherical tori. Strong magnetic shear just inside the separatrix, a feature of snowflake geometry, has a significant effect on peeling-ballooning modes. A relatively minor variation of the divertor coil current leads to a significant change of the shear, without affecting the overall magnetic configuration. This then provides an additional way of controlling the ELM activity. The increased shear causes a significant squeezing of neoclassical orbits for particles whose drift trajectories bring them close to the null point. We consider increasing the radiated power near the divertor target surfaces by puffing impurities directly into the "closed" divertor; the escape of the impurity ions into the main plasma can be inhibited by the enhancing the flow of deuterium toward the divertor plates, e.g., "puff-and-pum". In such a situation, it is possible that significant power can be radiated from the relatively large volume near the null-point, since this region has a larger volume than that of the standard geometry due to the stronger flux expansion. Conditions for detached regimes are identified both by analytical models and by simulations based on the UEDGE code.

IC/P4-9 · Thermonuclear Prospects of Modern Mirror Systems

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Abstract: At present, two of three modern magnetic mirror systems for plasma confinement (multi-mirror and gas dynamic) are studied in the Budker Institute. Both the systems are very attractive from the engineering point of view because of very simple axisymmetric geometry of magnetic configurations. Significant progress was achieved recently in understanding of phenomena of heating plasma ($n_e \sim 10^{21}$ m^{-3}) by powerful relativistic electron beam and its confinement in the multi-mirror facility GOL−3. In particular, several interesting collective phenomena are explained. Among them it should be mentioned a strong suppression (more than three orders of magnitude) of longitudinal electron heat conductance, fast collective effect of ion heating, observed only in the multi-mirror configuration, etc. The present day parameters of plasma in the GOL−3 are as follows: $T_e \approx T_i \approx 2$ keV, $n_e \approx 10^{21}$ m^{-3}, the energy confinement time of ions $\tau_i \approx 10^{-3}$ s. The second system presented here is the Gas Dynamic Trap (GDT). It is examined as a possible efficient 14 MeV neutron source for fusion materials tests. The status of the works is described. In particular, it should be mentioned that new NB injectors (25 keV, 1.5 MW, 5×10^{-3} s) worked out and partially installed on the GDT. The most important parameter responsible for the neutron power density is the electron temperature. At the moment, $T_e \approx 200$ eV is achieved due to placing the partially new NB injectors. To demonstrate a feasibility of a "moderate" NS (0.5 MW of 14 MeV neutrons), the value $T_e \approx 300$ eV is required. This value should be achieved after the GDT upgrade.

IC/P4-10 · Fusion Studies Using Plasma Focus Devices from Hundred of Kilojoules to Less than One Joule. Scaling, Stability and Fusion Mechanisms

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Abstract: The Thermonuclear Plasma Department of the Chilean Nuclear Energy Commission has, from some ten years ago, managed plasma production devices to study dense hot plasmas. The aim of the current research work has been to characterize the physics of these plasmas and also to carry out the design and construction of smaller plasma focus devices – in terms of both input energy and size – capable of providing dense hot plasmas. Fusion reactions are obtained in these devices when deuterium is used as working gas. The saga to push ahead the current status of the field has had several achievements. For instance, it was made and put into work a device of very low energy (less than 1 J, i.e., three or four orders of magnitude lower than the smallest device previously developed) that still produces radiating dense hot plasmas. In fact, recent experimental evidence shows that even the ultraminiature device is capable to produce neutrons from fusion reactions. The research work and the expertise developed in the group include transient electrical discharges going from 100 kJ to 0.1 J, plasma diagnosis and pulsed power technology. The diagnostics developed and used include: current derivative and voltage signals, neutron detections using silver activation counters and ^{3}He proportional counters; scintillators with photomultilier; and optical refractive diagnostics (schlieren, shadowgraphy and interferometry). In this work, recent results obtained from experiments in plasma focus devices whose stored energies are 0.1 J , 50 J, 400 J, and 70 kJ are presented. The similarity of the physical behavior and the scaling observed in these machines (experiments and theoretical simulations on actual devices show that scaling holds at least through six order of magnitude) is emphasized. In particular all of these devices, from the largest to the smallest, keep the same quantity of energy per particle. Therefore, fusion reactions are possible to be obtained in the ultraminiature device, as they are in the bigger devices. However, the stability regimes for the plasma depends on the size and energy of the device. The processes of neutron production (thermonuclear, beam-target) are discussed in relation with the stability regime.

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IC/P4-11 · Studies on Plasma Direct Energy Converters for Thermal and Fusion-Produced Ions Using Slanted Cusp Magnetic and Distributed Electric Fields

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Abstract: Two types of direct energy converters, cusp direct energy converter (CUSPDEC) and traveling wave (TW) DEC, used to produce electricity from thermal ions and fusion products, are investigated using small-scale devices. In CUSPDEC, magnetized electrons are deflected along the field lines of the cusp magnetic field to the line cusp region and collected by an electron collector, while weakly magnetized ions can traverse the separatrix and enter into the point cusp region. Thus, ions are separated from electrons, and flow into an ion collector to produce DC power. By using a normal cusp magnetic field, the particle separation is achieved for low energy electrons from a test plasma source, but not for electrons with much higher energies from the tandem mirror GAMMA10. The reason for this is found that the high energy electrons do not follow the field lines due to a high potential applied to the ion collector for ion deceleration. This difficulty is overcome by applying a slanted cusp magnetic field, which has the capability of separating electrons with energies as high as 8 keV. Efficiencies of energy conversion of separated ions with large thermal spread of energy in GAMMA10 are measured to be $\sim$ 55%. An additional lateral electrode, together with the existing collector, constitutes a two-stage ion collector that provides distributed ion-decelerating fields. From the measured voltage-current characteristics, the efficiency of this collector is estimated to be improved to $65-70\%$, which is consistent with the calculation assuming a parabolic shape of energy distribution function with 5% loss by secondary electron emission. Fusion-produced fast ions enter into TWDEC and are velocity-modulated by RF fields, bunched, and then decelerated by RF traveling-wave fields on the decelerator to produce RF power. The TWDEC device has shown that the energies of ions of $3-6$ keV can be decreased by $15-20\%$ for a one-wavelength decelerator. This would give a total efficiency of $60-70\%$ for a full-length decelerator. A novel system is being investigated for further improvement, in which the incoming ions are deflected transversely, according to each energy, to form a fan-shaped beam and a distributed electrode array for modulation and deceleration generates traveling waves appropriate to each ion path depending on the energy.

IC/P4-12 · Confinement Improvement with Magnetic Levitation of a Superconducting Dipole

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Abstract: The levitated dipole experiment (LDX) investigates plasma confinement and stability in a magnetic dipole. We report the first production of high β plasma confined in a fully levitated laboratory dipole using neutral gas fueling and electron cyclotron resonance heating (ECRH). The pressure results from a population of energetic trapped electrons that is sustained for many seconds of microwave heating provided sufficient neutral gas is supplied to the plasma. As compared to previous studies in which the internal coil was supported, levitation improves particle confinement and allows high-density, high-beta discharges to be maintained at significantly reduced gas fueling. Elimination of parallel losses coupled with reduced gas leads to improved energy confinement. Improved particle confinement assures stability of the hot electron component at reduced pressure and this self-consistency produces significantly improved dipole discharges. In the experiments reported, high-beta plasma discharges were studied when the LDX high-field superconducting dipole magnet was levitated by attraction to a coil located above the vacuum chamber. The resulting configuration utilizes a digital feedback system to obtain stable levitation of the floating coil. By eliminating supports used in previous studies, cross-field transport becomes the main loss channel for both the hot and the background species.

IC/P4-13 · FRCHX Magnetized Target Fusion HEDLP Experiments

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Abstract: We are testing the first liner implosions of high pressure field-reversed configuration (FRC) plasmas as a physics demonstration of magnetized target fusion (MTF)*. Integrated hardware on the new Field Reversed Compression and Heating Experiment (FRCHX) at the Air Force Research Laboratory Shiva Star facility, will form initial FRC's at 3 Tesla magnetic field, with high-β ($\beta \sim 0.9$, $T = T_e + T_i > 300$ eV, $n_e \sim 5 \times 10^{16}$ cm^{-3}) plasma pressures of $20 - 30$ atmospheres, and translate and capture them into an aluminum liner, and then compress them to kilovolt temperatures, forming a high energy density laboratory plasma (HEDLP). Magnetics design and $2 - 1/2$ dimensional MHD fluid (MACH2) simulations for the translated plasma and time-dependent magnetic geometries have been done, construction is nearly finished, and first 4.7 MJ implosion tests are expected in Spring 2008. modelling shows that FRC's should be formed after the liner implosion begins, to match plasma lifetime, and the liner compression timescales ~ 10 and 20 μs, respectively). Details of the hardware, diagnostics, and pre-compression plasma formation and trapping experiments, both from LANL and AFRL, will be presented.

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IF

*** Inertial Fusion Experiments and Theory***

IF/1-1 · Progress in Direct Drive Inertial Confinement Fusion

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Abstract: The direct-drive approach to inertial confinement fusion (ICF) ignition and high gain will be reviewed, focusing on results from experiments on the OMEGA laser. A general review of target designs and simulations will be presented together with implosion hydrodynamics and future directions. The target design effort has produced optimized designs for conventional isobaric ignition (including polar drive (PD)) for the National Ignition Facility (NIF), and two step ignition processes (shock ignition and fast ignition). Two step processes allow the possibility of higher gains due to the lower laser energy required for low velocity implosions to assemble the fuel. PD is a concept that allows direct-drive ignition experiments while the NIF is configured for indirect drive. This review will include the theory of adiabatic shaping to suppress the Rayleigh-Taylor instability, hydrodynamic scaling for current to ignition size target implosions, and the latest ignition designs for conventional, shock, and fast ignition targets. The Lawson criterion for ICF is presented in a form that can be experimentally determined. The experimental review will include adiabat shaping for high velocity, conventional ICF implosions, and low-velocity low-adiabat implosions, PD, and cryogenic implosion experiments on OMEGA. The effects of electron thermal transport models and the two-plasmon-decay instability will be highlighted. This intense research effort has culminated in OMEGA experiments producing the highest compression (i.e., highest value of the areal densities) ever achieved in cryogenic implosions, greater than 200 mg per square centimeter, and the related fusion product parameter ($n\tau T$), comparable or greater than achieved in magnetic fusion experiments. Results from the OMEGA implosion campaigns provide an important physics validation for the United States' direct and indirect drive ignition campaigns and increasing confidence that direct-drive ignition will be achieved on the NIF.

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IF/1-2 · Implosion and Heating Experiments of Fast Ignition Targets for FIREX-1 Project

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Abstract: Implosion and heating experiments of Fast Ignition (FI) targets for FIREX-1 project have been performed with Gekko-XII and PW/LFEX lasers at the Institute of Laser Engineering, Osaka University. Typical FI target has a hollow cone for guiding the short-pulse heating laser beam at the time of the maximum compression. The cone is mounted so as to in one-side penetrate the shell target. Detailed implosion hydrodynamics, FI heating and core plasma formation of plastic (CD) shell target with gold cone have been clarified by observing those with fast imaging x-ray spectroscopy and neutron diagnostics. A new Multi-channel Multi-Imaging X-Ray Streak Camera (McMIXS) was developed for observation of time-resolved two-dimensional x-ray images and time-resolved two dimensional temperature distributions. Also, a monochromatic x-ray imager coupled with 2D-SIXS (Two-Dimensional Sampling Image X-ray Streak camera) was developed for time resolved x-ray line-emission imaging. With these instruments, one can observe heating properties of the imploded core such as spatial distribution of the heated region and its temporal evolution. Synchronization of the heating beam injection to the implosion dynamics has been monitored with an x-ray framing camera. It was found that the shape of the core is neither spherical nor uniform mainly because of the existence of the cone and moving toward the tip of the cone and interacting with it. Experimental results are compared with two-dimensional hydrodynamic simulations. Target design taking into account of these phenomena is quite important because such core movement and jet formation can affect the condition of the cone.

IF/1-3 · Progress on the Development of Ion Based Fast Ignition

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Abstract: Results from research on fusion fast ignition (FI) [1] initiated by laser-driven ion beams is encouraging so far. Compared to electrons, FI based on a beam of quasi-monoenergetic ions (protons or heavier ions) has the advantage of a more localized energy deposition, which minimizes the required total beam energy, bringing it close to the theoretical minimum of ~ 10 kJ [2]. High-current, laser-driven ion beams [3, 4] are excellent for this purpose, and because of their ultra-low transverse emittance, these beams can be focused to the required dimension, ~ 10 s f.e.. Because they are created in ps timescales, these beams can deliver the power required to ignite the compressed D-T fuel, ~ 10 kJ/50 ps. Our recent integrated calculations of ion-based FI include high fusion gain targets and a proof of principle experiment. That modelling indicates the concept is feasible, and provides confirmation of our understanding of the operative physical processes, a firmer foundation for the requirements, and a better understanding of the optimization trade space. Three key requirements for the success of this scheme include the generation of a sufficiently monoenergetic ignitor ion beam (energy spread below $\sim 10\%$), with a sufficiently high ion kinetic energy (f.e. 400 MeV for C), along with a sufficiently high conversion efficiency of laser to beam energy. This paper describes the theory and experimental progress in our research program, which is concentrated on fulfilling these three requirements. An important benefit of the scheme is that such a high-energy, quasi-monoenergetic ignitor beam could be generated far from the capsule (1 cm away), so that the laser-target providing the beam may be protected from the implosion. This eliminates the need for a reentrant cone in the capsule, a tremendous practical benefit. A new scheme for laser-driven ion acceleration under experimental investigation at Los Alamos, the laser-breakout afterburner [5], promises to deliver the necessary ion-beam performance. This paper summarizes the ion-based FI concept; the progress in developing a suitable ignitor ion beam, and the integrated ion-based FI modelling.

[1] Tabak M, et al. 1994 Phys. Plasmas 1 1626 3291 [2] S. Atzeni, et al., Nucl. Fusion 42, L1 (2002) ; M. Temporal, et al., Phys. Plasmas 9, (2002). [3] E.g., see J. C. Fernandez, et al., Lasers & Part. Beams 23, 267 (2005); T. E. Cowan, et al., Phys. Rev. Lett. 9

IF/P7-2 · Microtarget Requirements, Production and Delivery for HiPER

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Abstract: HiPER - The proposed European Laser-Driven Fusion Facility – has received initial funding. An overview of the current status of the Targetry Work Package (WP11) will be given. 1) Baseline microtarget designs and other delivery parameters including a range of operational regimes. 2) Inter-relationships between the Targetry and other work packages. 3) Technical solutions for WP11 (comparing alternative technologies where relevant) including cryogenic targets, high repetition rate capabilities, tritium handling and microtarget delivery and tracking systems. The overview will show the strategic fit of WP11 within HiPER and examine how the work package has been structured to reduce risk in delivering on-time solutions.

IF/P7-3 · Studies on Targets for Inertial Fusion Ignition Demonstration at the HiPER Facility

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Abstract: Recently, a European collaboration has proposed the HiPER facility, with the primary goal of demonstrating laser driven inertial fusion fast ignition. HiPER is expected to provide 250 kJ in multiple, 3ω, nanosecond beams for compression and 70 kJ in $10-20$ ps, 2ω beams for ignition. Here, we report target studies performed in the framework of the HiPER project. The baseline approach is fast ignition by laser-accelerated fast electrons; cones are considered as a means to maximize ignition laser-fuel coupling. A design window has been identified by an integrated analytical model, including ablative drive, compression, ignition and burn, taking the ignition laser-fuel coupling efficiency η_{ig} as a free parameter. Adiabat shaping during the implosion is needed to achieve adequate compression, while limiting the growth of Rayleigh-Taylor instabilities, confirmed by perturbation code analysis. The efficiency η_{ig} must exceed 20%. This in turn demands that laser light is efficiently converted into a beam of fast electrons, with range comparable to the desired hot spot size, implying an average energy of $1-2$ MeV. Using the ponderomotive scaling for the average energy, a 2ω ignition laser seems necessary. Target studies have been based on 1D fluid simulations of compression and 2D fluid and hybrid simulations of ignition and burn. The baseline capsule concept is an all-DT shell, with a total mass of about 0.6 mg. Pulses of $150-250$ kJ (@ 3ω) can compress the fuel to a peak density above 500 g/cm^3 and an areal density ρR of about 1.5 g/cm^2. 2D simulations indicate that cone-guided targets can attain similar compression parameters. We have modelled the ignition of the compressed assembly by fast electrons using a Monte Carlo scheme, analysing the dependence of the ignition energy on average electron energy, energy spectrum, divergence and source location. Hybrid modelling also took into account the effect of self-generated fields. The ignited target releases about 13 MJ, giving an energy gain of $25-50$, depending on design margins and assumptions on coupling efficiency. In the final part of the paper we report results on alternative ignition schemes, such as laser accelerated ions, shock ignition, and impact ignition.

IF/P7-4 · Fast Electron Propagation In High Density Plasmas Created By Shock Wave Compression

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Abstract: Till now almost all the experimental results on fast electron generation and transport have been obtained in solid targets. It is clear instead that the fast ignitor approach to inertial fusion concerns the propagation of a fast electron beam in a dense hot plasma in the pellet surrounding the nuclear fuel. We present preliminary results from a recent experiment on fast electrons propagation and energy deposition in high density plasmas created by shock wave compression. The results were obtained at the PICO 2000 laser system at LULI, coupling a 400 J, 1.5 ns, 2ω compression beam with a counter-propagating 50 J, 1 ps high-intensity beam. Focused intensities larger than 1×10^{18} W/cm^2 generated intense currents of fast electrons through the in-depth regions of the target, which are shock-compressed at 3-4 times the initial solid density. We used multi-layered targets with thin copper and aluminium X-ray fluorescent layers embedded in either aluminium or plastic (CH) propagation layers of variable length. The geometry (divergence) and the range of the fast electrons propagation were diagnosed by Cu X-K α signal imaging and by X-ray spectroscopy, covering both Al (1^{st} diffraction order) and Cu (5^{th} order) K α emissions. Target heating was measured from ionisation-shifted Al K α emission. For each kind of target, the results are compared to the cold case (no compression beam).

IF/P7-5 · Theory and Simulation of Non-local Thermal Smoothing for Arbitrary Scale Length Modulation

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Abstract: We present a theory that describes the non-local heat transport and associated thermal smoothing modelling the non-uniform laser illumination by solving the steady state Fokker-Planck (FP) equation. A method that connects higher order Chapman-Enskog expansion for low energy electrons and convolution formula for high energy ones through the self-consistent determination of the electric field is developed. The theory explicitly expresses the heat flux for arbitrary periodic temperature modulation given by a sinusoidal type with moderate wave number $k\lambda_e \leq 0.1$, where λ_e is the electron mean free path. The theory is compared with one-dimensional Fokker-Planck simulations which investigate the relaxation of sinusoidal temperature perturbations. Reduction of the heat flux from the Spitzer-Härm (SH) theory and the hysteresis nature are found to be reproduced. As the wavelength of the modulation becomes shorter, the contribution from high energy electrons to the heat flux is found to increase whereas the total amount of the heat flux is reduced in proportion to $(k\lambda_e)^2$.

IF/P7-6 · Conceptual Design of a Fast-Ignition Laser Fusion Reactor FALCON-D

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Abstract: A new conceptual design of the laser fusion power plant FALCON-D (Fast ignition Advanced Laser fusion reactor CONcept with a Dry wall chamber) has been carried out. A fast ignition scheme is expected to achieve the sufficiently high pellet gain with a small fusion yield compared with a conventional central spark ignition scheme. To make full use of this property, the FALCON-D aims at designing with a compact dry wall chamber ($5-6$ m radius) and a high repetition of laser pulse (30 Hz). This design enables a simple cask maintenance method and a moderate electric output power (about 400 MWe). Core plasma design by using 1-D/2-D hydrodynamic codes shows that the achievement of the sufficient gain (around 100) is probable with relatively small laser energies; 350 kJ for implosion and 50 kJ for heating (20% coupling efficiency of heating laser to a compressed core was assumed). This enables the design of a dry wall chamber with $5-6$ m radius if considering the heat load limit of 2 J/cm^2. We selected tungsten-armed low activated ferritic steel (F82H) for the fast wall and supercritical water (623K) for the coolant. Thermomechanical analysis reveals that the increase in the temperature of the surface layer is not a concern but low cycle fatigue failure may take place due to a large plastic deformation. Blistering and exfoliation due to helium accumulation can also cause a great loss of the surface layer. Here the materials that have fine-structure or fine grain can provide a solution because they have high yield stress and can suppress the concentration of helium. We proposed the maintenance method of blanket system and final optics system of implosion laser. In FALCON-D, the cask method is adopted for a maintenance scenario. The first wall and blanket system are divided into 20 sectors so that all beam lines cross the edge between blanket sectors. The vacuum vessel (VV) is separated from the blanket system to enable the extraction of a large blanket sector without the interference with multiple beam ducts. 20 maintenance ports exist on the upper side of the VV, and the cask accesses to those maintenance ports. The final optics systems of implosion laser, which are built into the reactor room wall, are extracted horizontally into a cask-type replacement device, which can access the final optics system through 6 corridors placed along the reactor room.

IF/P7-7 · Modification of the Bobin-Chu Fusion Threshold for Laser Driven Block Ignition or for Spark Ignition

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Abstract: A unique anomaly at interaction of ps-TW laser pulses with targets was discovered [1] if the pulse had a contrast ratio of 10^8 for times up to 50 ps before the main pulse arrived. This elimination of the prepulse suppressed relativistic self-focusing in contrast to all usual experiments. Highly directed plasma blocks were generated by nonlinear (ponderomotive) force acceleration of skin layers to arrive at highly directed space charge neutralized 100 keV ions with current densities above 10^{11} Amps/cm^2. For using these pulses for fusion energy thresholds E_o^* flux densities of 100 MJ/cm^2 were calculated by Bobin and by Chu (1972). We revised and updated these calculations by taking into account later discovered phenomena as inhibition and collective effects and arrived at a reduction of the threshold by a factor of more than 20. This is interesting also for spark ignition and other laser fusion schemes. For the block ignition of DT, parameters are evaluated since conditions have to be met that the DT ions in the bocks should not have higher energies than in the range of 100 keV while the mentioned threshold requires sufficiently high intensities of the laser pulse. The discovered decrease of the threshold E_o^* simplifies the mentioned conditions.

[1] H. Hora, J. Badziak et al. Phys.Plasmas 14, 072701 (2007)

IF/P7-8 · "GENBU"-Laser Development with Cryogenic Yb:YAG Ceramic for Fusion Energy Reactor Driver

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Abstract: A new diode-pumped laser fusion driver has been conceptually designed at 16 Hz repetition rate and 17% electrical-optical efficiency with less than 3000 m^3 main amplifier size. Output power is 1.2 MJ for a blue compression laser and 0.1 MJ for a heating laser. One kJ pulse energy in near infrared is a basic unit to obtain such mega-joules energy. We have proposed two new technologies into the system. One is cryogenic Yb:YAG ceramics as a laser material. The other is a large-aperture active-mirror as amplifier architecture. Recently, 1 kJ laser system "GENBU" has been designed in details with these technologies as a milestone for the new driver. A 1 J system "GENBU-Kid" is under construction for power-scaling up to 1 kJ.

IF/P7-9 · Methods Improved Characteristics of Laser Source of Ions

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Abstract: At present there are number of works devoted to the study of laser-produced plasma as a source of ions for the inertial confinement fusion [1] together with heavy ion accelerators and the systems on the base of powerful impulse of electrical charge, i.e., Z-pinches. Laser ion source have been recently designed [2] to load the Heidelberg electron beam ion trap with a pulsed beam of lowly charged ions from solid elements. Number of theoretical [3] and experimental [4, 5] works have been carried out in order to optimize the performance of the laser ion source and to determine important operating parameters such as the velocity, mass and charge-state distribution of the generated ion beam and plasma temperature. In this work we discuss three methods to improve characteristics of laser source of ions, namely: i) the effect of the angle of interaction of laser radiation with targets on the plasma ions characteristics, ii) the use of targets of different densities to improve the parameters of plasma ions, and iii) influence of laser frequency on the plasma parameters. Our study will be based on the analysis of mass-charge spectrum of laser-produced plasma ions for different intensity of laser radiation.

[1] H. Hora, Laser and Particle Beams 22, 439 (2004). [2] M. Ogawa et al., Laser and Particle Beams 21, 633 (2003). [3] I.G. Brown et al., Applied Physics Letters 49, 1019 (1986). [4] R.T. Khaydarov et al., Laser and Particles Beams 23, 1-6 (2005). [5] R.T. Khaydarov et al., International Journal of Plasma Science and Engineering, (2008), doi:10.1155/2008/180950.

IF/P7-10 · Fast Electron Transport Studies in Petawatt Laser Irradiation of Solid Dielectric and Metallic Target Materials

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Abstract: During the interaction of high-intensity short-pulse lasers with solid target materials up to 30% of the laser energy is converted into so-called "fast" electrons, that is electrons with relativistic kinetic energy, much higher than the thermal plasma electron energy. The study of the transport of the high-current fast electron beam through the underlying target material is an active research field due to its crucial importance for the fast ignition approach to inertial confinement fusion. In the fast ignitor scheme the fast electron beam is required to travel through the outer region of low-Z dense plasma to reach the compressed fuel core, where the electrons deposit their energy to rapidly heat the fuel to ignition. Here we report on a joint experiment carried out within the HiPER project, for a systematic study of fast electron transport phenomena from the irradiation of multi-layered targets with the VULCAN Petawatt laser beam. The targets consisted of a propagation layerand a tracer layer for the X-ray measurements. High resolution spectroscopy and monochromatic X-ray source imaging of line emission from the tracer layer have been performed in the experiment, thus giving information on the spectrum and on the spatial distribution of the fast electrons after their passage through the propagation layer. The experiment was designed to study transport phenomena depending on the material resistivity, such as return current issues, and Z-dependent phenomena. Therefore, metallic and low and medium Z dielectric target materials have been used as propagation layers. The observed differences of the X-ray emission features for the different target materials will be discussed.

IF/P7-11 · Beam Combined Laser Fusion Driver Using Stimulated Brillouin Scattering Phase Conjugation Mirrors

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Abstract: The beam combination method using stimulated Brillouin scattering phase conjugate mirrors (SBS-PCMs) is a promising technique for realizing the laser fusion driver with an ultrahigh energy/power laser system operating with a high repetition rate over 10 Hz. For realizing the beam combined laser system, it is necessary to lock/control the phases of SBS beams. In our previous papers, the new phase control technique using the self-generated density modulation was proposed, and its principle has been demonstrated experimentally. However, all the previous works were done without amplifiers. In this work, it has been demonstrated that the phase is stabilized with $\lambda/51$ fluctuation by standard deviation during 5,000 laser shots (500 sec.) in the two-beam combination system with amplifiers with 200 mJ total output energy and 10 Hz repletion rate.

IF/P7-12 · Study on Fabrication and Manipulation of HEDgeHOB Cryogenic Targets

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Abstract: HEDgeHOB collaboration plans to carry out a set of experimental researches in the field of High Energy Density (HED) matter generated by the interaction of intense heavy ion beams. One line of the researchers was named LAPLAS (Laboratory PLAnetary Science) [1]. This scheme proposes low-entropy compression of a material like frozen hydrogen or deuterium ice that is enclosed in a cylindrical shell of a high-Z material like gold or lead. Such type of experiment is suitable for studying the problem of hydrogen metallization or for creating physical conditions that are expected to exist in the interiors of the giant planets. This paper presents a thorough analysis of different possible approaches to the cylindrical cryogenic targets preparation to the LAPLAS experiments, including targets fabrication, their delivery and positioning in the radiation area. The researches are performed in the following directions: (*i*) Target fabrication. For the fist time there was considered a possibility of cryogenic cylindrical target fabrication with a parameter $l/\oslash = 5.00 \div 6.25$ (where l and $\oslash$ are the length and the diameter of the cryogenic target core, respectively). There were also investigated different technical approaches for designing the target elements. The target quality criteria were stated as well. (*ii*) Target handling. There were investigated different variants for the target elements assembly, target delivery and positioning. There was made a thorough analysis of diagnostics possibilities for cryogenic cylindrical target placed inside the target chamber. (*iii*) Target survival. There was made a modelling of the target heating process up to the triple point of solid hydrogen or deuterium under the different heat-exchange conditions. There was given an estimation and made an optimization of the cryogenic target lifetime inside the chamber with hot wall. On the basis of the performed researchers there were defined the technical requirements to the corresponding specialized cryogenic system (SCS). The SCS conception aimed at the potential risks minimization during the system designing and functioning is in the research stage now.

[1] Technical Proposal for Design, Construction, Commissioning and Operation of High Energy Density Matter Generated by Heavy Ion Beams. HEDgeHOB collaboration, Jan. 14, 2005

IF/P7-13 · Study of Fast Electron Transport Dynamics in Solids Using Multispectral, Monochromatic X-ray Imaging

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Abstract: The issue of transport of high-current beams of relativistic electrons in solid matter or high-density plasmas is currently receiving a major attention due to its importance for the fast-ignition approach to the Inertial Confinement Fusion. Indeed, the Fast Ignitor in Inertial Fusion Energy requires high current, fast electron beams to propagate efficiently in high solid density collisional plasmas. Transient magnetic fields and neutralising plasma return current occur. Return current is expected to give rise to resistive thermal heating that will modify the spectral features of X-ray fluorescence. Significant shock-heating deep in the target may also occur at ultra-high intensities. Therefore, knowledge is needed on the state of the material in which f.e. propagation occurs. X-ray spectroscopy and imaging are the key tools for accomplishing this task. Here we report on the results of two recent experiments devoted to the study of the fast electron transport dynamics by means of a new experimental technique. The technique is based upon the simultaneous use of a 20×20 pinhole array and a cooled CCD camera operating in the single-photon regime. Two different short pulse laser interaction regime have been investigated, namely a relatively long, moderate intensity ($\sim$ 200 ps, $I = 10^{16}$ W/cm^2) and an ultrashort, relativistic intensity ($\sim$ 70 fs, $I = 5 \times 10^{19}$ W/cm^2) regime. In both cases, the fast electron transport was studied by looking at the X-ray emission (both $K\alpha/\beta$ and Bremsstrahlung) from specially designed, multi-layer targets. Results are presented concerning different substrate materials and layer thicknesses, allowing to gain insights on the role played by processes such as return current, electron refluxing, lateral transport on the dynamics of the fast electron transport.

IF/P7-14 · Effects of Magnetic Field, Viscosity and Shear Flow on the Richtmyer–Meshkov Instability

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Abstract: The effects of shear flow and transverse magnetic field on Richtmyer–Meshkov instability are examined and the expression of the interface perturbation is obtained by analytically solving the linear ideal magnetohydrodynamics equations. It shows that the perturbation evolves exponentially rather than linearly in the presence of shear flow and magnetic field. The shear flow acts as a destabilizing source, while the magnetic field is a stabilizing mechanism of the shocked corrugated interface problem. The whole analysis is based on the assumption that the fluid is incompressible. The effects of transverse magnetic field and viscosity on the Richtmyer–Meshkov instability are considered under the framework of two semi-infinite fluids with different densities, magnetic fields, and viscosities, respectively. The amplitude of the perturbations is analytically obtained. It is found that the magnetic field provides oscillation and damping, and viscosity provides damping. When both are present, the perturbations of the interface undergo damped oscillation provided that the magnetic field is strong enough; otherwise the perturbations will damp from the beginning.

IF/P7-15 · Advances in U.S. Heavy Ion Fusion Science

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Abstract: During the past two years, the U.S. heavy ion fusion science program, a consortium of heavy ion research at the Lawrence Berkeley National laboratory, Lawrence Livermore National Laboratory, and Princeton Plasma Physics Laboratory, has made significant experimental and theoretical progress in simultaneous transverse and longitudinal beam compression, ion-beam driven warm dense matter targets, high-brightness beam transport, advanced theory and numerical simulations, and heavy ion target physics for fusion. First experiments combining radial and longitudinal compression of intense ion beams propagating through background plasma resulted in on-axis beam densities increased by 700X at the focal plane. With further improvements planned in 2008, these results enable initial ion beam target experiments in warm dense matter to begin next year. We are assessing how these new techniques apply to higher-gain direct-drive targets for inertial fusion energy.

This work was performed under auspices of the U.S. Department of Energy by the University of California, Lawrence Berkeley and Lawrence Livermore National Laboratories under Contract Numbers DE-AC02-05CH11231 and W-7405-Eng-48, Princeton Plasma Physics Laboratory under Contract Number DE-AC02-76CH03073, and by the University of Maryland under contract numbers DE-FG02-94ER40855 and DE-FG02-92ER54178.

IF/P7-16 · Quest for Impact Fast Ignition

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Abstract: Impact Fast Ignition (IFI) scheme has the attractive features: simple physics, potential for high gain designs, and low cost. Since the first proposal of IFI in 2004, we have made some progresses: A preliminary experiment has demonstrated a highest velocity, 650 km/s, ever achieved in use of a planar target, which was ablatively driven by Gekko XII laser. Further we have performed experiments using an integrated target. It has turned out that the neutron emissions were isotropic and they were accompanied with no energy shift, indicating that the neutrons were generated by thermonuclear fusion due to the collision effect by the impactor. Here we describe the first proof-of-principle of impact ignition: We have observed an increase of two orders of magnitude in neutron yield at the right timing of the impact collision, demonstrating the high potential of impact ignition for fusion energy production.

IF/P7-17 · Encapsulation of Low Density Plastic Foam Materials for the Fast Ignition Realization Experiment (FIREX) — Control of Microstructure and Density –

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Abstract: Development of foam capsule fabrication for cryogenically cooled fuel targets is overviewed in the present paper. As a part of the Fast Ignition Realization Experiment (FIREX) Project at the ILE, Osaka University, the fabrication development was initiated in the way of bilateral collaboration between Osaka University and National Institute for Fusion Science (NIFS). For the first stage of FIREX (FIREX-I), a foam cryogenic target was designed where low-density foam shells with a conical light guide will be cooled down to the cryogenic temperature and will be fueled through a capillary. The required diameter and thickness of the capsule are 500 μm and 20 μm, respectively. The material should be low-density plastics foam. We prepared such capsules, and fueling test has started. For the second phase of FIREX (FIREX-II), much lower density foam materials (10 mg/cm^3) are required. Generally, lower density is from larger pore, then precise control of thickness and its encapsulation becomes more difficult. We have investigated the relation between pore size and preparation conditions using several precursor materials, and revealed how to control pore size of low density foams. Recently we found decananometer-pore foam which is one-order smaller than the previous one. On the other hand, we continue to fabricate lower density foam.

IF/P7-18 · Numerical Study of Advanced Target Design for FIREX-I

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Abstract: In the fast ignition (FI) scheme, at first high-density fuel core plasma is assembled by implosion laser, and it is then heated by peta watt laser to achieve fusion burning condition. The formation of high-density fuel core plasma is one of the key issues for FI. A typical target for FI is a shell fitted with a reentrant gold cone to make a pass for heating laser. The ablated plasma of gold cone interferes with the implosion dynamics, which is quite different from that of conventional central-hot-spot approach. Therefore, the dynamics of a non-spherical implosion must be controlled to assemble high density and high areal density. Numerical simulations are performed to study radiation hydrodynamics of cone-guided implosions. In the result, the effect of the cone on implosion dynamics is clarified. The cone surface is irradiated by the radiation, and ablated plasma affect the imploding shell. Coating on the cone, which tamps the gold plasma, is effective to improve the implosion performance. In addition, detail requirement for core plasma conditions are surveyed using 2-D multi-diffusion type heating simulation code, FIBMET. From these results and finding, we design an advanced target for FIREX-I.

IF/P7-19 · Elemental Research on Laser Fusion Reactor KOYO-F

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Abstract: We conceptually designed the laser fusion power plant KOYO-F based on fast ignition scheme and the result was reported at IAEA-FEC 2006. KOYO-F has a liquid LiPb flow as the first protective wall. After the activity we concentrated elemental researches on the issues to increase the reliability of the concept. The researches include two experiments on formation of aerosols and micro-particles, protection of laser beam ports using a magnetic field, stability of liquid LiPb flow. We used discharge method to contribute understanding formation of aerosols with aid of computer simulation. Energy deposition by alpha particles is volumetric heating that can be experimentally simulated more likely by electric discharge method than conventional laser heating. We observed lot of 30-nanometer-diameter aerosols on a witness plate, which agrees with numerical simulation. The other experiment is focused on the hydrodynamic phenomena of ablation process using backside illumination of a metal membrane on transparent substrate. Hydrodynamic instability was observed, which results formation of much larger droplets than aerosols. Beam ports that stick out of the LiPb flow need special scheme to protect from alpha particles. We found a 0.8 T magnetic field synchronized with laser irradiation was effective as the protective shield. A 3-mm-thick stable flow was formed on a mockup for cascade scheme of KOYO-F. The mockup was designed so that the Weber number of the film flow on the first wall was the same as the actual liquid LiPb flow from the flow stability point of view. The minimum flow rate to obtain the continuous film flow was confirmed experimentally and numerically.

IF/P7-20 · Chamber Responses and Safety and Fusion Technology in HiPER Facility

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Abstract: The HiPER project will design, inside the frame of the European ESFRI program, an Inertial Fusion Facility based in Fast Ignition concept where demonstration of ignition and gain with that option will be combined with a complementary exploration of longer term research in Inertial Fusion Energy and different spin-off in basic physics, astrophysics, nuclear physics, high density matter, etc. A key aspect in this work is to develop an appropriate design of the different chambers used for the envisioned goals. A main chamber for demonstration will use many of the knowledge already acquired in large Ignition Facilities under construction and almost in full operation such as NIF and LMJ. A review of available data consistent with HiPER will be presented. However, the particles emerging from burnup of the new set of fast ignition targets (different concepts) can change some of the aspect related to neutrons, X-rays, debris and shrapnel effects in the chamber walls. The change could be in the nature of materials, energy and directionality. That will be reviewed under the basis of present knowledge, and following the two-year collective groundwork activity within the HiPER community. An additional chamber proposed in HiPER is that related to accommodate repetitive operation with high-efficiency, diode-pumped driver laser. We will try to give the main parameters and potential consequences expected in such new system, where new physics can appear. In particular, assessment of previous developed work for consequences of repetition will be used. In addition, a chamber could be proposed for specific fusion technology experiments; the envisioned experiments will be described and the possibilities to accommodate in the facility reviewed. All these different aspect of design in considering damage effects (thermal, irradiation, ...), establishment of the layout of beams in the system, tritium handling, diagnosis accommodation, are accompanied with the necessity to evaluate in a proper manner the safety and environmental consequences coming from the neutron activation and potential use of tritium in some phase of the project. A review of methodology to carry out such calculations will be presented and reference numbers from previous experience will be discussed as input of the HiPER concept.

IF/P7-21 · Progress in Inertial Fusion and Fusion Technology at DENIM

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Abstract: Radiation hydrodynamics simulations have been performed for inertial fusion performing numerical tests of radiation transport to quantify the differences between the Sn model and the M1 mode. Radiative shocks experiments hold at high-energy laser installations relevant for astrophysical situations has also been simulated. Simulations using ARWEN are being performed for jet impact fast ignition and compared with present experiments. DENIM has been devoted for years to develop models and codes for computing plasma optical properties. New models (analytical) have been developed for computing NLTE populations, opacities and emissivities for optically thin and thick plasmas, which have been successfully compared with proven codes and experiments. A new methodology for 3D neutronic calculations suitable for complex and extensive geometries has been implemented and applied to the KOYO-FI, and also for description of ITER beam ports and Test Blanket modules. Primary damage behaviour using ACAB code of the low activation steel Eurofer was studied under irradiation in the high flux test module of IFMIF, in the first structural wall of the magnetic (ITER, DEMO) and inertial (HYLIFE-II, HAPL) fusion reactors. Main responses are temporal evolution and average gas production rates and gas to dpa production ratio. The effect of activation cross section uncertainties in the hydrogen and helium production is also analysed. Tritium dispersion study has been extended to new meteorological situations and doses, considering soil and vegetables desorption, are presented. A full system of multiscale modelling codes has been already implemented for study materials under irradiation and under extreme conditions (pressure, etc). Molecular dynamics considering magnetic effects, new parallel kinetic MonteCarlo algorithm and implementation and study of dislocation dynamics, allows studying He effects in Ni and FeCr, comparison with ion irradiation (Fe, FeCr) and modelling metallic foam and ultrahigh strength materials. In addition to work performed in IFMIF-EVEDA safety and test cells, a large effort is being performed to define a Laboratory for Fusion Technology in collaboration with CIEMAT, that includes Remote Handling testing under irradiation, materials irradiation and characterization, advanced materials processing, liquid metal loop and laboratory for numerical simulation.

IF/P7-22 · Integrated Simulations and Respective Simulation Modelling for FIREX-I

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Abstract: It was reported that the fuel core was heated up to ~ 0.8 keV in the fast ignition experiments with cone-guided targets at Osaka University (FI02), but efficient heating mechanisms and achievement of such high temperature have not been clarified yet. To attack this challenging problem, we have been promoting the Fast Ignition Integrated Interconnecting code project. Under this project, the radiation-hydro code (PINOCO), the PIC code (FISCOF), and the relativistic Fokker-Planck code (FIBMET) are integrated with data exchanges to understand overall complicated physics in the fast ignition. According to the integrated simulations for the FI02 experiments, it was found that the density gradient of preformed plasma, which is produced by pre-pulse of the laser at an inner-surface of the cone, strongly affects a generation mechanism of fast electrons, hence efficiency of core heating. As experimental conditions in the FIREX-I project are much different from those in the FI02 experiments, we should not only perform the integrated simulation but also model the respective physics to evaluate the performance of the fast ignition prior to FIREX-I experiments. For example, the heating laser in FIREX-I is designed to have the total energy of 10 kJ but to retain the same intensity in the FI02 experiments because higher intensity generates faster electrons that cannot heat the core efficiently. So the pulse duration is set up to be 10 ps instead of 750 fs, and it's long enough even for heavy Au preformed plasmas to be pushed and compressed by the longitudinal ponderomotive force. The profile steepening, therefore, occurs and the electron density is maximized at the laser front. While the electron density at the laser front goes up, the fast electron beam intensity decreases and the core heating efficiency is also reduced. We perform the integrated simulations in such conditions and evaluate the core temperature. It was found that the characteristic of the dependence of core heating on the preformed plasma parameter in long pulse lasers (FIREX-I) is completely different from that in short pulse lasers (FI02). We will also discuss respective simulation modelling for each code.

IF/P7-23 · Weibel Instability in a Bi-Maxwellian Laser Fusion Plasma

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Abstract: We are interested in this paper to analyse the Weibel instability driven by the plasma temperature anisotropy in the corona of a high intense laser created plasma. The unperturbed electronic distribution function f, of the anisotropic corona is supposed to be a bimaxwellian. That, $T_{par} = T_{per} + W$ where W is the averaged quiver energy of the electron in the laser electric field. The first and the second anisotropies of f projected on the Legendre polynomials are calculated as function of the scaling parameter, W/T_{per}. The Weibel instability parameters are explicitly calculated as function of the scaling parameter. For typical parameters of the laser pulse and the fusion plasma, it has been shown a very unstable Weibel mode is excited in the corona. For low values of the scaling factor, the Fokker-Planck simulation results are found.

IF/P7-24 · Progress in Studies of Ultra-intense Laser Plasma Interactions for Fast Ignition FIREX Project

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Abstract: It is significantly important to understand the interactions between ultra-intense laser and solid targets or plasmas relevant for fast ignition such as laser light self-focusing over critical density and hot electron behaviors leaving targets under strong electrostatic potential created by themselves. Both these features have been studied using ultra-intense lasers. A PW laser pulse 0.5 psec has been injected into an over-dense plasma and has penetrated through the plasm as a sinle channel. The total distance of the penetration was over 100 micron in the plasma which was estimate to have 10 times critical density. This penetration was simulated by a PIC simulation which has shown that the relativistic laser self-focusing was responsible for the penetration. Hot electrons can be produced efficiently with an ultra-intense laser pulse irradiation onto an target. The total number of hot electrons created on the target is always about 2 orders of more than the number measured by an electron spectrometer located a meter away from the target. Hot electrons going through the target are under the effect of strong B field as Alfvén current limit and electro-static potential. The electro-static potential could be controlled by creating a plasma at the rear side of solid planar target which could supply a return current and could increase the number hot electrons by a factor 2. The retardation of the electro-static potential due to the rear plasma is observed in a 1D PIC simulation and could allow hot electrons restricted by the Alfvén limit leaving the target.

IF/P7-25 · Studies of Phenomena Related to Fast Ignition of a Fusion Target

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Abstract: The experimental study and numerical modelling of generation of high-current proton beams of parameters scalable to those required for fast ignition of a fusion target have been performed. The possibility of production of collimated proton beams of MA currents and TA/cm^2 current densities at relativistic laser intensities above 10^{19} W/cm^2 was estimated using a hydrodynamic relativistic 2D code. In this method, ponderomotive forces induced by a short laser pulse near the critical plasma surface drive dense neutralized ion bunches of ion densities comparable to (or higher than) the plasma critical density. It was shown that the key parameter determining the spatial structure and angular divergence of the proton beam is the ratio of the laser beam diameter to the plasma density gradient scale length. The production of proton beams of such extreme parameters was demonstrated in the cooperative experiment performed with the use of 100 TW laser facility in LULI in Palaiseau, in which a subpikosecond laser pulse of intensity of 2×10^{19} W/cm^2 was used as the proton beam driver. Based on these results, a concept of ICF fast ignition using SLPA-produced proton beams was proposed.

IF/P7-26 · Development of Target Injection and Tracking for IFE in Japan

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Abstract: Target injection and tracking have begun to be developed for IFE (Inertial Fusion Energy) reactors. Demanding conditions of injected targets are 300 ± 2 m/s at 2 Hz repetition within ± 1.27 mrad pointing error. The accuracy of irradiated laser on an injected target is required in ± 30 mm. In an attempt to meet these requirements, a coordinated injection-tracking system is pursued by the co-research in Japan. Some of the unique features of this system are: 1) it is a combination of two-stage accelerator with the main acceleration pneumatic gun, and the additional accelerator coil gun, 2) it is a electromagnetic target separator from sabot, 3) it is high accurate target detection device based on Arago spot, 4) it is a beam stirring device of piezo-actuator driven mirror. Using prototype devices of the system, injection velocity exceeding 250 m/s and target detection accuracy of 0.2 micron have been achieved. The demonstration of beam stirring is underway.

IF/P7-27 · Experimental Studies in Mega Ampere Gas Embedded Z-pinch with Different Initial Conditions of Preionization

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Abstract: New experimental results of the gas embedded z-pinch research program of the Chilean Nuclear Energy Commission are presented. Different initial conditions of preionization are studied by using several geometries for the electrodes in combination with a pulsed laser (10 ns FWMH at 1064 nm). The initial conditions obtained correspond to a hollow plasma column and a coaxial double column. The details of the preionization schemes and the structure of the initial plasma are studied by using a small generator (1.2 μF, 30 kV, 180 kA in short circuit, 540 J, 300 ns rise time, dI/dt of 6×10^{11} A/s). Thus, experiments at mega ampere currents using deuterium as working gas are driven by the SPEED2 generator (4.1 μF equivalent Marx generator capacity, 180 kV, 2.5 MA in short circuit, 70 kJ, 400 ns rise time, dI/dt of 10^{13} A/s). The diagnostics used are: current derivative and voltage signals, neutron detections using silver activation counters, and ^{3}He proportional counters; scintillators with photomultilier; and optical refractive diagnostics (schlieren, shadowgraphy and interferometry), using a pulsed Nd-YAG laser (8 ns FWMH at 532 nm). Neutrons have been detected. The stability of the plasma column and the processes of neutron production (thermonuclear vs. beam-target) are discussed.

This work is supported by Bicentennial Program in Science and Technology, PBCT, by the grant ACT26, Center of Research and Applications in Plasma Physics and Pulsed Power Technology, P4 and a Postdoctoral PBCT-Chile grant.

IF/P7-28 · Sheared Flow Stabilization in the Z-Pinch

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Abstract: The stabilizing effect of a sheared axial flow is investigated in a Z-pinch in the ZaP Flow Z-pinch experiment at the University of Washington. Long-lived, Z-pinch plasmas are generated that are 100 cm long with a 1 cm radius and exhibit gross stability for many Alfvén transit times. Stability of the Z-pinch plasma is diagnosed with an azimuthal array of eight magnetic probes that measures the plasma's magnetic structure. Data from these probes are Fourier analyzed to determine the time-dependent evolution of the low order azimuthal modes ($m = 1, 2, 3$). Large magnetic fluctuations occur during pinch assembly, after which the amplitude and frequency of the magnetic fluctuations diminish. This stable behavior continues for an extended quiescent period. At the end of the quiescent period, the fluctuation levels again change character, increase in magnitude and frequency, and remain until the end of the plasma pulse. Plasma flow profiles are determined by measuring the Doppler shift of plasma impurity lines using an imaging spectrometer with an intensified CCD camera operated with a short gate. The spectrometer images 20 spatial chords through the plasma pinch at an oblique angle to the plasma axis to give the instantaneous, axial-velocity profile. Varying the trigger time between pulses provides a measure of the time-dependent evolution of the plasma flow profile throughout the plasma pulse. Experimental measurements show a sheared flow profile that is coincident with the low magnetic fluctuations during the quiescent period. The experimental flow shear exceeds the theoretical threshold during the quiescent period and the flow shear is lower than the theoretical threshold at other times. The observed plasma behavior and correlation between the sheared plasma flow and stability persists as injected neutral gas (and resulting plasma parameters) is varied over a wide range. The quiescent period is seen to increase with the amount of injected neutral gas. Computer simulations have been performed using experimentally observed plasma profiles and show a consistent sheared flow stabilization effect.

IF/P7-29 · Electromagnetohydrodynamic Rayleigh-Taylor Instability at the Ablative Surface of IFE Target Filled with a Poorly Conducting Fluid Layer Bounded Above by a Poro

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Abstract: The effects of combined electric and magnetic fields and densely packed porous lining made up of smart material of nanostructure on Electromagnetohydrodynamic Rayleigh-Taylor Instability (EMHRTI) at the ablative surface of IFE target filled with a poorly electrically conducting fluid is studied using Saffman (1971) slip condition at the interface between a poorly conducting fluid and fluid saturated porous layer. The dispersion relation for the growth rate is derived which is a function of electric parameter We, the Bond number B, the porous parameter σ_p, wave number l and the Hartmann number M. The maximum wavenumber l_m and the corresponding maximum growth rate n_m are obtained. The classical growth rate n_b and the corresponding n_{bm} in the absence of electric and magnetic fields are obtained. The ratio of maximum growth rate $G_m = n_m/n_{bm}$ to the classical growth rate is obtained and it is numerically computed for different values of electric parameter, We and Hartmann number M, for fixed values of the other parameters. We found that an increase in We and M decreases the G_m . From this we conclude that the smaller values of $We(\approx 0.1)$ is more effective in reducing the growth rate of about 70% compared to the classical growth rate. It is interesting to note that when $We = 1$ (i.e., equipartion of energy) the maximum growth rate tends to zero implying symmetry of the IFE target and hence maximum efficiency of IFE can be achieved.

IF/P7-30 · Fast Ignition Impact Fusion

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Abstract: We investigated the physics of historical impact fusion concept, and proposed a new approach, which is a fast ignition version of impact fuion. In this approach, a hypervelocity (500 to 2000 km/s) milimeter size macro particle is shot to a passive centimeter size target, to ignite and burn the target. The macro particle size is much smaller than accepted in earlier impact fusion schemes, so the argument ruling out the electrostatic acceleration is no longer valid. DT methane is used as fusion fuel, instead of common DT ice. Solid state DT methane have denser DT concentration than DT ice. Increased density and involvement of higher-Z ions reduces alpha particle stopping range by 80 percent. In typical temperatures, carbon ions can capture about 10 times more alpha particle energy than DT ions, but they are still in low energy regime, which will thermolize only with DT ions. We also considered the nuclear scattering and neutron energy deposition in DT heating. The larger size of the target means much more DT is burned, and produces much more fusion energy, which can fuel a power plant. In our approach, the fusion fuel is tossed to the reaction chamber, no device or component has to be near the fusion point, which means it is free of the notorious "stand-off" problem of ICF. Primary simulation shows that, with DT methane crystal as the fusion target, and 1 mg diamond as the bullet, the typic ignition energy of the bullet is about 1 MJ.

IF/P7-31 · Progress in Fast Ignition Studies with Electrons and Protons

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Abstract: The short pulse laser ignition process in the fast ignition (FI) concept for initiating burn in a fusion capsule is the critical area of this novel ignition scenario, and also the major area of uncertainty in an ICF concept that has substantial advantages relative to conventional ICF. In order to investigate critical issues for the short pulse ignition process in FI point designs, experiments were performed using the Titan laser facility at FI relevant intensities up to 10^{20} W/cm^2. Results will be presented from a large suite of diagnostics that monitored the electron source characteristics, including laser-to-electron conversion efficiency, and electron energy spectra from flat targets and hollow cones, including sensitivity to pre−pulse induced plasma. Conversion efficiency of laser energy to proton kinetic energy has also been measured and modeled using proton rich target layers. Conclusions will be drawn from both the electron and proton studies relevant to the short pulse ignition process in FI point designs. In particular the new data show a need to revise earlier estimates of the mean energy, or temperature of the electron source, with significant implications for point designs.

IF/P7-32 · Progess in Pulsed Power Driven Inertial Confinement Fusion at Sandia National Laboratories

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Abstract: Significant progress has been made in both the target physics and the driver technologies for the pulsed-power-driven approach to Inertial Confinement Fusion (ICF). Two different approaches to pulsed-power-driven ICF targets are showing promise. The traditional hot-spot ignition approach uses z-pinch-produced x-rays to implode a fusion capsule. Experiments and state-of-the-art 3D radiation magnetohydrodynamics simulations show a promising path for meeting the key requirements of the radiation symmetry, pulse shaping, and scaling of the x-ray output as a function of current. An alternate approach uses the magnetic pressure from the high current to directly compress the fusion fuel to ignition conditions. Quasi-spherical direct-drive configurations are as much as an order-of-magnitude more efficient than the x-ray drive approach at coupling driver energy to the fusion fuel. While direct-drive concepts are less mature, the higher coupling efficiency has the potential to greatly reduce the pulsed-power-driver requirements for ignition and high yield. A new approach to creating efficient, repetitively pulsed, high-current pulsed power devices (the Linear Transformer Driver) has been recently been demonstrated. This configuration uses simple components arranged in a modular way to produce high electrical currents at operating conditions nearly 50 times less stressful than previous pulsed power technologies. While a long term goal of pulsed-power ICF research is the realization of pulsed-power-driven inertial fusion energy (IFE), there may be significant advantages in taking an intermediate step. A recent study of a subcritical fission blanket assembly driven by a fusion source showed that this configuration can drop the fusion power requirements for electricity generation by as much as 150, and the resulting configuration can be used to transmute nuclear waste from fission power plants. This reduction in fusion power requirements could make it an easier first step on the path to fusion energy.

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IT

*** ITER Activities***

IT/1-1 · ITER on the Way to Become the First Fusion Nuclear Installation

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Abstract: At the beginning of 2008, the Director General of the ITER Organization (IO) sent out the request for the Authorization of Creation (DAC files) of the ITER Nuclear Basic Installation (INB) and therefore, following French law, ITER formally became a Nuclear Operator. This paper will present the new licensing process for ITER in application of the so-called "Transparency Law, TSN law", enacted in 2006, and will present examples of practical implications for ITER as a nuclear facility, such as operational domain and safety important components, used codes and standards, dust and tritium management, and waste processing. ITER will therefore be one of the first INBs in France to follow the TSN law and in fact the preparation of the needed safety files has followed the indications of the decree of the 2nd November 2007. In particular, one of the most important documents is the so-called Preliminary Safety Report, RPrS. The writing of this document started in 2006 in a collaborative effort between the ITER Organization and Agency ITER-France under an EU contract; it focuses on the safety demonstration that design provisions have been taken according to the importance of the risk. In particular, RPrS describes safety implications and technical choices ranging from the core of the tokamak to all equipment of the ITER installation. In the context of the ITER Design Review, a dedicated Working Group on Safety has addressed important subjects before submitting the files to the Safety Authorities and has recommended other subjects for future studies. Areas reviewed by the Working Group were dust management, tritium management, operating limits, magnet quenches, codes and standards, remote handling devices, waste processing and tritium releases. Without prejudice to the results of the on-going analysis of the safety documentation by the French Nuclear Safety Authority, it can already be claimed that the achieved integration of nuclear safety requirements in the design has been successfully accomplished, thus giving the basis for the construction, operation and decommissioning of ITER.

IT/1-2 · Principle Physics Developments Evaluated in the ITER Design Review

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Abstract: As part of the ITER Design Review and in response to the issues identified by the STAC, the physics requirements were reviewed and as appropriate updated. This was a worldwide effort, with major contributions from the ITPA. This synopsis will give an overview of the main issues that were addressed and are presently being considered by the IO. The following topics will be discussed: sensitivity of fusion power production to machine parameters (B_t, I_p, κ;); plasma shaping and vertical stabilization; heat load and requirements for plasma-facing components; effect of toroidal field (TF) ripple on fast ion and thermal confinement; Edge Localized Modes (ELM) control; Resistive Wall Modes (RWM) control; and disruptions and disruption mitigation.

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IT/1-3 · Results of ITER Superconducting Magnet R&D

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Abstract: The ITER magnets are among the most time-critical of the tokamak components. Over the last 2 years, as well as finalising the magnet design, the final elements of the magnet R&D have been almost completed. These elements include the resolution of the extreme Nb_3Sn strand performance degradation reported in some conductor samples, a basic demonstration of the performance of the uniaxial glass fibre precompression rings (an important element in the TF coil out of plane support) and the PF conductor performance demonstration in the PF insert coil. The focus of the work is now moving to qualification of components made under industrial conditions, the qualification of processes and tooling and fabrication process optimisation to reduce costs. Leaders in these areas are fabrication routes for the TF coil radial plates and structures, and design/manufacture of the TF coil winding, transfer and insulation tooling. Overall the progress in the work so far shows that a satisfactory performance of Nb_3Sn conductor could be reached; and the manufacture feasibility of ITER magnet is confirmed.

IT/1-4 · Progress in Design and R&D on ITER Plasma Facing Components

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Abstract: The ITER construction phase started last year with a design review covering the physics objectives and all of the major systems. One area where the review had a major impact was the in-vessel components, where it was recognized that a reliably replaceable First Wall is essential for the ITER experimental programme, and that further measures were required to minimise the damage due to plasma contact. The mechanical issues associated with First Wall components in ITER go beyond those previously experienced. For electromagnetic loads, previous scalings, and simple theory, have shown halo current driven forces on in-vessel components to increase with the square of the plasma current, while for induced currents the torques scale with the forth power of the linear dimension. Coping with these extreme loads is compounded by the need to allow for differential thermal expansion due to variable neutronic heating, as well as the direct affect of the neutron exposure on the material properties. This paper expands on the findings of the design review group, and describes how the design of the ITER plasma facing components, and the associated R&D, is being upgraded to fulfil the requirements, whilst coping with the extreme environment.

IT/1-5 · ELM physics and ELM Mitigation in ITER

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Abstract: Localised Modes are characteristic of the H-mode plasmas that are required for the attainment of $Q = 10/500$ MW fusion power in ITER. In these plasmas, type I ELMs are anticipated to release ~ 30 MJ over 250 μs, every 2 s. If 50% of the ELM energy arrives at the targets, the thermal load is 3.75 MJm^{-2} over μs. Dimensional analysis, experiment and modelling all show a limit of ~ 0.5 MJm^{-2} for 250 μs, where evaporation or sublimation of the surface occurs, independent of the material; eg. graphite or tungsten. The consequence of exceeding this thermal load limit is unacceptably rapid erosion of the divertor target, core plasma pollution and, possibly, increased disruption frequency. Thus it is mandatory that ELMs be eliminated or mitigated in ITER. It has been found that the injection of small pellets of solid deuterium, whose velocity is such that they just penetrate the barrier region, will trigger ELMs. The ELM energy is approximately inversely proportional to frequency. This method of ELM mitigation is known as "pellet pacing". A factor of two reduction in ELM energy has been obtained in AUG. An alternative is to use Resonant Magnetic Perturbations to ergodise region around the transport barrier edge region. This reduces the barrier pressure gradient and prevents ELMs by eliminating the MHD instability. Complete suppression of ELMs has been demonstrated for high performance plasmas in DIIID using RMP from the "I-coils". This talk will describe the experiments and modelling that have been undertaken to determine the effect of ELMs on the ITER divertor target. It will then take up the developments in pellet pacing and RMP for ITER. The latter will be emphasized, since much of the recent work has been undertaken as a part of the ITER Design Review and subsequent, related activities. The technical choices will be reviewed and the status of the design of an RMP system for ITER will be described. An aggressive programme of research should be undertaken to identify, develop and test new approaches to ELM control or suppression; especially if it proves to be impossible to implement RMP in ITER.

IT/2-1 · Progress in ITER Heating and Current Drive System

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Abstract: The plan of the ITER heating and current drive systems is based on a rated power of 73 MW delivered to the plasma as start up configuration. The commissioning of those systems is a key part of the early experimentation in the H/He and D phase. As a reference design, ITER heating system is composed of three kind of heating current drive (H&CD) systems, i.e., 20 MW of electron cyclotron (EC H&CD), 20 MW of ion cyclotron (IC H&CD) and 33 MW of negative ion beam (HB H&CD) systems. In this paper, the recent progresses in these heating and current drive systems are described.

IT/2-2 · Experimental Studies of ITER Demonstration Discharges

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Abstract: ITER requires routine operation at 15 MA to achieve $Q = 10$ in H-mode. The main constraints for ITER operation are the proposed additional heating power, the capabilities of the poloidal field set and the maximum power loads to the first wall components. An example of the restricted operational space in ITER is the specification to stay within a range of $l_i = 0.7 - 1.0$. In dedicated experiments all aspects of the discharge scenario have been studied ranging from the current rise phase, the performance during the flat top phase, and a safe ramp down of the plasma. New data are obtained from several devices (C-Mod, ASDEX Upgrade, DIII-D, JT-60U and JET), with key/unique contributions from each of the different machines. This overview summarises and compares the results obtained and documents the requirements for successful operation of the H-mode reference scenario at $q_{95} \sim 3$ and the hybrid scenario at $q_{95} = 4 - 5.5$. These dedicated experiments are supported by interpretation of the plasma discharges with several scenario modelling codes (ASTRA, CRONOS, JETTO, TSC, CORSICA and DINA). The new experimental data provide vital input in validating the transport models used in extrapolating the results to ITER.

IT/2-3 · Development of ITER 15 MA ELMy H-mode Inductive Scenario

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Abstract: Ensuring that ITER has the capability to sustain the range of plasma scenarios required to fulfill its mission has been a significant aspect of the Design Review carried out over the past 18 months. The poloidal field (PF) coil system on ITER, which provides both feedforward and feedback control of plasma position, shape, and current, is a critical element of this capability. These PF coils must support a range of baseline plasma configurations that include the 15 MA $Q = 10$ Inductive, 13.5 MA Hybrid, and $8 - 9$ MA Steady State scenarios. Analysis to date has focused on the 15 MA scenario with a 400 s flattop burn phase, since this is the ITER design basis and is the most challenging for the PF coils to maintain while remaining within all hardware limits. The primary task is to identify the operating space available for the 15 MA ELMy H-mode plasma discharges in ITER, assess its robustness, and identify upgrades to the PF coils or associated systems if required. Time dependent self-consistent free-boundary calculations were performed to examine the impact of plasma variability, discharge programming, and plasma disturbances. Based on these calculations a reference scenario was developed which included a large bore initial plasma, early divertor transition, low level heating in L-mode, and a late H-mode onset. Equilibrium analysis were performed to look at snapshot plasmas during the discharge and solve for coil currents that produce the desired plasma. These indicate that the PF coil limitations are not allowing low l_i (<0.8) operation or lower flux states, and the flattop burn durations are less than the desired 400 s. This motivated the expansion of the operating space in these directions, considering several upgrade options to the PF coils. Analysis was carried out to identify the feedback current headroom required in the CS and PF coils during a series of disturbances. The resulting headroom required on each coil ranges from $6 - 30\%$ of its maximum value. This analysis is critical to quantifying the margins required in coils. Other areas that will be reported include, 1) the plasma current rampdown phase of the discharge and the H to L transition, 2) 17 MA operation assessment, 3) ITER plasma operation and control strategies, 4) validation research and development necessary on existing tokamaks, and 5) impact of heating/CD in plasma rampup V-s savings.

IT/2-4Ra · ITER Plasma Vertical Stabilization

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Abstract: In this paper, we report the results of the modelling, analysis and assessment of the ITER Vertical Stabilization (VS) system design carried out in the last two years by the ITER IO and Domestic Agencies as well as its possible improvements. These include the use of an additional VS system connecting some of the Central Solenoid coils, a power supply upgrade from 6 kV to 9 kV, a re-configuration of some section of the PF coils, the use of passive stabilizers and in-vessel active coils. The assessment is performed on the basis of key figures of merit that are, in most cases, dimensionless and feedback control independent thus allowing ITER to be compared to present day experiments. On the basis of these indicators, different design solutions for possible VS system upgrades are compared. The effects of passive stabilizers connecting the blanket modules are studied by means of approximate 2D axi-symmetric models to assess the increased stability margin. In-vessel coils are modelled by a 3D code and then an equivalent 2D model is derived. This solution brings substantial advantages to plasma control being the vessel internal coils most effective to generate promptly the needed radial field over the plasma region. In the paper, recommendations are given on a strategy for integrated use of internal coils for the control of axi-symmetric and higher order modes. Experimental guidance is of key importance to understand the level to which present-day machines are able to push their VS systems without increased frequency of VDEs. Data are becoming available from several experiments and shall be presented in Ref. [1].

[1] D. Humphreys et al., this conference

IT/2-4Rb · Experimental Vertical Stability Studies for ITER Performance and Design Guidance

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Abstract: Axisymmetric stability control in ITER is expected to be challenging because the target operational scenarios can approach practical controllability limits, while the consequences of loss of control are potentially severe [1]. ITER scenarios require plasma $k_x = 1.85$ and correspondingly high growth rate, particularly at high values of internal inductance that can result during startup or in ohmic, L-mode, or high-q_{95} operations. Loss of vertical control in ITER could result in local forces on blanket modules which approach their allowable limits [2]. Sufficient control performance with adequate margins is thus critical to the success of ITER. We present results of experiments and analysis of operational experience in Alcator C-Mod, ASDEX-UG, DIII-D, JET, NSTX, and TCV. These results include ITPA joint experiments coupled with ITER modelling and model validation, and suggest that improving the vertical control capability of the ITER baseline design may be important to provide robustness comparable to that of operating devices. The present study focuses on "machine-independent" performance metrics that describe the proximity to practical controllability limits rather than ideal stability boundaries. modelling of ITER and analysis of multi-machine experimental data show that ITER is expected to control its baseline design point ($\kappa_x = 1.85$, $l_i = 0.85$) as robustly as operating devices. However, control performance for high-l_i equilibria in ITER, which can occur during rampup and rampdown, corresponds to a level of robustness which guarantees loss of vertical control in operating devices. For example, the maximum controllable vertical displacement of such equilibria in ITER corresponds to $\sim 2\%$ of the minor radius. modelling and experiments in DIII-D and Alcator C-Mod show that operation with this quantity at 2% in both devices corresponds to assured loss of control, while 4% corresponds to marginal controllability.

[1] M. Shimada, et al., Ed, Nucl. Fusion **47** (2007) S1. [2] M. Sugihara, et al., Nucl. Fusion **47** (2007) 337.

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IT/P6-1 · European ITER Site Studies: Lessons Learnt in Safety and Licensing

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Abstract: On June 28th 2005, an agreement was reached among the ITER parties on the selection of EU Site of Cadarache in France. In consequence, the ITER generic site safety assessments had to be adapted to the French requirements in order to get the required license. It is precisely in this point that Europe has been working through the European ITER Site Studies contract framework. The main objective: to contribute to the ITER Organization to fulfill in time the deliverable of licensing files to the French authorities. This paper details the gained lessons by the organizations involved in the writing of supporting studies for these licensing files. The lessons learnt tackle a very wide range of technical and strategic aspects, some of them never considered before for a fusion facility. It could be mentioned, for instance, the application of international codes and standards to typical fusion components, the integration of the approach for accident evaluation, the compilation of safety feed-back from fusion experiments, the definition of waste treatment strategies or the incorporation of the human factor in design and operation among others. The effort has involved EFDA, the current F4E organization, EURATOM associations and European industries. Two facts have made specially challenging the task of support the licensing process: ITER being the first fusion reactor to be licensed as Installation Nucléaire de Base and the entry in force of a new nuclear regulation in France. So original and challenging scenario has lead to adaptation strategies and innovative for fusion approaches in safety and licensing studies.

IT/P6-2 · Fast Ion Losses in ITER

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Abstract: The finite number of the toroidal field coils causes a periodic variation of the toroidal field called the magnetic ripple. This ripple can provide a significant channel for fast particle leakage, leading to very localized fast particle loads on the walls. Therefore ferromagnetic inserts (FIs) will be used in ITER to reduce the ripple. Furthermore, the magnetic inhomogeneity is locally enhanced by the presence of the test blanket modules (TBM). In this work, the ITER wall loads caused by fusion-born alpha particles, NBI-generated deuterons, and ICRH-heated minority ions were studied. This was achieved by simulating particle trajectories in different magnetic field configurations including FIs and none or more TBMs. The simulations were performed with the Monte Carlo guiding-center code ASCOT. The simulations were carried out for two different ITER plasmas: standard H-mode and steady-state operation. In all cases the alphas from thermal fusion overwhelmingly dominated the wall loads. In the absence of a TBM the loads were very small, about 100 $\mathrm{kW/m^2}$. Also the ripple-induced torque was evaluated and it was found comparable in magnitude but opposite in direction to that from the NBI. It must therefore be accounted for when estimating the plasma rotation in ITER. The TBM induces a perturbation to the magnetic field which increases the wall loads substantially. The highest loads were found on the limiters but even with the TBM the fast particle losses were relatively small. In the steady-state scenario, the losses were even smaller than in the H-mode one. This was due to steeper temperature and density profiles, which caused thermal fusions and ionization of NBI particles to occur only in the plasma core. The acceleration of collision time scale was used to reduce the number of time steps. Representing a multitude of orbits by a single orbit also reduces numerical drifts introduced by orbit-following. Acceleration is allowed when the orbits remain unchanged through several poloidal revolutions. In the presence of a toroidal ripple, the banana orbits become non-periodic, but selective acceleration of passing orbits appears still possible. However, when the magnetic field is further perturbed locally by the TBM, the symmetry needed for closed orbits is broken. Therefore the time scale acceleration is no longer allowed even for the passing ions.

IT/P6-3 · Fusion-born Alpha Particle Ripple Loss Studies in ITER

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Abstract: Alpha particles in ITER should be confined well during the burn phase of the discharge and not lost due to the effects of magnetic field ripple. The toroidal field ripple in ITER is designed to be small, 0.2% or less with optimized feritic inserts. However, the insertion of three test blanket modules (TBMs) increases the field ripple quite substantially with a notable increase in the loss of fusion-born alpha particles, according to simulations performed with two particle orbit following codes ORBIT and SPIRAL. we have investigated the fusion-born alpha particle losses with and without the TBMs and found that the losses increase by a factor of two overall with a marked increase in the localization of the losses when the TBMs are present. The heat load on the first wall with optimized ripple (0.2%) and no TBMs is spread uniformly over the outer wall due to the 18-fold symmetry of the toroidal field coil set. In the simulations with the TBMs, three intense hot spots are obtained with heat loads up to 1 MW/m^2 in the center of the TBMs.

IT/P6-4 · Integrated Modelling of Steady-state Scenarios for ITER: Physics and Computational Challenges

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Abstract: One of the primary goals of the ITER design is to demonstrate a long pulse operation prospective for the future tokamak reactor. This could be done with a fusion gain $Q \geq 5$ and pulse duration of 3000 s for a steady-state (SS) scenario. In this work, the modelling of the steady-state ITER scenarios is reviewed, as a subject of common work of the ITPA-SSO group. Focus is made not only on the basic physics issues, resulting from theory and experiments, but also on the difficulties and the needs of integrated modelling. The operational space in ITER is restricted by several physical and engineering limits which, for the reference ITER inductive scenario, were defined by extrapolation of a wide database of tokamak experiments. In contrast, the operational space for the SS operation has not been defined yet. Tokamak experiments indicate that the existence of a high performance long pulse operation regime and its MHD stability is sensitive to the details of the current density and pressure profiles. The complex nature of these plasma scenarios requires a substantial self-consistent integrated modelling effort in the next years to define the SS operational space for a variety of core and pedestal transport models. In fact, the SS scenarios in ITER combine a number of challenges. Long pulse operation requires replacement of Ohmic current by bootstrap and externally driven current. The fraction of the current driven externally is limited by power capabilities of CD systems designed for ITER. The bootstrap current fraction depends on the details of the profiles of the local plasma parameters. There are uncertainties connected with the transport in the plasma core and pedestal regions extrapolated to the ITER regimes. In this work, some specific issues connected with a few combinations of the core and edge transport models with high bootstrap fraction in the long pulse operation are addressed. The bootstrap current can be enhanced either by large pedestal temperatures, or by Internal Transport Barriers (ITB). Recent simulations for both high-pedestal scenarios and ITB scenarios are compared. Results of code benchmarking for typical parameters of ITER steady-state scenarios is also analyzed, and prospects for improvement of the integrated modelling capability will be discussed.

IT/P6-5 · Benchmarking of Neutral Beam Current Drive Codes as a Basis for the Integrated modelling for ITER

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Abstract: Neutral beam injection is a robust method for heating and current drive because it does not depend on any resonance conditions or coupling conditions at the edge. High-energy neutral beam current drive (NBCD) was experimentally validated for central current drive in JT-60U, giving a further confidence in ITER predictions. Recent progress in diagnostics, equilibrium solvers and analysis techniques enable rather detailed comparisons with NBCD codes. However, different codes give somewhat different results. Thus, we need to clarify physics implementations in NBCD codes, such as the beam model, ionization process, fast ion diffusion in the velocity space, orbit effects and electron shielding. Also from an integrated modelling viewpoint, an NBCD code benchmark is needed to establish a more solid basis for ITER operations. A benchmark of the Fokker-Planck code ACCOME has been performed against the orbit following Monte-Carlo code OFMC. Although calculated profiles agree rather well, the OFMC profile is slightly wider than the ACCOME one. The difference in the total fast ion current is $\sim 15\%$. We have examined fast ion diffusion in the 2D velocity space and observed difference in the diffusion in the pitch angle space. We have also examined orbit effects using a point source of the fast ions. Comparison of OFMC runs with and without the drift term in the orbit equation shows the finite banana width effect is not negligible. We have started a new NBCD code benchmark in the frame of the ITPA Steady-State Operation Topical Group with Fokker-Planck codes and orbit following Monte-Carlo codes such as OFMC, ACCOME, SPOT, NEMO, ASTRA, TRANSP/NUBEAM, ONETWO/NUBEAM, DRIFT and TOPICS.

IT/P6-6 · Assessment of Plasma Parameters for Low Activation Phase of ITER Operation

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Abstract: Assessment of plasma parameters is carried out for the low activation phase required for commissioning the basic ITER systems including plasma control, heating and current drive, etc. Such operation is analyzed for helium and hydrogen plasmas for full field and current as well as with magnetic field and plasma current reduced to half of their design values, $B_0 = 2.65$ T, $I_p = 7.5$ MA foreseen for ITER with hydrogen neutral beam injection (NBI). Here we assess the possible operating domain for providing safe operation and possible schemes for commissioning the NBI, electron cyclotron heating (ECH) and ion cyclotron heating (ICH) systems, taking into account the NB shine-through (NBST) loss. Calculations of the NBST loss are carried out taking into account the multi-step NB stopping cross sections for pure helium plasma and hydrogen plasma with 3% of the ^{3}He minority required for ICRH contaminated by carbon. Simulations with the ASTRA code show that injection of 33 MW of NBI in helium L-mode plasmas together with 20 MW of ECH and 20 MW of ICH may exceed the L-H power threshold by a margin sufficient to enter the H-mode and to provide a good H-mode confinement: $P_{NB} + P_{EC} + P_{IC} > 1.5P_{L-H}$ in a plasma density range, $n > 0.75n_G$, sufficient to keep P_{shth} below 0.5 MW/m^2 for half field, half current operation, $B_0 = 2.65$ T, $I_p = 7.5$ MA in a full bore helium plasma, a = 2 m, with hydrogen NBI with the energy of 870 keV and $P_{NB} = 16.5$ MW per injector. For the hydrogen plasma, the operational window for a transition to the H-mode without extra wall protection is expected to be narrower than for the helium plasmas, $0.85 < n/n_G < 0.95$. For low density helium plasmas with $n_e \sim 3 \times 10^{19}$ m^{-3} the NBST is too high for NBI commissioning at the full beam energy without additional measures for wall protection, but the RF power, $P_{IC} = 20$ MW and $P_{EC} = 20$ MW, may become sufficient for transition to the H-mode even without NBI.

IT/P6-7 · Integrated Modelling for ITER in EU

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Abstract: The ITER Scenario modelling Working Group (ISM WG) was organised within the European Task Force on Integrated Tokamak modelling (ITM-TF). The main responsibility of the WG is to advance a pan-European approach to integrated predictive modelling of ITER plasmas with the emphasis on urgent issues, identified during the ITER Design Review. Three major topics were selected, which are considered as and as those, where the WG has the best possible expertise: 1. Modelling of current profile control (including current ramp-up/down and maintaining desired current profile quasi stationary); 2. Modelling of density control in ITER (this includes integrated modelling of core plasma and SOL with both gas puffing and pellet injection); 3. Impurity control in ITER (also involves modelling of core and SOL plasma and deals with He ash removal and heavy impurity accumulation in the plasma core). Different methods of heating and current drive are tested as controllers for current profile tailoring during current ramp up in ITER. This includes Ohmic, NBI, ECRH and LHCD methods. Simulation results elucidate the available operational margins and rank different methods in their ability to meet different requirements. A range of "ITER-relevant" plasmas from existing tokamaks was modelled. Simulations confirmed that the theory-based transport model, GLF23, reproduces the density profile reasonably well and can be used to assess ITER profiles with both pellet injection and gas puffing. In addition, simulations of the SOL plasma were launched using both H-mode and L-mode models for perpendicular transport within the edge barrier and in the SOL. Finally, an integrated approach was also used for the predictive modelling of impurity accumulation in ITER. This includes helium ash, extrinsic impurities (like argon) and impurities coming from the wall (including tungsten). The relative importance of anomalous and neo-classical pinch contributions towards impurity penetration through the edge transport barrier and further accumulation in the core was assessed.

IT/P6-8 · Multimachine Extrapolation of Neoclassical Tearing Mode Physics to ITER

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Abstract: Coordinated experiments from several devices are reported advancing the understanding and extrapolation of Neoclassical Tearing Mode (NTM) physics for ITER. Strong falls in onset β thresholds are found for the most serious mode, the $m/n = 2/1$ NTM, as co-injected torque is reduced, both in conventional and spherical tokamaks. Interestingly, thresholds continue to fall as counter rotation increases. Analysis of mode behaviour suggests this dependence most likely arises from modification to Δ', rather than variations in coupling to other MHD or ion polarisation currents. Encouragingly, no evidence of increased error field sensitivity is found at lower rotation, in contrast to what might be expected theoretically. ρ^* scaling studies indicate a fall in metastable β thresholds for the 2/1 NTM with ρ^*, indicating ITER will operate deeply into the metastable regime, in which modes can be excited. Studies in hybrid scenarios, which rely on high β access, appear to confirm that this trend carries over to mode onset thresholds in some devices. However the trend is not borne out in cross-machine comparisons, indicating that other profile effects (current, rotation, changes fast particle content) can be beneficial in raising mode thresholds. Further studies have extended previous databases for the 3/2 NTM towards ITER-relevant parameters, both in terms of rotation and ρ^*, confirming falls in β thresholds with these parameters. However, progress has been made in controlling thresholds for this mode with extension of current drive sawtooth control techniques to high fast particle, high beta, ITER-like ideal sawtooth regimes and real time control. Results are confronted against theoretical models, with implications and opportunities for ITER identified. Analysis suggests that while overall global parameter trends are adverse for ITER, the most serious concerns for baseline scenario, arising from sawtooth triggered NTMs, may be ameliorated by core MHD control techniques. For higher β scenarios, it appears that falls in NTM thresholds as ITER-like parameters are approached may be limited, and may be beneficially influenced with profile control techniques. This is consistent with theoretical studies indicating that current profile shape can play a crucial role in governing NTM stability.

IT/P6-9 · Prospects for Stabilization of Neoclassical Tearing Modes by Electron Cyclotron Current Drive in ITER

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Abstract: The system planned for electron cyclotron current drive (ECCD) in ITER can mitigate the deleterious effects of neoclassical tearing modes (NTMs) provided that adequate alignment of the ECCD to the rational surface is maintained or too large a misalignment is corrected on a time scale shorter than the plasma response to "large" islands. Resistive neoclassical tearing modes (NTMs) will be the principal limit on stability and performance in the ITER standard scenario as the drag from rotating island induced eddy current in the resistive wall (particularly from the $m/n = 2/1$ mode) can slow the plasma rotation, produce locking to the wall, and cause loss of high-confinement H-mode and disruption. Continuous wave (cw) ECCD at the island rational surface is successful in stabilization and/or preemption of NTMs in ASDEX Upgrade, DIII-D and JT-60U. Modulating the ECCD so that it is absorbed only on the rotating island O-point is proving successful in recovering effectiveness in ASDEX Upgrade when the ECCD is configured for wider deposition as expected in ITER. The models for the effect of misalignment on both the cw and modulated ECCD effectiveness are applied to ITER. Tolerances for misalignment will be presented to establish criteria for both the alignment (by moving mirrors in ITER) in the presence of an island, and for the accuracy of real-time ITER MHD equilibrium reconstruction in the absence of an island, i.e., alignment to the mode or to the rational surface in the absence of the mode. The narrower ECCD with front steering makes the alignment more challenging even though the ECCD is still relatively broad, with current density deposition (full width half maximum) almost twice the marginal island width. This places strict requirements on ECCD alignment with the expected ECCD effectiveness dropping to zero for misalignments as small as 1.7 cm for cw and 2.5 cm for modulation. The system response time for islands transiently exceeding the critical value for locking is also provided for the plasma system controller to be developed. modelling for ITER based on DIII-D mode locking predicts that an $m/n = 2/1$ island 50% larger than critical would take "only" several seconds to lock in ITER. An alignment resolution error of no more than 1 cm and realignment rate of at least 1 cm/s are required.
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IT/P6-10 · Simulations of Material Damage to Divertor and First Wall Armour under ITER Transient Loads by Modelling and Experiments

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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 energy release (TE) from the confined plasma onto plasma facing components (PFCs), which can play a determining role in lifetime of these components. The expected fluxes on the ITER PFCs during transients are: Type I ELM $Q = 0.5 - 4$ MJ/m^2 in timescales $t = 0.3 - 0.6$ ms, and thermal quench $Q = 2 - 13$ MJ/m^2 with $t = 1 - 3$ ms. CFC and tungsten macrobrush armour are foreseen as PFCs for ITER divertor and Be - as FW armour. During the intense TE in ITER the evaporation (CFC, W, Be) and surface melting and melt splashing (W and Be) are seen as the main mechanisms of PFC erosion. A noticeable erosion of CFC PAN fibres and rather intense crack formation for the W targets were observed in plasma gun experiments at rather small heat loads at which the melt damage to W armour is not substantial. The expected erosion of the ITER PFCs TE can be properly estimated by numerical simulations validated against erosion experiments at the plasma gun facilities QSPA-T, MK-200UG and QSPA-Kh50. Within collaboration between EU fusion programme and Russian Federation, CFC and W macrobrush targets manufactured in EU were exposed to multiple ITER TE-like loads with $Q = 0.5 - 2.2$ MJ/m^2 and $t = 0.5$ ms at the QSPA-T. The measured erosion was used to validate the modelling codes developed in FZK (PEGASUS, MEMOS, and others), which are then applied to model the erosion of the divertor and main chamber ITER PFCs under expected transient loads in ITER. Numerical simulations performed for the expected ITER-like loads predicted: a significant erosion of the CFC target for $Q > 0.5$ MJ/m^2 was caused by the inhomogeneous structure of the CFC; the W macrobrush structure is effective in preventing gross melt layer displacement. Optimization of macrobrush geometry to minimize melt splashing is done. Different mechanisms of melt splashing are compared with the results obtained in QSPA experiments. The crack formation at W surface was modelled using the code PEGASUS and validated against the experiments. Simulation of carbon dust production has been performed using the PEGASUS code and validated against MK-200UG experiments. Simulations carried out for Be armour demonstrated that the Lorentz force generates the violent melt motion thus becoming the most dangerous cause of melt splashing

IT/P6-11 · Integrated Modelling of ITER Plasma Dynamics and Wall Processes Following Type I ELMs and Consequences for Tokamak Operation

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Abstract: The anticipated regime of the ITER is the H-mode in which the ELMs can significantly deteriorate the operation. The lost plasma is dumped into the SOL and then impacts on the target producing sputtering and vaporization erosion. The following contamination of core plasma by eroded atoms can interrupt the confinement. For tokamak modelling with account for the transients, the tokamak geometry MHD codes FOREV and TOKES have been developed. The FOREV models the processes on the time scale of 1 ms with the CFC wall. The TOKES is developed for the integrated simulation on the discharge scale (10^3 s), however permitting multiple ELMs. In this code the magnetic field can evolve together with the confined plasma and poloidal field coils automatically control plasma shape. The Pfirsch-Schlüter multi-fluid plasma model, corona radiation transport and the gyro-Bohm plasma transport are implemented. To control fusion reaction the auxiliary heating is automatically varied. The transport in the SOL is implemented based on the guiding center model for the dumped ions. The FOREV modelling of plasma evolution following the carbon influxes revealed that significant impurity contamination of the edge plasma can occur, which can cause large radiation losses and eventually lead to the collapse of the confinement at lesser ELM sizes than that determined by armour lifetime limitations. The TOKES modelled the cases without ELMs and that with ELMs of given energy, for the Be-C-W and Be-W walls. With B-C-W wall, the scattered atoms have significant chances to strike secondarily the dome and thus produce sputtered W-atoms even with the CFC wall at the separatrix strike point. The ELMs cause untimely discharge interruption if the ELM energy exceeds 4 MJ, which corresponds to the vaporization onset. Replacing CFC by tungsten resulted in rather fast termination of even ELMy-free discharge. To validate the codes, a dedicated research has been developed in frame of fusion programme of EU and RF. CFC targets manufactured for ITER have been exposed at the plasma gun MK-200UG in TRINITI. On the basis of these experiments, significant plasma contamination is expected for ITER transients at the target heat deposition above 0.5 MJ/m^2, which agrees with the FOREV and TOKES simulations.

IT/P6-12 · Simulation of ITER Transient Heat Loads to the Divertor Surfaces with Using the High Power Quasi-Steady-State Plasma Accelerator

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Abstract: The behavior of the surfaces of materials, relevant for ITER divertor plates (first of all tungsten), was investigated under the heat loads of wide range ($0.5 - 30$ MJ/m^2) typical for conditions of current disruptions and ELMs in tokamak-reactor. The experiments were carried out with using the powerful plasma streams generated by quasi-steady-state plasma accelerator QSPA Kh-50. Experiments were performed in longitudinal magnetic field of 0.54 T with various numbers of pulses of the time duration $\sim$ 0.25 ms and ion energy up to 0.6 keV. The influence of different forces (surface tension, pressure gradient, Lorenz force etc.) on the erosion properties of tungsten was investigated in experiments with irradiation by powerful plasma streams ($\sim$ 20 MJ per square meter) typical for current disruptions in ITER. The most pronounced erosion of the surface layer was observed due to the pressure gradient of plasma streams along the surface. The role of electric currents at the melt surface layer in the macroscopic melt movement was investigated also. To estimate the range of tolerable loads the effects of ELMs on the lifetime of plasma facing components have been experimentally simulated for large numbers of impacts with varying energy density (with up to 450 pulses of the duration of 0.25 ms and the heat loads in the range of 0.5 – 1.2 MJ per square meter). It was shown that the repetitive heat loads lead to the formation of cracks of two types at the tungsten surface: macro cracks (with the spacing of an order 1 mm) and the net of intergrannular micro cracks (with the step of $10 - 20$ microns). They have principally different physical nature. Under the irradiation of preliminary heated target up to the temperature of 650 centigrade degree (above the temperature of ductile to brittle transition) the macro cracks are not observed while the micro cracks are present at the surface, similarly to non heated targets. Under the irradiation of heated targets by hydrogen plasma streams with heat load of 0.75 MJ per square meter the formations of blisters was observed (it was considered earlier that the blister formation is typical for irradiation with helium ions). Some experiments were carried out with irradiation of combined, tungsten and graphite, targets. The influence of material redeposition at the target surface on its erosion characteristics was analyzed.

IT/P6-13 · Power and Particle Fluxes at the Plasma Edge of ITER : Specifications and Physics Basis

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Abstract: As a result of the ITER design review activities the specifications for power and particle fluxes onto plasma facing components (PFCs) have been revisited. The revised specifications for steady-state fluxes are based on 2-D/3-D SOL modelling code predictions with input from experiments in tokamaks, as well as on simple models and empirical extrapolations and they are provided for all phases of the discharges and scenarios in ITER. In particular, it has been identified that power fluxes along the field can be present for field lines in the far-SOL of ITER (more than $4-6$ cm from the separatrix at the outer midplane) with parallel power flux densities of ~ 5 MWm^{-2}. Evaluation of the expected characteristics of power fluxes to the divertor during ELMs in ITER and experimental measurements of plasma material erosion under such loads has shown that an acceptable divertor target lifetime can only be ensured for ELMs causing an energy loss of ~ 1 MJ or less. This is a factor of 20 smaller than that expected under uncontrolled ELMs and has determined the new requirements for ELM control in ITER. The expected range of energy and power fluxes on main chambers plasma facing components for controlled and uncontrolled ELMs in ITER is described on the basis of present models and empirical scaling of experimental data from various divertor tokamaks. Analysis of the energy outflow from the plasma to PFCs during the thermal quench of disruptions has shown that there is a significant degradation of plasma confinement for resistive-MHD disruptions expected in ITER scenario 2 at full performance. For ITER scenario 4, it is more likely that disruptions will be caused by exceeding ideal-MHD limits. In this second case, and that of VDEs, the plasma thermal energy remains close to its maximum value up to the thermal quench. As a consequence of this and the large broadening of the edge power flux profile at the thermal quench, the parallel energy flux during disruptions (mapped to the midplane) is expected to be in the range of $50-200$ MJm^{-2} for scenario 2 and $200-400$ MJm^{-2} for scenario 4. The specifications of the energy flux during VDEs for ITER have been refined on the basis of analysis of new experimental data that take into account the details of the plasma energy outflow up to the thermal quench of VDEs.

IT/P6-14 · Pedestal Stability Comparison and ITER Pedestal Prediction

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Abstract: The pressure at the top of the edge transport barrier (or "pedestal height") strongly impacts fusion performance, while large Edge Localized Modes (ELMs), driven by the free energy in the pedestal region, can constrain material lifetimes. Investigation of intermediate wavelength MHD modes (or "peeling-ballooning" modes) has led to improved understanding of important constraints on the pedestal height, and a mechanism for ELMs. High resolution pedestal diagnostics, including important recent upgrades, along with a suite of efficient stability codes, have made such analysis routine on several major tokamaks. Numerous comparison experiments between machines have elucidated the role of shape, aspect ratio, beta, collisionality, and rotation on edge stability. Here we present extensive comparisons of observations to predicted stability boundaries on several tokamaks. We find that the edge stability limit provides a constraint on the maximum pedestal height, which is reached in high performance Type I ELM discharges. A number of techniques have been developed to avoid or mitigate large Type I ELMs (e.g., magnetic perturbations, Quiescent H-Mode, Type II/Grassy ELMs), and stability studies on these discharges provide important insight into mechanisms for ELM mitigation. Projections for ITER require predictions of the edge barrier width, as well as stability calculations which provide a constraint on the height as a function of the width (typically height $\sim$ width $\frac{3}{4}$). Using stability calculations as a constraint allows precise testing of models of the pedestal width despite the challenges of accurately measuring the structure of the narrow edge barrier. Such comparisons rule out several types of models for the pedestal width. We present and evaluate candidate width models, leading to a working model which allows prediction of pedestal height on existing machines in the Type I regime with reasonable accuracy, and gives detailed projections for the ITER pedestal height, finding that strongly shaped ITER discharges are expected to have high pedestal temperatures. We also comment on criteria for successful ELM control in ITER.

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IT/P6-15 · Analysis of Performance of the Optimized Divertor in ITER

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Abstract: This paper reports current results of the final optimisation of the set of divertor cassettes to be installed in the initial phase of ITER. The complete re-optimised divertor configuration is investigated in order to determine the divertor performance and particularly the influence of the modifications on the divertor power load, the operating window for detachment, and the pumping capability with its implications for DT throughput and helium pumping. The assessment of performance is accompanied by an assessment of the gain in flexibility as well as of the compatibility of the modified divertor with the reference ITER operation.

IT/P6-16 · A New Look at the Specification of ITER Plasma Wall Interaction and Tritium Retention

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Abstract: The choice of plasma facing component (PFC) materials for ITER is strongly dependent on predictions of heat/particle fluxes as well as the tritium (T) inventory build-up in the vessel, which must be limited for safety reasons. The present knowledge of plasma-wall interaction and processes governing T retention gathered under the framework of the ITPA SOL/divertor group is presented. Estimates of the in-vessel T inventory for both the initial choice of ITER materials, carbon-fibre-composite (CFC), beryllium (Be) and tungsten (W), as well as alternative options. Co-deposition of T with eroded Be and C atoms is known to constitute the largest source for fuel retention when utilizing Be and C PFCs. Implantation and trapping of T in the bulk is the dominant retention process for W, with traps being produced either due to ion irradiation near the surface or throughout the bulk by irradiation with fusion neutrons. Combining all processes and taking into account different fluxes and temperatures on the various sections of the divertor and first-wall, a T level of 350 g is reached with the choice of CFC/W/Be within 200 to 600 discharges, compared to $25 - 250$ discharges for all-C PFCs, and $5000 - 10000$ discharges for a fully-W machine. The low T retention for fully-W PFCs is tempered by tokamak studies (C-Mod) of D retention in single discharges which can be a factor of 10 larger. Possible reasons for C-Mod's higher retention are higher ion flux densities (similar to ITER), and existence of impurities (B, C, O) on, and implanted in, the surface. For W the T retention due to neutron induced trap sites deep in the bulk was also estimated with large uncertainties. After operation times where trap saturation concentrations of 0.6 to 1 at% are reached enhancements of retention over that due to plasma ion irradiation alone were found to be in the range $5 - 8$. A major source of uncertainty of such T retention estimates is the particle fluxes to main chamber surfaces. Several independent analysis of current wall interaction data/understanding and scalings to ITER were made. These disparate methods gave similar values for the radial ion flux density at the limiter radius with total main chamber fluxes between 1 and 7×10^{23} s^{-1} deduced. The corresponding main chamber heat flux was estimated to be in the range $5 - 13$ MW.

IT/P6-17 · Simulations of ITER Disruption and VDE Scenarios with TSC and Comparison with DINA Results

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Abstract: Vertical Displacement Events (VDEs) and plasma current disruptions pose one of the major concerns for the lifetime of in-vessel components in ITER, as well as for machine robustness, as large electromagnetic and thermal loads will induced at such events. Hence, accurate modelling of such events is crucial for estimating disruption induced forces. In the past, ITER disruption modelling has been carried out for ITER using the DINA code. However, since predictive simulations of such events depend on a large number of model assumptions, there exists chances of large error bars on the model predictions. As such it is imperative to validate the code results with other models. Towards this objective, we have carried out the VDE and Disruption simulations using the TSC code and the results are compared with the earlier DINA predictions. A detailed electromagnetic model of the ITER vessel, blankets and the first wall components has been created in TSC. In both VDE and disruption cases, the initial plasma is taken as ITER reference scenario 2 end of burn (EOB) specifications with $I_p = 15$ MA, $B_t = 5.3$ T, $< T_e >= 8.8$ keV, $< n_e >= 1.1 \times 10^{20}$ m^{-3}. The plasma current disruption is initiated by dropping the plasma β in 1 msec, so that after the β crash $< T_e >= 6$ eV, following which the plasma position control is switched off, resulting in a plasma current quench in about 65 msec. On the other hand, in the VDE case, the plasma control is switched off which results in either upward or downward VDE depending on the initial position of the plasma current centroid. In this case the plasma current remains close to 15 MA for a much longer time, about 700 msec in the simulations till the edge safety factor (q) becomes less than 1.5, following which the β is crashed resulting in plasma current quench. Significant differences exist in the DINA and TSC models, for example, even though the plasma current quench rate predicted by the models matches well in till the halo currents start flowing, but in presence of the halo currents TSC predicts a much slower current quench. Details of the simulations and comparison with the earlier DINA results will be presented.

IT/P6-18 · Studies of Requirements for ITER Disruption Mitigation Systems

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Abstract: Recent work conducted under the aegis of the US Burning Plasma Organization related to establishing recommendations for requirements for ITER disruption mitigation systems is described. The recommendations and assessments of the resulting plasma and tokamak operations impacts of a massive-gas-injection disruption mitigation system have been developed in concert with the ITER Fusion Science and Technology Department. Reliability of the disruption mitigation system is identified as a critical concept selection consideration. In addition, the large quantity of deuterium or helium neutral gas ($\sim 5 \times 10^5$ Pa-m^3) needed to provide unequivocal collisional mitigation of runaway electron conversion is found to have significant after-injection impacts on the ITER torus vacuum pumping and exhaust processing systems. These impacts are reduced but not eliminated by employing neon injection. The impact assessments have highlighted needs to limit injected gas quantity to the minimum necessary for adequate runaway mitigation and to optimize the plasma uptake of injected gas or particles delivered to the torus. Use of solid pellet or liquid jet injection methods may have benefits in this latter regard. A variety of alternate or optimized gas, liquid and solid injection mitigation schemes that may offer mitigation efficacy or technology implementation advantages relative to presently-conceived basic-MGI options have been identified. The concluding section of the paper will identify implications of candidate alternate and optimized concepts for ITER and suggest corresponding development/test requirements.

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IT/P6-19 · Pellet Fueling, ELM Pacing, and Disruption Mitigation Technology Development for ITER

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Abstract: Plasma fueling with pellet injection, pacing of edge localized modes (ELMs) by small frequent pellets, and disruption mitigation with gas jets or injected pellets are some of the most important capabilities needed for successful operation of ITER. Tools are being developed at Oak Ridge National Laboratory that can be employed on ITER to provide the necessary core pellet fueling and mitigation of ELMs and disruptions. Here we present progress on the development of the pellet technology to provide reliable high throughput inner wall fueling, pellet ELM pacing with high frequency small pellets, and disruption mitigation with gas jets and pellets. Examples of how these tools can be employed on ITER are discussed. A novel high throughput twin-screw extruder is under development that can supply the 0.3 g/s (~ 1.5 cm^{3}s^{-1} solid DT) needed by ITER for fuel pellet formation. A prototype extruder of this type has been built and is under test for throughput and steady-state performance. A pellet dropper ELM pacing tool has been developed for testing the ELM pacing concept on the DIII-D tokamak. It has obtained 50 Hz 1-mm pellet formation for up to 5 seconds. The dropper achieves 10 m/s pellet speeds for shallow pellet penetration to minimize fueling while providing the necessary localized plasma edge perturbation to trigger rapid ELMs. The ITER pellet system is being designed utilizing these developments to have the flexibility in pellet size and repetition rate to be employed for fueling and as an ELM pacing tool. High flow rate single valve and multi-valve gas jet systems have been developed for gas jet formation and used to mitigate high forces and halo currents in disruptions on DIII-D. The high flow single valve has an orifice diameter of 22 mm and can obtain flows in excess of 1×10^7 mbar-L/s. The multi-valve jet has achieved flows in excess of 3×10^6 mbar-L/s and has a fast rise time of less than 1 msec. Gas jets systems such as these or a large pellet based system are under consideration for use on ITER.

IT/P6-20 · Key R&D Activities for ITER Diagnostics

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Abstract: The design of diagnostic systems for ITER requires active R&D in many areas. The International Tokamak Physics Activity Topical Group (TG) on Diagnostics has identified various topics as “high priority” (HP) and these form the focus of current work. This paper gives an overview of the progress that has been made in the field of diagnostics during the last two years, with emphasis on the various high priority topics (related to alpha particle measurements, neutron tomography, dust and erosion measurements). In addition, the progress that has been achieved in other diagnostic developments for ITER, for example, beam-aided spectroscopy, passive spectroscopy, neutron diagnostics, reflectometry and radiation effects) will be included.

IT/P6-21 · Measurement Requirements and the Diagnostic System on ITER: Modifications Following the Design Review

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Abstract: The requirements for plasma and first wall measurements on ITER, and the planned ITER diagnostic system, have been reviewed by a panel of experts experienced in operating modern tokamaks. The panel made recommendations for changes in both the measurement requirements and the diagnostic system. In subsequent work these changes have been largely adopted and a new baseline diagnostic system has been developed. In this paper we present the main changes arising from this process and we summarise the new baseline diagnostic system.

IT/P6-22 · Progress in Research and Development of Mirrors for ITER Diagnostics

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Abstract: Mirrors will be used as plasma-viewing components in all optical and laser diagnostics of ITER. Operation under impact of intensive particle and radiation fluxes will degrade the optical characteristics of diagnostic mirrors and, hence, the entire performance of the respective ITER diagnostics. R&D program on mirrors is underway on TEXTOR, DIII-D, JET, LHD, HT-7, HL-2A, EAST and ADITYA as well as in various laboratory devices worldwide under coordination and guidance of the Specialists Working Group on First Mirrors established under the wings of the ITPA Topical Group on Diagnostics. The aim of the R&D program is to ensure high-performance, robust and durable mirror solutions for ITER diagnostic systems. The general strategy of the R&D program on diagnostic mirrors will be presented in the paper along with recent experimental results of tests of candidate mirror solutions, technological and engineering developments and predictive modelling of mirror performance in ITER.

IT/P6-23 · Research and Development of Optical Diagnostics for ITER

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Abstract: In ITER, many optical diagnostic systems will be installed to measure plasma parameters. Compared to present day tokamaks, ITER will have greater fusion power and a longer plasma duration. Therefore, the optical components mounted close to the plasma will be exposed to a high amount of radiation from neutrons, gamma-rays and/or high energy particles. As such, the selection of materials and the arrangement of each component are important to prevent degradation of optical properties. The method of cooling is another key, mechanical design element for removing large amounts of nuclear heating power. Additionally, it is important to be able to measure reliably any changes in the sensitivity of the optical systems as a result of exposure to the harsh environment. This article presents design details of common components associated with optical diagnostics such as mirror holders and shutters. It also introduces a new calibration method based on research and development of the Impurity Influx Monitor (divertor). In the present design, the plasma is measured through a small aperture. A mechanical shutter protects a plasma-facing mirror from the bombardment of particles and the deposition of impurities. To reduce the degradation of transmission through windows due to neutron and gamma-ray irradiation, light from the plasma is reflected by metal mirrors through a labyrinth in the neutron shield. The mechanical design of the mirror holders and of the mechanical shutter has been optimized by analyzing expected heat deposition. As a result, the effects in the present design of nuclear heating power are sufficiently removed as a confounding factor. Because the installation of a light source in the vacuum chamber of ITER is not feasible, an in-situ calibration system using a newly developed micro retro-reflector array has been developed. During plasma measurement, the micro retro-reflector array, mounted on the shutter plate, retracts into a sheltered area to prevent particle bombardment and deposition. The micro retro-reflector array made of nickel is manufactured by an electro-forming method. Detected signals have been estimated through measured spectral reflectivity. A signal-to-noise ratio of more than 1000 is expected. These results indicate that the ITER optical diagnostic system may be realized using newly developed optical components and technologies.

IT/P6-24 · Progress in Development of Deposition Prevention and Cleaning Techniques for In-Vessel Optics in ITER

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Abstract: A lifetime of unprotected optical components can be very short due to contamination caused by carbon and beryllium originating from erosion of wall and in-vessel components. The eroded species will be transported inside the vacuum chamber and may deposit in the ducts equipped with diagnostic optical elements. Besides a significant reduction of optical transmission, even thin transparent films of pollutions can dramatically change a shape of reflectance spectra, especially for mirrors (Mo and W) with relatively low reflectivity. Obviously, a development of optics-cleaning and deposition-mitigating techniques is a key factor in the construction and operation of optical diagnostics in ITER. The optical elements positioned in the divertor region are the most vulnerable ones. It is apparent now that neither deposition prevention nor cleaning techniques provides the guaranteed protection of any in-vessel optics. To manage such a challenging task one should use pre-selected techniques, tailored for every optical surface. Last achievements in protection of in-vessel optics are presented on a combination of deposition prevention and cleaning techniques suggested for interior components of Thomson scattering system in the divertor. New techniques for mitigation of deposits have demonstrated the promising results. It has been shown that the blow-off technique is suitable for the first mirror and launcher protection from both dust particles and permanent diffusion flux of hydrocarbons at the conditions (exposure time and fluxes of contaminants) similar to ITER. Additionally, a complete suppression of hydrocarbon deposition has been demonstrated in cold plasma of DC discharge with hollow cathode when 5% N_2 gas was added in H_2/CH_4 mixture. The equivalent cleaning rate can reach more than 25 nm/min (deposition rate without inhibitor was 25 nm/min) in comparison of $\sim$ 3 nm/min etching rate of previously deposited a-C:H films in a H_2/N_2 discharge. Further considerations should be given to estimations of impurity fluxes for every diagnostic mirror, and also to verification of protective approaches, specific for each diagnostic assembly.

IT/P6-25 · Performance Evaluation of ITER Thomson Scattering Systems

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Abstract: Thomson scattering (TS) systems are being designed to be integrated to ITER where electron temperatures of up to 40 keV (substantially higher than any existing tokamak) and densities of up to several times 10^{20} m^{-3} are expected. They are needed for a wide range of physics studies, and some real-time plasma control, so high accuracy and, especially, high reliability is needed. Successful deployment of such a system requires that all components maintain adequate performance throughout the lifetime of the experiment. The parameters accessed by ITER lead to very different operating conditions from existing devices. These range from a high dose neutron environment to in-vacuum mirrors and the extremely long plasma discharges. This paper will assess the expected performance of the proposed systems and highlight critical areas.

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IT/P6-26 · Electrostatic Dust Detection and Removal for ITER

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Abstract: We present some recent results on the application of microelectronics technology to provide innovative solutions to dust inventory measurement and dust removal in ITER. A novel device to detect the settling of dust particles on a remote surface has recently been developed in the laboratory. A grid of two closely interlocking conductive traces on a circuit board was biased to 30˘50 V. Miniature sparks appeared when test particles, scraped from a carbon fiber composite tile, landed on the energized grid and created a transient short circuit. Typically the particles vaporized in a few seconds restoring the previous voltage standoff. The transient current flowing through the short circuit created a voltage pulse that was recorded by standard nuclear counting electronics and the total number of counts was related to the mass of dust impinging on the grid. The short circuit is temporary suggesting the device may be useful for the removal of dust from specific areas. Heating by the current pulse caused up to 90% of the particles to be ejected from the grid or vaporized an encouraging result for the management of dust inventories. A mosaic of these devices based on micro engineered traces on a low activation substrate such as SiO_2 could be envisaged for remote inaccessible areas in a next-step tokamak to ensure that these surfaces remained substantially dust free. A more advanced concept circuit grid is under investigation that would generate a miniature electrostatic traveling wave for conveying dust to a suitable exit port. We have fabricated a circuit board with 25 micron insulated traces that operates with voltages up to 200 V. Very recent results showed motion of dust particles with the application of only 50 V bias voltage. Such a device could remove dust continuously without dedicated interventions and without loss of machine availability for plasma operations. We will report on the further testing of this device.

IT/P7-1 · A Lower Hybrid Current Drive System for ITER

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Abstract: Lower Hybrid Current Drive (LHCD) is one of the four well proven H&CD systems on tokamaks. It is a key element of the present devices addressing the long pulse and steady-state issues, exhibiting a large current drive efficiency, and is fundamental for their inductive flux saving needs. All existing and new superconducting tokamaks (Tore Supra, HT-7, EAST, KSTAR, SST1, JT60-SA, ...) have or plan significant LHCD capability. Since the emergence of the close link between the current profile and the turbulent transport properties, a strong renewed interest was found for LHCD in Advanced Tokamak (AT) researches. JET, Tore Supra, HT-7, FTU, C-Mod make an extensive use of LHCD for AT research towards steady-state operation of fusion devices, as well as for the intermediate so-called Hybrid mode of operation, in which the plasma profile is slightly adjusted to mitigate the main limitations of the ELMy H-mode regime. LHCD will be crucial for the preparation of advanced modes of operation in ITER as the current ramp-up rates are expected to be slow, possibly preventing the establishment of AT scenarios. Finally, LHCD may help ITER increase the safety margins for long pulse operation of its superconducting poloidal coil system, as well as for the initial ramp-up phase of the low-q operation. Avoidance of high-li vertical stability limits is an additional contribution of LHCD under consideration. Though not part of currently planned construction phase, the ITER LHCD system, initially planned for AT and Steady-state studies during its second phase of operation, is now under consideration for an earlier delivery. This paper summarizes an appropriate work plan developed along this line, following recommendations by ITER STAC, after the ITER Design Review process.

IT/P7-2 · Simulating the ITER Plasma Startup Scenario in the DIII-D Tokamak

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Abstract: DIII-D experiments have investigated the ITER startup scenario, including the initial phase where the plasma is limited on low field side (LFS) poloidal bumper limiters. Both the original ITER "small-bore" (constant q_{95}) startup and a "large-bore" lower internal inductance (l_i) startup that avoids vertical disruption events (VDEs) have been simulated. In addition, l_i feedback control has been tested with the goal of producing discharges at the ITER design value, $l_i = 0.85$. These discharges have been simulated using the Corsica free boundary equilibrium code. High performance discharges ($\beta_N = 2.8$, $H_{98y2} = 1.4$) have been obtained in a scaled ITER shape during the flattop phase after the ITER-like startup. These discharges have a figure of merit, $G = \beta_{N^x} H_{89P}/q_{95}^2$, up to 0.40, approaching the ITER value of 0.42 calculated for a fusion gain, $Q = 10$. Important goals of this work are to demonstrate that the proposed ITER startup scenarios are feasible, and to benchmark modelling codes and to help develop future improvements to these scenarios. ITER startup presents unique challenges due to low inductive toroidal electric field (0.3 V/m), limits on poloidal flux due to power supply and poloidal field coil constraints, and plasma current ramp up near the $n = 0$ vertical stability limit. The original ITER startup scenario, beginning as a small volume and growing at constant q_{95}, reached values of normalized internal inductance, $l_i(3)$, much higher than the ITER design value of 0.85. A new scenario, beginning with a large volume plasma was developed that was stable and exhibited a lower $l_i(3)$. Although the discharge was limited on LFS bumper limiters during the initial phase of the current ramp, no deleterious effects from impurities or excessive wall fueling were observed. In order to further control the current profile, l_i feedback was developed and successfully implemented using the plasma current ramp rate as the actuator. Electron cyclotron heating in the breakdown and current initiation phases has also been evaluated producing faster burnthrough of low Z impurities.

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IT/P7-3 · Cryopump Design for the ITER Heating Neutral Beam Injector

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Abstract: Forschungszentrum Karlsruhe (FZK) is developing the cryopumps for the ITER neutral beam injectors (NBI). The cryopumps are pumping all gas fluxes from the ion beam source and the beam line components. The main gas source of all beam line components is the neutralizer with a protium flow of 25.4 Pa m^3s^{-1} next to the smaller gas flows from beam source and the residual ion dump. The cryopumps are based on sorption pumping by charcoal coated cryopanels at a temperature between 4.5K and 6.5K. During detailed investigations of the gas dynamics in the Neutral Beam System it showed up that in the close geometry of the beam line design a cryopump with a gas capture probability of at least 30% is needed to cover the needed low pressures. Therefore, a novel cryopump has been developed which is characterised by an increase in capture probability of 50% which could be achieved at a corresponding increase of the heat load of only 20% compared to a classical cryopump. The new cryopump design is now the reference design for the ITER Heating Neutral Beam System and the mechanical engineering has been started to come up with the detailed design in terms of a CATIA5 model. Two of these cryopumps will be integrated in the beam line vessel, each of them 8 m long and 2.8 m high resulting in an overall pumping speed for hydrogen of 5000 m^3s^{-1}. We discuss the investigations on the gas dynamic calculations for the ITER Heating Neutral Beam system and we summarize the resulting requirements to cover the needed gas profile. The development of the cryopump design is presented in detail accompanied by the results of calculated pumping properties and an outlook to the future work.

IT/P7-4 · High Energy, High Current Accelerator Development for ITER NBI at JADA

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Abstract: The ITER neutral beam (NB) system is constructed so as to inject 33 MW of D^0 beams from two NB injectors at the beam energy of 1 MeV. To meet the requirement, development of a high energy and high current accelerator is in progress at Japan Atomic Energy Agency (JAEA), utilizing two existing facilities. One is the MeV Test Facility (MTF) which is capable of 1 MeV and 0.5 A H^- ion acceleration. In the multi-aperture multi-grid (MAMuG) electrostatic accelerator, called as MeV accelerator, consisting of five stages, acceleration of high current density beams (836 keV, 0.2 A H^- at 146 A/m^2) was reported in the previous conference. The present paper reports countermeasure against backstream positive ion beam as a new issue under high power operation and consequent increase in the beam current to 0.32 A H^- at 796 keV, where the accelerated drain current including electrons was almost power supply limit. In a first beam acceleration test with a single-aperture single-gap (SINGAP) accelerator at the MTF under collaboration with JAEA and CEA, a notable difference of SINGAP from MAMuG was in a relatively high co-accelerated electron current. The mechanism of electron production and acceleration in the unique simple geometry is discussed in this paper. Another facility is the negative ion based NB system of the JT-60U, of which rated beam power is 500 keV and 22 A D^- ion acceleration. A result of three dimensional analysis of JT-60U beamlets shows a beamlet deflection by space charge repulsion. An aperture offset at exit of extractor is suggested effective as the compensation for all beamlet deflections.

IT/P7-5 · -1 MV DC UHV Power Supply for ITER NBI

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Abstract: A dc -1 MV ultra high voltage (UHV) power supply system is required for the ITER neutral beam injector (NBI) to accelerate negative ion beams up to the energy of 1 MeV with the beam current of 40 A for 3600 s. Domestic Agency of Japan (JADA) and Domestic Agency of EU (EUDA) have agreed the procurement sharing for the ITER NBI power supply system. JADA contributes procurement of dc -1 MV ultra-high voltage (UHV) components such as a -1 MV dc generator, a transmission line and a -1 MV insulating transformer. The dc UHV insulation is essential issue for the system, because dc long pulse insulation is different from conventional ac insulation. Voltage sharing is changed from capacitive distribution to resistive one by dc long pulse applying. Electric field distribution for multi-layer (oil/paper composites) insulation structure of the transformer has been studied by simulation for the long pulse operation up to 3600 s. The insulating structure has been designed and the overall dimensions of the dc UHV components have been finalized. In order to realize a stable NBI system, a surge energy suppression system is also essential to protect the accelerator from electric breakdowns. JADA contributes to provide an effective surge suppression system composed of a core snubber and resistors. Input energy from the power supply to the accelerator can be reduced to less than 20 joule which is smaller than design criteria of 50 joule at 1 MV breakdown. From these studies, JADA is ready for procurement arrangement for the UHV components.

IT/P7-6 · RF and Mechanical Design of the ITER Ion Cyclotron Resonance Frequency Antenna

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Abstract: The ITER Ion Cyclotron Resonance Frequency antenna must couple 20 MW at an antenna-plasma spacing of $\sim$ 15 cm for pulse lengths up to 1000 s at frequencies from 40 MHz to 55 MHz, using matching components mounted outside of the torus to allow powering through fast (sub-ms) changes in loading during Edge Localised Modes (ELM's) by the use of either 3dB couplers or a conjugate-T configuration. The chosen design comprises a port plug supporting a close-packed array of 24 straps which are connected in triplets to eight feed transmission lines. Rear sections of the antenna are removable from the rear of the port plug, to allow damaged windows or diagnostics to be replaced, and much of the interior comprises radiation shielding material. The RF specification poses substantial challenges. Computer modelling has been used to maximise the coupled power and/or reduce electric field strength for the straps, feeders and transmission lines, and is now being extended to minimise power loadings caused by sheath effects. The use of closely-spaced straps leads to significant levels of inter-strap mutual coupling that complicates the matching algorithm. Arc detection is also a key issue for this antenna, as recent JET and Tore Supra results have highlighted the need for parallel development of arc detection and ELM-tolerant systems. The mechanical design challenges lie even further beyond the range of present experience. Given the long pulse length, the thermal design dominates much of the detailed mechanical design as peak RF currents of $1-2$ kA will result in high thermal loads; a situation exacerbated by the power loading from the plasma. Resilience to disruption forces has required the design of RF windows that can transmit the forces on the central RF conductors to the port plug structure. The requirement that the rear transmission line section is removable considerably increases the complexity of the mechanical layout. Achievement of the required level of radiation shielding is challenging, given the need to both maintain the water/steel fraction close to the optimum value and to keep the total antenna weight below 45 tonnes. This paper details the RF and mechanical design features proposed for the antenna and outlines the manner in which the wider EU programme will feed into the design process.

IT/P7-7 · Status of the ITER Heating Neutral Beam System

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Abstract: The ITER neutral beam (NB) injectors are the first injectors that will have to operate under conditions and constraints similar to those that will be encountered in a fusion reactor. These injectors will have to operate in a hostile radiation environment and they will become highly radioactive due to the neutron flux from ITER. The injectors will use a single large ion source and accelerator that will produce 40 A 1 MeV D- beams for pulse lengths of up to 3600 s. Significant changes have been made to the ITER heating NB injector (HNB) over the past 4 years. The main changes are: (i) Modifications to allow installation and maintenance of the beamline components with an overhead crane. (ii) The beam source vessel shape has been changed and the beam source moved to allow more space for the connections between the 1 MV bushing and the beam source. (iii) The RF driven negative ion source developed by IPP Garching has replaced the filamented ion source from JAEA, Naka as the reference design. (iv) The ion source and extractor power supplies will be located in an air insulated high voltage (-1 MV) deck located outside the tokamak building instead of inside an SF6 insulated HV deck located above the injector. (v) Introduction of an all metal absolute valve to prevent any tritium in the machine to escape into the NB cell during maintenance. All the development of the ITER accelerators and ion sources has been carried out on relatively low powered test stands, making impossible the full demonstration of the ITER requirements. A Neutral Beam Test Facility (NBTF) will be built to allow the necessary R&D activity. The NBTF will consist of 2 test beds: one capable of operating a complete injector at full performance and a dedicated ion source test bed. The paper will describe the basic ITER heating neutral beam injector, the changes mentioned and a brief description will be given of the NBTF.

IT/P7-8 · Preparing ITER ICRF: Test of the Load Resilient Matching Systems on an Antenna Mock-up

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Abstract: The ICRF system of ITER must couple 20 MW to the plasma in the frequency band $40 - 55$ MHz through a surface of ~ 1.5 m $\times$ 1.9 m, with a wave spectrum appropriate for both heating and current drive. Its matching system must furthermore be resilient to the large load variations that are typically for ELMy discharges. A compact ICRH antenna concept where 24 radiating straps are grouped in 8 triplets by passive junctions has been selected. In view of its complexity the matching systems are studied and tested on a scaled down mock-up. When decreasing the lengths and increasing the frequency by the same scale factor the impedance matrix of the array remains identical realistic simulation of plasma-like load conditions can be obtained by facing the strap array by means of a large dielectric constant medium, such as water. The two selected matching network options to provide the needed load resilience are (i) 4 "conjugate T" (CT) circuits, or (ii) 4 hybrid junctions with reflected power dumped in dummy loads. Two challenges have to be faced hereby: (i) Due to the compactness of the array the mutual coupling effects between the radiating triplets are important and lead to a coupling between all matching actuators and power sources and to an asymmetry in the radiated power distribution. (ii) The low range of antenna loading resistance to be expected in ITER because of the large antenna-LCMS distance renders the adjustment of the matching circuits very critical and amplifies the mutual coupling effects. From the present study the following results and conclusions can be drawn: (i) Decouplers neutralizing the dominant coupling terms of the input admittance matrix are mandatory; (ii) Both matching options, if well adjusted, have good performances. The advantages of the hybrid option are its potential resilience for any value of the coupling resistance and its larger insensitivity to reactive load changes. The load resilience domain for the CT option becomes small if the mean plasma loading is too low. The advantages of the CT matching are its requirement of a lower number of components and the lower number of matching parameters requiring feedback control. (iii) For both options the generators must be combined to provide 4 power sources exceeding 5 MW. Matching procedures and automatic algorithms have been developed and were successfully tested on the mock-up.

IT/P7-9 · Physical Performance Analysis and Progress of the Development of the Negative Ion RF Source for the ITER NBI System

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Abstract: For heating and current drive the neutral beam injection system for ITER requires a 1 MeV deuterium beam for up to 1 hour pulse length. In order to inject the required 17 MW the ion source has to deliver 40 A of negative ion current at the specified source pressure of 0.3 Pa resulting in a large area source (1.8m $\times$ 0.9m). In 2007 the IPP RF driven negative hydrogen ion source was chosen by the ITER board as the new reference source for the ITER NBI system due to its in principle maintenance free operation and due the success in development over the last years. Although the IPP RF source has made substantial progress towards ITER's requirements there are still open issues to be addressed. Apart from the homogeneity of such a large RF source and the long pulse stability, a very critical factor is the amount of co-extracted electrons limiting also the maximum achievable ion current density. For all these issues, the control of the complex plasma chemistry in the source is the most critical item as cesium evaporation is needed for the production of negative hydrogen ions in sufficient quantities. Overall objectives are to identify tools for control of the source performance, to contribute to the design of the NBTF and ITER RF source, and to define a proper minimum set of diagnostics. Thus, the performance analysis of the IPP RF sources is strongly supported by an extensive diagnostic program and modelling of the source and beam extraction. The development efforts at the IPP test facilities are now focused on the achievement of stable pulses for up to 1 hour at the small size long pulse test facility MANITU and on demonstration of a sufficiently homogeneous large cesiated RF plasma operation with negative ion densities in the required range of few 10^{17} m^{-3} at the large ion source test facility RADI. MANITU is operating now routinely at stable pulses of up to 10 min with parameters near the ITER requirements; RADI demonstrated that a pure deuterium plasma (without Cs, bias, and filter field) is quite uniform. Due to the lack of extraction at RADI and in order to have an intermediate step between the MANITU and the ITER RF source, IPP is presently designing the new test facility ELISE for long pulse plasma operation and short pulse, but large-scale extraction from a half-size ITER source. Commissioning is planned for the end of 2009.

IT/P7-10 · ITER First Wall Fabrication Technology in China

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Abstract: China will manufacture 10% of the ITER first wall (FW) panels as her in-kind contribution to ITER. To ensure the manufactured panel in required quality, the fabrication technology should be qualified by fabrication and testing FW qualification mock-ups. One of the key issues for the fabrication is the formation of Be/Cu intermetallic phases that increase the brittleness of the joint. Therefore, the main objective of this study on the joining technology is to find ways to reduce the brittle phase formation, and the fabrication of a mockup satisfying the ITER-FW requirements. China started the study on the joining technology in 2005. Various interlayer of Ti, Al, AlSiMg alloy and Cu in forms of foil and coating were investigated with the joining method of HIPing at high temperature ($> 850^{\circ}$C) and relatively low temperature (580°C). By analysis of the interface microstructure and measurement of bounding strength, a relatively optimized joining technology was obtained in 2006. The technology was that (1) joining Be to CuCrZr alloy by HIPing at 580°C / 140 MPa/ 2 hr with Ti/Cu interlayer of coating and (2) joining CuCrZr to SS (stainless steel) plate and tubes by HIPing at 1040°C / 130 MPa/ 2 hr without interlayer material. It is required that the FW mockup and the semi-prototype should use qualified materials. Within one year, high purity Be pebbles ($> 99.5\%$) was produced in China. With a vacuum-hot-pressing (VHP) method of powder processing, small ingots of Chinese VHP-Be of $> 99\%$ purity was made. Its BeO content was less than 1%. Chemical composition and microstructure were analyzed; both meet the ITER's requirement. The physical and tensile properties of the material are similar to those of the VHP-S65C-Be made in Brushwelman in US. In the past two years, to improve the joining technology, several FW mock-ups have been fabricated. They were analyzed by NDT and destructively tested. Results showed that there was no interface defects larger than 2 mm in diameter. Ti_3Cu_4 inter-metallic phase was observed at the interface. Be/Cu bounding strength (shear test mode) reached up to more than 200 MPa at RT. The mechanical properties of CuCrZr alloy varied in the HIPing process. Both EU and China Cu alloy could satisfy the ITER requirement of the mechanical strength after the mock-up fabrication. However, only the alloy used in forged state could meet the requirement of grain size.

IT/P7-11 · Study on Characteristics of Dissimilar Material Joints for ITER First Wall

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Abstract: The first wall (FW) of ITER blanket includes the beryllium (Be) amour tiles joined to CuCrZr heat sink with stainless steel (SS) cooling tube and backing plate in order to improve plasma performance and reduce thermal stress. Since these joints must withstand thermal and mechanical loads caused by the plasma and electromagnetic force, it is important to evaluate the strength and fatigue life of the joints. Qualification tests for the FW fabrication technology are performed in order to maintain the ITER operation reliability enough. This paper describes the investigations of mechanical properties of the joints, namely the tensile strength and the characteristics of stress and strain distribution of the joints. When the dissimilar materials joints are subjected to external force, the stress distribution shows singularity and large stress concentration at the interface caused by the discontinuity of materials properties depending on the materials combination. The characteristics of stress singularity were analyzed for the Be/CuCrZr and CuCrZr/SS joints. As a result of tensile test for the joints, the Be/CuCrZr joint fractured at the interface because of large stress concentration at the interface. For the CuCrZr/SS joint, however, the joint fractured at CuCrZr apart from the interface. There is insignificant stress concentration at the interface of CuCrZr/SS joint, because the deformation properties between SS and CuCrZr are almost same. Elasto-plastic finite element analysis were also carried out to clarify the deformation characteristics and fracture of the tensile specimens for the joints. As a result of the analysis for the CuCrZr/SS joint, the deformation of CuCrZr at the interface was restricted from the SS and the stress and strain were small because of its triaxial tensile stress state. The maximum strain occurred at the CuCrZr side apart from the interface and the strain of CuCrZr was larger than that of the stainless steel, since the deformation stress of stainless steel was large rather than the CuCrZr. This result designates that the tensile specimen fractures at the CuCrZr apart from the interface.

IT/P7-12 · Progress on the Development of the Fabrication Technology for the ITER First Wall in Korea

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Abstract: Korea is responsible for the procurement of the ITER first wall (FW) blanket modules 1, 2 and 6, and has been developing the technology for their fabrication. Joining Be, CuCrZr and stainless steel (SS) is the most important aspect of the FW since it has to survive the normal and transient heat loads produced during the various ITER operating conditions. This paper addresses the optimum joining conditions, the results of the destructive test and the high heat flux test of the joints. Fabrication of the Be/CuCrZr/SS qualification mock-ups using the developed joining technology is progressing and a high heat flux test for mock-up qualification is planned for 2008.

IT/P7-13 · Qualification Tests and Facilities for the ITER Superconductors

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Abstract: All the ITER superconductors have been tested as short length samples in the SULTAN test facility at CRPP since over 10 years. The After the test of the Model Coil conductors, the design was updated in 2004. About 20 conductor samples with small layout variations were tested in the last three years with the double aim of verification of the design and qualification of the manufacturers. The assembly of the SULTAN samples from short lengths of ITER conductor sections manufactured all over the world, is done at the CRPP workshop, where the measurement techniques and the data acquisition/processing have been refined to improve the accuracy and reliability of the performance assessment. The qualification phase of the ITER conductors in SULTAN will likely extend into 2009. Upon confirmation of the target performance, the series production of the ITER conductors will start at the industry and the test in SULTAN of specimens taken from the large scale production will be used as quality control and acceptance test. Starting in 2010, another test facility for ITER conductors, named EDIPO, will be operating at CRPP to share with SULTAN the load of the samples for the acceptance tests. The preparation work to host the EDIPO facility is on-going at CRPP.

IT/P7-14 · EU Contribution to the Test and Analysis of the ITER Poloidial Field

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Abstract: The PFCI is a 45 m long ITER-type NbTi dual channel Cable-In-Conduit Conductor, designed to be representative of the one currently proposed for the ITER PF 1&6 coils, and wound into a single-layer solenoid. The PFCI, well instrumented from both the thermal-hydraulic and the electromagnetic point of view, is now installed in the bore of the ITER Central Solenoid Model Coil at JAEA Naka, Japan, and ready to be tested at nominal 45 kA and 6 T external field in June-July 2008. The test will concentrate on: current sharing temperature and critical current measurements, AC loss measurements, stability and quench propagation. The major test results of the PFCI are reported and discussed in the paper, based on the predictive/preparatory analysis and successive interpretation performed within the different EU laboratories, and a comparison with the short sample results is presented. Some emphasis will also be put on the validation of the models used to predict the experimental behavior of the PFCI. This point is obviously important, in view of the even more complex task of predicting the performance of the ITER PF coils and providing input to finalize the design of the PF conductor.

IT/P7-15 · Recent Progress of GIS/GDC Design and Manufacturing for ITER

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Abstract: Based on the Design Description Document (DDD) of Fueling and Wall Conditioning System, the design and manufacturing study of the GIS and GDC have been carrying on in CN PT. The recent progress of the GIS and GDC is introduced respectively: the progress for some key techniques has been obtained; the detailed designs of GDC electrode have been finished; some critical components in GDC system have been made; the key components of GIS have been investigated and the structure of GVB, which based on the candidate valve and components, has been designed; the detailed designs of the straight segment and adaptor in gas supply manifold are being done; the test schemes for the key components of GIS and GDC have been proposed; some tests of the critical components have been done and the preliminary data have been got.

IT/P7-16 · Implications of Increased Gas Throughputs at ITER on the Torus Exhaust Pumping System

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Abstract: The reference design of the ITER torus exhaust pumping system is based on 8 cryopumps, connected via 4 ducts to the torus. The exhaust gas flows will pass through the divertor into the pumping slots, at divertor pressures between 1 and 10 Pa. The pumping port and divertor geometry has limited conductance which reduces drastically the available pumping speed for the whole machine. Based on the transitional flow code ITERVAC, a conductance study is currently going on to model the complete divertor and pumping port geometry as a network in full detail and to assess the overall conductance at varied flow range conditions. This paper will report the relevant results of this study. The reference fueling rate has been recently increased; the resulting changes in the pump heat loads and operating envelopes are discussed. The maximum allowed hydrogenic gas accumulation is limited by oxy-hydrogen deflagration safety considerations: increased gas flows will lead to shortened pumping times. As the pumping system is operated in transitional flow regime, increased flows will change the overall conductance and directly re-influence the available divertor pressures. Higher gas flows will result in higher pressures inside the cryopump volume and lead to increased heat loads on the 4.5 K and 80 K cooling circuits of the cryopumps. Depending on the heat loads, the outlet temperatures of the coolant will tend to increase which, if unmitigated, could affect the pumping speed for helium, or, alternatively, the cryogenic flows might needed to be increased, which then has an impact on thermohydraulics and pressure losses across the pumps cryocircuits. To mitigate the potential deleterious influence on helium pumping speed, means to provide a lower coolant inlet temperature are being studied by ITER. Massive gas injection is the proposed concept to safely terminate a disrupting plasma discharge. Many aspects of this type of disruption mitigation need more investigation, but the first results of modelling the effects of massive gas injection are given for the current design of the ITER pumping system. This paper will present the results of a study on the different effects and their mutual influences and consequences. As ITER is an experimental device, it would be desirable to add as much flexibility to the pumping system as possible without increasing cost significantly.

IT/P7-17 · Provisional Procurement Acitivity and R&D's on Divertor HHF Components in JADA

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Abstract: Japan domestic agency (JADA) will procure the ITER divertor components in the cooperation with other DAs in EU and RF after qualification processes. In the qualification processes that started in 2007, JADA need to demonstrate its capability of the divertor procurement through a specific divertor mock-up, called a qualification prototype (QP), which includes most of the technical specifications of the ITER divertor and a quality assurance system of material and fabrication process is applied to it. JADA is now carrying out further provisional activity on the divertor procurement under close collaboration with the ITER Organization (IO), especially joining plasma-facing materials to heat-sink material in order to simplify the fabrication process that is more suitable for the series production of the ITER divertor components compared with the past mock-ups[1]. The mock-up fabricated by using a new joining process shows its durability against the repetitive heat load of more than 20 MW/m^2. Based on this, JADA starts to fabricate QPs.

IT/P7-18 · R&D Activities for ITER Blanket Remote Handling equipment

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Abstract: Maintenance of ITER blanket is carried out in the vacuum vessel (VV) by the remote handling equipment. For this purpose, in-vessel transporter (IVT) with a rail mounted vehicle manipulator, which consist of vehicle manipulators working on rail transporter forming a toroidal ring structure, has been adopted. The number of the blanket modules is about 400 and installed in the VV. The dose rate of gamma ray is expected about 500 Gy/h during blanket maintenance. R&D to clarify the specifications of the detail design for IVT in JApan Domestic Agency (JADA) has been performed with feasible outputs, i.e., force sensor to avoid the overload between blanket and keys during blanket installation, dry lubricant to prevent the lubricant oil from spreading in the VV. In addition, rail connection and cable handling in the transfer cask, which are critical issues for IVT system, are under preparation of the demonstration tests to finalize the design of the IVT system. These test facilities will be assembled by the end of March 2008, and the performance tests will be carried out from April 2008. The present paper describes the recent progress of the R&D of the IVT system for blanket maintenance.

IT/P7-19 · DTP2 - Verifying the Divertor Remote Handling Equipment for ITER

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Abstract: For the long term availability of the ITER reactor, one of the key issues is the reactor maintenance. An essential tool to meet this goal is the use of virtual techniques and virtual prototyping in design phase. The environment and tools are presented as 3D-CAD models. By adding kinematic and dynamic features, the maintenance tasks, robots and tools can be designed and tested through the complete modeled tasks in real-like virtual environment. However, the virtual prototyping shows just the considered and modeled behavior. The real hardware is needed to prove the clearances, show the flexibility of structures, to point out the weak points of the proposed design, like the combination of clearances, flexibility and the control accuracy. The DTP2 facility is hosted and operated by the Finnish Fusion Association Tekes. The DTP2 facility provides necessary elements for verifying the operations for replacing the ITER divertor, from the reactor port opening to the point maintenance port is sealed again. After installation and commissioning of the DTP2 components, the second cassette replacing trials will be conducted. Later, the DTP2 facility will be extended to provide testing of the most critical divertor maintenance operations. During the test program in DTP2 , the maintenance sequence will be developed and recorded, potential hazards are analysed and recovery schemes are trained. The DTP2 will produce a lot of information impossible to obtain otherwise. In the paper, the DTP2 structure, principles and operations will be explained more closely.

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IT/P7-20 · Progress on the Design of the ITER Tokamak Assembly Tools

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Abstract: ITER tokamak assembly is mainly composed of lower cryostat activities, sector sub-assembly, sector assembly, ex-vessel activities and in-vessel activities. Conceptual design of the main tools for lower cryostat activities, sector sub-assembly and sector assembly has been developed to satisfy the ITER basic assembly concept. The upending tool, sector sub-assembly tool, sector lifting tool and vacuum vessel (VV) support and bracing tool for sector sub-assembly procedures have been developed and are described herein. The basic structure of the upending tool has been developed with the assumption that lifting will be performed with a crane which will be installed in the tokamak building. The sector sub-assembly tool is composed of special adjusting frames for fine position control with 6 degrees of freedom. The sector lifting tool is designed to adjust the position of a sector to minimize the difference between the center of the tokamak building crane and the center of gravity of the sector. Finally, the VV support and bracing tool has been developed to support four kinds of 40 degree VV sectors. Also, structural analysis has been performed on the assembly tools using ANSYS for the situation of an applied load with the same dead weight multiplied by $\frac{4}{3}$ in order to take uncertainty into consideration. The results of the structural analysis for the assembly tools show tolerances within allowable values. Work continues to develop the conceptual design of the ITER Assembly Tools for ex-vessel and in-vessel activities by September 2008.

IT/P7-21 · Design Progress and Analysis for ITER Thermal Shield

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Abstract: The Thermal Shield (TS) system of ITER plays the role of reducing the heat load transferred by thermal radiation and conduction from warm components to the components and structures that operate at 4.5K. The ITER Thermal Shield consists of a central TS, an upper TS and a lower TS. Thermal radiation to the superconducting magnets is minimized by operating the thermal shields at low temperature and by providing surfaces with low emissivity using silver coating. The detailed design of joints is done to satisfy both structural stability of the TS and assembly and manufacturing feasibility. Structural analysis under various design loads was conducted in order to support the new joint design. Detailed design of cooling panels was performed to minimize panel deformation due to tube welding by applying a new tube attachment scheme and cooling line layout. The results of thermal analysis and calculations of the pressure drop in each cooling line are reported in this paper.

IT/P7-22 · Status of R&D Activities on the ITER Tritium Storage and Delivery System

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Abstract: ITER is designed to operate at a nominal fusion power of 500 MW using deuterium and tritium as fuels delivered from a tritium plant. The tritium storage and delivery system (SDS) is one of the major components of the tritium plant. The SDS is designed to store and deliver the tritium and deuterium inventories in the ITER tritium plant. For ITER construction, Korea Domestic Agency (KO-DA) will deliver the tritium storage and delivery system. The system is currently in the detailed design stage to prepare for its procurement. This includes finalizing system composition and layout, completing detailed design of system composition, equipments including the metal-hydride getter bed and layout, and detailed definition of testing (factory and on-site) testing, on-site installation and commissioning. The on-going R&D activities are presented in this paper. R&D activities for the detailed design of the ITER SDS have progressed in two directions: improvements of the getter beds to increase hydrogen recovery and delivery rates from ZrCo under various conditions, the other to investigate the performance of several unit processes that are related to the main operation procedures, both for which experiments have been or will be performed.

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*** Overviews***

OV/1-1 · Overview of Physics Research on the TCV Tokamak

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Abstract: The TCV tokamak is characterised by high-power (4.5 MW), real-time-controllable ECRH and flexible shaping, and plays an important role in fusion research by broadening the parameter range of reactor relevant regimes, by exploring fundamental tokamak physics questions and by developing new control tools. Steady-state discharges are achieved, in which the current is entirely self-generated by the plasma through the bootstrap mechanism, a fundamental ingredient for ITER steady-state operation. The discharge remains quiescent over several current redistribution times, demonstrating that a self-consistent, "bootstrap-aligned" equilibrium state is possible. Electron ITB regimes sustained by EC current drive have also been explored. MHD activity is shown to be crucial in scenarios characterised by large, slow oscillations in plasma confinement, which in turn can be modified by small Ohmic current perturbations altering the barrier strength. In studies of the relation between anomalous transport and plasma shape, the observed dependences of the electron thermal diffusivity on triangularity (direct) and collisionality (inverse) are qualitatively reproduced by nonlinear gyrokinetic simulations and shown to be governed by trapped-electron-mode turbulence. Parallel SOL flows are studied for their importance for material migration; by changing from lower to upper single null diverted equilibria and shifting the plasmas vertically, flow profiles are measured using a reciprocating Mach probe. The dominant, field-direction-dependent Pfirsch-Schlüter component is found to be in good agreement with theoretical predictions. A field-direction-independent component is identified and is consistent with flows generated by transient over-pressure due to cross-field, ballooning-like interchange turbulence driven flux. Initial high-resolution infrared images confirm that ELMs have a filamentary structure, while fast, localised radiation measurements reveal that ELM activity first appears in the X-point region. Real-time control techniques are currently being applied to ECRH control, given the availability of multiple independent power supplies and beam launchers, e.g., to control the plasma current in fully non-inductive conditions and the plasma elongation through current broadening by far-off-axis heating at constant shaping field.

OV/1-2 · Overview of JET Results

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Abstract: Since the last IAEA conference three experimental campaigns have been executed on JET. The programme focussed on the qualification of the integrated operating scenarios for ITER and on physics issues essential for the efficient exploitation of ITER. The ITER baseline scenario (ELMy H-mode with Type I ELMs) has been investigated up to 3.5 MA / 3.2 T ($q_{95} \sim 2.9$, plasma stored energy $W \sim 9.5$ MJ, ELM energy $W_{ELM} \sim 0.6$ MJ). The hybrid scenario has been extended in parameter space ($2.7 \leq q_{95} \leq 4.5$, β_N up to 3.6, density up to the Greenwald limit n_G at $\beta_N = 2.7$, discharge duration up to $20s$ at $\beta_N = 2.5$) and a systematic comparison with the ITER baseline scenario has been made at high triangularity (~ 0.45) and $\beta_N \sim 3.0$. High β_N (~ 3.0 for one resistive time), high triangularity Advanced Tokamak experiments have demonstrated the importance of the q profile shape for accessing operational domains beyond the experimental "no-wall MagnetoHydroDynamic (MHD) limit". Specific ITER-relevant studies using the unique JET capability of varying the Toroidal Field ripple demonstrated that increased toroidal field ripple in ELMy H-mode plasmas have a detrimental effect on plasma stored energy and density, specially at low collisionality ($H_{98}(y,2)$ is reduced by $\sim 20\%$ for $\delta_{BT} = 1\%$, $\sim 10\%$ at $\delta_{BT} = 0.5\%$; steady state density lower by $\sim 40\%$), with confinement loss due to pedestal degradation.These results suggest that $\delta_{BT} < 0.5\%$ is required on ITER to maintain adequate confinement to allow $Q_{DT} = 10$ at full field. Various ELM mitigation techniques have been assessed for all ITER operating scenarios on JET using active methods such as resonant magnetic field perturbation, vertical stabilisation and pellet pacing. For instance, the amplitude and frequency of Type I ELMs have been actively controlled over a wide parameter range ($q_{95} = 3-4.8$, $\beta_N \leq 3.0$) by adjusting the amplitude of the $n = 1$ external perturbation field induced by Error Field Correction Coils (EFCCs), with an increase in the ELM frequency up to ~ 120 Hz and simultaneous decrease in the D_α signal by one order of magnitude, the pedestal electron temperature drop $\Delta T_{eped} \sim 100 - 200$ eV and normised ELM energy loss $< 2\%$.

OV/1-3 · Overview of JT-60U Results towards Establishment of Advanced Tokamak Operation

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Abstract: The development of steady-state advanced tokamak (AT) plasmas is important to realize economical fusion reactor. High confinement, high normalized β (β_{N}), high bootstrap current fraction (f_{BS}) and heat/particle handling are important parameters for keeping AT plasmas steadily. Recent JT-60U experiments have been focused on the expansion of operational regime of AT plasmas using improved flexibility of NBIs by the modification of power supply and on the investigation of important issues for steady-state AT plasmas using new diagnostics. In this paper, recent JT-60U experimental results since the last IAEA conference are presented. High β_{N} of 2.6 at high $H_{\mathrm{H98(y,2)}} > 1$ was sustained for 25 s in the positive shear plasma. Fully non-inductive plasma having relaxed current profile with high $f_{BS} = 0.5$ was obtained for 2 s in the weak shear plasma. These plasmas contribute to establish AT plasma operation in ITER, JT-60SA and DEMO. In ELMy H-mode plasmas, high density ($\bar{n}_e/n_{GW}$=0.75-0.84) and high confinement ($H_{\mathrm{H98(y,2)}} \sim 0.77 - 0.84$) was sustained for about 13s at high radiation loss fraction ($f_{rad} = 0.82 - 0.90$) by utilizing Ar injection with real-time feedback system. In this plasma, type I ELMs were replaced with type III ELMs. Thus, no detectable ELM energy loss was observed. A suppression of RWM by a toroidal plasma rotation is demonstrated. With the plasma rotation exceeding a critical rotation velocity of $\sim$ 20 km/s for suppressing RWM, higher β_{N} of 2.8 than no wall ideal MHD limit of $\sim$ 2.4 was sustained for $\sim$ 2 s. In the plasma, the sustained duration was limited by the degradation of the plasma confinement after an $m/n = 2/1$ mode was locked. In other discharge terminated by the disruption due to growing of RWM, a precursor of RWM ($n = 1$) was observed for $\sim$ 40 ms. As the precursor grew gradually, the plasma rotation and/or its shear near the $q = 2$ surface decreased.

OV/1-4 · DIII-D Research in Support of ITER

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Abstract: DIII-D research is providing key information for the design and operation of ITER. Recent results have led to improved understanding of several critical issues, including stability control, plasma rotation effects, mitigation of disruptions, and plasma-wall interactions. In addition, DIII-D experiments that simulate the various ITER operating scenarios provide a platform for projections of fusion performance and tests of plasma control. Progress has been made in understanding the physics of instabilities that may occur in ITER and in developing strategies to control them, including edge localized modes, neoclassical tearing modes, resistive wall modes and core-localized Alfvén eigenmodes. Disruption mitigation experiments with a new fast risetime gas injection system show the importance of delivering the gas to the plasma before the thermal quench ends, in order to achieve the high densities needed for suppression of a runaway electron avalanche in ITER. Internal control of the plasma is being developed using a range of actuators; recent results include simultaneous feedback control of plasma β and rotation by independent control of neutral beam power and torque. Using these tools, a more detailed scientific basis for extrapolating key performance measures to ITER is being developed, including studies of energy confinement and L-H transition with low neutral beam torque and T_e/T_i approaching unity. DIII-D research is developing approaches to the control of plasma-wall interactions in ITER, including issues of fuel retention and the reduction of divertor heat flux by radiation and strike-point spreading. Conventional H-mode, advanced inductive, hybrid, and steady state regimes have been developed that achieve the normalized performance goals of ITER with the ITER shape and aspect ratio. A large-bore startup scenario was demonstrated in an ITER-like shape, using feedback control of the internal inductance to maintain good vertical stability. High fusion performance consistent with $Q = 10$ in ITER has been demonstrated in hybrid mode discharges with low neutral beam torque, and hybrid mode operation is shown to be compatible with ELM suppression. High-beta wall-stabilized scenarios compatible with steady-state operation have been sustained for 2.5 seconds.

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OV/2-1 · The Status of the ITER Design

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Abstract: In parallel with a fast build up to almost 300 people within the International Organization (IO) in Cadarache, the project team, including many members from the member countries represented by their Domestic Agencies (DA), has concentrated its effort on an overall ITER design review. An updated technical baseline was presented to council at the end of 2007. Several additional improvements were included during spring 2008 and it is probable that the review will be accepted by Council. As a result the ITER design today provides a robust basis for a technical design that allows operation over a wide range of physical parameter, that can operate stably with high gain, and can exploit the full scientific potential of the machine. In the technical areas, design changes have been integrated to improve performance, provide more robust subsystems and minimize technical or operational risk. All required adaptations have been made with respect to safety and licensing to support the licensing process as a Nuclear Facility in France (INB). In parallel major components are already under construction within the DA's. A full overview of the status of ITER design and construction will be given including the detailed discussion of the 2007 ITER baseline.

OV/2-2 · Ignition on the National Ignition Facility: A Path Towards Inertial Fusion Energy

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Abstract: The National Ignition Facility (NIF), the world's largest and most powerful laser system for inertial confinement fusion (ICF) and experiments studying high energy density (HED) science, is nearing completion at Lawrence Livermore National Laboratory (LLNL). NIF, a 192-beam Nd-glass laser facility, will produce 1.8 MJ, 500 TW of ultraviolet light. The NIF Project is scheduled for completion in March 2009. Currently, 136 of the 192 beams have been commissioned, producing over 2.88 MJ of 1ω light and making NIF the world's first megajoule laser. The plan is to begin 96-beam symmetric indirect-drive experiments early next year. These first ICF experiments represent the beginning of NIF experiments for the National Ignition Campaign (NIC). This national effort for ignition experiments is coordinated through a detailed plan that includes the science, technology, and equipment such as diagnostics, cryogenic target manipulator, and user optics required for ignition experiments. Participants in this effort include General Atomics, LLNL, Los Alamos National Laboratory (LANL), Sandia National Laboratories (SNL), and the University of Rochester Laboratory for Energetics (LLE). The primary goal for NIC is to have all of the equipment operational and integrated into the facility soon after project completion to begin ignition experiments in 2010. When the NIF is complete, the long-sought goal of achieving self-sustaining nuclear fusion and energy gain in the laboratory will be much closer to realization. The focus of this talk will be on NIF technical capabilities, the National Ignition Campaign, and challenges for inertial fusion energy. In particular, the talk will address the scientific and technical accomplishments for NIC, the roles and responsibilities of the various partners comprising this national effort, as well as the NIC experimental plans. Successful demonstration of ignition and net energy gain on NIF will represent the first step towards demonstrating the feasibility of inertial fusion energy and will be a transforming event for inertial fusion, likely focusing the world's attention on the possibility of an ICF energy option. The first NIF experiments to demonstrate ignition and gain will use central hot

OV/2-3 · Overview of ASDEX Upgrade Results

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Abstract: The ASDEX Upgrade programme is directed towards physics input to critical elements of ITER design and preparation of ITER operation. The main recent hardware enhancement was the final step to a fully tungsten (W) coated wall, allowing to study W as a wall material in a divertor tokamak under ITER relevant conditions. Start-up in 2007 and 2008 was carried out without any further coating and the 2007 campaign was run without boronisation. H-modes with $H_{98} = 1$ and $\beta_N = 2$ at 0.7 nGW can be readily achieved in steady state with W-wall. The operational space is restricted by impurity accumulation in case of low central heating or low density/gas puff. The use of ICRH with W-coated ICRH limiters is restricted, with impurity ions accelerated in the rectified antenna sheath leading to substantial W-influx. An overarching aim of the W-programme is to study compatibility with ITER operational scenarios; hence discharges aiming at improved H-mode operation were carried out in unboronised conditions, showing clear improved H-mode behaviour. In transport, GAMs extend over finite radial regions with constant frequency, with discontinuous radial frequency change. Turbulent k-spectra from Doppler-reflectometry indicate variability of the spectrum with radius in H-mode, while L-mode spectral indices are largely constant. Ion heat transport is generally close to necolassical, with ratio of ion density to temperature length around 1. Filaments expelled by ELMs show velocities consistent with toroidal rotation of the pedestal from which they detach, with no radial acceleration. A new core localised non-Alfvénic mode ("Sierpes") that enhances fast ion losses was discovered and the loss mechanism for fast ions induced by TAEs and NTMs was clarified. Using a field line mapping technique, stochastisation may play an important role in fast crash phenomena (sawteeth, FIR-NTMs). Study of pellet triggering shows that type I ELMs may be metastable through most of the cycle. Using high pressure injection of different gases, efficient disruption mitigation is achieved optimising the injection valve parameters. Hardware enhancements under way are the ECRH enhancement and the installation of in-vessel coils and a conducting wall for ELM mitigation and RWM control. Also, plans for an LHCD system have been elaborated, exploring the possibility to install such a system in support of ITER.

OV/2-4 · Development of Net-Current Free Heliotron Plasmas in the Large Helical Device

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Abstract: Remarkable progress in the physical parameters of net-current free plasmas has been made in the Large Helical Device (LHD) since the last Fusion Energy Conference, 2006. LHD is based on the heliotron employing a pair of superconducting helical coils. The heating capability of neutral beam injection (NBI) has been significantly upgraded in these two years. A new perpendicular NBI with a low accelerating voltage of 40 keV (7 MW) enables efficient ion heating. Combined with the 3 existing tangential beams (180 keV), the heating capability of the NBI has exceeded 20 MW. The high β regime has been extended to the β of 5% by optimizing the magnetic configuration. A high β state beyond 4.5% is maintained for longer than 100 times the energy confinement time. The density and temperature regimes also have been extended. The central density has reached 1.1×10^{21} m^{-3} due to the formation of an Internal Diffusion Barrier (IDB). In contrast with an Internal Transport Barrier, the IDB is not accompanied by an improvement in thermal transport in the relatively flat temperature profile in an IDB region while temperature gradient is established in the mantle outside an IDB. In spite of a peaked density profile and a negative electric field in the periphery, no significant impurity contamination has been observed. An impurity screening effect in the ergodic layer intrinsic to LHD exists. These high density plasmas lie in the plateau regime where the present collisionality is only approximately a factor of 2 higher than that of a super high density reactor. The ion temperature has reached 6.8 keV at the density of 2×10^{19} m^{-3}. The discharge with a high ion temperature exhibits a change of temperature gradient in the intermediate region of its profile and the ion thermal diffusivity decreases to the level of the estimate from neoclassical theory. This new regime accompanies a large toroidal rotation driven by tangential NBI and by a spontaneous rotation due to the ion temperature gradient. Also, a large outward convection of impurities has been found coincidentally in spite of a negative electric field. All these high-performance plasma parameters have elucidated the potential of net-current free heliotron plasmas. Diversified studies in recent LHD experiments are reviewed in this paper.

OV/2-5 · Theory and Observations of Magnetic Islands

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Abstract: Magnetic perturbations with frequency much less than the Alfvén frequency interact strongly with the plasma near resonant surfaces where their wave-vector is parallel to the magnetic field, giving birth to magnetic islands and zonal flows. Examples of low-frequency MHD modes include the Neoclassical Tearing Mode (NTM) in tokamaks, conventional tearing modes in Reversed Field Pinches (RFP), and various Resonant Magnetic Perturbations (RMP), such as the Low-Density Locked Modes caused by error fields, the ergodic magnetic divertor in tokamaks, and the island divertor in Stellarators. All of these perturbations affect local confinement directly. Furthermore, they have an important influence on plasma rotation and thereby on global stability and confinement properties. The theory of low-frequency Alfvén perturbations has advanced to include neoclassical damping of the plasma rotation outside of the resonant layer as well as the effects of symmetry breaking, diamagetic drifts, polarization currents, and turbulence inside the layer. At the same time the increasing resolution of the diagnostics has provided exquisitely detailed information on the profiles inside magnetic islands and on the effects of the perturbations on global plasma rotation. Experimental observations have confirmed expectations, based on two-fluid theory, that the relative velocity between the NTMs and the surrounding plasma lies between the ion and electron diamagnetic drift velocities. In this range of velocities the polarization current has a healing effect, in agreement with observations. Recent numerical investigations have also shown that turbulence leads to a slowing down of the island, contributing to NTM stabilization. Resonant magnetic perturbations drive magnetic reconnection resulting in the formation of islands in otherwise tearing-stable plasmas. modelling of the effects of edge-resonant magnetic perturbations has explained the observations of density pump-out as being caused by convection cells that are significantly wider than the islands. Two-fluid effects contribute to broadening the cells through coupling with drift waves. Mode penetration can lead to island overlap and field-line stochasticity. Successful comparison between theory and experimental observations of electron transport in the RFX experiment has advanced the understanding of transport in stochastic fields.

OV/3-1 · Overview of Results from the National Spherical Torus Experiment (NSTX)

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Abstract: The mission of NSTX is the demonstration of the physics basis required to extrapolate to the next steps for the spherical torus (ST), such as a plasma facing component test facility (NHTX) or an ST based component test facility (ST-CTF), and to support ITER. Key issues for the ST are transport, and steady state operation. To better understand electron transport, a new high-k scattering diagnostic was used extensively to investigate electron gyro-scale fluctuations in a variety of plasma conditions including: as a function of toroidal field, with varying electron temperature gradient scale-length, during ELMs, and versus of magnetic shear. New results from electron Bernstein wave emission measurements from plasmas with lithium wall coating applied indicate transmission efficiencies near 70% in H-mode. The improved transmission is associated with reduced edge collisionality - a result of lithium coating. Improved coupling of High Harmonic Fast-Waves is due to reduced edge density relative to the critical density for surface wave coupling. In order to achieve high bootstrap fraction, future ST designs envision running at very high elongation. Plasmas have been maintained on NSTX at very low internal inductance $l_i \sim 0.4$ with strong shaping ($\kappa \sim 2.7$, $\delta \sim 0.8$) for several energy confinement times. Instabilities driven by super-Alfvénic ions are an important issue for all burning plasmas, including ITER. Fast ions from NBI on NSTX are super-Alfvénic. Linear TAE thresholds and appreciable fast-ion loss during multi-mode bursts are measured and these results are compared to theory. RWM/RFA feedback combined with $n = 3$ error field control was used on NSTX to maintain plasma rotation with β above the no-wall limit. The impact of $n > 1$ error fields on stability is a important result for ITER. Other highlights are: results of lithium coating experiments, measurements of the beta-scaling of confinement - an important issue for ITER, momentum confinement scaling studies, and demonstration of divertor heat load mitigation in strongly shaped plasmas. These results advance the ST towards next step fusion energy devices such as NHTX and ST-CTF.

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

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Abstract: Important progress has been made on MAST towards advancing the physics basis for ITER and DEMO, and future devices based on spherical tokamaks. This is being facilitated by improvements, such as the two long-pulse neutral beam injectors, 12 internal coils for ELM mitigation and TAE excitation ($n = 3$), a 28 GHz gyrotron for electron Bernstein wave (EBW) start-up, and new diagnostics and analysis techniques. The MAST confinement database was extended to higher heating power and plasma current indicating a stronger scaling of the energy confinement time with toroidal field and a weaker scaling with plasma current compared to the IPB98(y,2) scaling. A study of the fuel retention time for pellet fueling with ITER relevant pellet deposition from the high-field side scales to a particle throughput on ITER of about 70% of the design value. An increase of the ELM frequency by a factor of 2 in a new low collisionality ELMy H-mode was achieved with magnetic perturbations from external mid-plane coils in $n = 2$ configuration. First results using the new internal $n = 3$ coil sets above and below the mid-plane are presented. The particle and energy transport due to filaments has been studied, and a new approach in scrape-off-layer (SOL) modelling based on OSM-EIRENE has been developed to account for the intermittent nature of the transport. Core transport studies were focussed on momentum transport using a statistical approach as well as dedicated experiments, suggesting a Prandtl number of the order of 0.5 for MAST. The core studies were extended to the edge with particular emphasis on the role of the balance between SOL flows and core momentum transport on the generation of the edge radial electric field. In addition the impact of core rotation and fast particles on MHD and vice versa is studied. A rich spectrum of fast particle modes has been observed on MAST from 10 kHz fish-bone modes to 4 MHz compressional Alfvén eigenmodes. In addition a broad range of stable fast particle modes can be excited with the new TAE antennas with frequencies up to 500 kHz. TRANSP analysis indicates 30% off-axis ($r/a = 0.4$) neutral beam current drive with an anomalous fast particle diffusion of 0.5 m^2s^{-1}. EBW current drive start-up is demonstrated for the first time in a tokamak generating plasma currents up to 55 kA.

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OV/3-3 · Investigation of Steady-State Tokamak Issues by Long Pulse Experiments on Tore Supra

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Abstract: Twenty years after the start of its operation (April 1988), Tore Supra is still the only large tokamak fully equipped for steady-state operation, i.e., combining superconducting magnetic coils, actively cooled plasma facing components and non-inductive current drive capability. Its primary mission, the investigation of physics and technology issues of steady state tokamak operation, retains all its scientific interest and relevance to the fusion programme. After the attainment of the symbolic target of discharges with 1 GJ of injected energy, the following step has been the increase of the injected power ($\sim$ 10 MW), on shorter discharges (still lasting $\sim 20 - 30$ s, significantly longer than the current redistribution time). This phase allows the exploration of an operational domain of the highest interest for ITER and reactor applications. The bonus of this type of experiments is related to the simultaneous presence of electron and ion tails in plasmas at $T_i{\sim}T_e$. This work reports the main achievements of the Tore Supra programme in the last two years. Significant results have been obtained in three areas: ITER relevant technology developments and tests in a real machine environment, tokamak operational issues for high power and long pulses, and fusion plasma physics. The potential role of Tore Supra in the worldwide fusion programme before the start of ITER operation will also be discussed. Technology developments: (*i*) Tests of a new ITER-like prototype ICRH antenna (*ii*) Tests of a new articulated inspection arm on Tore Supra Operational issues: (*iii*) Deuterium inventory and carbon deposits (*iv*) Scenarios at high power level (*v*) Real-time control of sawteeth by ECCD (*vi*) Plasma start-up assisted by ECRH Fusion plasma physics: (*vii*) Dependence of transport and turbulence on non-dimensional quantities (β, ρ^*, ν^*) (*viii*) Transport studies using active perturbation methods (*ix*) Ion temperature scaling in the Scrape Off Layer (*x*) Non-linear double tearing dynamics (*xi*) MHD instabilities associated to fast particles

OV/3-4 · Recent Experiments in the EAST and HT-7 Superconducting Tokamaks

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Abstract: First divertor plasma configuration in EAST was obtained in the second campaign after the last IAEA meeting. The physical engineering capability on the superconducting magnetic system of EAST was assessed by simulating discharges. A number of operational issues, such as plasma initiation, ramp up and configuration control with constraints of superconducting coils were successfully investigated. New capabilities including the fully actively water cooled in-vessel components, current drive and heating power, diagnostics, real-time plasma control algorithm were developed to support the long pulse diverted plasma discharges. Since the last IAEA meeting, experiments in HT-7 focused on long pulse discharges under different scenarios and high power heating to support EAST experiments both physically and technically. The long pulse discharges up to 400 s renews the records in HT-7. Both AC operation and high performance regimes have been greatly extended. The wave-particle decorrelation on the cross-filed electron transport and zero-frequency long distance correlation were investigated in the HT-7 by a Langmuir probe array at plasma edge and in core by ECE image.

OV/4-1 · Plasma Physics Study and Laser Development for the Fast Ignition Realization Experiment (FIREX) Project

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Abstract: Thermonuclear ignition and subsequent burn are key physics for achieving laser fusion. Laboratory ignition with very large laser systems is now anticipated with the National Ignition Facility (NIF) in the US and Laser Mega Joule (LMJ) in France. Fast ignition has a potential to achieve ignition and burn with about one tenth of laser energy required for these programs. With the fast ignition, the fuel compression and heating are separated, with ignition initiated by a short very high power laser pulse incident on the already compressed fuel. The fast heating of a compressed core [1], together with the scalability to high-density compression [2], has provided the scientific basis for the start of the Fast Ignition Realization EXperiment (FIREX) project. The goal of the first phase (FIREX-I) is to demonstrate ignition temperature of $5 - 10$ keV, followed by the second phase to demonstrate ignition and burn. Coupled with the achievement of central ignition on NIF and LMJ, the research focus would then move to the demonstrations of high gain and of the inertial fusion energy technology. These programs would converge onto a laser fusion test reactor that can deliver net electric power.

[1] R. Kodama et al., Nature 418 (2002) 933; [2] H. Azechi et al., Laser Part. Beam 9 (1991) 193

OV/4-2 · Turbulence Theory and Gyrokinetic Codes

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Abstract: This overview will present an assessment of the present understanding of the theory of turbulence in magnetized fusion plasmas. It will present the state of the art in this field by presenting the main physical issues, and the most recent results of gyrokinetic simulations. After a broad introduction to the basics of turbulent transport, the main techniques that are used for solving gyrokinetic equations will be presented. The second part of this overview will be devoted to the main open physical issues in turbulent transport, with some emphasis on the results brought by gyrokinetic simulations regarding these issues. This part will cover the determination of dimensionless scaling laws of confinement, the characterization of the various transport channels, and the development of reduced transport models. The final part will be dedicated to the validation of gyrokinetic codes by comparing the simulations to measurements such as thermal fluxes in steady-state and transients, and fluctuation measurements. The last part of this overview will be dedicated to the question of improved confinement and transport barriers, with some focus on the role of shear flows and magnetic shear. The most recent results related to this subject will be presented.

OV/4-3 · Overview of Experimental Results on HL-2A

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Abstract: The HL-2A tokamak programme is to address key physics issues relevant to ITER. Since last FEC, recent experimental campaigns have been focused on studying and understanding the physics of turbulence, transport, MHD instabilities and energetic electron dynamics. Significant advances have been made in these fields. In particular, the three dimensional spectral structures of the theory and simulation predicted low frequency zonal flow and quasi-mode have been observed and analysed simultaneously for the first time. A spontaneous particle transport barrier has also been observed for the first time in Ohmic discharges without any external momentum input or core particle source, and this is evidenced by particle perturbation study using modulated SMBI technique and microwave reflectometry measurement. The non-local transport effect with new features induced by SMBI fueling has been investigated. In addition, the mode structure of the density fluctuation at GAM frequency is identified. The diffusion induced by the large amplitude intermittent bursts in the scrape-off layer is found to account for $45-50\%$ of the total outward particle flux. It is also found that the suppression of $m/n = 2/1$ tearing modes can be sustained by ECRH with low modulation frequency of about 10 Hz, resulting continuous confinement improvement following the mode suppression. Besides, the electron fishbone instability excited by energetic electrons has been observed and investigated with 10-channel CdTe hard X-ray detectors. The results show that the e-fishbone is correlated with the existence of energetic electrons of $30-70$ keV. All these advances have benefited from substantial improvements and developments of hardwares, including installation of modulated ECRH (2 MW/68 GHz), NBI (2 MW/45 keV), modulated SMBI system, plasma control and diagnostic systems.

OV/4-4 · Overview of the Alcator C-Mod Research Program

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Abstract: Recent research activities on Alcator C-Mod have addressed many issues important for the success of ITER and future reactors. A study of density peaking in C-Mod plasmas at low collisionality, enabled after the installation of a divertor cryopump, has broken the covariance between collisionality and density limit fraction that wasn't separable from observations on other devices; it now appears that the collisionality is the relevant control parameter, and density peaking factors of about 1.5 may be expected for ITER. Pedestal density profiles exhibit a remarkable stiffness, with resiliency to core fueling and magnitude of the plasma current, suggesting an edge critical gradient regulated by transport, similar what is seen in the plasma core. There is also evidence for a critical gradient in the near SOL profiles. From gas puff imaging, turbulent structures in the plasma edge are found to propagate poloidally with phase velocities of a few km/s, in the electron diamagnetic direction in the pedestal, in the ion diamagnetic direction in the far SOL, and in both directions in the near SOL. The characteristics of edge turbulence also change depending on the distance from the X-point. LHCD experiments at ITER relevant magnetic field and density have demonstrated off-axis current drive, with good agreement between hard x-ray profile and MSE measurements, and code modelling. ITBs have also been formed with LHCD. Measurements of the k spectrum of fluctuations in ITB plasmas generated with ICRF heating demonstrate the role of density gradient driven trapped electron modes through comparison with nonlinear gyrokinetic simulations. The radial structure of Reverse Shear Alfvén Eigenmodes, driven unstable by ICRF heating during current ramping, has been measured with Phase Contrast Imaging and compares favorably with modelling. ICRF sheaths have been identified as a primary mechanism for impurity generation during auxiliary heating. Studies of deuterium retention in metal (Mo) PFCs suggest that high pressures in the near surface distort the lattice and trap hydrogen isotopes. The ITER-relevant vertical stability control margins of C-Mod equilibria and the regions of operation space (q, internal inductance, elongation) accessed and robustly controlled, have been shown to be consistent with theory.

OV/4-5 · Overview of TJ-II Experiments

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Abstract: The last TJ-II experimental campaigns have focused on improving and understanding plasma confinement. Spatially resolved balance studies in ECH plasmas with Boronised wall show that (i) heat confinement increases linearly with density while particle confinement increases sharply a factor four above a certain density threshold associated with the positive electric field created by fast electron ECH pump-out; (ii) lowest order magnetic resonances, even in a low shear environment, can impose locally smaller effective diffusivities; and (iii) there is an overall increment of transport with ECH power in agreement with the known "power degradation", although the core region should be considered differently due to the presence of CERCs. The inherently strong plasma wall interaction of TJ-II has been successfully reduced after Lithium coating by vacuum evaporation. Li was chosen by its high efficiency for H-retention and low Z and, moreover, because there exists a reactor-oriented interest in this element (e.g., liquid divertors). This has led to important changes in plasma performance. Particularly, the effective density limit in NBI plasmas has been extended reaching central values of 8×10^{19} m^{-3} and $T_e \approx 250 - 300$ eV, with peaked density, rather flat T_e profiles and two-fold ion temperatures. Dramatic increases of particle (up to 40 ms) and energy (~ 10 ms) confinement times happen as expected for the collisional regime of Heliacs. No radiative instability at the edge has been observed. Studies on flux expansion divertor to improve the plasma-wall interaction in TJ-II are underway. The studies of electric fields, plasma rotation and turbulence show that the inversion of the perpendicular rotation velocity of the turbulence from ion to electron diamagnetic direction depends on collisionality and can be produced also by plasma biasing. Edge measurements with probes confirm the onset of associated sheared flows, which enhance confinement. It is found that long-range toroidal correlations of potential fluctuations are amplified by the onset of radial electric fields, which happen to play a dual role as a turbulence stabilizing mechanism and as an agent affecting the parallel momentum balance. An increase in non-linear mode coupling during forced confinement transitions is also observed during the biasing-induced transition.

OV/5-1 · Overview of the FTU Results

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Abstract: New ohmic high-density regimes (1.6 times the Greenwald density at $I_p = 0.5$ MA) with peaked profile and H(97-L) ~ 1.2 are accessed on FTU with Lithizated wall and compared with boronised and metallic wall operations. The space parameters for the existence of fishbone-like modes destabilized by supra-thermal electrons generated by LHCD is being studied. Precursor events, identified as $m/n = 1/1$ mode localized around $r/a = 0.3$, are already found with minimum q about 1 and LH power as low as 0.5 MW i.e., well below the threshold for the strong mode destabilization. Theory predicts the existence of new magnetic-island induced Alfvén Eigenmodes (MiAE). The dependence of MiAE continuum accumulation point on mode number and magnetic island size is used to analysis FTU experimental data with the aim of demonstrating a novel magnetic island diagnostic. Disruption avoidance is obtained with ECRH localized at any of the magnetic islands produced by the MHD modes. 0.4 MW of absorbed ECRH power are sufficient to avoid disruptions at 0.5 MA (deposition at $q = 2$ surface) in agreement with calculations based on Rutherford equation. Metallic dust of micrometer size, characterized by hypervelocity (10 km/s) have been detected by Langmuir probes, the number of recorded events is in agreement with post mortem analysis. New oblique ECE and a refractometer diagnostics have been also tested.

OV/5-2Ra · Overview of the RFX-mod Results

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Abstract: With the exploration of the MA plasma current regime in up to 0.5 s long discharges, RFX-mod has opened new and promising perspectives for the Reversed Field Pinch (RFP) magnetic configuration. RFX-mod has made a significant progress in understanding and improving confinement and in controlling stability since the last 2006 IAEA FEC. We present results, which are relevant for the understanding of the RFP physics and fusion perspectives, and contribute to the solution of physics and technology open issues for the optimization of ITER construction and operation. RFX-mod is a large ohmically heated RFP ($R = 2$ m, $a = 0.457$ m, volume 10 m^3), equipped with an advanced system of feedback controlled amplifiers and saddle coils, based on 192 independently driven coils covering the whole plasma boundary. Thanks to this system an axisymmetric magnetic boundary is produced, with differential rotation of the tearing modes. Improved confinement has been accessed both using the quasi-stationary Oscillating Poloidal Current Drive and through optimization of the spontaneous helical state (single helicity-SH state) appearing at high plasma current. In 1 MA discharges the power losses decrease by about 10 MW. The electron temperature shows peak values in the 1 keV range, linearly increasing with plasma current with no degradation of beta. As current is increased, the plasma self-organizes such that the growth of the helical structure is accompanied by a synergic decrease of magnetic turbulence throughout the plasma (SH states). SH states are more frequent and persistent at high current (at 1.4 MA they occupy 60% of the discharge and last up to 50 ms) with a global reduction of transport (increase of the global electron energy confinement up to 50%). The paper reports results on tools for multi-MHD mode control to cope with mode non-rigidity, avoidance of the harmful effects of sideband harmonics produced by the control coils, effective mode identification and tracking capability, robust controller stability and strong dynamical coupling between saddle-coils and corresponding sensors. Multiple RWMs have been addressed in detail and their full stabilization for many growth times has been achieved.

OV/5-2Rb · Overview of Results in the MST Reversed-Field Pinch Experiment

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Abstract: We report on results achieved in the MST reversed-field pinch in the following three key areas. In the area of plasma confinement and β we have achieved improved-confinement plasmas with simultaneously high $T_e \approx 2$ keV and high $T_i > 1$ keV at MST's highest current capability ~ 0.5 MA. Both temperatures increase with plasma current. In addition, we have achieved plasmas with simultaneously high n_e and high β tot up to 26 %, which exceed local and global stability limits. In the area of auxiliary heating and current drive we report on two complementary rf approaches, using lower hybrid and electron Bernstein wave injection at the power level up to 250 kW. In the area of transport and fluctuation studies we report on experimental and theoretical results of particle and momentum transport resulting from MHD tearing fluctuations. The results indicate that the Maxwell and Reynolds stresses are both important for momentum transport. We also report on experimental and theoretical studies of anomalous ion heating from magnetic reconnection.

OV/5-3 · Recent Studies in Fast Electron Energy Transport Relevant to Fast Ignition Inertial Fusion

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Abstract: The results of the first of a series of experiments on the VULCAN laser facility are reported that is intended to demonstrate that the fast electron beam produced at irradiances in excess of 10^{19} W/cm^2 can be artificially collimated even when the beam enters the target with a large divergence angle. This campaign has major implications for the future – if this artificial collimation can be demonstrated, then it will be a major landmark on the road to the realization of fast ignition inertial fusion and will influence most aspects of high intensity laser-plasma interactions physics, from ion acceleration, isochoric heating of matter to high temperatures, etc.. Indeed, artificial collimation will greatly enhance the probability of success for the high gain ignition campaign on HiPER. New two-dimensional hybrid Vlasov-Fokker-Planck / fluid modelling has confirmed that this is a real possibility. The artificial collimation occurs because of a pre-generated magnetic field that is produced by a laser pulse of 10^{18} W/cm^2 that precedes the main pulse. Preliminary analysis of the first experiment suggest that the fast electron beam from the generator pulse did not provide the expected magnetic field structure. A number of possible explanations for these observations will be discussed and future experiments outlined. A new theoretical model will also be presented, based on energy flux and momentum flux conservation, that provides a scaling law consistent with experimental data by Beg et al., generalised to the fully relativistic regime. The electron temperature is much less than ponderomotive scaling and (with the inclusion of reflected laser light) predicts up to 90% absorption at intensities above 6×10^{19} W/cm^2. Experimental evidence in support of the model, based upon electron spectra observed some distance from the target, will also be provided.

OV/5-4 · Overview of Results Obtained at the Globus-M Spherical Tokamak

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Abstract: High values of plasma parameters were reached at the Globus-M spherical tokamak ($A = R/a = 0.36/0.24 = 1.5$ m) at moderate heating power ($0.2-0.8$ MW) and high specific power of auxiliary heating ($1-1.5$ MW/m^2). Greenwald and Murakami density limits were achieved. Sykes-Troyon coefficient was up to $\beta_N = 6\%$ m·T/MA. The regimes with $\beta_T = 15\%$ and $\beta_P \sim 0.7$ were accessed. Important feature of most regimes is the low edge safety factor, which is unusual for spherical tokamaks, $2.7 < q_{95} < 4.5$ ($0.8 < q_{cyl} < 1.3$) and small plasma-outer wall clearance ($3-5$ cm). High ion heating efficiency was demonstrated with NB injection. Ion temperature reached 0.75 keV and exceeded electron temperature 1.5 times. Numerical simulation of fast ion trajectories is performed and it is shown that heating efficiency increases in the inwardly shifted plasma due to reduction of first orbit losses. Results of ion cyclotron heating at the fundamental harmonic of hydrogen minority in deuterium plasma ($7.5-10$ MHz, $100-300$ kW) are discussed. Reasons for achievement of high IC heating efficiency are outlined. Important are space position of IC resonance in the poloidal cross section, high vessel wall reflectivity and minority concentration. Reliable H-mode regime was achieved when ion ∇B drift was directed towards the lower X-point. L-H transition is recorded in OH, NB and IC heating regimes in both limiter and diverter plasmas. Transport ASTRA modelling demonstrated that ion heat diffusivity remains neoclassical during H-mode and the particle diffusion coefficient decreases down to $(0.2-0.5)$ m^2/s inside transport barrier. Injection of plasma jet with high kinetic energy into the plasma leads to nearly instantaneous increase of density in the center of the column. The plasma density increase is recorded on the magnetic axis of target 0.2 MA plasma with 5×10^{19} m^{-3} within 90 mks after injection start. At the same time electron temperature drops significantly. RGTi diverter tiles analysis was performed after irradiation by plasma during big number of shots (10000 shots in average). Mixed layers contained absorbed deuterium and its depth distribution was analyzed. Important result is that deuterium is absorbed only inside the mixed layers and its concentration is vanished in the bulk of material.

OV/P1-1 · Results of Joint Experiments and other IAEA Activities on Research Using Small Tokamaks

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Abstract: Joint Experiments (JE) in the framework of the IAEA Coordinated Research Project (CRP) on "Joint Research Using Small Tokamaks" have been carried out on tokamaks CASTOR at IPP Prague, Czech Republic, T-10 at RRC "Kurchatov Institute", Moscow, Russia and ISTTOK at IST, Lisbon, Portugal, in 2005 – 2007. Experimental programmes were aimed to diagnose and characterize the core and the edge plasma turbulence in a tokamak in order to investigate correlations between the occurrence of transport barriers, improved confinement, electric fields and electrostatic turbulence using advanced diagnostics with high spatial and temporal resolution. On CASTOR and ISSTOK, electric fields were generated by biasing an electrode inserted into the edge plasma and an improvement of the global particle confinement induced by the electrode positive biasing has been observed. Geodesic Acoustic Modes were studied using HIBP diagnostics on T-10 and ISSTOK and correlation reflectometry on T-10. It was shown that the GAM may have a complex structure, (not similar to conventional periodical oscillations with a single frequency), which is mainly manifested in the plasma potential and not much pronounced in the plasma density fluctuations. Both T-10 and ISTTOK results suggest the existence of GAMs in a narrow region just inside the LCFS. ISTTOK is equipped with the gallium jet injector and a technical feasibility of gallium jets interacting with plasmas has been investigated, in pulsed and AC operations. The first JEs have clearly demonstrated that small tokamaks are suitable for broad international cooperation to conduct dedicated joint research programmes. The contribution of small tokamaks to the mainstream fusion research can be enhanced through coordinated planning. The following Joint Experiment is scheduled for March 2009 on the TCABR tokamak at the University of São Paulo, Brazil. Other activities within the IAEA CRP on RUST will be also overviewed.

This work was partly funded by the IAEA technical contract under the CRP on the Joint Research Using Small Tokamaks.

SE

*** Safety, Environmental and Economic Aspects of Fusion***

SE/1-1 · Economic Consequences of Fusion Materials Development

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Abstract: The programme of qualification and improvement of materials for use in a fusion power station is crucial to the introduction of fusion power into the energy market and power plant studies are used to guide the targets for materials development. It is common to mis-interpret these targets as minimum values that must be achieved for fusion power to be a viable energy source, implying that failure to fully meet these targets will prevent fusion from playing a role in future. It remains important to improve materials properties but this must always be considered in context by an integrated power plant design in which all aspects are consistent. Rather than making fusion an impossibility, different levels of materials performance will lead to a different design in which their use is optimised, for instance compensating reduced lifetime neutron fluence by power density and machine size. Such trade-offs are an economic issue and relatively large variations in materials properties can be accommodated with small changes in overall cost of electricity. The full material impacts are much more complex than just the tolerance to neutron damage, and these must all be considered in a fully integrated way. Of particular importance are the tolerance to heat load, the combined effects of operating temperature and neutron load, which can impact on thermodynamic efficiency, the compatibility of different materials and their phase stability over the characteristic timescale of continuous operation of a power plant.

SE/P2-1 · Burning Plasma Simulation and Environmental Assessment of Tokamak, Spherical Tokamak and Helical Reactors

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Abstract: Burning plasma simulation in tokamak (TR), spherical tokamak (ST) and helical (HR) reactors were carried out focusing on Internal Transport Barrier (ITB) plasma operations using the TOTAL-T (Toroidal Transport Analysis Linkage - Tokamak) code coupled with GLF23 turbulent transport code and NCLASS neoclassical transport codes, and TOTAL-H (Helical) code with multi-helicity helical ripple transport analysis code. The effectiveness of these ITB transport coefficients is checked using experimental data of JT-60U and LHD. It clarified the requirement of deep penetration of high-field-side (HFS) pellet injection fueling to realize steady-state advanced burning operation in TR and ST. The neoclassical ripple transport plays an important role on the ITB operation in HR. Moreover, economical and environmental assessments were performed for these three type reactors by the PEC (Physics Engineering and Cost) system code in the case of four blanket designs (Li/V, Flibe/FS(Ferritic Steel), LiPb/SiC, FF(Fission-Fusion) Hybrid). In the present analysis, maximum field of superconducting coil is assumed 13 T, instead of maximum normal conductor strength of 8T in ST reactor. The tolerable neutron wall fluence is assumed 20 MW·Yr/m^2 in the case of LiPb/SiC blanket system, which determines the replacement cycle of blanket modules. As for cost analysis, the fusion island (FI) cost of ST-1 is lowest. However, its fusion thermal power is largest and the TR is superior in cost of electricity (COE). Among four blanket designs Flibe/FS is superior in cost, because ferritic steel (FS) is much cheaper than vanadium (V). The life-cycle CO_2 emission amount per output electric power and the energy payback ratio are also evaluated. The ST reactor is favorable in CO_2 emission reduction, because rather compact and simple normal conducting coil system is adopted here. The ST and TR need more frequent blanket exchanges than HR with lower neutron wall load. However, HR is still expensive and has lower energy payback ratio and higher CO_2 emission within the present evaluation model. These burning plasma and systematic environmental analysis for tokamak, spherical tokamak and helical reactors have been done comparatively for the first time by the help of the above transport and system codes.

TH

*** Magnetic Confinement Theory and modelling***

TH/1-1 · Physics of Non-Diffusive Turbulent Transport of Momentum and the Origins of Spontaneous Rotation in Tokamaks

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Abstract: Here, we report on recent progress in the theory of turbulent momentum transport (emphasizing primarily toroidal momentum) and show that the flux of toroidal velocity takes the generic form of the sum of a Fickian diffusive flux, a pinch and a residual stress (driven by pressure gradient). We extensively discuss the physics of the non-diffusive constituents and show that both the convective pinch and the residual stress are required to successfully confront the phenomenology of profile evolution and spontaneous rotation. We present a unified model which recovers many features of the ITPA database scalings. We specifically address: (i.) theory of the pressure gradient-driven residual stress and its effects; (ii.) the TEP (turbulent equipartition) and thermoelectric pinch of toroidal momentum and their implication for rotation profile evolution and (iii.) a unified model of spontaneous rotation scaling and rotation profile evolution.

TH/2-1Ra · Physics of Penetration of Resonant Magnetic Perturbations Used for Type I Edge Localized Modes Suppression in Tokamaks.

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Abstract: Resonant Magnetic Perturbations (RMP) generated by external coils have been shown to be effective in eliminating Type I ELMs in DIII-D or significantly mitigating them in JET. At present, ELM control by RMP is strongly recommended for ITER since it could significantly increase the lifetime of the ITER divertor by reducing surface erosion caused by Type I ELMs. The present days extrapolations to ITER are mainly based on the experimental criterion and "vacuum" field modelling suggesting that Type I ELMs are suppressed when edge plasma is ergodized. However, depending on plasma parameters and RMP spectrum and amplitude, the actual RMP field could be very different, especially in rotating plasmas where the generation of the current perturbations near rational surfaces could prevent reconnection leading to effective shielding of RMP. Also it is known from experiment that RMP penetration could produce significant slowing down of toroidal plasma rotation which potentially leads to MHD modes locking. The understanding of the rotating plasma response on RMPs is important in the optimization of the RMP coils spectrum in order to avoid a significant loss of plasma rotation in ITER The present paper describes physics of penetration of RMP into the toroidally rotating plasma. A new development of the non-linear cylindrical reduced MHD (RMHD) code was done to take into account toroidal rotation, resonant and global plasma braking due to the Neoclassical Toroidal Viscosity (NTV). In the present paper it is shown that RMP penetration time significantly increases for ITER-like parameters ($\sim$s) compared to present day tokamaks ($\sim$few ms in DIII-D). RMHD modelling for specific RMP coils in ITER and DIII-D demonstrated that central islands on the resonant surfaces are expected to be shielded due to the stronger rotation and lower resistivity compared to edge plasma which is expected to be ergodised in the pedestal region for both machines. It was demonstrated that non-resonant harmonics which do not produce islands in plasma are not screened by plasma rotation, hence they could play an important role in NTV braking mechanism. Latest RMP coil designs for ITER are analysed with respect to the minimisation of non-resonant NTV braking effect and the optimisation of the design.

TH/2-1Rb · MHD Simulation of Resonant Magnetic Perturbations and ELMs

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Abstract: Resonant magnetic perturbations (RMP) applied by non-axisymmetric coils placed outside the plasma have been found effective in suppressing Edge Localized Modes (ELMs) in the DIII-D experiment. The experimental temperature profile is found to be less affected by the RMP than the density profile. Simulations with the M3D code indicate that plasma toroidal rotation has an essential effect on the RMP. When the rotation is below a threshold value, the magnetic field including the RMP is stochastic in the outer part of the plasma. The temperature is strongly affected by parallel thermal conduction, while the density is modified much less. At higher rotation speed, the RMP is screened from the plasma, and the temperature is hardly affected by the RMP. The density is perturbed at the plasma edge and accumulates near the magnetic separatrix x - point. When a poloidal rotation is added, or two-fluid diamagnetic drifts, the effect is further enhanced. The scaling of the screening effect with rotation shear and Lundquist number S will be discussed. In the absence of an RMP, unstable ELMs show a difference between temperature and density behavior during the nonlinear evolution, similar to that seen with the RMP.

TH/3-1 · Behaviour of Turbulent Transport in the Vicinity of a Magnetic Island

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Abstract: The impact of a magnetic island on electrostatic turbulence in a tokamak is studied by means of gyrokinetic particle-in-cell simulations performed with the ORB5 code [1]. The equations of motion of the markers (representing the ion population) are modified by adding a small radial component to the magnetic field. Electrons are treated as adiabatic. Zonal flows are not included. Since the streaming along field lines is very fast, the reconnected region exhibits very high radial transport and consequently poor confinement. On the other hand, the drive for the turbulence is drastically reduced by the fact that density and temperature profiles are flattened within the island by the fast parallel dynamics. Moreover, the shape of the turbulent structures is modified by the geometry of the magnetic field of the island. The simulations show that turbulent eddies break across the island separatrix and that the convective cells are deformed by the reconnected field. Inside the island, potential fluctuations associated to low transport are observed. The turbulence spectrum also reflects the presence of the island. Toroidal mode numbers which are a multiple of the island mode number are favoured in both the linear and the nonlinear phase. Moreover, fields with low mode numbers are amplified by nonlinear coupling to the turbulent modes. The radial heat flux due to the turbulence velocity is compared to the radial heat flux due to the streaming along the perturbed field lines. The width of the layer around the island separatrix where both fluxes are comparable shows a weak scaling with the initial temperature gradient. The analytic prediction (based on the assumption of a constant diffusion coefficient) of a proportionality between the temperature jump across the island and the temperature gradient is not verified in the simulations. The space dependence of the transport coefficients and the more basic question about the nature of the transport across the island are addressed by means of a test-particle analysis.

[1] S. Jolliet, A. Bottino, P. Angelino et al., Comp. Phys. Comm. 177, 409 (2007)

TH/3-2 · Temperature Gradients are Supported by Cantori in Chaotic Fields.

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Abstract: With the tantalizing prospect that localized regions of chaotic magnetic field can be used to suppress ideal instabilities in fusion devices, as suggested by the resonant magnetic perturbation (RMP) experiments on DIIID, it becomes necessary to understand the impact of chaotic fields on confinement, particularly so considering that RMP fields are being considered as an ELM mitigation strategy for ITER. Using a model of heat transport for illustration, this paper will show that chaotic fields can support significant temperature gradients, despite the fact that flux surfaces may be destroyed by applied error fields. The remnants of the irrational flux surfaces, the cantori, present extremely effective partial-barriers to field-line transport, and thus present effective barriers to any transport process that is dominantly parallel to the field. We extend the concept of magnetic coordinates to chaotic fields, and show that the temperature, generally a function of three-dimensional space, takes the simple form T(s), where s labels the chaotic-coordinate surfaces.

TH/3-3 · Kinetic Modelling of Impurity Transport in Detached Plasma for Integrated Divertor Simulation with SONIC (SOLDOR/NEUT2D/IMPMC/EDDY)

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Abstract: To investigate the plasma and impurity transport in peripheral plasmas, 2D multi-fluid divertor codes have been developed, where the impurities are treated as fluid species. The fluid modelling for the impurity transport contains the improper descriptions; assumption of instantaneous thermalization of impurity ions, neglecting the kinetic effect on the thermal force and simplification for the complicated dissociation process of hydrocarbons. The MC approach is suitable for such effects to be taken into account. However it has the disadvantage for long computational time, large MC noise, and assumption of steady state. We solved the first and the second problems of MC modelling by developing a new diffusion model for scattering process and optimizing on the massive parallel computer. Recently, we solved the last problem by extending the IMPMC to a time-dependent simulation code, where increasing number of test particles with time is suppressed by a particle reduction scheme. Thereby we have accomplished a coupling of non-steady IMPMC code into a 2D divertor code (SOLDOR/NEUT2D) for the first time. An integrated divertor code SONIC enables us to investigate the details of impurity transport including erosion/redeposition processes on the divertor plates by further coupling of an MC code EDDY. The dynamic evolution of X-point MARFE observed in JT-60U is investigated. The simulation results indicate that the hydrocarbons sputtered from the dome contribute directly to the enhanced radiation near the X-point. The kinetic effect of the thermal force is found to improve the helium compression by a factor of ~ 2, compared with the conventional (fluid) evaluation. This effect possesses the significance for the helium exhaust of tokamak reactors.

TH/3-4 · Energetic Particle Physics Issues for 3-dimensional Toroidal Configurations

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Abstract: Energetic particle driven shear Alfvén wave (SAW) instabilities are frequently observed in both stellarator and tokamak experiments. Three-dimensional equilibrium variations are present to some extent in all toroidal devices and can significantly influence both the stability properties of energetic particle populations and their loss patterns on the first wall. Insuring the integrity of the first wall will be an important issue for future ignited devices. Three-dimensional equilibrium variations in stellarators and broken symmetry tokamaks provide new couplings that increase the complexity and density of the Alfvén mode spectrum. Methods have been developed for calculating Alfvén mode structures in such configurations and identifying the most likely modes for resonant energetic tail destabilization. Specific applications have been made to LHD where estimates of the associated phase space island widths based on stellarator Alfvén modes are consistent with experimental observations. For stellarators such as HSX, which have low shear rotational transform profiles, fast particles can also potentially destabilize acoustic modes, global Alfvén (GAE) modes and variants that couple both branches. Coupling to the sound wave continuum has been included in order to properly treat these lower frequency modes. In the case of tokamaks with symmetry breaking effects, a rippled, finite β equilibrium model using the VMEC code has been developed for ITER. This has indicated that finite β can amplify ripple levels internal to the plasma. Monte Carlo slowing down studies have indicated higher prompt alpha losses for the finite β equilibria. Although the ripple is not expected to significantly modify the linear Alfvén mode structure, it can be at a similar level and wavelength as nonlinearly saturated SAW modes and can influence saturation levels of these instabilities.

TH/3-5 · Critical Problems in Plasma Heating/CD in Large Fusion Devices and ITER

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Abstract: We outline identified problems resolution for ECRF, ICRF, NBI and LHH methods by: 1) elaborated ECRF 3D full wave code modelling to decrease broadness of EC power deposition and proper EC power launcher(s) positions choice; STELEC code and ECRF similarity laws benchmarking against ECH experiments and ray tracing modelling at second harmonic, related to the ITER hydrogen phase of operation, and well resolved modelling at fundamental harmonic for small tokamak(s) to demonstrate reality of STELEC predictions, including ITER active phase of operation. 2) Proposing new ICRF far off axis Fast Wave current drive operation scenario for non active (hydrogen) ITER operation phase using minority ions scenarios like H(He-4) which naturally overcomes the LHH method coupling problem. This new scheme is especially important to provide reality for worldwide broad modellings for large tokamaks and for ITER hybrid scenarios. 3) proposing High Frequency Fast Waves (HFFW) numerically modelled scheme for DIII-D and ITER, with commercially available 1 MW CW sources at 200 MHz , and waveguide type antennae, much more electrically strong, being as a back up for NBI method; 4) considering principally new approach for ICRH/CD method, especially in conditions of transient ELM activity, making use toroidally broad multi loop Travelling Wave Antenna (TWA) concept (prospective ITER design will be given; TWA consequently also uses previous magnetic loop antenna world theory/experiment experience), naturally incorporates antenna's loops inter coupling through a plasma (being principally unresolved problem for classical multi loop powerful antennae) with elegant control of antenna-plasma coupling through a small generator frequency change to properly control toroidal wave's spectrum during plasma edge density profile reconstruction; 5) developing new ITER-like ICRF scenarios at fundamental deuterium harmonic, partially recently explored on JET.

TH/4-1 · The Physics of Sawtooth Stabilisation in Tokamak Plasmas

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Abstract: For the first time, comprehensive numerical modelling can now quantify the relative roles of fast anisotropic ions (including the hitherto neglected stabilisation from energetic passing ions), the toroidal rotation and changes to the magnetic shear profile in determining sawtooth behaviour. The control of sawteeth is important for baseline scenario operation of burning plasmas, since plasmas with large amplitude sawteeth are more susceptible to neo-classical tearing modes, resulting in substantial confinement degradation. The stabilising effects of alpha particles are likely to exacerbate this, so recent experiments have identified various methods for amelioration. This paper first presents a series of experimental results from auxiliary heated tokamak plasmas in various devices, before showing how numerical modelling has accurately explained the observed behaviour. Shorter sawtooth periods than in Ohmically heated plasmas have been achieved in JET, MAST and TEXTOR. Each experiment exhibits an asymmetry of sawtooth period with respect to NBI direction. In JET, the asymmetry is explained by combining the destabilisation arising from counter-passing ions with the effect of flow shear on the stabilising trapped ions. The sawtooth behaviour in TEXTOR is explained through a subtle combination of both gyroscopic effects and kinetic effects. Furthermore, the application of off-axis NBI can destabilise sawteeth which had previously been strongly stabilised by simultaneous on-axis heating. This is explained qualitatively through the role of passing energetic ions with a positive pressure gradient at the $q = 1$ surface. modelling of the effects of toroidal rotation and anisotropic fast particle distributions in the presence of sheared flows has significantly advanced the understanding of the physical mechanisms that determine sawtooth stability. These improvements in our understanding have also been used to assess the possibility of employing off-axis N-NBI for sawtooth control in ITER. Partly funded by EPSRC and the EURATOM/UKAEA Contract of Association and partly carried out under EFDA.

TH/4-2 · Turbulent Transport and Flow Effects on NTM Evolution and Trigger Mechanisms

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Abstract: We report on detailed numerical studies of two problems related to the excitation and nonlinear evolution of neoclassical tearing modes in a tokamak. The issue of seed island creation is addressed through an investigation of the sawteeth instability in the presence of toroidal sheared flows. The simulations are carried out within the framework of a generalized reduced MHD model and employing a fully toroidal initial value code NEAR. In the absence of flow we observe the characteristic widespread stochasticity in the magnetic field after a sawteeth crash and also the nonlinear excitation of higher poloidal mode number islands. In the presence of sheared flow the stochasticity is found to reduce considerably and there is also a significant change in the reconnection rate. We also find flow-induced changes in the sawteeth period. A detailed parametric study for various tokamak equilibria (including some standard ITER scenarios) and flow profiles has been undertaken to delineate the conditions for generation or suppression of the seed islands. For the NTM simulations we have focused our investigations towards exploring the effects of anomalous transport on the threshold and growth rates of the (2,1) mode. The basic aim is to study the evolution of the NTM in the presence of a background of microturbulence generated, for example, by short scale length instabilities such as the Ion Temperature Gradient (ITG) or the Electron Temperature Gradient (ETG) modes. Model transport coefficients taken from past theoretical studies have been appropriately incorporated in the NEAR code in the evolution equations for the vorticity, parallel velocity and the poloidal flux function. Our numerical results are further compared with analytic model calculations based on the generalized Rutherford equation. Finally, we also carry out a set of studies to test the validity of various neoclassical closure schemes for the viscous stress tensors and assess their impact on the evolution of the neoclassical tearing mode.

TH/5-1 · Particle Simulation of Energetic Particle Driven Alfvén Modes

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Abstract: The mutual nonlinear interactions of shear Alfvén modes and alpha particles can enhance their transport in burning plasmas. Theoretical and numerical works have shown that rapid transport of energetic ions can take place because of fast growing Alfvén modes (e.g., Energetic Particle driven Modes, EPMs). Rapid transport of energetic ions related with Alfvén modes has been observed in experiments as well as in numerical simulations. Hybrid MHD-Gyrokinetic codes can investigate linear and nonlinear dynamics of energetic particle (EP) driven modes, retaining the mutual interaction between waves and EPs self-consistently. Self-consistent nonlinear wave-particle interactions (both in configuration and velocity space) are crucial for a correct description of the mode dynamics in case of strongly driven modes; thus, a non-perturbative approach is mandatory. The knowledge of the threshold characterizing the transition from weakly to strongly driven regimes is of primary importance for burning plasma operations (e.g., for ITER), in order to avoid EPM enhanced EP transport regimes. Recently, the hybrid MHD gyrokinetic code (HMGC) has been applied to the interpretation of phenomena observed in present experiments with Neutral Beam (NB) heating. In JT-60U, negative NBs generate an EP population which is responsible of Abrupt Large amplitude Events (ALEs). Numerical simulation results compare well with experimental observations, with respect to mode frequency, time duration of the bursts and radial transport induced by ALE. In reversed-shear beam-heated DIII-D discharges, a large discrepancy between the expected and measured EP radial density profile has been observed in presence of large Alfvénic activity. Initializing HMGC simulations with EP radial profiles expected from classical NB deposition gives rise to strong EPM activity, resulting in relaxed EP radial profiles at EPM saturation close to experimental measurements. The frequency spectra obtained from several simulations with different toroidal mode numbers compare well with experimental observations both in absolute frequency and in radial localization of the modes. Detailed synthetic diagnostics, documenting the power exchange between waves and EPs and the evolution of the EP distribution function, give deeper insight into the observed phenomenology.

TH/5-2 · Theory and Observations of Low Frequency Eigenmodes Due to Alfvén Acoustic Coupling in Toroidal Fusion Plasma

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Abstract: Investigating the interaction of energetic ions with MHD modes is important not only for planning future self-sustained burning plasmas but it is an area where ideal MHD and kinetic theories are put to the test with great accuracy. We report on the theory and observations of a new class of global MHD solutions resulting from coupling of the Alfvén and acoustic fundamental MHD oscillations due to geodesic curvature. These modes, predicted theoretically and numerically and called Beta-induced Alfvén-Acoustic Eigenmodes (BAAEs), have been recently observed in low β JET, DIII-D and high β NSTX plasmas. They are capable of inducing strong radial transport of beam ions in NSTX, especially together with multiple TAE instabilities. Acoustic branch coupling in high β NSTX plasma also affects the low frequency Alfvénic eigenmodes, such as Reversed Shear Alfvén Eigenmodes (RSAEs). Both RSAEs and BAAEs are expected in next step fusion experiments and can potentially deteriorate the fast ion confinement in burning plasmas. At the same time they can be used as a diagnostic tool for fast ion and the safety factor profiles a technique known as the MHD spectroscopy. A theory and numerical analysis of such modes are developed for the case when their interaction with the continuum is strong.

TH/6-1 · Theoretical Modelling of Transport Barriers in Helical Plasmas

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Abstract: The understanding of the turbulence-driven transport and the improved confinement (transport barriers) is the key issue in fusion research. Many kinds of the improved confinement mode in the core plasma of toroidal helical plasmas have been reported; e.g., the electron Internal Transport Barrier (e-ITB) with the strong positive radial electric field. The radial transition of the electric field was predicted to induce the internal transport barrier due to the shear of the radial electric field in the helical plasmas. The transition of the radial electric field was found on the Compact Helical System (CHS), and the improvement of confinement was found inside of the transition point for the radial electric field. In the previous study of the e-ITB, we have shown the reduction of the heat diffusivity because of the effect of zonal flows, which qualitatively predicts the e-ITB experimentally observed in the whole region of the strong positive electric field. On the other hand, the Internal Diffusion Barrier (IDB) in Large Helical Device (LHD) was recently discovered with the strong gradient of the density in a super dense core (SDC) plasma when a series of the pellet is injected. In this article, we present the unified transport model to explain the e-ITB and the IDB. At first, we examine the parameter regime in which the formation of the e-ITB can be predicted in helical plasmas with the effect of zonal flows. Next, the theoretical model for the IDB observed in LHD is shown. The mechanism, which is based on the transport reduction due to the shear of the radial electric field, is newly examined in the formation of the IDB. In the case of the particle fueling, the density rapidly increases. Therefore, the ion temperature temporally decreases and the positive gradient of the ion temperature is found to appear. From the ambipolar condition to determine the radial electric field profile, the positive electric field is found. As the result, the strong gradient of the electric field and the reduction of the anomalous particle diffusivity can be shown. When we study the physics of the transport barrier, the investigation of the density limit including the radiation loss for the case of the IDB in LHD is necessary. We discuss the critical density for the thermal stability in the case of the IDB in helical toroidal plasmas.

TH/7-1 · Molecular Dynamics Simulation of Chemical Sputtering of Hydrogen Atom on Layer Structured Graphite

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Abstract: In the experiment of plasma confinement, a portion of hydrogen plasma flows into divertor walls, which are shielded by the tiles of polycrystalline graphite. Chemical sputtering of hydrogen atom on graphite was simulated using molecular dynamics. Especially, the layer structure of the graphite was kept by interlayer intermolecular interaction. In the incidence of hydrogen atoms perpendicular to graphenes, the graphite intercalation of the hydrogen atoms appeared. As the hydrogen atoms increased, the graphenes were peeled off one by one and yielded molecules had chain structures. When the hydrogen atoms were injected into the armchiar and zigzag edges of graphite parallel to the graphenes, many C_2H_x and H_2 molecules were generated, respectively.

TH/8-1 · Validation of Gyrokinetic Transport Simulations Using DIII-D Core Turbulence Measurements

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Abstract: It is now widely recognized that the development of a predictive modelling capability for ITER and beyond requires the validation of transport codes against experimental measurements at multiple levels. Towards this end, the first direct comparisons of drift-wave turbulence amplitudes, power spectra, and correlation lengths predicted by the gyrokinetic code GYRO [1] against experimental observations are presented. Both local and non-local fixed-gradient simulations are used in this study, while the experimental measurements were obtained in a series of repeatable, steady low-power L-mode discharges. Spatially localized measurements of density fluctuations were obtained via beam emission spectroscopy (BES) [2]. Similarly localized measurements of electron temperature fluctuations were obtained via a newly implemented correlation electron cyclotron emission (CECE) diagnostic [3] on DIII-D. The development and use of synthetic diagnostics to conduct accurate comparisons between experiment and simulation is described in detail. Using these newly implemented synthetic diagnostics with local, fixed-gradient simulations, good agreement between GYRO and experiment is found for fluctuation amplitudes, power spectra, and correlation lengths at $r/a = 0.5$, in addition to close agreement with heat fluxes calculated via the ONETWO code. At $r/a = 0.75$, fluxes and fluctuation levels are under predicted by a factor of 2 and 4, respectively, but very good agreement in correlation functions and power spectra "shapes" is still found. To address the issue of profile stiffness, comparisons using results from the new TGYRO code [4] are presented. The TGYRO code is used to run GYRO simulations in a fixed-flux mode, which predicts the equilibrium profiles needed to match the fluxes obtained via power balance, rather than the traditional fixed gradient mode which predicts fluxes for a given set of profiles.

[1] J. Candy and R. E. Waltz, J. Comput. Phys. **186**, 545 (2003). [2] G.R. McKee, et al., Rev. Sci. Instrum. **77**, 10F104 (2006). [3] A.E. White, et al, "Measurements of Core Electron Temperature and Density Fluctuations in DIII-D and Comparison to Nonlinear Gyrokinetic Simulations," submitted to Phys. Plasmas (2007). [4] R. E. Waltz, et al., Nucl. Fusion **45**, 741 (2005).

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TH/8-2 · Conservative Global Gyrokinetic Toroidal Full-f 5D Vlasov Simulation

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Abstract: A new global gyrokinetic toroidal full-f five dimensional (5D) Vlasov simulation (GT5D) is developed using a novel non-dissipative conservative finite difference scheme, which guarantees numerical stability by satisfying relevant conservation properties of the modern gyrokinetic theory. By using GT5D, robust and accurate long time simulations of tokamak micro-turbulence are enabled for the first time based on a full-f approach with self-consistent evolutions of density and temperature profiles sustained by sources. GT5D is verified through comparisons of zonal flow damping tests, and linear and nonlinear analysis of ion temperature gradient driven (ITG) modes against a global gyrokinetic toroidal δf particle-in-cell simulation. In the comparison, global solutions of the ITG turbulence are first identified quantitatively using particle and mesh approaches, and it is found that in sourceless plasmas, the temperature profiles are relaxed to the nonlinear critical gradients given by the Dimits shift, and the heat transport is quenched in the steady state.

TH/8-3 · Gyrokinetic Simulations of Impurity, He Ash and α Particle Transport and Consequences on ITER Transport Modelling

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Abstract: The behaviour of minority species, like light and heavy impurities, He ash, as well as energetic alpha particles, in response to the background plasma turbulence has not been assessed yet with the same degree of reliability as in the case of the main plasma components. To this purpose, linear and nonlinear simulations with three gyrokinetic codes, GKW, GS2 and GYRO, are performed in concert with analytical derivations, in order to elucidate the basic transport mechanisms of impurities and energetic alpha particles. In addition, the ASTRA transport code and the GLF23 model are applied to the modelling of the ITER standard scenario, in which the transport of minority species is included by means of appropriate formulae which fit the gyrokinetic results. A method to separate diffusive and convective terms in the flux of a trace particle is introduced and elegantly formulated analytically through the introduction of a Green function in the solution of the gyrokinetic equation. Mechanisms leading to the experimentally interesting conditions of outward convection of impurities are introduced. Their effect is validated against experimental observations of impurity transport in ASDEX Upgrade by means of physics comprehensive gyrokinetic simulations. Interestingly, light and heavy impurities can have opposite behaviours due to the different size of the thermodiffusive contribution at small and large electric charge. Gyrokinetic simulations as well as specific transport modelling, including neoclassical transport, are performed in order to assess the behaviours of He ash and heavy impurities in the conditions foreseen in the plasma core of the ITER standard scenario. These are predicted to have a centrally peaked density profile by the GLF23 transport model, in agreement with observations in several tokamaks in low collisionality H-mode plasmas. Finally, gyrokinetic calculations of the transport of energetic alpha particles with a slowing down equilibrium distribution function in the trace limit are presented. For electron temperatures below 20 keV, the alpha particle diffusivity is at least 20 times smaller than the diffusivity of He ash. The transport of energetic alpha particles is integrated in the transport modelling of the ITER standard scenario by means of appropriate fitting formulae of the gyrokinetic results.

TH/P3-1 · Off-Axis Neutral Beam Current Drive for Advanced Scenario Development in DIII-D

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Abstract: Modification of two existing DIII-D neutral beam lines is proposed to allow off-axis current drive with neutral beam injection (NBI) vertically steered to drive current as far off-axis as half the plasma radius. New calculations indicate very good current drive with good localization off-axis as long as the toroidal magnetic field, B, and the plasma current, I, are in the same direction (for a beam steered downward). For a given set of density and temperature profiles, the off-axis NBCD efficiency is comparable to that of ECCD near the axis, and increases off-axis due to the increased trapped electron fraction. These simulations indicate that even in the presence of fast ion diffusion, NB current drive remains peaked off-axis, although the magnitude of CD is reduced. Self-consistent Advanced Tokamak (AT) scenario modelling using both the scaled, experimental transport model and the theory-based [gyro-Landau fluid (GLF23)] transport model shows that the proposed 10-MW off-axis neutral beam current drive (NBCD) and high power electron cyclotron current drive (ECCD) will allow demonstration of fully noninductive, high β scenario with flat safety factor and $q(\rho)$ above 2. The modification of the DIII-D NB system will provide flexible scientific tools for understanding transport, energetic particles, heating and CD physics, and validating the off-axis NBCD in support of scenario development for ITER and future tokamak reactors.

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TH/P3-2 · Fokker-Planck Modelling of NBI Deuterons in ITER

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Abstract: Investigation of the NBI impact on the processes in tokamak plasmas requires detailed information about the distribution function of beam ions since it appears strongly anisotropic in velocity and inhomogeneous in spatial coordinates. Here we present the modelling results of distribution functions of energetic deuterons generated in ITER as a result of 1 MeV neutral beam injection. Considered are beam deuterons in energy range 100 keV $< E <$ 1 MeV for basic ITER scenarios. The main attention is paid to steady-state distributions formed by steady NBI (at least over a 1 MeV deuteron slowing time), though for the prospect of gamma diagnostics of fast deuterons in ITER the temporal evolution of their distribution function after NBI switch-off is investigated as well. Calculations of steady-state beam deuteron distributions are performed using the 3D in constants-of-motion (COM) Fokker-Planck approach applied for interpretive modelling of beam ions in Trace Tritium Experiments (TTE) on JET, while the examination of the relaxation of the deuteron distribution is based on a time-dependent 3D COM Fokker-Planck code previously applied for analyzing the evolution of alpha particle induced gamma-emission in JET TTE. The thus modelled 3D COM distribution functions of beam deuterons are the founding element for simulations of poloidal profiles of beam deuteron density, of NBI power deposition to bulk plasma electrons and ions, as well as of the NBI generated current. In addition, the time evolution and the R, Z profiles of beam deuteron induced gamma-emission is modeled. Also the energy dependent neutral deuterium fluxes are calculated as induced by beam deuterons via various neutralization processes. Our modelling of NBI deuterons in ITER demonstrates the sensitivity of their distribution functions to different scenarios and proves the potential of gamma-diagnostics based on nuclear reactions of fast ions with Be and C impurities. Further validated is the capability of neutral particle diagnostics via analysis of energetic deuterium fluxes from ITER plasmas.

TH/P3-3 · Impact of ICRH on the Measurement of Fast Ions by Collective Thomson Scattering in ITER

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Abstract: Collective Thomson scattering (CTS) is a many-sided diagnostic which has been proposed for determining the phase space distribution of confined fast ions in ITER plasmas. These will have substantial non-thermal fast ion populations in the MeV range due to fusion reactions and due to auxiliary heating such as neutral beam injection (NBI) and ion cyclotron resonance heating (ICRH). The measurement of phase space distributions of confined fast ions in ITER is vitally important for understanding alpha particle physics in the burning plasma regime, for example energetic particle modes or interaction of collective plasma modes with fast ions. Predicted consequences of such effects can be tested against experimental data obtained by CTS. The proposed CTS system for ITER operates with 60 GHz gyrotrons with a power level of 1 MW each and is designed so as to meet the measurement requirements for ITER. The dynamics of the fast ion feature can thus be studied on a time scale of a few milliseconds with the proposed CTS diagnostic. In the present study, we compare the CTS signal due to fast ions generated by ICRH to the CTS signal generated by fusion alphas for typical ITER burning plasma scenarios. This comparison of predicted CTS signals reveals that fusion born alphas dominate the total CTS signal in the frequency range of interest. The assumed heating scenarios for ICRH were off-axis tritium heating at the second harmonic resonance and ^{3}He minority heating at the fundamental resonance with various ^{3}He concentrations (0.1% to 3%). The frequency was set to 50 MHz. The heating power was varied from 20 to 40 MW. The distribution functions of the fast tritons and ^{3}He in parallel and perpendicular velocity space were obtained by simulations using the PION code. It was found that the CTS signal due to scattering from alpha particles is more than an order of magnitude larger than due to scattering from fast ions generated by ICRH in the relevant frequency band, even in the narrow region which contains most of the fast resonant ions and even for velocities in the perpendicular direction in which the resonant ions are preferentially heated by ICRH. For the parallel direction as well as for other stations in configuration space, the contribution due to fast ions generated by ICRH is even smaller.

TH/P3-4 · Criteria for Runaway Electron Generation in Tokamak Disruptions

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Abstract: In tokamak disruptions a beam of relativistic runaway electrons is sometimes generated, which can cause damage on plasma facing components due to highly localized energy deposition. This problem becomes more serious in larger tokamaks with higher plasma currents, such as ITER, and makes it important to understand what processes may limit or eliminate runaway electron generation. The experimentally observed parametric dependence of the number of runaway electrons generated is still not understood. This paper reviews two mechanisms that can limit runaway electron generation. One mechanism is the interaction of the runaway electron beam with whistler waves. These waves in the post-disruption plasma can be destabilized through the anomalous Doppler resonance by the anisotropy of the runaway electron distribution function. Once destabilized, whistler waves affect the runaway electron distribution by a rapid pitch-angle scattering of the resonant electrons. Therefore, the threshold of this instability imposes a constraint on the generation of a runaway electron beam. Another constraint arises from the solution of two coupled differential equations for the runaway electron density and plasma current. This zero-dimensional model describes the time-dependence of the electric field induced by the decaying plasma current and takes into account the combined effect of primary and secondary runaway generation due to the induced electric field. These constraints delineate regions of parameter space where only few runaway electrons can be generated.

TH/P3-5 · Shear Alfvén Wave Continuous Spectrum in the Presence of a Magnetic Island

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Abstract: The continuous spectrum of Shear Alfvén waves is calculated for finite-beta tokamak equilibrium in the presence of a finite-size magnetic island. The beta-induced Alfvén Eigenmodes continuum accumulation point (BAE-CAP) is found to be positioned at the separatrix flux surface, while the frequency is not changed. The most remarkable feature is found to be the presence of new CAPs at the O-point of the island, which give rise to gaps in the continuous spectrum. This could make the existence of new magnetic-island induced Alfvén Eigenmodes (MiAE) possible, excited via wave-particle resonances if the island is sufficiently wide with respect to the mode radial localization. Experimental data of modes observed in Frascati Tokamak Upgrade (FTU) are presented, which agree with theoretical scalings. Due to the frequency dependence on the magnetic island size, the feasibility of utilizing MiAE as novel magnetic island diagnostic is also discussed.

TH/P3-6 · Minority Ions Acceleration by ICRH: A Tool for Investigating Burning Plasma Physics

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Abstract: Minority ions, accelerated by radiofrequency waves in the range of MeV energies, predominantly transfer their energy to plasma electrons via collisional slowing down. The use of ICRH in the minority scheme (H or ^{3}He) in D plasmas 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 ρ_{fast} and β_{fast} 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 address a number of burning plasma physics issues, e.g., fast ion transport due to collective mode excitations and cross-scale couplings of micro-turbulence with meso scale fluctuations due to energetic particles themselves. The aim of the present paper is to determine the characteristic fast-ion parameters that are necessary for addressing the mentioned above burning plasma physics issues and to present a stability analysis of collective modes excited by the ICRH induced energetic ion tail. 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 Fusion Advanced Studies Tokamak (FAST) conceptual study. Moreover, a bimaxwellian distribution for energetic particles, which takes into account the anisotropy in the velocity space ($T_\perp > T_\parallel$) due to ICRH, has been used as initial velocity space distribution function for a parametric study, based on density and temperatures profiles given by the TORIC code, investigating the destabilization and saturation of fast ion driven Alfvénic modes below and above the EPM (Energetic Particle Modes) stability threshold. Numerical simulation results, based on the HMGC hybrid code, will be presented and discussed

TH/P3-7 · Kinetic Theory of Geodesic Acoustic Modes: Radial Structures and Non-linear Excitations

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Abstract: Geodesic Acoustic Modes (GAM) are shown to constitute a continuous spectrum due to radial inhomogeneities. The existence of a singular layer causes GAM to mode convert to short-wavelength kinetic GAM (KGAM) via finite ion Larmor radii; analogous to kinetic Alfvén waves (KAW). KGAM are shown to propagate radially outward; consistent with experimental observations and numerical simulation results. The degeneracy of GAM/KGAM with β induced Alfvén Eigenmodes (BAE) is demonstrated and discussed. Furthermore, it is shown that KGAM can be nonlinearly excited by drift-wave (DW) turbulence via 3-wave parametric interactions, and the resultant driven-dissipative nonlinear system exhibits typical prey-predator self-regulatory dynamics. KGAM are preferentially excited with respect to GAM because of the radial wave-number dependence of the parametric excitation process. Plasma non-uniformity effects on nonlinear KGAM excitations are discussed.

TH/P3-8 · Advanced Modelling of Cyclotron Wave Heating and Current Drive in Toroidal Plasmas Based on Integro-Differential Full Wave Analysis

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Abstract: Self-consistent and accurate modelling of wave-plasma interactions is one of the key issues in producing and sustaining burning plasmas. We have upgraded our full wave component of the integrated modelling code TASK, taking account of the finite gyroradius effects indispensable for correctly describing the absorption of ICRF waves by energetic ions and the behavior of electron Bernstein waves. The integro-differential full wave analysis was coupled with the Fokker-Planck analysis and the GNET code to describe the modification of the momentum distribution function and to calculate the power deposition profile including the finite orbit size effects. These advanced modelling provides more accurate evaluation of the efficiency of wave heating and current drive in tokamaks and helical configurations. The newly-updated TASK/WM component is first applied to the analysis of the electron Bernstein waves in a small-size spherical tokamak plasmas and the driven current is evaluated from the Fokker-Planck analysis. Next the power absorption by alpha particles during the ICRF heating in tokamak plasmas is evaluated with the finite gyroradius effect taken into account. Finally the power deposition profile during the ICRF heating in the three-dimensional LHD configuration is calculated by taking account of the finite gyroorbit effect.

TH/P3-9 · Simulation Study of Interaction between Energetic Ions and Alfvén Eigenmodes in LHD

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Abstract: Interaction between energetic ions and Alfvén eigenmodes (AE) was studied with computer simulation on LHD shot #47645, where the creation of holes and clumps in the energetic ion energy spectrum associated with AE bursts was observed with the neutral particle analyzer (NPA). The difference in slowing-down time between the holes and clumps suggests that the energetic ions were transported over 10% of the plasma minor radius. The NPA measurement gives the energy and pitch-angle of the energetic ions. Then, we can investigate in detail the interaction between the energetic ions and the Alfvén eigenmodes inside the plasma. We examined the LHD plasma with the AE3D code and found two toroidal Alfvén eigenmodes (TAE) with toroidal mode number $n = 1$. The numerical analysis demonstrated the energetic ions with the orbit frequencies same as the AE frequencies. The phase space structures of the energetic ions were investigated with the Poincaré plots in an axisymmetric plasma comparable to the LHD plasma. An oscillating Alfvén eigenmode is employed for each plot. It was found that Alfvén eigenmodes with amplitude $\delta B_r/B \sim 10^{-3}$ transported energetic ions over 10% of the minor radius. Comparison between the theoretical prediction of the fluctuation amplitude and experimental data in future is crucial to the understanding of the interaction between energetic particles and AEs not only in LHD plasmas but also in general toroidal plasmas. Furthermore, a simulation code where AE spatial profiles are assumed constant has been newly developed to study time evolution of energetic particles and AEs in three-dimensional equilibria such as LHD. In this code, the time evolution of AE amplitude and phase is followed in a way consistent with the energetic-particle evolution. The three-dimensional MHD equilibrium calculated with the HINT code and the Alfvén eigenmodes calculated with the AE3D code are used in this simulation. TAE instability in the LHD plasma was simulated with this code. The linear growth and saturation of the instability was observed. The amplitude oscillation took place after the saturation, suggesting the particle trapping by the TAE. Dependence of the TAE burst interval in the LHD experiment on TAE damping rate will be investigated with simulation where neutral-beam injection and collisions are taken into account.

TH/P3-10 · Electron Cyclotron Power Losses in Fusion Reactor-Grade Tokamaks: Scaling Laws for Spatial Profile and Total Power Loss

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Abstract: Potential importance [1] of electron cyclotron (EC) wave emission in the local electron power balance in the steady-state regimes of ITER operation suggested analyzing in more detail the accuracy of calculating the 1D distribution, over magnetic flux surfaces, of the net radiated power density, P_r. Recent comparison [2] of numeric codes SNECTR [3], CYTRAN [4], CYNEQ [5] and EXACTEC [6] for different electron temperature profiles and average temperatures of relevance for fusion reactor-grade magnetoplasmas, has shown good agreement of results within two different tasks: (A) specular reflection in a circular cylinder (SNECTR, EXACTEC) and (B) diffuse reflection in any geometry or any type of reflection in noncircular toroids (SNECTR, CYTRAN, CYNEQ). These tasks were shown to provide, respectively, the lower and upper bounds for P_r, including that for the modulus of P_r, inverted in sign in the plasma column periphery. Here we extend the analysis [1] to show the following approximate scaling laws on the example of calculations with CYNEQ: (i) new formula for the volume-integrated EC power loss, P_{tot}, which modifies the Trubnikov's formula, suggested for homogeneous electron density and temperature profiles, to the case of N_e and T_e profiles typical for ITER and DEMO, and depends both on the volume-averaged values, N_eAv and T_eAv, and peak values; (ii) universal form of the normalized profile, P_r/P_{tot}, which appears to depend only on the shape of temperature and density profiles, $T_e(r)/T_eAv$ and $N_e(r)/N_eAv$; (iii) similarity of the profile P_r as a function of an effective wall reflection coefficient, which allows for the type of reflection (diffusive or specular) and for the geometry of the vacuum chamber, and is defined to describe the degree of distributing the EC wave ray trajectories over plasma volume and to draw a quantitative parallel between the above-mentioned tasks A and B.

[1] Albajar F., et al., Nucl. Fusion 45, 642 (2005). [2] Albajar F., Bornatici M., Engelmann F., Kukushkin A.B., EC-15 workshop, USA, March 2008. [3] Tamor S., Nucl. Technol. Fusion 3, 293 (1983). [4] Tamor S., Rep. SAI-023-81-189-LJ/LAPS-72, La Jolla, CA: Science Applications, 1981. [5] Kukushkin A.B., IAEA-1992, v.2, pp. 35-45; Cherepanov K.V., Kukushkin A.B., IAEA-2004, TH/P6-56. [6] Albajar F., Bornatici M., Engelmann F., Nucl. Fusion 42, 670 (2002).

TH/P3-11 · Sub-GAM Modes in Stellarators and Tokamaks

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Abstract: The work deals with fast-ion-driven instabilities characterized by frequencies below / about the frequency of the Geodesic Acoustic Mode (GAM). These instabilities were observed in experiments on tokamaks, stellarators, and spherical tori. In this work, a conclusion was drawn that they can hardly be described in terms of the ideal MHD modes. Therefore, new eigenmode equations valid for both stellarators and tokamaks were derived. These equations contain two variables (the electrostatic potential and plasma compressibility, coupled due to the presence of the field line curvature) and take into account effects of finite Larmor radius. The derived equations are based on a collisionless two-fluid magnetohydrodynamic with anisotropic plasma pressure and non-vanishing thermal fluxes. The modes described by them are not strongly damped even at very low frequencies. Because the frequency of these modes is below / about the GAM frequency, it is natural to refer to them as sub-GAM modes. A code calculating sub-GAM modes by using the derived equations was developed. This code, as well as previously developed ideal MHD codes (COBRAS, BOA), was used to model Wendelstein 7-AS discharges #39029 and #40173 where several modes with various frequencies (both less and more than the geodesic acoustic frequency) were destabilized simultaneously. The considered modes are very sensitive to the magnitude of the rotational transform. Therefore, experimental observation of them can be used for diagnostics providing the knowledge of iota in the points where the modes are localized with high accuracy. Possible sub-GAM modes in tokamaks are discussed.

TH/P3-12 · Effect of the Toroidal Asymmetry on the Structure of TAE Modes in Stellarators

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Abstract: It was found recently with the use of the ballooning formalism that even weak toroidal asymmetry can significantly change the structure of Torodicity-induced Alfvén Eigenmodes (TAE) with high toroidal mode number n, producing wave functions strongly anharmonic in the toroidal coordinate. In this work, the effect of the toroidal asymmetry on TAE modes with finite n in stellarators is studied with the code BOA-fe. A sequence of TAE-modes with almost the same frequency and localization but with different n is considered. The asymmetry consecutively couples the modes in the sequence. As the asymmetry is gradually increased to realistic values, the modes become strongly mixed (that is, mixed wave functions arise in which the contributions of separate TAE-modes are comparable). The mixed wave functions are strongly anharmonic, having a tendency to localization along field lines. The strong mixing is achieved most easily for modes with large n; then it spreads to lower n as the asymmetry increases. An analytical estimate for the level of asymmetry required to mix the modes is derived, which results from a relationship between the difference of the mode frequencies and the matrix element of their interaction due to asymmetry. The estimate is in agreement with the numerical results. The critical value of n for the mode mixing is evaluated analytically, depending on the profile of the rotational transform and parameters of the magnetic configuration. Possible influence of non-ideal factors on the obtained solutions is discussed. The circumstances under which strongly anharmonic TAE-modes can be observed experimentally in stellarators are outlined.

TH/P3-13 · Influence of Anisotropy on Radiation of Any Linear Antenna System in Magnetoplasma

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Abstract: Radiation of the antenna system located in magnetoplasma is considered. The system consists of two thin linear impedance antennas any way oriented concerning a direction of an external magnetic field. Thus antennas are not neither parallel, nor perpendicular among themselves. Plasma is in a strong external magnetic field and is anisotropic. Dielectric properties of plasma are described diagonal tensor dielectric permeability which components can be positive, and negative, and generally are complex functions of frequency. This problem was solved by the method of integral equations of macroscopic electrodynamics. The problem solution consisted of two stages. At the first stage the equations for current in thin antennas with skin impedance were obtained and solved by average method. At the second stage, knowing function of distribution of a current in the antennas, fields of radiation are defined. Conditions of effective transfer of electromagnetic energy from antenna system to plasma are defined from the received dispersive equations connecting geometrical parameters of antenna system (the sizes, orientation concerning an external magnetic field, superficial impedance) and parameters of surrounding plasma at a resonance. The received results can be useful at the decision of problems of high-frequency heating plasma, the analysis of electrodynamic compatibility of the devices working in plasma, and also to diagnostics of plasma.

TH/P3-14 · Gyrokinetic Simulation of Energetic Particle Turbulence and Transport

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Abstract: The fully self-consistent simulation of the energetic particle turbulence and transport must incorporate three new physics elements: kinetic effects of thermal particles at the thermal ion gyroradius (micro scale), nonlinear interactions of many meso scale (energetic particle gyroradius) shear Alfvén waves (SAW) induced by the kinetic effects at the micro scale, and meso-micro couplings of the microturbulence and SAW turbulence. The large dynamical ranges of spatial-temporal processes further require global simulation codes that are efficient in utilizing massively parallel computers at the petascale level and beyond. Therefore, the studies of energetic particle physics in ITER burning plasmas call for a paradigm shift to the gyrokinetic turbulence approach. In this paper, the progress of the gyrokinetic simulation using GTC and GYRO codes of the energetic particle turbulence and transport in tokamaks is reported. Specifically, the nonlinear gyrokinetic simulation finds that the energetic particle transport induced by the microturbulence decreases rapidly when the energy of the particles is an order of magnitude higher than the electron temperature. The linear global gyrokinetic simulation of the excitation of the toroidal Alfvén eigenmode (TAE) shows that the most unstable mode number increases with the tokamak size. A linear flux-tube gyrokinetic simulation demonstrates the energetic particle excitation of discrete Alfvén eigenmodes in high-beta plasmas.

TH/P3-15 · Energetic Particle-induced Geodesic Acoustic Mode

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Abstract: A new energetic particle-induced Geodesic Acoustic Mode (EGAM) is shown to exist. The mode frequency, mode structure, and mode destabilization are determined non-perturbatively by energetic particle kinetic effects. In particular we find the EGAM frequency is substantially lower than the standard GAM frequency. The radial mode width is determined by the energetic particle drift orbit width and can be fairly large for high energetic particle pressure and large safety factor. These results are consistent with the recent experimental observation of the beam-driven $n = 0$ mode in DIII-D. The new mode is important since it can degrade energetic particle confinement as shown in the DIII-D experiments. The new mode may also affect the thermal plasma confinement via its interaction with plasma micro-turbulence.

TH/P3-16 · Electron Cyclotron Current Drive in Spherical Tokamaks with Application to ITER

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Abstract: The high-beta spherical tokamaks (ST), such as NSTX and MAST, are attractive fusion devices for studying the physics of current drive by electron cyclotron (EC) waves. While ST plasmas are overdense to conventional EC waves, electron Bernstein waves (EBW) can be used to generate plasma currents. Besides providing better confinement, EBW driven current can also help suppress neoclassical tearing modes. This paper examines the characteristic features of EBW current drive. It is shown that the propagation and damping of EBWs and their interaction with electrons in STs provides useful insight into the propagation and damping of O waves and their interaction with electrons in ITER. The physics of current drive has also many similar features. A new relativistic wave-particle diffusion operator in toroidal plasmas that includes spatial and momentum transport due to RF waves is derived. It is suitable for numerical implementation and could explain the observed broadening of the current profile due to ECRF waves. The diffusion operator is relevant for studies on heating and current drive by EC waves in present day fusion devices and in ITER.

TH/P3-17 · ITER Relevant Simulations of Lower Hybrid and Ion Cyclotron Waves with Self-Consistent Non-Maxwellian Species

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Abstract: The next step toward fusion as a practical energy source is the design and construction of ITER. ITER relies in part on ion-cyclotron radio frequency (ICRF) power to heat the deuterium-tritium plasma to fusion temperatures and well as to provide some current during flat-top operations. Lower hybrid (LH) is under consideration as an upgrade to the baseline heating and current drive system in order to provide control over the current profile and additional current during start up. We have applied a suite of mature radio frequency (rf) full wave codes using self-consistent particle distributions from a Fokker-Planck code to ITER ICRF scenarios and ITER relevant LH experiments on Alcator C-Mod. We will present three dimensional full-wave simulations showing that the ICRF waves propagate radially inward in ITER with strong central focusing and little toroidal spreading. Fokker-Planck coupled rf simulations show that because of the high plasma density, energetic ion tail formation in ITER is typically weak, with the exception of the minority deuterium heating scheme where strong tails can develop on the minority ion distribution. Absorption by the fast alpha particles can approach five to ten percent of the injected power. Massively parallel full wave simulations in the lower hybrid range of frequencies using 2000 poloidal modes and 1000 radial elements have shown that large amounts of diffraction occur at caustic surfaces and in resonance cones resulting in localized power deposition at those surfaces. We demonstrate that this linear mechanism is sufficient to bridge the spectral gap (the difference between the high injected phase velocities and the slower phase velocity at which damping on electrons occurs) and explains the efficient damping of lower hybrid waves. This is seen to affect the amount of broadening in the phase velocity and the current drive location.

TH/P3-18 · Kinetic Simulation of Heating and Collisional Transport in a 3D Tokamak

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Abstract: The microwave plasma heating has a strong influence on collisional transport, experimentally observed both in stellarators and tokamaks. The estimate of the interplay between heating and collisional transport implies solving a 5D kinetic equation (2D in momentum space and 3D in real space), which is a difficult task. We attempt to solve this problem using a recently developed code (ISDEP: Integrator of Stochastic Differential Equations for Plasmas) in a tokamak with ripple as a test device. Previously, ISDEP has been successfully applied to the TJII stellarator [1]. Our Monte Carlo method is based on the equivalence between the linear Fokker-Planck and Langevin equations. This allows us to describe the system by taking average values over many independent test particle trajectories. The dynamics of these ions are determined by the guiding center approximation, ion-ion collision [2] and the interaction with microwaves (Ion Cyclotron Heating, ICH, since we are dealing with ion transport). We also have developed a selfconsistent method to update the background temperature in order to introduce nonlinear terms. Thus, we have modified ISDEP to include the geometry of a tokamak with ripple and the Langevin equations for ICH [3]. The European Computer Grid (EGEE) has been used to perform the calculations. The main results of this work are the calculation of transport quantities and the velocity probability distribution function. We have compared these results in the cases with and without heating, and investigated the differences between them. There are three main conclusions: i) the increment of the kinetic energy, with a consequent increment of the temperature, and an increase of the outward fluxes, which implies a reduction of the particle confinement. ii) The deviations of the distribution function from the Maxwellian, both in the bulk and in the wings in the presence of ICH. iii) The included ripple is not enough to generate toroidal asymmetries. This computer code can be adapted to other geometries and allows to consider other features that can be taken into account in the ion dynamics (NTM, ee collisions and Alfvén instabilities).

[1] F. Castejón et al, Nucl. Fusion 42, 271 (2002); [2] Z. A. Pietrzyk et al. Physical Review Letters 86, 1530 (2001); [3] S. Murakami, Nucl. Fusion 45, 221 (2003)

TH/P4-1 · Operation Window with Mutually Consistent Core SOL Divertor Conditions in ELMy H-Mode: Prospects for Long Pulse Operation in ITER and DEMO

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Abstract: The operational space of ITER in ELMy H-mode is explored with a simulation model that ensures consistent conditions throughout the plasma from the center through pedestal and SOL to the divertor plates. We concentrate on the long-pulse possibilities of ELMy H-mode operation in ITER and on the adequacy of such operation to perform the technical testing mission. In an analogous fashion, the operating space for DEMO is also examined, with a view to evaluating the range of output power achievable in reactor operation.

TH/P4-2 · Turbulence and Flow Interplay in the Tokamak Edge Plasma

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Abstract: The interplay between large scale flows and turbulence is a matter of growing attention. This particular physics is observed to play a significant role in the transport properties that govern the edge operational scenarios in present devices as well as their projection to ITER. Regarding core physics, the plasma rotation driven by turbulence can impact the MHD stability as well as trigger the external H-mode barrier. The generation of plasma rotation by turbulence is recognized as most important since the external torque is expected to be too small in next step devices such as ITER. Regarding the Scrape-Off Layer and divertor physics, large scale parallel flows have been reported for both limiter and divertor configurations. These flows are particularly large when the so-called ion ∇B drift is oriented towards the X point in the divertor configuration or towards the limiter as on Tore Supra. These SOL flows appear to be governed by the particle sources, in particular by the ballooned particle outflux from the core plasma. In turn, they are expected to govern material migration from the low field side to the high field side. The propagation of such a flow pattern from the SOL into the edge plasma via turbulent momentum transport is regarded as one possible player in the onset of the H-mode transport barrier in the framework of turbulent eddies shearing by flows. Transport simulations with no poloidal asymmetry of the diffusive transport, parallel flows reach a Mach number of $M \sim 0.2$ from the low field side to the high field side. When the toroidal field is reversed, this neoclassical flow is reduced, and only weakly reverses. In the turbulent regime analysed in the edge plasma, the ballooning nature of the transport enhances the parallel flow, $M \sim 0.4$ near the top. The simulation of the edge plasma coupled to the SOL indicates that turbulence triggers ballistic transport events deep into the SOL plasma that spread the previously described parallel Mach number pattern from the core into the SOL. These results are in agreement with the Tore Supra experiments and the data base of edge flows in X-point devices

TH/P4-3 · Recent Results from Edge Modelling on ASDEX Upgrade

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Abstract: In this paper we cover recent results obtained from modelling the edge plasma of ASDEX Upgrade (AUG): (1) detailed simulations of a few discharges where upstream and target data from the experiment were compared with the code results; (2) detailed simulations of a series of AUG discharges targeted at understanding the processes of divertor detachment; (3) the simulation of an AUG discharge where a transition from L-mode to H-mode was triggered solely by a drop in the toroidal magnetic field; (4) simulations of the penetration of Massive Gas Injection (MGI) gas puffs used for disruption mitigation; (5) simulations to examine the effect of ELM "size" on the whole plasma. The simulations were performed with the 2d fluid plasma, 3d Monte-Carlo neutrals code B2-Eirene, part of the SOLPS suite of plasma simulation codes. Key results include: (1) a difference between well diagnosed AUG discharges and the modelling possibly pointing to kinetic physics not included in the fluid plasma description; (2) qualitative but not yet fully quantitative agreement between the experiment and modelling results of detachment; (3) modelling of the L- and H-mode phases of an AUG discharge showed a change in the particle transport coefficients but not the energy transport coefficients.

TH/P4-4 · Divertor-Transport Study for Helical Devices

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Abstract: Finding a divertor solution for controlled plasma exhaust is one of the key issues for helical devices on the way towards a reactor. Unlike the standard poloidal-field divertor in tokamaks, divertor concepts presently investigated in stellarators are based on specific edge magnetic field structures intrinsically available in each device. Typical examples are the island divertor for the advanced low-shear stellarators W7-AS and W7-X, and the helical divertor for the high-shear, largest heliotron-type device LHD. The former utilizes the divertor potential of inherent edge magnetic islands while the latter is based on a stochastic field resulting from island overlapping. In view of the large differences in field and divertor geometry among helical devices, it is interesting to see whether there exist certain common physics issues in terms of divertor transport or divertor functionality. Recently a collaboration work between IPP and NIFS was started for this purpose and the paper presents a comparative divertor transport study for the island divertors (ID) in W7-AS and the helical divertor (HD) in LHD. In particular, it is concentrated on understanding the elementary, global transport processes associated with specific field topologies, as already extensively studied for the W7-AS ID, aiming at forming a comparison basis for two typical helical devices of completely different divertor concept and geometry.

TH/P4-5 · Mitigation of ELMs and Disruptions by Pellet Injection

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Abstract: Heat loads caused by disruptions and ELMs may severely damage fusion devices, but this can be mitigated e.g., by pellet injection. ELMs are triggered by deuterium pellets reaching the pedestal top region. Disruptions can be mitigated by strongly radiating impurity pellets. Deuterium, impurity and impurity doped pellets will be studied in the light of their use for ELM and disruption mitigation. Pellets are always able to trigger an ELM if the scenario is prone to such instabilities, but the underlying mechanism is not yet understood. Based on ELM pace making studies at ASDEX Upgrade and JET, the minimum pellet size required to cause an ELM, has to be determined in ITER for the foreseen pellet velocities. Disruption induced problems may be reduced by killer pellet injection. Killer pellets cools the plasma by radiation, but they might generate runaways. Impurity doped deuterium pellets cool the plasma effectively and increase its density avoiding runaways. The number of runaways will be determined for JET like and ITER plasmas for different pellets. To calculate the ablation rate and to describe the cloud dynamics we used the Hybrid code which considers the formation of the neutral cloud according to the NGS ablation model and the dynamics of the ionized cloud part is treated by an one-dimensional Lagrangian cell code. In case of deuterium pellets the heat absorbed from the plasma is mainly consumed by cloud expansion, while in case of carbon and carbon doped deuterium pellets it is radiated. These short time issues are especially important when we perform simulations of ELM triggering scenarios. On a longer time scale, the interaction between cold pellet particles and the background plasma, the radiation, Ohmic heating and heat diffusion are taken into account to calculate the change in the temperature. The resistive diffusion of the electric field is followed, and the resulting number of Dreicer, burst and avalanche runaway electrons are calculated during the current quench. By comparing experimental results from ASDEX Upgrade and JET with the simulation, predictions for the minimum pellet size and velocity pairs able to trigger an ELM in ITER has been estimated. Furthermore a tool has been created to test the suitability of different pellets for disruption mitigation.

TH/P4-6 · Edge Plasma Issues of the Tokamak FAST (Fusion Advanced Studies Torus) in Reactor Relevant Conditions

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Abstract: Among the R&D missions for possible new European plasma fusion devices, the FAST project will address the issue of "First wall materials & compatibility with ITER /DEMO relevant plasmas". FAST can operate with ITER relevant values of P/R (up to 22 MW/m, against the ITER 24 MW/m, inclusive of the alpha particles power), thanks to its compactness; thus it can investigate the physics of large heat loads on divertor plates. The FAST divertor will be made of bulk W tiles, for basic operations, but also fully toroidal divertor targets made of liquid lithium (L-Li) are foreseen. Viability tests of such a solution for DEMO divertor will be carried out as final step of an extended program started on FTU tokamak by using a liquid lithium limITER. To have reliable predictions of the thermal loads on the divertor plates and of the core plasma purity a number of numerical self-consistent simulations have been made for the H-mode and steady-state scenario by using the code COREDIV. This code, already validated in the past on experimental data (namely JET, FTU, Textor), is able to describe self-consistently the core and edge plasma in a tokamak device by imposing the continuity of energy and particle fluxes and of particle densities and temperatures at the separatrix. In the present work the results of such calculations will be illustrated, including heat loads on the divertor. The overall picture shows that at the low plasma densities typical of steady state regimes W is effective in dissipating input power by radiative losses, while Li needs additional impurities (Ar, Ne). In the intermediate and, mainly, in the high density H-mode scenarios impurity seeding is needed with either Li or W as target material, but a small (0.08% atomic concentration) amount of Ar, not affecting the core purity, is sufficient to maintain the divertor peak loads below 18 MW/m^2 that represents the safety limit for the W monoblock technology, presently accepted for the ITER divertor tiles. The impact of the ELMs on the divertor in the case of a good H-mode with low pedestal dimensionless collisionality will be discussed too.

TH/P4-7 · Generation of Dust Seeds by Sputtering of Carbon-based Plasma Facing Materials under Low-energy H/D/T Ion Bombardment

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Abstract: Low energy ion impact (with injection energy less than 50 eV or so) on the plasma facing wall materials of a magnetic confinement fusion device such as a tokamak can generate a significant amount of sputtered species that may grow into dust particles through agglomeration in the gas phase. In this work, we have studied sputtering properties of graphite and amorphous carbon substrates due to hydrogen (H), deuterium (D), and tritium (T) ion bombardment at low incident energies using classical molecular dynamics (MD) simulations. The classical interatomic potentials that we have used in this work are Brenner-type multi-body potential functions with weak Van der Waals interactions. MD simulations have been performed in such a way that we allow significant accumulation of incident species up to 1.25×10^{17} cm^{-2}. Our simulation results indicate that a high level of H/D/T dose accumulation on the top surface is prerequisite for the formation of relatively large-sized sputtered hydrocarbon species. Significant isotopic dependence of sputtering yields has been also observed after the dose of incident D or T reaches about 10^{16} cm^{-2}. It has been clearly shown that the sputtering yield can be lower at higher incident energies in the low energy range for D and T injections. These simulation results are consistent with some of earlier published experimental observations of carbon-based material sputtering.

TH/P4-8 · Modelling of Hydrocarbon Redeposition in the Gaps of Castellated Structures

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Abstract: Castellated armor tiles are proposed for the first wall and the divertor area in ITER. However, there is a critical issue of fuel (tritium) retention and impurity transport in the gaps between the castellated tiles. The previous calculation reproduces the tritium profiles observed on the gap sides of the TFTR bumper limiter [1]. In this study, we have performed a simulation calculation of transport and deposition of hydrocarbons on the castellated structure in order to understand the mechanisms of co-deposition and to mitigate carbon deposition in the gaps by changing castellated geometry. Reflection/sticking coefficients of the tile surface in realistic conditions were investigated by using molecular dynamics (MD) of collisions with all hydrocarbons that result from chemical sputtering. A new castellation geometry of unit cells with a tilted surface is proposed and the optimization of the shape of the cell is very likely to work to minimize the redeposition rate in the gaps. The tile geometry is important to reduce the in-vessel tritium inventory for the safety operation of fusion reactors.

[1] K. Inai, K. Ohya, J. Nucl. Mater. 363-365, 915 (2007)

TH/P4-9 · Two-dimensional Full Particle Simulation of the Flow Patterns in the Scrape-off-layer Plasma for Upper- and Lower- Null Point Divertor Configurations in Tokamak

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Abstract: The plasma flow in the scrape-off layer (SOL) plays an important role for the particle control in magnetic fusion reactors. The flow is expected to expel Helium ashes and to retain impurities in the divertor region, if it is directed towards the divertor plate. It has been experimentally observed, however, that the flow direction is sometimes opposite; from the plate side to the SOL middle side in the outer (low-field-side) SOL region of tokamaks. This backward flow is seen when the single null point is located in the ion ∇B drift direction, while it vanishes for the reversed null-point location. Physics mechanisms of this backward flow have not fully been known, though many simulation studies have been carried out with the fluid model. Kinetic simulations are considered to bring a breakthrough on this subject. A full particle code, PARASOL, is applied to the two-dimensional axisymmetric system of a tokamak plasma with the upper-null-point (UN) or lower-null-point (LN) divertor configuration, where the ion ∇B drift is downward. Orbits of ions are fully traced, while guiding-center orbits are followed for electrons. All the drift effects and the finite-banana-size effects are correctly simulated. In addition to the PIC method for the self-consistent electrostatic field calculation, a Monte-Carlo binary collision model for Coulomb collisions and a random-walk model for anomalous transport are adopted. Hot particles are supplied in the central core region, and the plasma is diffused across the magnetic field to the SOL region. PARASOL simulations for the medium aspect ratio reveal the following variation of the flow pattern. For the UN case, the flow velocity $V_\parallel$ parallel to the magnetic field is directed to the diverter plate both in the inner and outer SOL regions and the stagnation point ($V_\parallel = 0$) is located symmetrically at the bottom. On the other hand for the LN case, $V_\parallel$ in the outer SOL region has a backward flow pattern. The stagnation point moves below the mid-plane of the outer SOL. These simulation results are very similar to the experimental results. Simulations are carried out by changing the plasma parameters and by artificially cutting the electric field. It is found that the banana motion of trapped ions is very important for the formation of the flow pattern in addition to the $E\times B$ drift motion.

TH/P4-10 · What is the RMP Driven Transport and How Does it Affect ELMs?

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Abstract: Ion and electron transport in RMP field is highly kinetic phenomena. A proper inclusion of the realistic toroidal geometry and the self-consistent radial electric field dynamics in a kinetic code is a necessary component of a reliable prediction. We use an edge particle code XGC0. A realistic tokamak magnetic field equilibrium is read in from a g-eqdsk file, together with a three dimensional perturbed vacuum magnetic field data (I-coil, DIII-D, provided by I. Joseph). It is found that the vacuum RMP field is not of Rechester-Rosenbluth (R-R) type. The electron thermal transport rate in a self-consistent XGC0 simulation is an order of magnitude less than R-R. The increased ion density flux is found to be mostly convective, and driven by the ambipolar radial electric field in non-axisymmetric B-field. The pedestal density drops significantly in response to the enhanced particle convective loss in RMP. However, a strong heat flux from the core can maintain the pedestal temperature. The simulation results from XGC0 with vacuum RMP qualitatively agree with the experimental observations. A more complete agreement is obtained with a plasma screening factor, providing the first quantitative understanding of the RMP driven transport at detailed level. At the same time, this study predicts a form of the plasma response function to RMP penetration. M3D resistive nonlinear MHD code is used to study the RMP screening factor in coupling with XGC0 code for self-consistent plasma rotation profile. Collaboration with the ideal MHD code IPEC will also be utilized to get an ideal MHD screening form factor into XGC0. The mechanism of ELM mitigation by an RMP modified plasma pressure profile has been most successfully studied by an MHD peeling-ballooning stability code. XGC0 code has a unique capability of coupling the kinetic plasma information to an MHD code. For this purpose, XGC0 is coupled to the Elite linear stability code (in collaboration with P. Snyder of General Atomics). The RMP reduced bootstrap current is found to play an important role in stabilizing or de-stabilizing the RMP affected edge, and its optimization will be a key factor in ITER physics study. Validation against the known DIII-D experimental data will be presented.

TH/P4-11 · Vortex Nucleation in Strongly Sheared Poloidal Rotation and Effects on Velocity Saturation and Generation of ELM Modes

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Abstract: The fact that a two-dimensional plasma in strong magnetic field has an intrinsic tendency to evolve to self-organized profiles of vorticity suggests to look for the preferred states of vorticity radial distribution in tokamak. The method which we have developed for this problem uses a field-theoretical formulation of the continuum limit of the discrete set of point-like vortices interacting in plane by a short range potential. The Lagrangian shows that there are privileged states consisting in particular radial profiles of vorticity, for which the action functional is at the lowest extremum. Vortex nucleation is the process that appears in layers of strongly sheared rotation when the velocity is greater than a threshold (it is common to fluids, superfluids, Bose-Einstein condensates, protoplanetary disks, etc.). We show that the hydrodynamic nature of the process of vortex nucleation may explain the saturation of rotation via transport of angular momentum. A strong perturbation induces in a sheared layer (typical for H mode) a Kelvin-Helmholtz event consisting of a double spiral of vorticity. The local radial collapse stabilizes the vortex, generating a filament of vorticity. We find the threshold in terms of stable ring-type solutions with topological content. The enhanced parallel current triggers a filamentation of the current density whose maximum coincides with the maximum of vorticity. This leads to the break up of the poloidal velocity sheared layer into a set of periodic filaments that represent significant perturbation of the profiles of the vorticity and the current density. This process is equivalent to the ELM relaxation.

TH/P4-12 · Numerical Modelling of Li Limiter Experiments in T11-M Tokamak

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Abstract: The modelling of the T11-M tokamak discharges for Ohmic heating plasma experiments with Lithium and Carbon limiters is carried out by means of DINA and SOL-DINA codes. The investigation results of the radial and poloidal lithium distributions in the SOL of the T11-M plasma are presented. The values of calculated radiated power are compared with the experimental data. Radial dependencies of anomalous radial diffusion coefficients of Lithium ions and heat conductivity are chosen to fit the calculated heat and particle fluxes to the experimental radial dependence data.

TH/P4-13 · Computation of Radial Electric Field in the Turbulent Edge Plasma of the T-10 Tokamak

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Abstract: Numerical computation of turbulent dynamics of edge plasma was carried out. The computation is based on the solution of nonlinear MHD equations in the frame of reduced two-fluid Braginskii's hydrodynamics. It is shown that under transition from OH regime to ECR regime the amplitudes of the turbulent fluctuations decrease due to the increase of longitudinal dissipation related to the increase of the electron temperature. However, phase relations of the potential fluctuations change. It results to the increase of the Reynolds turbulent force. The growth of the force leads to the generation of higher poloidal velocity. If this velocity is negative, then the value of radial electric field decreases. It follows from the equation for radial balance of ions. Data of computations qualitatively confirm the experimental results obtained in T-10 device.

TH/P4-14 · Characteristics of Intermittency and ELM Dynamics in the Edge Region of Magnetic Confinement Devices

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Abstract: In the approach presented here we combine theoretic aspects and experimental results in order to obtain threefold information on intermittency and edge localized mode (ELM) dynamics using: a) stochastic catastrophy theory (CT), b) multifractal analysis and c) level-crossing properties of plasma density time series. We study intermittency properties of two different devices, the MAST spherical tokamak (L-, H- and dithering H-mode) and the Tore Supra tokamak with limiter configuration (L-mode). In the first approach we obtain a reliable estimate of the number of equilibrium states and transitions (bifurcations) between these states. This is of great importance for determining the L-H transition in the dithering mode of tokamak data as well as for identifying zonal flow formation. In addition we show that one may infer from experimental data that the ELMs are catastrophic bifurcation events. Multifractal analysis suggests that each magnetic confinement device has distinct spectral characteristics and that confinement modes (L-, H- and dithering H- modes) exhibit distinct multifractal spectral features. Also, cascade processes in different tokamak devices are distinct in spite of some universal common features. In the third approach, intermittency is studied by considering only the clustering of zero-crossings of turbulent signals and we show that it is related to particles clustering and possibly to accumulation of vorticity.

TH/P4-15 · Flux-expansion Divertor Studies in TJ-II

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Abstract: The quest for the stellarator reactor needs a robust divertor concept. The island-based divertor has worked efficiently in W7-AS and is suitable for devices like W7-X with a robust magnetic configuration that hardly changes with plasma pressure and ensures a small bootstrap current. These conditions guarantee that the island positions and widths do not change substantially during plasma operation. Such divertor is not appropriated either for devices that rely their magnetic configuration on the bootstrap current, like QPS (ORNL, USA) or NCSX (Princeton, USA), or for devices that present high flexibility in their rotational transform values, like TJ-II. For these cases, the flux expansion divertor is a good candidate. This concept is based on intercepting the particle and energy fluxes with a plate located where the magnetic lines are well separated, so that the power flux onto the plates is small. NCSX team is performing studies to develop a divertor based on this concept, consisting of following field lines with an ad-hoc diffusion coefficient of 1 m/s^2. TJ-II presents several magnetic configurations suitable for such a divertor concept. Exploration of TJ-II capabilities for having a flux-expansion divertor are done by following particle trajectories rather than field lines due to the large drifts that appear in this device. ISDEP code, which follows ion guiding-center orbits including collisions and radial electric field, has been used to study the 3D fluxes in ECRH and NBI plasmas. Beyond the effect of thermal ions, the trajectories of fast ions coming from NBI are considered, using the Monte Carlo Fafner code to estimate the velocities and birth points of fast ions. Comparisons of ISDEP calculations with the results from Langmuir probes, bolometer and Soft X Ray detector arrays are underway. The present results are valuable to explore the viability of the flux expansion divertor in future devices.

TH/P4-16 · Bifurcation Behaviour of Rotation Velocities in Collisional Edge Plasma with Steep Gradient

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Abstract: Both poloidal and toroidal plasma rotations of tokamak edge plasma are known to interact with various other mechanisms and thus are related to plasma stability and transport. Solutions for rotation velocities were studied, using a two-time scales system comprising the ambipolarity constraint and the parallel momentum balance equations of the revisited neoclassical theory, with the corrected contribution from also the gyro-viscosity tensor. Temperature and density profiles with realistic pedestal forms etc were considered known and were controlled parametrically. Similarity of this equation system to reaction-diffusion equations were utilized in the numerical simulation and study of critical points on the bifurcation diagram. It was found that the steepness of the density and temperature gradients have important effects on the rotation stability and on its bifurcative behaviour.

TH/P4-17 · Edge Turbulence, Blob Generation, and Interaction with Sheared Flows

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Abstract: A program of 2D simulations of edge turbulence and blob generation has been carried out using the Lodestar SOLT code. Results have been obtained on the transport of particles and poloidal momentum in the edge and SOL, the processes by which momentum transport drives sheared poloidal flow layers in the edge plasma, the linear instability saturation mechanisms producing the blobs, and the interaction of blob generation with sheared flows. Coppi has conjectured that blobs transport plasma momentum away from the core region towards the wall, and hence provide a momentum "source" that can induce net core plasma rotation. Our simulations illustrate that idea for poloidal momentum transport and poloidal flows. If the sheared flows are sufficiently strong, they can stabilize the underlying instability and saturate the turbulence. If the flows are weak, the saturation occurs by wave-breaking. A synthetic gas-puff-imaging (GPI) diagnostic in the code has been used to compare the simulations with GPI data and with probe data for the NSTX experiment and qualitative agreement has been obtained on several features.

TH/P4-18 · Intermediate Nonlinear Regimes of Line-tied g Mode and Ballooning Instability

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Abstract: A theoretical framework has been developed to describe the nonlinear regimes of line-tied g modes in slab geometry and ballooning instabilities in toroidal configurations. This work is motivated by the correlation of edge localized mode (ELM) activity in H-mode plasmas with ideal MHD peeling-ballooning instability. Recent experimental observations and numerical simulations demonstrate a persistence of ballooning-like filamentary structures well into the nonlinear stage. The theory is based on an expansion using two small scale lengths, the mode displacement amplitude normal to the magnetic flux surface and the mode width in the most rapidly varying direction, both normalized by the equilibrium scale length. When the mode displacement across magnetic flux surfaces is much less than the mode width in the most rapidly varying direction, the mode is in the linear regime. When the mode displacement grows to the order of the mode width in the rapidly varying direction, the plasma remains incompressible to lowest order, and the Cowley-Artun regime is obtained. The detonation regime, where the nonlinear growth of the mode could be finite-time singular, is accessible when the system is sufficiently close to marginal stability. At higher levels of nonlinearity, the system evolves to the intermediate nonlinear regime, when the mode displacement across the magnetic surface becomes comparable to the mode width in the same direction. During this phase, the nonlinear growth of the mode in parallel and perpendicular directions are coupled, and sound wave physics contributes to nonlinear stabilization. The governing equations for the line-tied g mode and the ballooning instability in the intermediate nonlinear regime have been derived. For the line-tied g mode case, the solution of the equations agrees with direct ideal MHD simulations, indicating the dominance of the intermediate nonlinear regime in the nonlinear evolution of the line-tied g mode. Similar comparison studies are being carried out for the nonlinear ballooning mode case. We are also applying the nonlinear ballooning equations to the edge pedestal region of an H-mode tokamak, taking into account two characteristic features of that region, the narrow pedestal width and the magnetic field structure associated with diverted tokamaks.

TH/P4-19 · RMP Enhanced Transport and Rotational Screening in DIII-D Simulations

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Abstract: The application of resonant magnetic perturbations (RMP) to DIII-D plasmas at low collisionality has achieved ELM suppression, primarily due to a pedestal density reduction. The mechanism of the enhanced particle transport is investigated in 3D MHD simulations with the NIMROD code. The simulations apply realistic vacuum fields form the DIII-D I-coils, C-coils and measured intrinsic error fields to an EFIT reconstructed DIII-D equilibrium, and allow the plasma to respond to the applied fields while the fields are fixed at the boundary, which lies in the vacuum region. A non-rotating plasma amplifies the resonant components of the applied fields by factors of $2-5$. The poloidal velocity forms $E\times B$ convection cells crossing the separatrix, which push particles in to the vacuum region and reduce the pedestal density. Low toroidal rotation at the separatrix reduces the resonant field amplitudes, but does not strongly affect the particle pump-out. At higher separatrix rotation, the poloidal $E\times B$ velocity is reduced by half, while the enhanced particle transport is entirely eliminated. Various energy transport models are implemented in NIMROD, to determine which produces the closest experimental match the observed temperature profiles. Multiple initial equilibria, and a range of dimensionless parameters are simulated to determine the role of the initial stability properties, and to obtain a scaling to realistic experimental parameters for DIII-D and ITER.

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TH/P4-20 · Theory and Modelling of Edge Plasma Transport, Plasma-Wall Interactions, and Dust Dynamics

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Abstract: The physics of edge and Scrape-off Layer (SOL) plasmas are very complex and multifaceted. It requires the understanding of physics of anomalous plasma transport and plasma flows, plasma-wall interactions, including dust production and transport. Here, we present our recent results of theoretical studies and simulations of the following edge and SOL plasma issues: A) It is known that intermittent convective-like transport associated with meso-scale coherent structures (e.g., blobs, ELMs) are often dominant in the cross-field transport at the edge. The generation of ELM is typically attributed to the peeling-ballooning instabilities, while the mechanism of blob formation is not clear. We show that the interplay of the interchange drive and nonlinear effects associated with drift wave turbulence can lead to the blob formation. B) Near-sonic parallel plasma flows has been measured in the SOL of several tokamaks. We demonstrate the dominant role of ballooning-like asymmetry of cross-field transport in the large plasma flow formation and present the UEDGE simulations of parallel plasma flows and radial electric field including plasma drifts. Synergistic effect of transport asymmetries and drifts on large plasma flows in the core plasma and near separatrix SOL regions will be discussed. C) Sophisticated integrated modelling tools are needed for assessment of plasma-wall interactions. Here we present newly developed code WallPSI capable of modelling the wall processes including erosion rates and concentration of absorbed, mobile and trapped particle species in the wall material. We simulated with WallPSI the hydrogen transport in the wall impacted by plasma for various materials. We study thermal instability, which may occur due to plasma-wall coupling, by coupling the 1-D plasma transport code to WallPSI. D) In reactors like ITER dust can pose safety hazards and impact plasma performance. In this work we analyze dynamics and statistics of dust particles with the DUSTT code: dust trajectories experimentally measured on NSTX with fast cameras; dust radius distribution, resulting from laser scattering measurements on DIII-D, taking into account non-Rayleigh regimes of light scattering as well as dust evaporation due to heating by laser radiation. Dust wall collisions we analyze with the LS-DYNA code.

TH/P4-21 · Effects of Transport and Non-thermal Particles on Kinetic H-mode Pedestal Evolution

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Abstract: Poloidal and toroidal plasma rotation plays a critical role that includes: The quenching of the turbulent transport; the formation of the edge and internal transport barriers, where anomalous transport is strongly reduced so that neoclassical effects play an important role; and the transition to the enhanced confinement mode. It is important to have a self-consistent model of the tokamak plasmas that includes neoclassical and anomalous effects as well as neutral beam dynamics in order to describe the physics of intrinsic and imposed plasma rotation. The focus of this paper is the application of this model to understand the effect of transport and non-thermal particles on kinetic H-mode pedestal evolution in order to optimize tokamak performance. The beam geometry package from the NTCC NUBEAM module has been recently implemented in the XGC0 code. Neutrals in the beam geometry package are started with random angles from the injector plate and tracked to the tokamak plasma edge. Once the neutrals enter the plasma, they are handled by a model for neutrals in the XGC0 code. Turbulence-driven transport is modeled as a radial random walk diffusion of particle orbits with diffusivity computed using a predictive transport model. The GLF23 gyro-Landau-fluid and the Multi-Mode fluid based drift-wave transport models are coupled with the XGC0 code. The dynamic evolution of the plasma edge region is computed in self-consistent simulations using the XGC0 code. These simulations include the formation of sheared velocity flows and $\boldsymbol{E} \times \boldsymbol{B}$ flow shear, turbulence transport suppression, and formation of the H-mode pedestal. It is demonstrated that, in the H-mode pedestal region, the turbulence is strongly reduced by the strong $\boldsymbol{E} \times \boldsymbol{B}$ flow shear. The dependence of the pedestal width and height on plasma parameters such as plasma density, elongation and triangularity is investigated. Scaling dependencies derived from the simulation results are compared with the scaling dependencies derived from experimental observations. The effects of neutral beam torque and non-thermal particles are demonstrated. A projection of the simulation results for ITER scenarios is presented.

TH/P4-22 · Fully Nonlinear Edge Gyrokinetic Simulations of Kinetic Geodesic-Acoustic Modes and Boundary Flows

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Abstract: We present edge gyrokinetic neoclassical simulations of tokamak plasmas using the fully nonlinear (full-f) continuum code TEMPEST. Either a drift kinetic or non-linear Boltzmann model is used for the electrons. The electric field is obtained by solving the 2D gyrokinetic Poisson Equation. We demonstrate the following: (1) Higher harmonic resonances significantly enhance geodesic-acoustic mode (GAM) damping at high-q (tokamak safety factor), and are necessary to explain both the damping observed in our TEMPEST q-scans and experimental measurements of the scaling of the GAM amplitude with edge safety factor q_{95} in the absence of obvious evidence that there is a strong q dependence of the turbulent drive and damping on the GAM. (2) The kinetic GAM exists in the edge for steep density and temperature gradients in the form of outgoing waves, its radial scale is set by the ion temperature profile, and the ion temperature inhomogeneity is necessary for GAM radial propagation. (3) The development of neoclassical electric field evolves through different phases of relaxation, including GAM, their radial propagation, and their long-time collisional decay. (4) Natural consequences of orbits in the pedestal and scrape-off layer region in divertor geometry are substantial non-Maxwellian ion distributions and flow characteristics qualitatively like those observed in experiments.

TH/P8-1 · Cross-field Transport Driven by Turbulent Scattering of Particles in Tokamaks

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Abstract: The experimental evidence for wave-particle decorrelation dominates the cross-filed electron transport is presented in this paper. The experiment was carried out in the HT-7 superconducting tokamak using a Langmuir probe array. Several time scales related to the turbulent electron transport were directly measured, such as the parallel kinetic decorrelation time 5.3 μs, the eddy turn over time 8.7 μs, the turbulence local autocorrelation time 10 μs, the turbulence autocorrelation time in wave frame 273 μs. The experimental results indicate that the parallel kinetic decorrelation time is the most relevant dissipation time scale underlying the anomalous electron transport in high-temperature fusion plasmas. The importance of resonant scattering on cross-field transport was recently pointed out by Liu Chen. This new idea directly challenges the traditional mixing-length-estimation based transport paradigms. This problem has also been discovered by Lin, et al. in their recent particle simulation of electron temperature gradient mode turbulence: the underlying mechanism responsible for electron anomalous transport is not fluid eddy mixing but kinetic wave-particle decorrelation. An analytical theory about resonant scattering of particles by turbulence in tokamaks is formulated using drift kinetic equations and resonant broadening theory in this paper. The mechanism underlying the electron heat transport is found dominated by nonlinear kinetic wave-particle decorrelation rather than by fluid eddy mixing. The overlapping of neighboring phase-space islands and stochastic disturbance of orbits by turbulence leads to a diffusion in real space as well as in velocity space, with a time scale determined by the resonant scattering time. The dominant decorrelation mechanism in sheared magnetic field and two toroidal effects, i.e., one for passing particles and the other for trapped particles, are discussed here. It is indicated that the transport estimation based on the traditional mixing length conjecture is often misleading, and the nonadiabatic response of electrons induced by resonant scattering is an important nonlinear effect, should not be ignored in the study of turbulent transport in tokamaks.

TH/P8-2 · ITER Simulations with Internal and Edge Transport Barriers

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Abstract: Simulations of ITER with the presence of both an edge transport barrier (ETB) and an internal transport barrier (ITB) are carried out using the BALDUR integrated predictive modelling code. In these simulations, the height of the pedestal is calculated using a new version of the pedestal temperature model based on the magnetic and flow shear stabilization concept together with the infinite-n ballooning stability concept. To enhance the predictive capability of the pedestal model, values of the safety factor and magnetic shear are taken directly from the calculation in the BALDUR code. In this manner, the effect of the bootstrap current is properly included. A version of semi-empirical Mixed Bohm/gyroBohm (Mixed B/gB) core transport model is used that includes ITB effects. In this model, the anomalous transport in the core can be stabilized by the influence of $\varpi_{E\times B}$ flow shear in regions with low or reversed magnetic shear. The Mixed B/gB transport model with ITB effects together with the pedestal model are used to simulate the time evolution of temperature, density and current profiles for ITER discharges. The presence of both transport barriers results in complicated scenarios that yield improved performance relative to the standard H-mode discharges. In addition, the interactions between both transport barriers and sawtooth crashes are investigated.

TH/P8-3 · Performance Analysis of Compact Tokamak Reactors

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Abstract: A new approach to fusion power has been recently considered – to demonstrate early power production in a compact reactor with low first wall load. The reduced load allows using presently available first wall technologies, until new materials are developed and tested by ITER and future component-testing facilities. However, the use of the small fusion power output of the pilot plant has to be maximized either by energy multiplication methods (fuel breeding) or in applications such as hydrogen production at high temperature, high-level waste transmutation, and testing of fusion nuclear technology components. Low aspect ratio tokamaks with increased toroidal field seem to be the ideal candidates for these applications, either by using replaceable central copper rods or high temperature superconductor technology. In this paper the performance of compact tokamak reactors is analyzed, considering the fusion power, power gain and average wall load. Stability issues related to the toroidal β limit, safety factor and density limit are taken into account. The analysis is based on the solution of the global power balance equation with the convection and conduction losses modeled by empirical scaling laws (ITER scaling law in particular). The plasma model includes geometrical aspects, profiles and impurities effects, neoclassical effects, and stability constraints. A convenient normalization of the plasma temperature and density, and of the auxiliary power, is introduced, which leads to the definition of a simple figure-of-merit parameter. This figure of merit sets the operating conditions of the tokamak reactor according with the required performance. In this way, it is possible both to search for a set of machine parameters that satisfies the performance goal and to classify different tokamaks by their figure-of-merit value.

TH/P8-4 · Plasma Shaping Effects on Temperature Gradient-Driven Instabilities and Geodesic Acoustic Modes

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Abstract: Plasma shaping effects on temperature gradient-driven instabilities and geodesic acoustic oscillations are investigated. Based on the local equilibrium of noncircular tokamak plasmas, a gyrokinetic model with integral eigenmode equations is developed to study the effect of shape parameters such as finite aspect ratio, elongation, triangularity, and Shafranov shift gradient on temperature gradient-driven modes. Finite aspect ratio has a general stabilizing effect, while the elongation can be stabilizing or destabilizing, depending on the poloidal wavelength region of the mode and other parameters. Since the growth rate spectrum is greatly shifted towards shorter poloidal wavelength regions when the elongation increases, the region of the fast-growing linear modes will depart from that of $k_\theta \sim 0.2 - 0.4$, where critical gradients are calculated previously. Therefore, lower critical gradients are obtained from our calculation and shows weaker dependence on elongation than the scaling from previous studies. The geodesic acoustic mode is also investigated with a gyrokinetic theory in noncircular, finite aspect ratio, toroidal plasmas. Dependences of the GAM frequency on finite aspect ratio and elongation are investigated when trapped particle effects are neglected in the large aspect limit. It is found that the GAM frequency sharply decreases with an increasing elongation by dependence of $\sqrt{2/(\kappa^2+1)}$, and slightly decreases as a parabola with an increasing inverse aspect ratio.

TH/P8-5 · A Possible Model for Non-Local Electron Heat Transport

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Abstract: In the present model, the non-local electron heat transport is theoretically studied from both microphysics (electromagnetic electron temperature gradient instabilities) and macrophysics (global power relation). The conditions of core electron temperature rise in low density plasmas or drop in high density plasmas are obtained and compared with experiments. Qualitatively, the present results are in good agreement with those observed in experiments.

TH/P8-6 · Full-f Gyrokinetic Simulation of Tokamak Plasma Turbulence Using ELMFIRE

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Abstract: A global gyrokinetic full-f particle-in-cell code ELMFIRE exploiting advanced and efficient data storage means has been designed for first principles transport simulation in tokamaks including both neoclassical and turbulent physics. It is based on an implicit gyrokinetic model where the coefficient matrix in the field equations is constructed and updated at each time step from particle data. Both linear and nonlinear benchmark tests are discussed. Experimental benchmarking is performed with the Doppler reflectometric measurements at FT-2 tokamak for plasma rotation and correlations and spectra in fluctuations. First results of edge plasma simulation in conditions of L-H transition with ASDEX Upgrade like parameters are presented.

TH/P8-7 · Validity of Quasi-Linear Transport Model

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Abstract: In order to gain reliable predictions on turbulent fluxes in tokamak plasmas, physics based transport models have to be improved. Nonlinear gyrokinetic electromagnetic simulations for all species are still too costly in terms of computing time. On the other hand, interestingly, quasi-linear approximation seems to retain the relevant physics for fairly reproducing both experimental results and nonlinear gyrokinetic simulations. Quasi-linear fluxes are made of two parts, the quasi-linear weight and the squared electrostatic potential. The first is shown to follow well non-linear predictions; the second is based on both non-linear simulations and turbulence measurements. A new quasi-linear transport model, QuaLiKiz, based on a fast linear gyrokinetic code has been recently developed. The validity of the quasi-linear approach is tested against nonlinear gyrokinetic simulations, using the Eulerian code GYRO and the semi-Lagrangian code GYSELA. A good agreement of the de-phasing between the potential and the transported quantities (particles, ion and electron energy) in coupled ITG-TEM turbulence is observed between quasi-linear and non-linear regimes. On the other hand, the quasi-linear flux weights for various ITG-TEM cases are affected by a slight, but constant, over prediction with respect to the nonlinear values. The chosen saturation level is discussed versus results from nonlinear GYRO/GYSELA simulations. The frequency and wave vector shapes obtained by reflectometry and laser backscattering measurements in Tore Supra are confronted to GYRO/GYSELA simulations. For example, the density fluctuations spectrum scaling as $k_\theta^{-3.5}$ observed with laser backscattering experiments seems to be reproduced by collisionless GYRO simulations, while strong levels of collisionality lead the fluctuations to scale rather as $k_\theta^{-4.7}$. The impact of both types of slopes is tested for the quasi-linear flux estimations. Finally, to further test the quasi-linear model for turbulence ranging from ITG to TEM dominated, the impact of T_i/T_e is tested for parameters ranging from hot-ion mode in TFTR to dominant electron heating in Tore Supra. The ratios between ion heat flux, electron heat flux and particle flux for quasi-linear and non-linear simulations are shown to agree well for coupled ITG-TEM turbulence.

TH/P8-8 · Influence of RF Fields on Anomalous Impurity Transport in Tokamaks

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Abstract: The transport of impurities by Ion-Temperature-Gradient (ITG) and Trapped-Electron (TE) mode turbulence in the presence of RF fields in the ion cyclotron range of frequencies is investigated using an electrostatic, collisionless fluid model. The trace impurity diffusivity and convective velocity (pinch) are calculated and their scaling with plasma parameters are investigated. Two separate effects of the RF field on the ITG/TE stability and transport are considered: a four wave parametric process involving a fast magnetosonic source wave and ion cyclotron sidebands; and the ponderomotive force associated with the RF field of the fast magnetosonic wave. It is found that the inward impurity convective velocity can be reduced by the RF field, in particular close to the wave resonance location where the RF ponderomotive force may be significant.

TH/P8-9 · Electromagnetic Self-Organization and Transport Barrier Relaxations in Fusion Plasmas

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Abstract: In hot magnetized plasmas, the cross-field transport is dominated by the presence of instabilities which give rise to both, electrostatic and magnetic fluctuations. Experimental measurements of fluctuation levels on typical fusion devices reveal that magnetic perturbations are typically much smaller than electrostatic perturbations. However, as even small magnetic fluctuations can locally modify the topology of the magnetic surfaces, they play an important role with respect to the transport properties of the plasma. In this work, we show results from turbulence simulations at a tokamak plasma edge, realized with EMEDGE3D, a three dimensional global code in toroidal geometry. This code evolves self-consistently electromagnetic fluctuations as well as the pressure and $E\times B$ velocity profiles. The level of magnetic fluctuations is linked to the β parameter (the ratio between the kinetic pressure and the magnetic pressure). Using statistical tools (structure functions, Extended Self-Similarity), we analyze the impact of the electromagnetic fluctuations and we show that finite β effects may cause an increase of the irregularity in the redistribution of the energy in the turbulent cascade, leading in this way to an increased degree of intermittency. The competitive mechanisms (Reynolds and Maxwell stresses) that are responsible for the generation of the large scale flows, which regulate the level of the turbulent fluctuations in the plasma, are investigated. It is found that with increasing beta, the steady state component of the flow is reduced and replaced by an oscillatory behavior. We also focus on the dynamics of transport barriers at the plasma edge. The presence of a transport barrier is a key element of high confinement modes (H-modes) in fusion devices. In the present work we report the first barrier relaxations obtained in the general electromagnetic case including magnetic fluctuations effects. The main mechanisms inducing these relaxations in a purely electrostatic case are also present here, i.e., a transitory growth of a mode localized at the barrier center. In addition, peaks of the magnetic flux and strong magnetic fluctuations are associated to such relaxations events, in agreement with experimental observations.

TH/P8-10 · Results From the International Collaboration on Neoclassical Transport in Stellarators (ICNTS)

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Abstract: This contribution describes work carried out within the IEA Implementing Agreement for Cooperation in Development of the Stellarator Concept, with the ultimate goal of providing a comprehensive description of neoclassical transport processes in stellarator experiments. Such a description is mandatory for analyzing experimental results and carrying out predictive simulations as the performance of high-temperature stellarator discharges generally conforms to neoclassical expectations for confinement, bootstrap current and parallel electric conductivity. Within the ICNTS, a thorough benchmarking of the various numerical methods used to calculate mono-energetic neoclassical transport coefficients in the complex 3-D magnetic-field topologies of planned and existing stellarators has been performed. An overview of these results is presented here along with the theoretical and numerical tools required for practical application of these results.

TH/P8-11 · Diamagnetic GAM Drive Mechanism

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Abstract: Geodesic Acoustic Modes (GAM), poloidal flows oscillating at the characteristic acoustic frequency of a tokamak or stellarator, are an ubiquitous edge plasma phenomenon. In recent years, they have dramatically gained experimental interest and are candidates for applications ranging from plasma diagnostics to transport control. GAMs oscillate between states of strong rotation and up-down asymmetric plasma compression. Hence, at first glance the natural turbulent drive (or damping) mechanisms for them is either a direct boost of the rotation – via Reynolds stress – or the creation of asymmetric pressure distributions – via transport. However, an up-down asymmetric pressure may also be created by the divergence of the diamagnetic drifts, if there is a perturbation in the diamagnetic drift velocity, i.e., of the overall pressure gradient. That in turn may fluctuate due to any modulation of the flux surface averaged turbulent transport. On the other hand, a modulation of the turbulent transport due to the GAMs themselves is expected to happen in the tokamak edge, and has been observed early on in simulations and recently in many experiments. In turbulence simulations for edge parameters, the described effect tendencially is a strong driver of the GAMs of equal importance to the other two. As a striking consequence, the coupling of diamagnetic velocity and GAM can produce propagating fronts of high flow velocity and transport, which closely resemble avalanches – without necessity of a critical gradient. The diamagnetic flow drive is strong enough to advance the flow and transport layer in radial direction – although the linear dispersion relation would just result in a localized oscillation! An interesting consequence of the diamagnetic drive mechanism is that it offers the possibility of direct excitation of GAMs by resonantly modulated external heating (replacing turbulent transport with heating power). If the GAMs are detected by Doppler reflectometry, the achievable efficiency is certainly enough for diagnostic purposes such as to *actively* probe the GAM frequencies or to measure the turbulence response to the GAMs. Particularly exciting however is the prospect of a way to *artificially* set up a GAM pattern to control the transport.

TH/P8-12 · Multi-Scale, Multi-Mode Gyrokinetic Simulations and Implications for Transport Modelling of Tokamaks and Optimized Stellarators

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Abstract: Gyrokinetic turbulence simulations have reached a level of maturity which allows for direct comparisons with experiments. Here, we present recent GENE results for the tokamaks ASDEX Upgrade, JET, DIII-D, and NSTX, as well as for the optimized stellarators W7-X and NCSX. In this context, new effects are discovered which have an impact on our understanding and modelling of turbulent transport in present-day and future fusion devices. In particular, we find that: (i) different microinstabilities (like ITG modes and TEMs) can interfere destructively, exhibiting significant deviations from conventional simple predictions but better agreement between simulations and experiments (ii) based on this effect and the notion of attractors in a multi-dimensional gradient space, one can better understand the relaxation of the experimental temperature and density profiles towards their equilibrium states (iii) for realistic ion heat (and particle) flux levels and in the presence of unstable ETG modes, there tends to be a scale separation between electron and ion thermal transport, enforcing the inclusion of sub-ion scales in comprehensive transport models (iv) near the tokamak edge, microtearing modes and ion-scale ETG modes peaking near the X-point (both of which are not included in present-day transport models) are important, with the latter setting the base level of the electron heat diffusivity in the edge barrier region (v) the first gyrokinetic simulations of two-species ITG/TEM turbulence for the optimized stellarators W7-X and NCSX point to the possibility of turbulence control in 3D systems via geometric optimization Our studies are based on the nonlinear gyrokinetic code GENE which is electromagnetic, includes pitch angle and energy scattering for each particle species, and is able (via a newly developed field-line tracing procedure) to extract the required geometrical information from actual MHD equilibria, without any model assumptions. Radially nonlocal effects will be investigated by means of a global version of GENE which is currently under development.

TH/P8-13 · Global Electromagnetic Gyrofluid/Gyrokinetic Computation of Turbulence and Self Consistent Rotation in Large Tokamaks

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Abstract: We report on the theory and computation of gyrofluid and gyrokinetic turbulence in large tokamaks with special emphasis on the self consistent equilibrium including the poloidal rotation profile. The use of both gyrofluid and gyrokinetic models provides important control cases against the role of trapped electrons and ions. The models are necessarily global and electromagnetic, with strict energy and entropy conservation laws used as diagnostics. A control method for the particle weights in the PIC model provides correct thermodynamic saturation. A Hamiltonian discretisation scheme in the phase space continuum model provides correct long term nonlinear responses. For small tokamaks the turbulence can compete with neoclassical processes to determine the rotation profile but in large tokamaks the latter are dominant due to the comparative size scalings of the various processes. Edge turbulence is investigated with all the models, with regard to profile self consistency and also ELM scenarios. In all scenarios converged cases are found only if the spectrum reaches down to the ion gyroradius, so even the ELM scenario saturation is well outside the MHD regime.

TH/P8-14 · Nonlinear Excitation of Zonal Flow and Geodesic Acoustic Modes in the Edge of Tokamak Plasma

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Abstract: A self-consistent model of the multiscale interaction of zonal flows (ZFs), and geodesic acoustic modes (GAMs) with edge turbulence is presented. In the collisionless and collisional regimes, the dominant short scale modes in the background turbulence at the edge of tokamak plasmas are ion temperature gradient (ITG) mode and high-m drift resistive ballooning modes (DRBM), respectively. The modulational instability of ZFs and GAMs in the background of ITG and DRBM driven turbulence has been studied. The kinetic wave equation is used to study the adiabatic interaction between long scale ZFs, and GAMs, and the small-scale background turbulence. We have also determined conditions under which ZFs saturate by different mechanics such as by (i) collisional damping (ii) instability to tertiary modes (iii) nonlinear trapping of ITG mode turbulence in ZFs, giving coherent structures etc. A self-consistent, simplified low-dimensional model of these interactions is also constructed. This model has the characteristic form of a 'predator-prey' system in which the population of the drift wave quanta grows via the finite-beta DRBM and ITG instabilities, generates ZFs, and GAMs via the modulational instability and reacts back on the confinement of the plasma.

TH/P8-15 · Self-consistent Simulation of Torque Generation by Radial Current due to Fast Particles

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Abstract: Toroidal rotation plays an important role in transport and stability of burning plasmas, and clarification of the mechanism of rotations spontaneously generated by perpendicular NBI and ICRF can help evaluate the rotations in the plasmas quantitatively, contributing to accurate control of them. In order to study the properties of plasma rotations, one-dimensional multi-fluid transport code TASK/TX has been developed, which is the first code to self-consistently solve a tokamak plasma evolution including a plasma rotation and a radial electric field E_r. TASK/TX automatically generates a return current j_{bulk} due to the charge separation originated from the difference in electron and ion banana orbit widths, resulting in a $\vec{j}_{\text{bulk}} \times \vec{B}$ torque on a bulk plasma. Through the simulations, we confirm the generated torque drives a toroidal rotation accompanied by the modification of a radial electric field without having to add extra torque input term in equations of motion, while a poloidal rotation almost remains unchanged. It is found that the toroidal rotation decreases with a weaker dependence than inversely proportional as the turbulent viscosity increases.

TH/P8-16 · Transport due to Electromagnetic Turbulence in Externally Heated Plasma

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Abstract: Turbulent transport caused by multi-scale interactions is investigated by numerically solving a reduced set of two-fluid equations. We propose a new three-dimensional numerical simulation of transport phenomena in an open system controlled by external heat source and sink by virtue of self-consistent calculation of interactions among electromagnetic micro-turbulence, zonal flows, and macro-MHD instabilities. We have found fast turbulent thermal diffusion by a strong external heating. In addition we have found a global ion temperature profile flattening when the turbulence coexists with macro-MHD instabilities.

TH/P8-17 · Multi-scale Transport Dynamics Dominated by Multiple Dissipation Mechanisms near the Critical Gradient

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Abstract: We found a new class of transient transport near the critical gradient (CG) referred to as GAM growing intermittency [1], which is caused by the collision-less GAM damping and leads to dynamical establishment of the Dimits shift. Here, we present a new predator-prey model which includes the effect of anisotropic pressure perturbation (GAM) and parallel ion sound velocity and successfully reproduces the essential features of the growing intermittency. We have also extended the simulation model by taking into account the collisional zonal flow (ZF) damping. Due to the mixture of two kinds of damping mechanisms, i.e., the GAM damping and collisionnal damping, the growing intermittency is found to recursively appear accompanied with complex envelope modulation to ZFs over collisional (or transport) time scale. Furthermore, we have investigated the effect of zonal pressure (ZP) near the CG, which also works as a dissipation mechanism. The ZP changes the temperature scale length through the coupling with GAMs and causes a sudden termination of the growing intermittency. Thus, the multiple dissipation mechanisms are found to synergetically couple each other and lead plasmas to complex dynamical transport over long time scale.

[1] K. Miki etal. Phys. Rev. Lett. 99, 145003 (2007).

TH/P8-18 · New Characteristics of Zonal Flows in Multi-scale Plasma Turbulence

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Abstract: The nonlinear interaction in multi-scale plasma turbulence including resistive kink-tearing mode (RKTM) and ion temperature gradient (ITG) mode is studied based on a 5-field gyrofluid simulation aiming to understand the underlying physical mechanisms. We discuss specifically the zonal flow(ZF) dynamics in mixed MHD and ion-scale turbulence as well as the dual stabilization effects. Here we report two new findings on the ZF characteristics: (1) A robust oscillatory ZF with finite frequency in mixed MHD and ITG turbulence is created due to the multi-scale interaction so that the ion heat transport is not efficiently suppressed. This implies probable degeneration of the favorable role of ITG ZFs by MHD fluctuations. (2) A coupled ZF eigenmode is found in Hasegawa-Mima (HM) model due to the radial spectral effect of ZFs, where all components have the same enhanced growth rate. It is identified that this new eigenmode originates from successive cross coupling between ZFs and turbulent sidebands.

TH/P8-19 · Multi-scale Transport Simulation of Internal Transport Barrier Formation and Collapse

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Abstract: The mechanism of internal transport barrier (ITB) collapse in the reversed shear configuration is investigated using a global Ion Temperature Gradient driven drift wave (ITG) turbulence code. The toroidal momentum source (TMS) is introduced to follow the self-consistent evolutions of the ion temperature and flow profiles. Using a new code, we investigated the Toroidal Flow Shear (TFS) effect on the dynamics of ITB evolution. It is found that the enhancement of lifetime of ITB occurs by TFS effect.

TH/P8-20 · Regulation of Turbulent Transport in Neoclassically Optimized Helical Configurations with Radial Electric Fields

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Abstract: Gyrokinetic Vlasov simulations are performed on the Earth Simulator by using the GKV code in order to investigate effects of helical magnetic configurations and equilibrium radial electric fields on the ion temperature gradient (ITG) turbulence and zonal flows. The simulation result obtained for the inward-shifted configuration of the Large Helical Device (LHD) manifests generation of large-amplitude stationary zonal-flow structures leading to significant turbulent-transport reduction, which was not so obviously shown in our previous simulations using simpler model configurations. The present result confirms the theoretical prediction that helical configurations optimized for reducing neoclassical ripple transport can simultaneously improve the turbulent transport with enhancing zonal-flow generation, and also provides a possible explanation to the confinement improvement observed in the LHD experiments of inward plasma shift. Recent theoretical analysis predicts further enhancement of zonal flows in the presence of equilibrium radial electric fields which produce poloidal $E{\times}B$ rotation of helical-ripple-trapped particles with decreased radial displacements. Thus, the anomalous transport is expected to be more effectively reduced by the equilibrium electric fields in neoclassically optimized helical configurations such as the inward-shifted LHD case. The predicted favorable effects of the radial electric fields are also examined by means of the newly-extended GKV simulations.

TH/P8-21 · Transport Simulations of KSTAR Advanced Tokamak Scenarios

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Abstract: Predictive modelling of advanced tokamak (AT) scenarios is performed for the first half operation phase of KSTAR programme. Various transport codes are used for the simulations employing a first principle transport model. Firstly, steady state operation capabilities are investigated with time dependent simulations using the free-boundary transport code. Secondly, high performance steady state operation scenario reproducibility is investigated using the experimental data from the other tokamak device. Thirdly, DEMO-relevant AT capability is investigated for reactor-relevant conditions which are able to be established in KSTAR. The stability analysis is performed for the target plasma profiles obtained in steady state conditions. Lastly, the possibility of achieving normalised $\beta \sim 5$ is investigated in KSTAR.

TH/P8-22 · Nonlinear Dynamics of Impurities in Turbulent Tokamak Plasmas

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Abstract: Several aspects of impurity transport in turbulent plasmas are presented. We show that the gradient of the confining magnetic field generates a pinch (average velocity) in turbulent plasmas. It is a ratchet type process that appears in test particle approach due to the modification of guiding center trajectories. It determines the contamination of the plasma from the source of impurities localized at the border. Particle collisions and plasma poloidal rotation are included in the test particle model. We show that strong nonlinear effects appear when trajectory trapping or eddying is effective. An additional effect of the magnetic field gradient appears when particle density is considered: the divergence of the $E\times B$ drift produces a pinch velocity, the curvature or turbulent equipartition pinch. The compressibility effect does not appear in the test particle approach and it makes the density pinch different of the test particle pinch. We show that the density pinch is not equal to the curvature pinch and that it is strongly influenced by the ratchet effect. Impurity accumulation (density peaking) can appear only in the presence of trajectory trapping and of a slow poloidal rotation, with velocity of the order of 10^3 m/s for JET plasmas. In these conditions, the presence of collisions determines a dependence of the peaking factor p that is similar to the JET H-mode database for the range of the effective collision frequency appearing there, and that p decays at weaker collisionality. Numerical simulations of the Hasegawa-Wakatami turbulence for inhomogeneous magnetic field were performed in order to verify these theoretical results. The statistical properties of impurity density passively advected by the drift turbulence modeled by the Hasegawa-Wakatami equation are investigated using the structure function analysis. We show that the impurity density and the vorticity of the $E\times B$ drift exhibit similar multifractal behavior. A good agreement is found between the impurity density relative exponent and the She-Leveque model, which shows that intense vortex filaments are responsible for impurity transport. These studies strengthen the idea that the impurity transport in tokamak turbulent plasmas is a nonlinear process with characteristics far from the Gaussian ones, with intermittent behavior and memory effects.

TH/P8-23 · Canonical Profile Transport Model for H-mode Shots in Tokamaks

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Abstract: Self-consistency of temperature and pressure profiles is used to to construct an equation for the heat conductivity in terms of so called 'critical temperature gradients'. The developed Canonical Profiles Transport Model (CPTM), which includes both the heat and particle transport equations, is used to simulate sets of JET, MAST and T-10 shots. Also we found the asymptotic analytical expressions for the pedestal temperature T_{ped}, the width of the transport barrier D and their ratio. The slope of the temperature profile T_{ped}/D depends only on the plasma current I_p. The calculated values of the density pedestal also weakly depend on plasma parameters. To estimate the quality of simulation, we introduce the RMS deviations of calculated values of temperature from experimental ones. For the 10 JET shots with $I_p > 1.5$ MA, RMS are less than 12%. The model predicts an increase of the density profile peaking at low collisionalities. Comparison of the experimental and calculated density profiles for MAST shot #13035 has shown that the model reproduces complicated details well: the H-mode density barrier at the periphery and flat density profile at the plasma core.

TH/P8-24 · Approach to Canonical Pressure Profiles in Stellarators

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Abstract: It was assumed during a long time that the energy and particle transport in a stellarator is defined by a neoclassical transport coefficients, and in this sense the stellarator changes radically from a tokamak. However, observation of the H-mode and internal transport barriers, the proximity of the stellarator energy scaling to the tokamak scaling, all such features lead to the idea that the turbulent transport in stellarators plays an important role. Albeit there is no temperature profile self-consistency in stellarators, nevertheless, the pressure profile is self-consistent for several stellarator devices. For example, NBI heating of the target ECRH plasma leads to dramatic changes of the plasma density and temperature profiles in TJ-II. Values of n_e and T_e are varied up to an order of magnitude, the profiles varied from hollow to peaked (density), and from peaked to flat (electron temperature), but their product, the plasma pressure P_e, presents much stronger profile resilience and the normalized pressure profiles $P^{norm} = P(\rho)/P(0)$ is practically unchanged. The similar behaviour of pressure profiles was observed in CHS, W7-AX and ATF in spite of difference in the magnetic configurations. The normalized pressure profile has universal shape for normal confinement (L-mode) and can be fitted by a quasi-linear function $P(0)^{-1} \cdot dP/d\rho \sim const = k = 1.3 \pm 0.1$. To describe the self-consistent (canonical) pressure profile, we use the variation procedure similar to developed one for tokamaks. We consider the variation problem to minimize the energy functional W under the constraint that the total plasma current J is a constant. As a result, the canonical equilibrium equation and corresponding pressure profile for low-pressure stellarators with model shear has been obtained. The maximal slope of normalized pressure profile was close to maximal slope of profile observed in TJ-II.

TH/P8-25 · Modelling of Tokamak Discharges with the Fast Central Response to the Boundary Plasma Perturbations

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Abstract: One of the fundamental phenomena in tokamak plasmas is the fast propagation of electron temperature perturbation from the boundary to the plasma core (faster than the characteristic transport time). This phenomenon has been observed in different tokamak (and stellarator) experiments: fast plasma core response to the L-H mode transition, fast cold pulse propagation after laser impurity ablation or deuterium pellet injection and some others. Standard diffusive model of local turbulent transport fails to describe these phenomena. It was the reason for the conclusion on a non-local nature of transport in tokamaks. Results of detailed simulations of some discharges with the fast central responses to the boundary perturbations of different nature are presented in this work. Role of different transport mechanisms is investigated. modelling was performed by the ASTRA transport code. The model takes into consideration the behavior of the plasma neutral component and neutrals recycling at the wall. It is shown that the effect of fast central response in discharges under consideration can be explained without the assumption on the non-local character of transport processes in tokamaks. This effect may be attributed to the behavior of the neutral plasma component, which propagation time across the plasma column is sufficiently short (less than 100 μs). A rapid change in the neutral flux into the plasma column may somewhat affect the plasma energy balance in the whole plasma cross section almost simultaneously. For example, the rapid electron temperature increase in the plasma core after the fast L-H mode transition can be explained by the reduction of the cold electron source caused by the decrease of neutral flux in plasma. An opposite effect of fast drop of the core electron temperature after deuterium pellet injection may be explained by the rise of deuterium atom density in the plasma core with corresponding increase of ionization source of cold electrons. So, the neutral plasma component is responsible for the visible coupling plasma edge and the core in considered experiments in time interval just after events.

TH/P8-26 · Reduction of Cross-Field Plasma Transport in Tokamaks due to Power Input Redistribution and Sheared Flow Profile Modification

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Abstract: Reduction of low-frequency (LF) turbulence and the associated anomalous cross-field plasma transport in tokamaks due to redistribution of ECH power input and controlled formation of sheared flow profiles is studied theoretically. Various scenarios of transitions to improved confinement regimes were analyzed and simulated numerically using a cylindrical model of tokamak. A fast decrease of heat flux in the core plasma region to the neoclassical value was observed in a scenario when an appreciable part $(40 - 70\%)$ of ECH power input initially localized near the surface $q = 1$ was switched over closer to the plasma edge. A considerable weakening of LF turbulence and an appreciable increase of plasma life-time were observed as well, while the relative plasma pressure gradient remained almost unchanged. Transient regimes with controlled modification of sheared flows profiles were also simulated. It is shown that presence of high vorticity layer in the sheared flow profile can radically modify the global structure of LF turbulence leading to a transport barrier formation.

TH/P8-27 · Quasilinear Transport Fluxes Driven by Microinstabilities in Tokamaks

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Abstract: Quasilinear transport fluxes driven by microinstabilites in weakly-collisional tokamak plasmas are calculated by a semi-analytical approach based on a solution of the gyrokinetic equation, where collisions are modelled by a pitch-angle scattering operator. Scalings with collisionality, magnetic drift frequency, diamagnetic frequency and ratio of the density and temperature scale lengths have been determined. The particle transport due to ion temperature gradient (ITG) modes can be reversed from inward to outward as electron collisions are introduced, if the plasma is far from marginal stability. However, if the plasma is close to marginal stability, collisions may even enhance the inward transport. Comparison with transport fluxes calculated when collisions are modelled by an energy-dependent Krook operator shows that the sign and the magnitude of the fluxes are very sensitive to the form of the collision operator. Unlike the ITG-driven flux, the trapped electron (TE) mode driven flux is expected to be outwards for conditions far from marginal stability and inwards otherwise, and the presence of collisions contributes to the inward flow.

TH/P8-28 · Transport in ITER-like Plasmas in Neoclassical, Fluid and Gyrokinetic Descriptions

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Abstract: Turbulent and neoclassical transport in ITER-like plasmas is studied using a hierarchy of modelling tools. Transport from ITG/TEM instability is studied with nonlinear gyrokinetic simulations (using GYRO) and compared with results from an electrostatic fluid model. GYRO simulations include full kinetic ion (D,T,^{4}He,C) and electron dynamics. Key issues ignored in standard transport modelling, such as D-T flow separation, as well as electromagnetic effects on electron transport, are considered and consistency of the ASTRA transport simulations is examined. Neoclassical transport of particles and energy is also computed using a new first-principles kinetic code with an advanced collision operator. Simulations are based on ITER-like profiles derived from ASTRA modelling. For the ASTRA profiles used in the simulations, the fluid and gyrokinetic simulations are in qualitative agreement, with inward particle flux is observed for all species. Importantly, we find that the ion and electron energy fluxes are significantly larger in the core than the values obtained by ASTRA in steady state. This implies that the ASTRA profiles may be overpredicting the ion temperature and thus the fusion power.

TH/P8-29 · Symmetry breaking effects of toroidicity on toroidal momentum transport

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Abstract: A derivation of symmetry breaking toroidicity effects on toroidal momentum transport has been made from the stress tensor. This effect is usually stronger than the symmetry beaking caused by the flows themselves on the eigenfunction. The model obtained generalizes a recent derivation of diagonal transport elements from the stress tensor to convective elements [1, 2] of turbulent equipartition or thermoelectric types. This gives possibility for interpretation of the same type of effects previously obtained from a phase space conserving nonlinear gyrokinetic equation [2, 3] and means that we have a consistent fluid derivation of a model that has already given promising results in predictive simulations.

[1] D. Strinzi, A.G. Peeters and J. Weiland, Accepted for publication in Phys. Plasmas. [2] A. G. Peeters, C. Angioni and D. Strinzi, Phys. Rev. Lett. 98, 265003 (2007). [3] T.S. Hahm, P.H. Diamond, O.D. Gurcan and G. Rewoldt, Phys. Plasmas 14, 072302-1 (2007).

TH/P8-30 · Global Nonlinear Simulations of Ion and Electron Turbulence Using a Particle-In-Cell Approach

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Abstract: Understanding turbulence driven by micro-instabilities such as Ion Temperature Gradient (ITG) or Trapped Electron Modes (TEM) is one of the key issues to accurately predict the performance of future Tokamak devices such as ITER. Simulations with gyrokinetic codes require extremely large computer resources, especially for TEM turbulence where the fast electron time scale needs to be resolved simultaneously with the ions time scale. So far, physical studies of TEM turbulence have been essentially obtained using the flux-tube approach but very little is known on global TEM turbulence. This work presents global nonlinear simulations performed with the Particle-In-Cell code ORB5. In order to counteract profile relaxation due to the turbulent transport, a heat source in combination with a noise control algorithm has been implemented in the code, which restricts the accumulation of numerical noise in the system, and allows long simulations to be run which reach a quasi-steady state. The electron model retains kinetic trapped electrons but considers Boltzmann passing electrons. The reliability of TEM simulations will be addressed both numerically and physically by a careful look at the numerical noise and the late time fluxes. While in ITG turbulence, the main saturation mechanism is the $E \times B$ shearing due to the zonal flow, the main saturation mechanism for TEM turbulence is still unclear and, according to flux-tube simulations, seems to depend on the physical parameters. Global simulations of TEM turbulence at moderate rho-star with Cyclone-like parameters will be presented. Results will be compared with GENE flux-tube simulations for different equilibrium gradients values. The second part of this work will focus on the rho-star scaling for ITG turbulence. Gyrokinetic simulations predict a transition of the transport from Bohm to Gyro-Bohm, but results are still controversial. With the help of the noise control algorithm, nonlinear simulations with heat sources and noise control are performed in which quasi-steady state values of the ion diffusivity and temperature gradients are obtained in order to study such transition.

TH/P8-31 · Pedestal Temperature Models with Self-Consistent Calculation of Safety Factor and Magnetic Shear

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Abstract: In this work, three pedestal temperature models in the work by T. Onjun [1] are improved by using a real time self-consistent values of safety factor and magnetic shear, which are acquired from 1.5 D BALDUR integrated predictive modelling code. This modification results in a better estimation of pedestal width and pedestal pressure gradient since the geometrical and bootstrap current effects are properly included. The modified pedestal temperature models are used together with a core transport model to describe the H-mode discharges obtained from DIII-D and JET tokamak experiments. The core transport used in this work is either the Mixed Bohm/gyro-Bohm (Mixed B/gB) core transport model or the Multimode (MMM95) core transport model. For each discharge, profiles of electron density, electron temperature and ion temperature from simulations, as well as their values at the top of the pedestal, are compared with the corresponding experimental data from the DIII-D and JET tokamaks. It is found that the predicted pedestal values are in the range of the corresponding experimental data and the profiles yield reasonable agreement with the data near the pedestal, but show high deviation near the plasma-core region. The RMS errors of each profile from each discharge, as well as the offset, are calculated to quantify the agreement.

[1] T. Onjun, et al., Physics of Plasma 9, 5018 (2002).

TH/P8-32 · Non-perturbative Models of Intermittency in ITG Drift Wave Turbulence with Zonal Flows

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Abstract: There has been overwhelming evidence that coherent structures play a critical role in determining the overall transport in magnetically confined plasmas. These coherent structures cause intermittent, bursty events, which can mediate significant transport of heat and particles, for instance, imposing a large heat load on container walls. A crucial question in plasma confinement is thus the prediction of the PDFs of the transport due to these structures and of their formation. Here, we report on a first analytical result on these two closely related problems by using a novel non-perturbative method. We first compute the PDF tails of global momentum flux and heat flux in the ion-temperature-gradient (ITG) turbulence, by assuming that a short-lived modon (a bipolar vortex soliton) is a coherent structure responsible for bursty and intermittent events, contributing to the PDF tails. The tails of PDF of global momentum flux and heat flux are shown to be stretched exponential, which is broader than a Gaussian. This result suggests that rare events of large amplitude due to coherent structure are crucial in transport (similarly to what was found in the previous local studies), offering a novel explanation for exponential PDF tails of momentum flux found in recent experiments at CSDX at USCD. We show the crucial dependence of the overall amplitude of the PDFs on the temperature and density scale lengths. We then present a consistent theory of the PDFs of momentum flux and of the formation of shear flow, by incorporating the interplay between ITG turbulence and shear flow. While the PDF tails of momentum flux have similar stretched exponential, the amplitude of the PDF tails are significantly reduced by shear flows. Furthermore, the PDF tails of shear flow formation have different exponential behaviour, with the overall amplitude severely quenched by strong flow shear. These results highlight the key role of structures on intermittent transport through the feedback on turbulence. Implications for transport in tokamaks will be discussed.

TH/P8-33 · Gyro-Kinetic Study of Toroidal Momentum Transport

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Abstract: This paper will report on a gyro-kinetic study of toroidal momentum transport. Both linear as well as non-linear results will be presented, and both the diagonal (proportional to the velocity gradient) as well as the pinch velocity contribution (proportional to the toroidal velocity) of the radial momentum flux are investigated. All simulations use the flux tube geometry, and are performed with the gyro-kinetic code GKW. The study aims at both clarifying the physics processes involved in toroidal momentum transport as well as accurately predicting the rotation profiles. The diagonal part of the momentum transport is expressed in a Prandtl number, i.e., the ratio of the momentum diffusivity and the ion heat conductivity coefficients. The Prandtl number (Pr) is found to be in the range $0.6 - 1.2$ over a wide range of plasma parameters in both linear as well as non-linear simulations. The parallel velocity shear drive is found to be small. The paper then reports on the momentum pinch velocity due to curvature effects on micro-instabilities in a toroidal plasma. This pinch can be elegantly derived in the co-moving system, and is then connected with the "Coriolis drift". A simple fluid model is used to highlight the physics mechanism. The derived pinch velocity will be split into a part due to the gradients, and the part due to the turbulent equipartition. Turbulent equipartition alone will be shown to be unable to generate peaked rotation profiles in low collisionality H-mode plasmas. The accuracy of the analytic models is found to be limited through the calculations using the full gyro-kinetic model. The normalized pinch is smaller by roughly 40% compared with the analytic model, and its dependence on the density gradient is weaker. Non-linear simulations reproduce the pinch found in linear theory. The results indicate that the pinch is somewhat weaker than the linear theory suggests, and follows roughly the same scaling. The results of this study predict a moderately peaked rotation profile for ITER assuming zero momentum input.

TH/P8-34 · An Angular Momentum Source in Tokamaks

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Abstract: Although the torque that drives a net toroidal mass flow in tokamaks can sometimes be attributed to an external heating source such as unbalanced neutral beams, it has been long recognized that there has to be an intrinsic momentum source that operates even in purely Ohmic plasmas [1] without an obvious external drive. This paper is concerned with one such intrinsic source of momentum in toroidal plasmas that has its origin in neoclassical physics. The main ideas that have been already discussed in earlier works [2, 3] are clarified and extended here, with appropriate scaling studies to indicate their relevance to next-step devices like ITER. Results of numerical calculations with our CTD code are compared with experimental results, in particular with those from C-Mod [4].

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[1] I. H. Hutchinson, J. Rice, R.S. Granetz, et al., Phys. Rev. Lett 84, 3330 (2000). [2] A. Y. Aydemir, Phys. Plasmas 14, 056118 (2007). [3] A. Y. Aydemir, Phys. Rev. Lett. 98, 225002 (2007). [4] B. LaBombard, J. E. Rice, A. E. Hubbard, et al., Nucl. Fusion 44, 1047 (2004).

TH/P8-35 · Integrated Modelling Simulations of Toroidal Momentum Transport in Tokamaks

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Abstract: Simulations of existing tokamak and ITER H-mode discharges are carried out using the PTRANSP predictive integrated modelling code to compute the time evolution of the plasma toroidal rotation frequency profile as well as the temperature and magnetic q profiles. Momentum and thermal transport coefficients are computed using either the recently advanced Weiland model [1] or the GLF23 model [2], for transport driven by drift modes, together with the NCLASS model [3] for neoclassical transport. Additional momentum transport can be driven by convection of ions. In neutral beam injected discharges, the source of torque in the plasma core is computed using the NUBEAM module [4], while charge exchange provides a sink of angular momentum near the plasma edge. Simulations are validated by comparing simulation results with experimental data for H-mode and L-mode tokamak discharges with a wide range of values for heating power, plasma density, current, magnetic field and geometry. It is found that flow shear driven in the plasma core by neutral beam injection can have a significant effect on the plasma profiles and fusion power production in ITER simulations. Simulation results indicate that the fusion power production is significantly higher when the toroidal momentum diffusivity is computed by the transport model rather than being taken equal to the ion thermal diffuvisity. In simulations using the GLF23 transport model, the fusion power production is 500 MW when the pedestal temperature is 2.7 keV and neutral beam injection power is 30 MW, or when the pedestal temperature is 3.5 keV and neutral beam injection power is 20 MW.

[1] J. Weiland et al., Phys. Plasmas 12, 012505 (2005); 12, 092509 (2005); 11, 3238 (2004); 33rd EPS Conference on Plasma Physics, 30I, P2.186 (2006); F.D. Halpern et al., Phys. Plasmas 15, 012304 (2008). [2] R.E. Waltz et al., Phys. Plasmas 4, 2482 (1997). [3] W.A. Houlberg et al., Phys. Plasmas 4, 3220 (1997). [4] A.Y. Pankin et al., Computer Phys. Commun. 159, 157 (2004).

TH/P8-36 · Toroidal Rotation in Tokamak Plasmas

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Abstract: Determining the magnitude and evolution of toroidal rotation in ITER plasmas is an important issue – for $E \times B$ flow shear control of anomalous transport, prevention of locked modes, ELM control via RMPs etc. Many effects influence the toroidal rotation in tokamak plasmas. Momentum sources and radial transport due to axisymmetric neoclassical and paleoclassical, and anomalous processes are usually considered. In addition, the toroidal rotation can be affected by magnetic field errors, which this paper concentrates on. Small, non-axisymmetric field errors in tokamaks arise from coil irregularities, active control coils and collective plasma magnetic distortions (e.g., NTMs, RWMs). Nonresonant field errors cause magnetic pumping (TTMP), ripple-trapping and radial drifts of bananas; they thus lead to radial non-ambipolar particle fluxes and neoclassical toroidal viscosity (NTV) flow damping over the entire plasma. Resonant field errors (FEs) cause localized $J \times B$ electromagnetic torques near rational surfaces in toroidally rotating plasmas. Toroidal flow inhibits penetration of resonant field errors into the plasma by producing a shielding effect on rational surfaces. Sufficiently large resonant FEs can lock plasma rotation at rational surfaces to the wall leading to magnetic islands and reduced plasma confinement or disruptions. Many of these processes can also produce momentum pinch and intrinsic rotation effects. A comprehensive picture of all the effects of field errors along with the usual radial plasma transport effects on plasma toroidal rotation has been developed using a fluid moment approach. The toroidal rotation equation results from setting to zero the net radial plasma current induced by the sum of all the non-ambipolar components of the particle fluxes in the plasma from these effects. In general, the NTV due to nonresonant field errors generates a global torque that attempts to rotate the plasma at an “intrinsic” rate that is in the “counter” (to the plasma current) direction and depends on the ion temperature gradient. Inclusion of NTV flow damping effects in a resonant field error penetration model for ohmic tokamaks predicts locking thresholds in better agreement with experimental results from a wide variety of tokamaks, especially with regard to the scaling of the threshold with electron density.

TH/P8-37 · Limitations, Insights and Improvements to Gyrokinetics

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Abstract: We focus on (i) the limitations of gyrokinetic quasineutrality to evaluate the axisymmetric radial electric field; (ii) a gyrokinetic entropy production restriction on the ITER ion temperature pedestal; and (iii) a hybrid gyrokinetic-fluid treatment valid on slowly evolving time scales. (i) Standard gyrokinetics incorrectly determines the axisymmetric, long wavelength electrostatic potential to leading order in gyroradius over major radius as demonstrated by considering a steady-state theta pinch with a distribution function correct to second order. We argue gyrokinetic quasineutrality often improperly determines the potential in the long wavelength, axisymmetric limit for a distribution function evaluated to within numerical resolution [1]. (ii) Using canonical angular momentum as the radial variable allows strong gradients to be treated gyrokinetically. Entropy production then requires a physical lowest order banana regime ion distribution function to be nearly an isothermal Maxwellian [2] with the ion temperature scale much greater than the poloidal ion gyroradius. Thus, the background ion temperature profile in ITER cannot have a pedestal like that of the density. Weak ion temperature variation with subsonic pedestal flow requires electrostatically restrained ions and magnetically confined electrons. (iii) Simulating tokamaks on transport time scales requires evolving drift turbulence with axisymmetric neoclassical and zonal flow radial electric field effects retained. Coupled flux tube gyrokinetic simulations evolving density and temperature are becoming available [3]. However, full electric field effects are more difficult to keep since they require evaluating the ion distribution function to higher order than standard gyrokinetics - well beyond forseeable numerical resolution1. An electrostatic hybrid gyrokinetic-fluid treatment using moments of the full Fokker-Planck equation removes the need to go to higher order. This hybrid description evolves potential as well as density, temperatures and flows, and models all electrostatic turbulence effects with wavelengths much longer than an electron gyroradius.

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[1] M. Barnes and W. Dorland, Intl. Sherwood Fusion Theory Conf. 2D2 (2007). [2] P. J. Catto and R. D. Hazeltine, Phys. Plasmas 13, 122508 (2006). [3] M. Barnes and W. Dorland, Bull. Amer. Phys. Soc. 52, 90 (2007).

TH/P8-38 · Non-local Models of Perturbative Transport: Numerical Results and Application to JET Experiments

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Abstract: Perturbative experiments in fusion plasmas have shown that edge cold pulses travel to the center of the device on a time scale much faster than expected on the basis of diffusive transport. An open issue is whether the observed fast pulse propagation is due to non-local transport mechanisms or if it could be explained on the basis of local transport models. To elucidate this distinction, perturbative experiments involving ICRH power modulation in addition to cold pulses have been conducted in JET for the same plasma. Local transport models have found problematic to reconcile the fast propagation of the cold pulses with the comparatively slower propagation of the heat waves generated by power modulation. In this paper, a non-local model based on the use of fractional diffusion operators is used to describe these experiments. The proposed non-local model is able to reproduce the profiles of the amplitude and the phase of the dominant harmonics of the electron temperature corresponding to the propagation of heat waves excited by the ICRH modulation in JET. Most importantly, for the same model parameter values, the model can successfully accommodate the propagation of pulses with time delays comparable to those in the experiment, ~ 4 ms. We also present a detailed numerical study of the parameter dependence of the transport properties of the fractional model. In addition, we present preliminary results on the role of non-locality in the propagation of pulses in the presence of internal transport barriers.

TH/P8-39 · Role of Zonal Flows in TEM Turbulence through Nonlinear Gyrokinetic Particle and Continuum Simulation

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Abstract: Trapped electron mode (TEM) turbulence exhibits a rich variety of collisional and zonal flow physics. This work explores the parametric variation of zonal flows and underlying mechanisms through a series of linear and nonlinear gyrokinetic simulations, using both particle-in-cell (the GEM code) and continuum (the GS2 code) methods. The two very different and independent codes are compared in studies of TEM turbulence, providing verification. Zonal flows are important in TEM turbulence near the critical density gradient. Our previous work revealed a new nonlinear upshift in the TEM critical density gradient, associated with zonal flow dominated states, which increases strongly with collisionality [1]. In contrast [2], zonal flows have little effect on the TEM saturation level in cases with strong electron temperature gradients and $T_e = 3T_i$. We have addressed this apparent contradiction in a series of linear and nonlinear simulations using GS2 [3] and GEM [4]. Zonal flows in TEM turbulence are sensitive to the electron temperature gradient, T_e/T_i, and other parameters. The link between zonal flows and the strength of the electron temperature gradient can be understood using both TEM linear stability properties and secondary instability theory. We have developed a comprehensive TEM/ETG stability diagram to illustrate that strong electron temperature gradients tend to drive fine scale structure, weakening zonal flow drive. When zonal flows are not the dominant saturation mechanism, a simple mode coupling model shows that stable zonal density fluctuations are driven to large amplitudes, at twice the growth rate of the dominant "primary" mode [5]. Simple estimates of the saturation level agree with GEM simulations.

[1] D. R. Ernst *et al.*, in Proc. 21st IAEA Fusion Energy Conference, Chengdu, China, 2006, IAEA-CN-149/TH/1-3. (http://www-pub.iaea.org/MTCD/Meetings/FEC2006/th_1-3.pdf) [2] T. Dannert and F. Jenko, Phys. Plasmas **12** 072309 (2005). [3] M. Hoffman and D. R. Ernst, Bull. Am. Phys. Soc. (2007). [4] J. Lang, Y. Chen, and S. Parker, *Phys. Plasmas* **14**, 082315 (2007). [5] J. Lang, S. Parker and Y. Chen, accepted for publication in *Phys. Plasmas*, May 2008.

TH/P8-40 · Core and Edge Full-f ITG Turbulence with Self-consistent Neoclassical and Mean Flow Dynamics Using a Real Geometry Particle Code XGC1

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Abstract: We report new core and edge physics of the ITG turbulence including the turbulence-driven mean and neoclassical background dynamics self-consistently, using a full-f gyrokinetic code XGC1 in a real tokamak geometry with magnetic X-point and conserving nonlinear Fokker-Planck Coulomb collisions. Some of the new findings in the core plasma are 1) the initial ITG modes growth is rather quasilinear than linear. Accordingly, highly anisotropic growth of streamers may not be observed experimentally. 2) Strong interaction of the ITG turbulence with the mean and neoclassical background dynamics results in an ion heat conductivity closer to the experimental observations ($\sim 1\ \mathrm{m}^2/\mathrm{s}$), even in the cyclone plasma, than the much higher values obtained from the conventional delta – f simulation. 3) Plasma temperature gradient is relaxed to marginally stable η_i and maintained throughout the nonlinear stage, thus exhibiting temperature profile stiffness. Some surprising new findings in the edge are 1) The ITG turbulence exists in a mild pedestal where the local η_i is lower than the conventionally known critical values from the core plasma profiles. 2) Stronger interaction of ITG with mean/neoclassical field in the pedestal does not allow any streamer-like structures in the growth stage. 3) There is a strong co-current parallel flow of the background plasma in the scrape-off region, which is stronger at the high field side. 4) At the same time, there is a mean $E\times B$ rotation in the positive poloidal direction (carrying the plasma toward the inner divertor) near the separatrix. $E\times B$ rotation is in the negative poloidal direction near the wall due to the wall-sheath effect. Thus, a large convective $E\times B$ circulation of the plasma is formed in the scrape-off layer, during which many particles hit the divertor plate. Other important issues to be presented include radial momentum pinch and diffusion, internal transport barrier formation in a central reverse sheared magnetic geometry, L-H transition study, role of zonal and mean flow to the nonlinear saturation, the role of neoclassical field on turbulence, neutral effect on the edge turbulence, and experimental validation.

TH/P8-41 · Gyrokinetic Turbulence Simulation of Physics Basis for Transport Modelling

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Abstract: Most of existing transport models for whole discharge simulation are based on some form of quasilinear theory (QLT), the validity of which depends on the ordering of several characteristic time scales of the turbulence. In particular, the Chirikov stochasticity parameter $S > 1$ and the turbulence Kubo number $K < 1$ are required for the validity of the QLT. Determination of the time scales to verify the QLT requires detailed diagnosis of the physical processes underlying nonlinear decorrelation, saturation, and anisotropic scattering. First-principles, gyrokinetic turbulence simulations have matured enough to address the key physics basis of the transport modelling and to validate the turbulence simulation models through comparisons with experimental measurements. In this paper, we report a verification of the quasilinear transport model through global gyrokinetic particle simulations using the gyrokinetic toroidal code (GTC). Relevant kinetic and fluid time scales which enter K and S are systematically examined in a comparative study of the turbulence driven by electron temperature gradient (ETG) instability, ion temperature gradient (ITG) instability, and collisionless trapped electron mode (CTEM). Transport processes for both active and passive species are studied. The simulation results provide the foundation for the ordering of time scales in the transport models and clarify the validity regimes of the quasilinear transport models.

TH/P8-42 · Testing the Trappes Gyro-Landau Fluid Transport Model With Data From Tokamaks and Spherical Tori

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Abstract: Results of the first tests of the trapped gyro-Landau fluid (TGLF) [1-2] transport model with experimental data from the Spherical Tori MAST and NSTX will be reported and compared with recent tests with the conventional tokamaks DIII-D, JET, and TFTR [2]. The TGLF transport model is a reduced theoretical model that has been constructed to fit a large database of nonlinear gyrokinetic turbulence simulations using the GYRO code [3]. The TGLF model generalizes the methods of its predecessor GLF23 [4] to a more accurate system of moment equations that is valid for shaped magnetic geometry and finite aspect ratio. Recent test of TGLF [3] with a data from 96 discharges and three tokamaks found that the improved fidelity of TGLF to exact gyrokinetic theory resulted in an improved prediction of the tokamak temperature and density profiles compared to GLF23, strengthening the case for gyrokinetic driftwave turbulence being the dominant energy transport mechanism in tokamaks. The plasma-stored energy incremental to the boundary energy was predicted by TGLF with an RMS error of 19%. The GLF23 model prediction had an RMS error of 36% for this same dataset. High wavenumber electron temperature gradient (ETG) modes are predicted to play a dominant role in electron energy transport in NSTX [5]. In the highest $E\times B$ velocity tokamak discharges it is found that the ion transport is reduced to neoclassical and nearly all of the electron energy transport is due to high-k modes. Sperical tori have a naturally high $E\times B$ velocity shear suppressing the low-k turbulence and providing a good test of the high-k gyrokinetic physics modeled in TGLF. TGLF is the first theory-based transport model to include finite aspect ratio shaped magnetic equilibria. This capability results in a much more uniform accuracy of TGLF between circular (TFTR) and shaped (DIII-D, JET) tokamaks and is expected to be essential for spherical tori.

[1] G.M. Staebler, et al., Phys. Plasmas **12** (2005) 102508. [2] J.E. Kinsey, et al., Phys. Plasmas to be published. [3] J. Candy and R. E. Waltz, J. Comput. Phys. **186** (1003) 545. [4] R.E. Waltz, et al., Phys. Plasmas **4** (1997) 2482. [5] S.M. Kaye, et al., Nucl. Fusion **47** (2007) 499.

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TH/P8-43 · Role of Impurity Cyclotron Damping in Ion Heating and RFP Turbulence

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Abstract: In the reversed field pinch ions are heated anomalously relative to collisional equipartition with Ohmically heated electrons. The process channels electron energy to ions in a way that is still not understood. Recent observations suggest that impurities are preferentially heated. A theory for ion heating via impurity ion-cyclotron-resonant damping of the turbulent energy cascaded from unstable global tearing modes is presented. The theory treats the magnetic turbulence as an Alfvén wave cascade and calculates the rate of damping to impurity ions via cyclotron resonances. The transient temperature rise in sawtooth crashes is modeled from a 1-D transport model that accounts for the resonant cyclotron heating, anomalous heat losses, and collisional equipartition between impurities, bulk ions, electrons, and parallel and perpendicular temperatures. The larger fluctuation level at lower impurity cyclotron frequencies leads to heating rates that are consistent with experiment. Dependencies on temperature, density, and magnetic fluctuation level are described. The magnetic fluctuation spectrum is modeled, first for dissipation associated with viscosity and resistivity, and then for dissipation from cyclotron resonances. Viscosity and resistivity produce a dissipation range spectrum with exponential decay beyond a dissipative wavenumber. This spectrum is fit to the magnetic fluctuation spectrum of MST, which shows exponential decay. The inferred values of viscosity and resistivity are much larger than the true values, suggesting that the spectrum is subject to a damping mechanism that is stronger than viscosity or resistivity. Cyclotron resonant dissipation leads to a spectrum with an asymmetry in toroidal wavenumber, consistent with a resonance of an Alfvén wave frequency with impurity ion cyclotron frequencies. The MST spectrum has an asymmetry in toroidal wavenumber with missing energy in the wavenumbers of the Alfvén/cyclotron resonance. This suggests that anomalous ion heating from cyclotron resonance absorption has observable consequences on the spectrum.

TH/P8-44 · Interaction between Turbulence and Neoclassical Dynamics and Its Effect on Tokamak Transport: Gyrokinetic Simulations and Theory

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Abstract: To deepen our understanding of burning plasma transport properties, it is required to take into account the interaction between turbulence and neoclassical dynamics. 1) The relationship between momentum and energy transport is investigated for burning plasmas with focus on understanding recent experimental observations, using global gyrokinetic simulations with proper coupling between turbulence and neoclassical dynamics. First, from our neoclassical simulations it is found that the effective toroidal momentum diffusivity is much smaller than the ion heat diffusivity, i.e, $\chi_\phi/\chi_i \sim 0.01 - 0.1$ due to the absence of banana orbit enhancement. The neoclassical contribution to the toroidal momentum transport is negligably small compared to experimental levels for NSTX and DIII-D plasmas. On the other hand, our turbulence simulations verify that there exists strong coupling between ion momentum and heat transport for ion temperature gradient (ITG) driven turbulence, and the effective χ_ϕ/χ_i is on the order of unity. This is in broad agreement with experimental observations in conventional tokamaks where low-k fluctuations are believed to be responsible for a high level of plasma transport. Further, we have found that residual turbulence can survive the dissipation of a strong mean $E\times B$ flow shear and drive a significant momentum flux. Moreover, the equilibrium $E\times B$ flow shear is found to reduce the turbulence driven transport for energy more efficiently than for momentum. These findings may offer an explanation for recent puzzling experimental observations that the toroidal momentum transport remains highly anomalous, even while the ion heat flux is reduced to a neoclassical level. Also reported are interesting new results of large inward momentum fluxes found in the transient phase of ITG turbulence, which lead to core plasma rotation spin up. This robust inward flux may relate to the residual stress due to the symmetry breaking induced by self-generated zonal flow shear. 2) We revisit the neoclassical physics in the presence of microturbulence. The turbulence can impact the level of bootstrap current via two processes: i) turbulence generated "zonal" pressure corrugation and ii) potential fluctuation induced effective random "kicking" of drift orbits.

TH/P8-45 · Study of ITER Perfomance Based on Different Plasma Geometry

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Abstract: Effects of plasma geometry on nuclear fusion power production in ITER tokamak are investigated using the 1.5D BALDUR integrated predictive modelling code. In this work, shaping parameters, which are elongation and triangularity, are varied, and their impacts on nuclear fusion power production is observed. These simulations are explored with two different core transport models: an empirical based Mixed Bohm/gyro-Bohm (Mixed B/gB) and a theoretical based Multimode model (MMM95). It is found that as elongation is increased, the simulations based on MMM95 model show a steady rise of the fusion performance until at the elongation of 1.9. On the other hand, the ITER performance decreases steadily with elongation when Mixed B/gB model is used. Regarding triangularity dependence, the simulations based on both MMM95 and Mixed B/gB models show a decrease in the nuclear fusion power production as the plasma becomes more triangulated. Furthermore, the simulations using MMM95 consistently yield higher plasma performance (approximately by a factor of 3) than those using Mixed B/gB. In addition, when MMM95 is used, it appears that the ion temperature gradient (ITG) and trapped electron modes (TEM) are the dominant modes. When the Mixed B/gB is used, it appears that the Böhm contribution is the most dominant term.

TH/P8-46 · Global Gyrokinetic Simulations of ρ^* and ν^* Scalings of Turbulent Transport

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Abstract: Predicting the confinement time in next step controlled fusion devices such as Iter requires to know how the turbulent transport depends on key scaling parameters, more specifically on ρ^*, the ratio of the ion Larmor radius to the machine size a, and on the collisionality nustar. Such a scan is performed with the global gyrokinetic code GYSELA, which solves the standard gyrokinetic equation for the full distribution function of the ions with adiabatic electrons. The code achieves more than 80% efficiency on 4 096 processors. Gyrokinetic simulations have predicted the transition from Bohm to gyroBohm scaling when decreasing ρ^*, the open issues remaining the precise value of this transition and the underlying physics. Such a scan is performed with GYSELA. While the correlation time is roughly independent of ρ^*, the correlation length λ_c looks consistent with the gyroBohm scaling at small ρ^* and well above the linear threshold. Conversely, at large values of ρ^* and close to the threshold, λ_c exhibits a dependence on the system size, consistent with the Bohm scaling. When examining the latter regime, one finds that the gyroBohm scaling is more relevant at small ρ^*, suggesting that the transition value between Bohm and gyroBohm scaling is close to $\rho^* = 0.01 - 0.02$ and might depend on the distance to the threshold. Interplay between neoclassical and turbulent transport is addressed with ν^* scans. A Fokker-Planck operator has been added to GYSELA, acting in the parallel velocity space only. It is shown analytically that the neoclassical transport coefficients are recovered with such an ion-ion collision operator in the three collisional regimes, banana, plateau and Pfirsch-Schlütter. Below the critical Ion Temperature Gradient for the onset of turbulence, the poloidal equilibrium velocity is found to reverse sign when increasing ν^* from the banana to the Pfirsch-Schlütter regime, in good agreement with theoretical predictions. Collisions modify the response of the zonal flows, as well as the turbulent heat flux. The dependence of the turbulent heat flux on ν^* is presently being analyzed for these simulations. It is found to combine the reduction of the linear growth rate, the damping effect on zonal flows as well as the spreading into the stable boundary regions and the profile relaxation.

TH/P9-1 · Global MHD Stability Study of KSTAR high β Plasma Equilibria Under Passive and Active Mode Control

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Abstract: The Korea Superconducting Tokamak Advanced Research, KSTAR, is designed to operate a steady-state, high β plasma while retaining global magnetohydrodynamic (MHD) stability to establish the scientific and technological basis of an economically attractive fusion reactor. Global MHD mode stabilization can be achieved by equilibrium profile optimization, passive control via plasma rotation, and active control. Equilibria spanning the expected operational range of KSTAR and constrained by coil system capabilities are computed using EFIT for various pressure and q profiles including generic L-mode and DIII-D experimental H-mode pressure profiles. Ideal MHD stability of toroidal mode number of unity using DCON shows a factor of two improvement in the plasma β limit over the no-wall β limit at moderate to low plasma internal inductance. KSTAR equilibria will be computed using EFIT and VALEN based on the planned experimental start-up scenario and subsequently from experimental data when available in the summer of 2008. Evaluating and verifying advanced MHD control systems gives greater confidence for stabilization of future devices, including the advanced scenario operation of ITER. The planned stabilization system in KSTAR comprises passive stabilizing plates and actively cooled in-vessel control coils (IVCC) designed for non-axisymmetric field error correction and stabilization of slow timescale MHD modes including the RWMs. VALEN analysis using standard proportional gain shows that active stabilization near the ideal wall limit was reached with feedback gain 1.0 – 10 V/G using the mid-plane segment of the IVCC. For a fast IVCC coil circuit with $L/R = 1.0$ ms the upper limit of the RMS power required using noise taken from NSTX stabilization experiments is computed to be on the order of 1 MW for β near the ideal wall limit. Advanced state-space control algorithms are theoretically capable of reaching higher β limits at reduced RMS system power than allowed by classical control techniques. Present computations using an advanced Linear Quadratic Gaussian (LQG) controller, including balanced truncation to as few as 3 states of the VALEN state-space yield a factor of 2 power reduction assuming white noise while remaining robust with respect to variations in plasma β. (Supported by US DOE Grant DE-FG02-99ER54524)

TH/P9-2 · Exploration of Configurational Space for Quasi-isodynamic Stellarators with Poloidally Closed Contours of the Magnetic Field Strength

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Abstract: In a quasi-isodynamic (qi) stellarator [1] with poloidally closed contours of B [2] the confinement of reflected particles is exclusively determined by the behavior of the contours of the second adiabatic invariant J if the gyro radius of the energetic particles is sufficiently small for J to be a valid invariant. In this work, the configurational space of such configurations is further investigated for finding very-high-beta MHD-stable configurations with a small equivalent ripple characterizing the neoclassical transport. The qi configuration considered earlier exhibits a parallel current density with a dipole component which impairs MHD stability at very high β in the qi situation considered here with shallow magnetic well in the associated vacuum magnetic field and it also impacts the compatibility of very high β with small neoclassical ripple. It has been clarified that quasi-isodynamicity is compatible with vanishing dipole current density in a more elaborate structure of the field period exploiting the possibility of its detailed toroidal design in 3d configurations, and that configurations of this type indeed promise MHD stability at very high β values, average β approximately 0.2. One restriction, inherent in the previous type of configuration studied, has still been present: only one minimum of the magnetic field strength, so that only one class of reflected particles exists. Two tendencies have been discovered: i) it still seems to be difficult to achieve small equivalent ripple, ii) during the optimization process a tendency to form two minima in the field strength has preliminarily been avoided for simplicity. Currently, the behavior of the first tendency within a sequence of configurations with larger number of periods and aspect ratio is studied and it appears that a further structural change in the strength of the magnetic field occurs before small effective ripple is compatible with very high stable beta. In addition, the second tendency is exploited and appears to enable small effective ripple, too. A detailed description of these configurations will be presented.

[1] Gori S., Lotz W., Nührenberg J., Theory of Fusion Plasmas (International School of Plasma Physics), Bologna: SIF (1996) 335. [2] Subbotin A.A. et al, Nucl. Fusion 46 (2006) 921.

TH/P9-3 · Passive Stabilization of the Vertical Mode in Tokamaks by Localized Non-axisymmetric Fields

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Abstract: We find that vertical instability of tokamak plasmas can be controlled by nonaxisymmetric magnetic fields localized near the plasma edge at the bottom and top of the torus. The required magnetic fields can be produced by a relatively simple set of parallelogram-shaped coils. By providing stable equilibria with more highly elongated cross-sections, the addition of these nonaxisymmetric fields can potentially lead to devices with improved β limits (as predicted by Troyon scaling) and improved confinement (as predicted by empirically derived global confinement scaling laws). The nonlinear effect of the coils, with the stabilizing field increasing exponentially in the vacuum regions just above and below the plasma, can potentially reduce the susceptibility to vertical displacement events (VDEs). Although the coils are not stellarator coils, in the sense that they do not produce vacuum flux surfaces or vacuum rotational transform, they do modify the rotational transform of the tokamak near the edge, reducing or eliminating the need for RF current drive in that region for steady-state operation. An analytical calculation with a number of simplifying assumptions is performed for the purpose of demonstrating the physics of the stabilization, and to obtain an estimate of the required magnitude of the nonaxisymmetric field for stabilization. The analytical calculation assumes a large aspect ratio plasma that is well approximated by a cylinder, zero beta, and a uniform equilibrium current density. Stability is determined by a δW calculation, using the stellarator approximation for both the equilibrium and stability calculations. It is estimated that a nonaxisymmetric field with a maximum magnitude at the plasma edge of about 10% that of the toroidal field is needed to see a substantial stabilization effect. A set of nonaxisymmetric coils proposed by Furth and Hartman are calculated to have essentially the same vertical stabilization effect as the simpler parallelogram-shaped coils, so that the vertical stabilization demonstrated experimentally by Furth-Hartman coils supports the feasibility of stabilizing vertical modes by the simpler coil set. The physical mechanism of the stabilization suggests that the stability properties do not depend on the precise shape of the coils, so that curvature can be introduced to optimize relative to other considerations.

TH/P9-4 · A Generalized Relaxed MHD Model for 3D Equilibria with KAM Surfaces

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Abstract: The calculation of three-dimensional magnetohydrodynamic (MHD) toroidal plasma equilibria is a challenging one conceptually because of the problems of Hamiltonian field-line chaos and singular currents. Combined with the need to achieve high accuracy for plasma stability calculations, this makes the problem very difficult, but important for advanced fusion experiments. A new approach based on multiple relaxed regions separated by ideal-MHD toroidal barrier tori is presented. This is cast as a variational principle, with the second variation giving necessary stability conditions for both ideal MHD and resistive MHD (in the limit of small resistivity).

TH/P9-5 · Fast Growth and Sheared Flow Generation in Nonlinear Development of Double Tearing Modes

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Abstract: The nonlinear development of double tearing modes (DTMs) in a large aspect ratio torus is simulated. The emphasis is placed on the mechanisms for the fast growth and sheared flow generation in the development of parallel electron viscosity DTMs. Four nonlinear developing stages: the early growth, transition, fast growth and decay, are found. In comparison with the helical magnetic flux contour, the fast growth is revealed to be associated with the second reconnection and annihilation of the magnetic islands formed by the first reconnection of equilibrium magnetic field in early stage of the mode development. The quantitative scaling of the growth rate and the flow shear with respect to electron viscosity is presented. The flow effects on ion temperature gradient turbulence are analyzed. The relation between low safety factor q rational surfaces, MHD activities, flow layer and ITB formation in reversed magnetic shear configurations is discussed.

TH/P9-6 · Analysis of the Relaxed States for the Rotating Plasmas with no Momentum Input

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Abstract: Plasma rotation has many beneficial effects on tokamak operation including stabilization of MHD modes and suppression of microturbulence to sustain high β equilibria and improve energy confinement. Plasma rotation has been observed in NBI, ICRF heated plasmas as well as in pure Ohmic plasma, in which case there is no momentum input. In this paper, the relaxation state with a mass flow but no momentum input is investigated. The analysis for the dissipation, injection and transition of generalized helicity shows that the pure ohmic plasma with an AC magnetic helicity injection may relax to a state with flow duo to the generalized helicity transition, which gives a possible explanation for plasmas rotation without momentum input. The relaxed state of plasma with rotation in Ohmically driven tokamaks with an arbitrary aspect ratio is explored using the principle of minimum total energy dissipation rate subject to the generalized helicity balance and the energy balance. The resulting Euler–Lagrange equations for plasmas with flow are solved analytically. The solutions describe the structure, transition, and sensitive parameters of the relaxed state with plasma rotation. It is found that there exist different types of relaxed states in the different regions of the parameter space for a specific device. The different plasma current profiles include the typical experimental profiles and the profiles with hole or reversed current in the central region. The results show that there exists a key parameter $(E_o\nu)/(B_o\eta)$ (where η is plasma resistivity, ν is plasma viscosity, B_o and E_o related to boundary toroidal magnetic field and electric field respectively) in determining the final relaxed states and there exists the critical value of this key parameter to induce the abrupt state transition. The results indicate the rotation characteristics for the minimum dissipation state. It is shown that plasma fluid vorticity is in parallel the plasma current density in co-current or counter-current directions. The flow can even be reversed from co- to counter- (or counter- to co-) current direction during transitions.

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TH/P9-7 · Optimization of Magnetic Perturbation Spectra for the COMPASS Tokamak

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Abstract: The present contribution describes preparatory calculations for producing controlled resonant magnetic perturbations (RMPs) using the set of saddle coils of the COMPASS tokamak, recently transferred from UKAEA Culham to IPP Prague. In the future experimental programme of COMPASS we plan to focus on studies of RMPs, especially in view of their application as an ELM control mechanism and their planned use in ITER. We computed the spectra of perturbations for several different equilibra simulated by MHD equilibrium and transport codes and determined the positions and sizes of the resulting islands. The adaptation of coil configuration to a particular equilibrium is necessary because of varying safety factor profiles among the equilibria. It is shown how the saddle coils of COMPASS can be adapted to our equilibria to obtain good island overlap at the edge, which is believed to be a key component in the ELM mitigation effect. The mutual relation between the nonlinear plasma response and the perturbation field is discussed, using results of a cylindrical reduced MHD code.

TH/P9-8 · Nonlinear Dynamics of Magnetic Islands Imbedded in Small Scale Turbulence of Edge Tokamak Plasmas

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Abstract: In tokamaks, macro-scale MHD instabilities (magnetic islands) coexist with micro-scale turbulent fluctuations and zonal flows. Although several works were devoted to the study of macro-scale and micro-scale instabilities separately, only few investigations were devoted to explore the mutual interaction between these instabilities. In this work, we study the interaction between a tearing mode and a pressure gradient instability (interchange's like instability). This can lead to the generation of an island poloidal rotation. The properties and the origin of such rotation are investigated. The model used to describe the simultaneous evolution of the interchange turbulence and the tearing instability is a RMHD one. The numerical study of the latter model shows the existence of four regimes. First, we observe a linear growth of the magnetic island driven by an interplay between the pressure gradient and the perturbed magnetic field (this result has been recovered by analytic calculations). In the second phase, the system reaches a nonlinear saturated state where the island width saturates. In the third regime, a secondary instability triggered by interchange modes destabilizes the saturated state and allows the generation of turbulent small-scales. The latter produce in return zonal flows and nonlinear diamagnetic effects which produce the rotation of the magnetic island. The parameter dependence of the dynamics of the interchange-tearing system will be explored. A special attention will be given to the link between small dissipation parameters and amplitudes of the stresses as well as the influence of the β parameter on nonlinear rotation of the island.

TH/P9-9 · Computational Study of Magnetic Islands in the W7-X and NCSX Stellarators

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Abstract: The magnetic fields of stellarators and, in general, all not strictly axisymmetric toroidal fusion devices exhibit magnetic islands. They strongly influence the confinement properties and are exploited in divertor concepts. So, the existence and the structure of the magnetic islands are an important issue in configuration design. Codes that determine finite-plasma-beta stellarator equilibria while fully accounting for their island structures exist, e.g. the Princeton Iterative Equilibrium Solver [1]. An alternative approach to the assessment of magnetic islands in finite-beta stellarator equilibria has been developed with the method of perturbed equilibria. Since a perturbed equilibrium represents a small deviation from an equilibrium, ideal magnetohydrodynamic (MHD) stability theory and ideal MHD stability codes, e.g., the Code for the Analysis of 3-dimensional Equilibria [2], can be used to determine a perturbed equilibrium. A widespread tool for the calculation of finite-beta stellarator configurations is the Variational Moments Equilibrium Code [3] which is based on the assumption of nested surfaces. With the ideal MHD picture and this assumption surface currents on the surfaces with rational rotational transform are used to model islands. The width of an island can be estimated from the strength of such a surface current which in turn depends on the discontinuity of the ideal MHD normal displacement computed by the CAS3D code. The PIES code and this alternative method have been applied to the National Compact Stellarator Experiment [4] and the Wendelstein 7-X stellarator [5] and yield comparable island sizes.

[1] PIES, Reiman et al. Comput. Phys. Commun. 43:157, 1986. [2] CAS3D, Nuehrenberg et al., Phys. Plasmas, 10:2840, 2003. [3] VMEC, Hirshman et al., Comput. Phys. Commun. 43:143, 1986. [4] NCSX, Zarnstorff et al., Plasma Phys. Control. Fusion, 43:A237, 2001. [5] W7-X, Lotz et al., Nucl. Fusion Suppl. 2:603, 1991

TH/P9-10 · Three-dimensional Effects in Tokamaks - How Tokamaks Can Benefit from Stellarator Research

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Abstract: We give an overview of 3-d effects in tokamaks, emphasising, in particular situations where synergies exist with stellarator research, in particular in the development and use of computational tools. We restrict ourselves to situations, where the 3-d perturbations grow slowly ($\gamma \ll R/v_A$), so that the plasma passes through a sequence of ideal-MHD equilibrium states, with the time-dependence determined by the resistive decay of plasma or wall currents (excluding thereby Alfvén-type modes). In particular we investigate the influence of 3-d structures like magnetic field ripple and NTMs on the confinement of fast ions and the effect of 3-d wall structures on the growth of resistive wall modes (RWMs). 3-d magnetic field perturbations lead to losses of fast particles, generally not associated with resonances in velocity space, but similar to those in non-optimized stellarator equilibria. They can now be well diagnosed with fast particle analyzers having high resolution in time and velocity space. We have followed the orbits of NBI ions in 3d ASDEX Upgrade equilibria with NTMs and magnetic ripple. Comparing the code results with fast ion loss measurements on ASDEX Upgrade we found good qualitative agreement. The combined effect of magnetic field ripple and NTMs and/or field perturbations generated by external error fields applied for ELM suppression might have a significant effect on the fast particle confinement at ITER. We are therefore investigating the effect of magnetic field ripple, external field perturbations and an (2,1) NTM on the fast particle orbits in ITER both for NBI generated fast ions or α particles. The consideration of realistic wall structures (with port-holes) in case of unstable RWMs leads to a coupling of different toroidal modes, effectively suggesting the use of stellarator codes for their analysis. We have therefore developed a full 3d linear MHD stability code that deals with realistic resistive wall structures (including holes) and applied it to the planned ITER and (modified) ASDEX Upgrade wall geometry. Plasma rotation has a strong effect on the predicted mode growth rate, but the theoretical investigation of rotation damping requires a kinetic treatment. We have therefore utilized the truly (gyro-)kinetic stability code LIGKA (developed originally for fast particle driven modes) to the analysis of resistive wall modes.

TH/P9-11 · Temperature-Gradient-Driven Tearing Modes in the Semicollisional Tokamak Regime

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Abstract: The observation of spontaneous islands growth in tokamak experiments indicates that the tearing mode could also be linearly unstable. However, in the framework of resistive magnetohydrodynamics, previous works have demonstrated that drift effects, electron heat transport along magnetic field lines and ion Larmor radius effects have stabilizing effects. Recently, numerical calculations and analytical theory in the collisional regime have demonstrated that inclusion of perpendicular electron heat transport carries out a new type of linear tearing mode driven by a large electron temperature gradient. In this paper, we present analytical and numerical studies, which extend previous results to more realistic high-temperature regimes in which, besides perpendicular thermal conduction, ion Larmor radius effects have to be taken into account. We obtain that the perpendicular electron heat transport produces a strong destabilizing term proportional to the electron temperature gradient in the expression of the island growth rate.

TH/P9-12 · Effects of a Toroidal Rotation on the Stability Boundary of the MHD modes in the Tokamak Edge Pedestal

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Abstract: Effects of a toroidal rotation on the stability of the MHD modes in the edge pedestal, which relate to the edge-localized mode (ELM), are investigated numerically. The new linear MHD initial value code MINERVA is developed for analyzing the effects of a toroidal rotation on the edge MHD stability with high accuracy. As the numerical results, a toroidal rotation increases the achievable maximum pressure gradient. Moreover, it is found for the first time that the edge MHD mode whose toroidal mode number is intermediate (>5) is stabilized, and this stabilization realizes to sustain a steep pressure gradient even in the low edge current density case. From the result in this paper and the previous theoretical studies about the effect of the sheared toroidal rotation on the ballooning mode, the toroidal rotation is thought to stabilize the edge MHD mode due to suppressing the destabilizing contribution from the pressure-driven component of the MHD mode.

TH/P9-13 · Suppression of Error-Field-Induced Magnetic Islands by Alfvén Resonance Effect in Rotating Plasmas

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Abstract: Theory for error-field-induced magnetic islands is developed with inclusion of the Alfvén resonance effect for rotating plasmas. The Alfvén resonance effect changes the magnitude of tearing mode parameter (δ)‘ dramatically and makes (δ)‘ complex. This allows a new stationary state of non-rotating magnetic islands given by the balance between the convection term and (δ)‘ in the extended Rutherford equation. We found that the plasma rotation can reduce the island size compared to previous studies and even eliminate the islands completely when the error field is small. The Alfvén resonance effect generates a significantly larger torque and makes the torque an increasing function of plasma rotation. This indicates the modification of the so-called forbidden band picture adopted to explain the low critical rotation frequency for resistive wall mode.

TH/P9-14 · Integrated simulation of ELM Energy Loss and Cycle in Improved H-mode Plasmas

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Abstract: Integrated transport code with a stability code for peeling-ballooning modes and a transport model of scrape-off-layer (SOL) and divertor plasmas has been improved to take account of the density dynamics, which realizes for the first time the comprehensive study from energy and particle losses due to an edge localized mode (ELM) crash to its cycle. The integrated code reproduces a series of ELMs with experimental characteristics of type-I ELMs, which include the ELM frequency increasing linearly with the input power and the collisionality dependence of the ELM energy loss, and clarifies the following mechanisms. The steep pressure gradient inside the pedestal top as well as the low collisionality, required for improved H-mode plasmas with the HH98y2 factor above unity, is found to enhance the ELM energy loss and reduce the ELM frequency so that the power loss due to ELMs keeps almost constant. The steep pressure gradient inside the pedestal top broadens the region of the ELM enhanced transport. It is found that the collisionality dependence of the ELM energy loss is caused not only by the bootstrap current and the SOL transport but also by the equipartition effect, which enhances ion convective and charge-exchange losses in the low collisionality.

TH/P9-15 · Plasma Rotation Effects on the Trigger of Reversed Shear Plasma Disruptions

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Abstract: Nonlinear responce of a rotating plasma to an external perturbation is investigated in order to prpose a new scenario of a low β disruption of a reversed shear plasma. It is found that, in rotating plasma, magnetic islands formed around inner and outer magnetic resonant surfaces, which are stable for the tearing mode, by an external perturbation, (driven magnetic island) evolve with different growth rates during an initial growth phase. After an initial growth phase, an outer magnetic island grows explosively earlier and triggers an explosive growth of an inner one. At a final phase, enlarged magnetic islands flatten a q-profile in wide radial region including the plasma center. Though this final phase resembles closely a nonlinear destabilization of a spontaneous double tearing mode (DTM), this process can explain time delay of a plasma edge oscillation to trigger an internal MHD events and disruption, that is one of longstanding questions in a low β reversed shear plasma disruption. This process also shows a way that reversed shear plasma passes intermediate rational surfaces, though it collapses when the minimum of q value passes a low rational surface.

TH/P9-16 · MHD Simulation of High Wavenumber Ballooning-like Modes in LHD

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Abstract: MHD stability is one of the key issues of plasma dynamics in magnetic confinement devices. Linear stability analysis of the LHD predicts short-wave instabilities such as the ballooning instability, while in the experiments low m/n modes signals are identified as MHD activities though high modes are not because of difficulties to detect them. A typical understanding is that influences of the growth of high modes are limited in a narrow area of a poloidal cross-section and their influences are not very large. Confirming this understanding is crucial, since it can essentially determine the resolutions required for the theoretical analysis and/or numerical simulations.In our earlier works of MHD simulations of LHD, the lower modes are shown to be superior to higher modes and the growth of unstable modes are not destructive because the impact of the instability is considerably weakened by (1) flow generation to release the free energy to the toroidal direction, (2) the compressibility and (3) nonlinear couplings. However, the influences of the higher modes are not clarified because relatively large viscosities are adopted. Direct numerical simulations of full 3D MHD equations are carried out for the magnetic field with the vacuum magnetic axis position 3.6 m and a peaked pressure profile with the peak β value 3.7%. In our simulations with sufficiently small viscosities, growth and dominance of high m/n pressure-driven unstable modes over lower modes are realized. High m/n (m is around 30) ballooning-like modes cover wide area of a poloidal cross-section so that the growth of higher modes can influence the growth of lower modes, because the higher modes consumes the free energy (pressure gradient) for their growth and the lower modes can not fully make use of it. The higher modes can grow to the levels comparable to the lower modes. The higher modes might stabilize the lower modes by consuming the free energy, presumably being helped by the mechanism (1)-(3) shown in the above. Because of the possible influences on the lower modes, including higher modes in simulations (either via larger number of grid points or via some models) might be important for an advanced numerical studies for quantitative comparisons with the experiments.

TH/P9-17 · Nonlinear Dynamics of Collapse Phenomena in Heliotron Plasma with Large Pressure Gradient

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Abstract: We have executed nonlinear magnetohydrodynamic(MHD) simulations in a helical system of the heliotron-type configuration with a large pressure gradient to reveal the nonlinear dynamics of collapse phenomena of plasma. The simulation results reproduce the basic characteristics of the experimental observation on the so-called core density collapse (CDC) events in the Large Helical Device (LHD) plasma with the internal diffusion barrier (IDB). A long-term nonlinear behavior on the event, including the flushing mechanism of the core pressure, is clarified. The simulation result shows the linear growth of the ballooning-like resistive instability modes with the intermediate poloidal wavenumbers. The growth of the modes are saturated soon, and the system experiences the energy relaxation in about 1 msec. It should be noted that the linear mode structures are localized in the edge region, whereas the core pressure rapidly falls as the system reaches the relaxed state. The co-existence of the edge perturbation and the core collapse is consistent with the experimental observations, in which a rapid flushing of the core density toward the edge is observed. The lost pressure forms a wider tail in the peripheral region. The core pressure is remarkably reduced at a certain period, while it had withstood the disturbance before it. The most salient feature on this period is the disordering of the magnetic field structure. The system keeps the nested-flux-surface structure well in the beginnings, whereas part of them are abruptly lost at this period. At this period, finger-like structures appear due to the growth of the instabilities. Such a situation can induce a magnetic reconnection between the high-pressure region in the finger and the external low-pressure one, so that a considerable amount of plasma flows out along the reconnected field line. By checking the place where the plasma loss due to this mechanism occurs, such plasma outlets are found to be located mainly on the disordered region. Thus, one can conclude that the core collapse can be caused by the disturbance of the magnetic field.

TH/P9-18 · Study of MHD Stability β Limit in LHD by Hierarchy Integrated Simulation Code

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Abstract: Based on LHD experimental result of MHD stability analysis instead of "conventional" criteria based on ideal linear theoretical prediction, so-called "MHD stability β limit" for helical plasmas is studied by using a hierarchy integrated simulation. The calculations for the LHD configuration, in which the volume averaged β value of 5% is achieved experimentally, shows that the interchange modes limit the pressure gradient only in the periphery region not in the core region. The achievable β value based on the experimental results through the Mercier parameter is not limited by MHD instabilities and it is considered to increase up to the equilibrium β limit.

TH/P9-19 · Theoretical Studies of Equilibrium β Limit in Heliotron Plasmas

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Abstract: In the conventional theory, the MHD equilibrium limit is defined by the Shafranov shift. Achieving the shift to 1/2, the separatrix appears in the plasma core because the vertical field by the equilibrium current cancels the vacuum poloidal field by the external coils. The separatrix limits the increase of the beta. However, it is not still clear that how MHD equilibrium is determined in stellarators. In this study, the MHD equilibrium limit of LHD plasmas is studied using a 3D MHD equilibrium calculation code, HINT2, which does not assume nested flux surfaces. For finite β equilibria, flux surfaces become stochastic in the peripheral region. However, though the axis shifts until the conventional limit, the separatrix to limit the equilibrium does not appear. For high β equilibria, the force balance starts breaking in the stochastic region. To keep the force balance, the pressure gradient in the stochastic region decreases and the fixed profile is reduced. As the result, the volume averaged β saturates. The β value, where the force balance starts breaking inside the stochastic region, is proposed as an index of equilibrium β limit in heliotron plasmas. The generation of the stochastic region in the peripheral region is an intrinsic property in heliotron plasmas. Since the transport increases in the stochastic region, it seems that the stochasticity leads to the β limit. As the first step to study effects of the stochasticity on the transport, the diffusion coefficient of magnetic field lines and the Kolmogorov length are studied in the stochastic region.

TH/P9-20 · A New Matching Scheme for Resistive Wall Mode Analysis

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Abstract: A new matching scheme is proposed in a form that the numerical analysis of linear MHD stability analysis, such as RWMs, is highly tractable. The scheme divides the plasma region into outer regions and inner layers, as in the conventional matching method. However the outer regions do not contain any rational surface as their terminal points; an inner layer contains a rational surface as an interior point. The Newcomb equation is therefore regular in the outer regions. The matching condition is given as a linear equation on the values of the solution at the matching points, which can be easily solved. The proposed scheme facilitates our analyzing the effect of plasma rotation around rational surfaces on RWMs, which gets attention in order to realize a stationary high performance tokamak.

TH/P9-21 · Multi-scale Interaction Among Neoclassical Tearing Mode and Drift Wave Turbulence

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Abstract: The multi-scale interaction among the neoclassical tearing mode (NTM) and the drift wave (DW) is investigated using 4-field reduced MHD equations including neoclassical electron viscosity. It is newly found that the fluctuating bootstrap current term in the Ohm's law gives rise to a new instability for $\delta' <0$. The drift wave coupling stabilizes this instability. Our result indicates not only β and ν^* but also the normalized ion skin depth δ or density profile is an important ingredient to consider the NTM dynamics.

TH/P9-22 · Stability Analysis of NTMs and Self-consistent Dynamic Simulation for the Control of Neoclassical Tearing Mode in KSTAR

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Abstract: Neoclassical tearing modes (NTMs) will be a key instability to be suppressed to achieve high confinement discharges in KSTAR. To simulate temporal behavior of NTM suppression by electron cyclotron current drive (ECCD), a new NTM simulation model is developed and tested for the KSTAR environment. In the model, a non-linear plasma response model is combined with on-line running TORAY and NTM growth model for the evaluation of island widths by using ECCD deposition parameters calculated at each time by TORAY and corresponding time-varying plasma parameters from non-linear plasma model. The NTM control model is developed to use ECCD launch angles as a main control parameter to align ECCD deposition on NTMs and the model gives an actuation signal to TORAY. As a theoretical basis for the NTM suppression simulation under the KSTAR environment, a series of prediction of current drive performance and corresponding NTM stability in KSTAR plasma is carried out. The optimal ECCD launch angles are found to drive the most highly localized ECCD profile on NTMs under the reference equilibrium having 3.5 Tesla toroidal magnetic field. An optimized plasma equilibrium having lower toroidal field around 2.5 Tesla is proven to exhibit highly enhanced current drive performance on both $m/n = 3/2$ and 2/1 NTM resonant surfaces. Under the 3.5 Tesla reference equilibrium, 1 MW continuous ECCD power is proven to be sufficient for 3/2 NTM suppression and about 3 MW is required for 2/1 NTM. In case of the optimized equilibrium having lower toroidal field, the stability analysis shows that even less than 1 MW ECCD power can suppress 2/1 NTM. Based on the theoretical predictions for KSTAR, NTM suppression simulations are successfully performed for the various ECCD conditions. The results show that the developed simulation model is capable of simulating the time varying ECCD deposition very reliably and consistently with the stability predictions.

TH/P9-23 · Magnetohydrodynamic Equilibrium with Flow in Toroidal Plasmas and its Relevance to Internal Transport Barriers

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Abstract: The appearance of transport barriers in plasmas is usually taken to be the result of bifurcations in the transport equations, in which the possible control parameters are the radial electric field in association with shear flow, which may contribute to the reduction or suppression of turbulent fluctuations. Yet, most calculations start from magnetohydrodynamic (MHD) equilibira calculated form the Grad-Shafranov equation, in which no flows are taken into account. The purpose of this work is to study the effect of toroidal and poloidal flows in the quantities related to the build up of internal transport barriers. The force balance equation, gives rise to a modified Grad-Shafranov equation, which must be solved along with a Bernoulli equation. In order to understand the role of input power, it is useful to understand the behaviour of the MHD conserved quantities as well, which are related to the topology of the fields. The electrostatic field potential remains a surface quantity, so far as resistivity is ignored. It is found that, while most of the equilibrium quantities show small variations, the most important changes are observed in the electric field.

TH/P9-24 · Negative Energy Waves and Stability of Rotating Plasmas

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Abstract: For static plasma equilibria, thresholds and structures of unstable magnetohydrodynamic (MHD) modes can be found from the well known energy principle. However, in the case of rotating plasmas typical for present-day fusion devices, energy principle is too “strong” giving sufficient stability condition that is usually far away from necessary one. In this report, we reveal the physics of this discrepancy and suggest new variational approach appropriate to study stability of rotating plasma. We carry out the analysis of the energy of modes and conclude the following: (*i*) energy of unstable $\gamma \neq 0$ eigenmodes is always zero; (*ii*) energy of stable oscillatory modes can be both positive and negative; (*iii*) negative energy waves always correspond to non-symmetric eigenmodes, i.e., to the modes with non-zero wave vector component along the direction of plasma flow; (*iv*) energy of symmetric modes is never negative, and their stability can be studied by use of standard energy principle; (*v*) non-symmetric instabilities in ideal MHD systems with flows are always associated with coupling of positive and negative energy waves. The above statements explain the lack of applicability of energy principle for moving medium, e.g., for a rotating tokamak plasma. More appropriate variational approach has not based on the energy analysis only. We present such an approach suitable to investigate stability of rotating plasma and demonstrate its productivity using numerical examples.

TH/P9-25 · Rotating Wall, the Error-field-induced Torque and the Problem of the Error Field Shielding in Tokamaks

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Abstract: The electromagnetic torque on the toroidal plasma is calculated analytically. The derivations basically follow the line described in [1]. The main difference is that the torque is calculated now with account of the resistive wall rotation. The model is based on the Maxwell equations for the magnetic perturbation and Ohm's law for the currents in the resistive rotating wall. The model assumes a thin wall, which is a standard approach, a linear plasma response to the applied resonant magnetic perturbation, which is a good model for naturally small error fields, a stationary rotation of the wall and, finally, adopts the cylindrical geometry. The obtained formulas are used for analysis of the error field shielding by a liquid metal wall in tokamaks predicted in a similar model [2] with a conclusion there that a flowing liquid metal wall can prevent resonance amplification of the error field by the plasma near its no-wall stability limit. Our theory does not support this concept. Instead, it gives the expressions for the torque without singularities near the no-wall stability limit, with or without the wall rotation. Such singularities have been a reason for the "dramatic difference" of these two cases in the mentioned theory. In contrast, our model does not allow severe peaking of the torque below the RWM stability limit, even if the most pessimistic estimate will be used for the torque. The reason of the differences is explained, the consequences are discussed and the new predictions are compared with available experimental data. Also the experiments are proposed to model the addressed effects on the existing tokamaks with conventional non-rotating walls. The competing theories can therefore be validated against the available data using the developed experimental technique. Finally, the question is discussed whether or not we need a liquid wall for mitigating the error field effects.

[1] Pustovitov V.D., Nucl. Fusion 47 (2007) 1583; [2] Zheng L.-J., Kotschenreuther M., Nucl. Fusion 46 (2006) L9

TH/P9-26 · Modelling Resistive Wall Modes with Self-consistent Inclusion of Drift Kinetic Resonances

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Abstract: The resistive wall mode instability limits the achievable plasma pressure in present and future advanced tokamak scenarios. We investigate the effect of drift kinetic resonance damping, due to plasma thermal particles, on the stability of this mode. A self-consistent toroidal kinetic damping model is developed and incorporated into the MHD code, MARS-F. The new code (MARS-K) allows study of the kinetic effects in both a perturbative and non-perturbative manner. The primary difference between these two approaches is that the former normally uses the no-wall ideal kink eigenmode structure to compute the kinetic energy perturbation, whilst the latter takes into account self-consistently the modification of the RWM eigenfunction by the kinetic effects. The perturbative approach in MARS-K is benchmarked against another, also perturbative calculation using the ideal MHD code MISHKA and the drift-orbit particle-following code HAGIS. The perturbative approach normally predicts a significant modification of the perturbed total energy (fluid + vacuum + wall + drift kinetic) by the kinetic contribution, which in many cases leads to a complete stabilization of the mode. However, the non-perturbative approach in many cases studied results in a partial stabilization, due to the fact that the drift kinetic energy is sensitive to the mode structure, that is modified by the kinetic effects. Therefore, the predicted RWM stability can be significantly different, depending on whether the kinetic terms are included self-consistently. Detailed comparison of the self-consistent modelling with the experimental results (JET and DIII-D) is in progress. Predictive simulations of the RWM stability will be made for the ITER advanced plasmas, using the new damping model.

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TH/P9-27 · The Interaction between Transport and Reconnection Processes

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Abstract: The interaction between turbulent transport and reconnection is important for understanding the threshold for neoclassical tearing modes (NTMs) and the influence of magnetic islands on transport (eg in the pedestal region for control of edge localised modes). In the first part of the paper, we address the impact of magnetic islands on ion temperature gradient (ITG) stability in sheared slab geometry. Using a simplified model we show (a) the islands have a stabilising effect, and (b) the ITG mode structure is modified. In particular, the mode is localised in a flux surface at a position which is shifted relative to the island X-points (a local theory predicts maximum instability at the X-point position). This is interpreted through a WKB theory as a consequence of a complex wavenumber, k_y (y is the direction along the island length). Both the localisation and the stabilisation of the ITG mode would suppress turbulent transport in the vicinity of the island due to this mode. The net effect on the transport would be a competition between the ITG mode suppression and the stochastic magnetic fields induced by overlap of multiple island chains. The second part of the paper addresses the impact of transport processes on the distribution of density and temperature around the magnetic island, and the resulting impact on the polarisation current: a candidate for the threshold of NTMs. Cross-field transport is important near to the island separatrix, so that the plasma density is not a constant on the magnetic flux surfaces in its vicinity. We employ a kinetic model to quantify the impact of the competition between free streaming along magnetic field lines and diffusion across them. Far from the island, the particle distribution function is both constant on the flux surfaces and Maxwellian in velocity space. The separatrix region is matched to this distant region through an intermediate layer where collisions are important. Having derived the form of the distribution function, we are now in the process of deriving the resulting polarisation current to assess the impact on tearing mode stability.

TH/P9-28 · Hollow Current Profile Scenarios for Advanced ITER Operations

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Abstract: Advanced tokamak scenarios are necessary to boost reactor performances. Such schemes usually trigger current holes in existing tokamaks, a particular magnetohydrodynamics equilibrium where no current or pressure gradients exist in the core of the plasma. While such equilibria have large bootstrap fractions, flat pressure profiles may not be optimal for a reactor. However, small modifications of the current profile can lead to diamagnetism. Such configurations have very good confinement properties. Most of the pressure gradient is now balanced by poloidal currents and the toroidal magnetic field. Since current holes already exist experimentally, it is possible to extrapolate such configurations to ITER. In this paper, we consider the properties of diamagnetic current holes, also called "dual equilibria", and demonstrate that fusion throughput can be significantly increased in such scenarios. Their stability is investigated using the DCON code. Plasmas with a β peak of 30% and an average β of 6% are found stable to both fixed and free-boundary modes with toroidal mode numbers $n = 1$, 2 and 3, as well as Mercier and high-n ballooning modes. This is not surprising as these scenarios have normal β close to 3. While α particle confinement is usually degraded in standard current holes, dual equilibria have better confinement due to the presence of a magnetic well. Alpha particle orbits are explored numerically to assess the ability of such configurations to re-establish acceptable confinement. Finally the parameter space scan of dual equilibrium families highlights the best possible scenarios for advanced ITER operations.

TH/P9-29 · Two-Fluid and Resistive Nonlinear Simulations of Tokamak Equilibrium, Stability, and Reconnection

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Abstract: The NIMROD [1] and M3D [2, 3] codes now each have both a resistive MHD and a two-fluid (2F) capability including gyro-viscosity and Hall terms. We describe: (1) a new 3D verification and validation test in the resistive MHD regime using an applied loop voltage in which the two codes are in detailed agreement and they each match the experimental sawtooth period to within 10%, (2) new studies that illuminate the effect of 2F physics on spontaneous rotation in tokamaks and the effect on linear stability, and (3) nonlinear re-connection in regimes of relevance to fusion plasmas with reconnection rates that are in-dependent of the resistivity, and (4) linear two-fluid tearing mode calculations including electron mass that agree with analytic studies over a wide range of parameter regimes.

[1] C. Sovinec, A. H. Glasser, et al., J. Comput. Phys., 195 355 (2004) [2] W. Park, et al., Phys Plasmas 6, 1796 (1999) [3] S. C. Jardin, J. Breslau, N. Ferraro, J. Comput. Phys, 226 (2007) 2146

TH/P9-30 · Critical Toroidal Rotation Profile for Resistive Wall Modes and Control of Magnetic Islands in Tokamaks

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Abstract: The critical toroidal rotation speed required to stabilize the resistive wall modes is an important quantity to tokamak operations. A three-mode model including the effects of toroidal coupling is developed for resistive wall modes. The enhancements of the plasma inertia and the dissipation resulting from the neoclassical effects are included in the layer physics. The toroidal rotation speeds at rational surfaces and the mode frequency appear in the dispersion relation. The toroidal rotation profile is calculated by solving toroidal momentum diffusion equation. These coupled equations determine a critical toroidal rotation profile for stabilizing resistive wall modes. The results indicate that the toroidal rotation profile is more important than the rotation speed at a given radius in determining the stability of the resistive wall modes. The appearance of magnetic islands in tokamaks degrades plasma confinement. It has been known that using radio frequency waves to drive a current inside the island can heal the island. Here, it is proposed to inject pellets at the island O-point to heal the island. Island evolution equation including the bootstrap current driven by the plasma density inside the island resulting from the pellet injection is derived. It is shown that the saturated island width can be reduced or the island can be healed depending on the magnitude of the pellet driven bootstrap current. It is also noted that by controlling the plasma profiles one can also control magnetic islands.

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TH/P9-31 · Interaction of Turbulence and Magnetic Islands

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Abstract: Magnetic islands are important in several connections. They may arise as a consequence of the Neoclassical Tearing Mode (NTM) instability in tokamaks, where they limit the duration of high β discharges. Alternatively, they may arise as a result of currents flowing in external conductors, either applied intentionally as in the ergodic magnetic divertor or as result of errors in the alignment of the coils. Whenever external currents give rise to islands, they bring about qualitative changes in the profiles of plasma velocity and thus on confinement. Previous efforts at predicting the onset and growth of magnetic islands have modeled the effects of turbulence through the use of anomalous transport coefficients. In most cases of interest, however, the widths of the magnetic islands and their phase velocities are comparable to the corresponding scales for plasma turbulence, so that the mutual interaction between islands and turbulence cannot properly be described through the use of macroscopic transport coefficients. We have investigated the role of turbulence for two problems, the effect of the polarization current on the NTM and the effect of diamagnetic drifts on mode penetration. For the NTM problem, we find that the fluctuations exert a drag force on the island that results in a stabilizing slowdown in the propagation velocity. The drag force arises from the emission of drift waves by the island and the flattening of the density by these waves. For the mode penetration problem, we have investigated the influence of polarization and diamagnetic rotation in the error field penetration threshold by comparing the results of electromagnetic simulations using a two-fluid model with theory. For moderate rotation velocities drift-waves are excited and the solution becomes delocalized. We find that the existing linear theories that neglect the drift wave radiation effect underestimate the penetration threshold.

TH/P9-32 · Gyrokinetic Theory for Kinetic Analysis of Resistive Wall Modes in ITER

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Abstract: Resistive wall mode (RWM) stability is a major concern for ITER. It has been shown previously by several authors that the kinetic and Alfvén resonances can play a significant role in stabilizing RWMs. However, a fully kinetic analysis of the RWMs is a challenging issue. First, as shown in Ref. [1], the ordering in the conventional gyrokinetic equation is inconsistent, causing the MHD limit to be lost. Second, the coupling of Alfvén continuum damping requires high-resolution computation of the mode near the singular surfaces, which is difficult to achieve with usual non-adaptive codes. Our current effort is aimed at resolving these two difficulties. To begin with, we present our newly derived gyrokinetic theory [1]. Then, we describe our numerical work to implement our newly developed gyrokinetic theory by extending our existing AEGIS code to AEGIS-K. Since the basic AEGIS formalism is based on the adaptive numerical scheme, with our new gyrokinetic theory implemented we are able to analyze RWM stability in ITER fully kinetically. To understand how our analysis differs from other treatments, other than our use of the adaptive numerical scheme, we summarize the salient features of our underlying kinetic equation:(1) the $\boldsymbol{J_0} \times \boldsymbol{\delta B}$ effect in the perpendicular momentum equation is retained self-consistently; (2) additional source terms in the gyrophase-averaged gyrokinetic equation are recovered, so that the MHD parallel equation of motion can be retrieved in the proper limit; and (3) additional FLR effects are recovered. The success in recovering full MHD with our newly derived gyrokinetic theory now allows the possibility to study RWMs in a self-consistent kinetic manner. Numerical results for RWM stability in ITER will be presented.

[1] L.J. Zheng, M. T. Kotschenreuther, and J. W. Van Dam, Phys. Plasmas 14 (2007) 072505.

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