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Book of Abstracts

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Overview

OV/1-1 · Overview of JT-60U Progress towards Steady-state Advanced Tokamak

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Abstract: Toward realization of an advanced tokamak (AT) fusion reactor, key issues are sustainment of high confinement with high normalized beta ($\beta_{\rm N}$), high bootstrap current fraction ($f_{\rm BS}$) to the total plasma current ($I_{\rm p}$) and heat/particle handling to be compatible with the divertor. Recent JT-60U experiments have been focused on sustainment of AT relevant plasmas, and sustainment of $\beta_{\rm N} \sim 3$ and $f_{\rm BS} \sim 75\%$ for longer than the current relaxation time ($\tau_{\rm R}$) are demonstrated. A standard ELMy H-mode plasma has also been extended into a long time scale. For this purpose, the JT-60U system is modified so as to perform a discharge up to 65 s with 30 s high power heating by positive ion source NB (P-NB). Although the power is limited, $\beta_{\rm N} \geq 1.9$ is sustained for 24s. Change in wall recycling in such a longer time scale has been unveiled. In this paper, recent JT-60U experimental results with the emphasis on evolution and saturation of plasma characteristics in a long time scale towards steady state advanced tokamak are presented.

OV/1-2 · Overview of JET results

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Abstract: The JET facilities are extensively used in preparation of ITER by research teams from 16 European countries and associated collaborators from China, Japan, the Russia Federation and USA. Since October 2002, JET operated for more than 230 days, including 170 days of scientific experimentation in 8 Campaigns. Significant progress has been made in developing plasma scenarios in a range of parameters and configurations as close as possible to ITER. Transient phenomena, such as ELMs and disruptions, have been studied to improve the scaling to ITER and develop mitigation strategies. One highlight is the Trace Tritium Experiment conducted during a five-week campaign in October 2003: unprecedented experimental data was collected on particle transport in all the plasma configurations presently foreseen for ITER. Significant advances were achieved on fast particle physics and diagnostics for future burning plasma experiments. Progress achieved was made possible by several technical developments, including i) Real-Time Control systems, ii) increased plasma shaping capability up to ITER-like triangularities iii) upgraded neutral beam power and iv) upgrades of several diagnostics for core and edge.

OV/1-3 · Development of Burning Plasma and Advanced Scenarios in the DIII-D Tokamak

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Abstract: The DIII-D research program emphasizes integrated scenarios for burning plasma experiments and investigation of elements in those scenarios that are critical for the success of proposed devices such as ITER. Stationary discharges have been developed which exceed the ITER baseline design figure of merit by 50% for durations much longer than the current relaxation time. Following the initial phase of burning plasma experiments, steady- state operation of a tokamak is attractive. DIII-D has demonstrated the first fully noninductive discharges that project to high fusion gain in an ITER-sized tokamak. This requires simultaneously high beta, high confinement, and high bootstrap fraction, plus the capability in DIII-D to drive current where required to sustain a stable q profile by electron cyclotron current drive. A vigorous program of basic physics studies is maintained in the areas of stability, transport, plasma boundary, and current drive physics. New results in areas such as dynamics of the sawtooth instability, the absence of a critical gradient in electron energy transport, and expulsion of impurities by ELMs will be presented.
*Work supported by U.S. DOE under DE-FC02-04ER54698.

OV/1-4 · Confinement and MHD stability in the Large Helical Device

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Abstract: The Large helical Device (LHD) is a heliotron device with l=2 and m=10 continuous helical coils with a major radius of 3.5-3.9 m, a minor radius of 0.6 m. There has been significant progress in extending plasma operational regime in various plasma parameters by neutral beam injection (NBI) and electron cyclotron heating (ECH). The electron and ion temperature have reached up to 10 keV in the collisionless regime and the maximum electron density, the volume averaged beta value and stored energy are $2.2 \times 10^{20} m^{-3}$, 4.1% and 1.3 MJ, respectively. In the last two years, intensive study of the

MHD stability to access the high beta regime and healing of the magnetic island in comparison with the neoclassical tearing mode in tokamaks has been conducted. Local Island Divertor (LID) experiments have been also done to control the edge plasma aimed at confinement improvement. The high ion temperature plasma was obtained by adding impurities to the plasma to keep the power deposition to the ions reasonably high even at very low density. By injecting 72kW of ECH power, the plasma was sustained for 756 second without serious problems of impurities or recycling.

OV/1-5 · Overview of ASDEX Upgrade Results

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Abstract: During the last two years ASDEX Upgrade has made substantial progress towards attractive operating scenarios for ITER and the understanding of the underlying physics. This was made possible by its flexible heating system (on and off-axis NBI, ICRH, ECRH/ECCD), improved diagnostics, and extensions of its operating range (flat top extension, shaping capability enhancement). Highlights of the progress concern: understanding of the observed particle, energy and impurity transport with a consistent model, development of ELM control schemes (e.g., by pellet triggering), successful operation with increased tungsten surface coverage (about 65 %), tools for active control of core and edge impurity transport, control of core MHD instabilities (sawteeth, NTMs), and the extension of operational space for the improved performance "Hybrid Scenario".

OV/2-1 · Overview of Zonal Flow Physics

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Abstract: Zonal flows, by which we mean azimuthally symmetric bend-like shear flows, are ubiquitous phenomena in nature and the laboratory. It is now widely recognized that zonal flows are a key constituent in virtually all cases and regimes of drift wave turbulence, indeed, so much so that this classic problem is now frequently referred to as "drift wave-zonal flow turbulence". In this theory overview, we present new viewpoints and unifying concepts which facilitate understanding of zonal flow physics, via theory, computation and their confrontation with the results of laboratory experiment. Special emphasis is placed on identifying avenues for further progress.

OV/2-2 · Steady-state operation of Tokamaks: key physics and technology developments on Tore Supra.

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Abstract: The Tore Supra tokamak routinely addresses the physics and technology of very long duration discharges. Progresses in RT controls have allowed establishing new world records in fully non-inductive 6 min. discharges (ITER discharge length in standard scenarios) with a coupled energy of 1 GJ. Such capabilities allow a thorough investigation of new and exciting physics at vanishing loop voltage in a situation where the resistive current diffusion has fully taken place, and where the plasma facing components (PFC) work at steady-state temperature. Synergy effects between electron cyclotron and lower hybrid current drive have been unambiguously demonstrated with prospects toward an improved current profile control. Turbulent particle transport has been clearly observed and is shown to be consistent with micro stability analyses. Gas balance on consecutive long discharges exhibits steady-state hydrogen retention. All of them display the same fuelling dynamic with three different time constants, suggesting various mechanisms of retention. Such observations are of direct relevance to ITER and benefit from a strong effort in modelling (predator/prey modelling of transport/current profile coupling observed at 0 loop voltage...) and progress in long pulse technology for ITER (plasma test of a load resilient ICRH antenna, PFC development and assessment in plasma device and high heat flux test bed...).

OV/2-3 · Progress Towards High Performance Plasmas in the National Spherical Torus Experiment (NSTX)

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Abstract: The objectives of the National Spherical Torus Experiment (NSTX) are to understand the physics of low aspect ratio toroidal confinement and to produce high performance plasmas in this geometry. Research over the past two years has indicated the importance of toroidal flow shear in achieving high confinement and high beta plasmas. The rapid rotation observed in NSTX NBI discharges affects both the equilibrium and stability properties of these plasmas. The highest beta plasmas (over 35%, and normalized betas up to 6.9) extend beyond the ideal no-wall even in the presence of 1/1 MHD activity. Results from the M3D code indicate that the toroidal flow shear reduces the linear growth rates of the internal kink by up to a factor of 3. H-mode plasmas on average have higher thermal confinement times than do L-modes, reaching values of ∼30% over the ITER98pby,2 scaling. Electrons clearly dominate transport while the ion thermal diffusivity is at or just above the NCLASS neoclassical value. A regime of improved core electron confinement is found to be associated with a region of low or negative magnetic shear, consistent with gyrokinetic calculations. Reproducible H-mode access was obtained with high field side fueling which is associated with higher edge rotation, and this is consistent with a recent theory considering the effect of charge-exchange on momentum loss. L-H transitions have also been shown to be associated with loss of fast ions by bounce fishbones, also consistent with a modification of edge Er. During High Harmonic Fast Wave heating experiments, a coupling between launched RF waves and edge ions has been observed with a hot (∼0.5 keV) poloidally rotating (50 km/sec) thermal ion component developing during the period of HHFW heating, possibly due to parametric decay of the waves. The NSTX research during the upcoming campaign will focus on establishing physics understanding in individual topical areas as well as on integrating the best features of these components. Integration efforts up to now have produced a plasma with confinement time exceeding the ITER98pby,2 scaling, normalized beta of 6.5, over 50% non-inductive current and with a current flattop exceeding the current diffusion time. Scenario development calculations underscore the importance of integrating these techniques to achieve the ultimate research goals of NSTX.

OV/2-4 · Overview of MAST results

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Abstract: Activities on the MAST tokamak are focused on the core goals of improving first principles understanding of transport, stability and edge physics and investigating technological solutions to heating, current drive and plasma exhaust, both for ITER and future spherical tokamak (ST) devices. The study of transients is a key topic for MAST. Observations during the early phase of ELMs are consistent with filamentary structures expanding beyond the LFS separatrix at a radial velocity of 1 km/s. Observations during the thermal quench in locked mode disruptions reveal a phase where the divertor target heat flux width is significantly broadened and another where up to half the core thermal energy is released with no broadening. Efforts to dynamically control both static and transient power loads in MAST are being tested. In particular, toroidally asymmetric active (external power supply) and passive (plasma induced) electrical biasing of divertor components are being employed to induce broadening of the divertor wetted area and rapid sweeping of the strike-points. Adding confinement data from quasi-steady state MAST H-mode plasmas to the international confinement database suggests a stronger aspect ratio dependence than indicated by the IPBH98y2 scaling and provides support for a collisionality dependence absent from this scaling. Edge density profiles at the high and low field side provide support for an analytical model of the density pedestal width dependent on the neutral penetration depth. Regimes with strong ITBs are now also routinely accessed in MAST. An ion ITB is typically formed with co-NBI, whilst an electron ITB is formed with counter-NBI. A strongly peaked density profile and high toroidal velocity accompanies the electron ITB. Transport modelling of core and edge plasmas is undertaken for all plasma regimes. B2SOLPS5.0 L-mode modelling suggests that the neo-classical ion heat flux at the edge is comparable with the anomalous radial ion heat flux, while neo-classical contributions to the particle and electron heat fluxes are negligible. TRANSP analysis indicates that heat diffusivities are close to the ion neo-classical level over a significant region of the plasma in NBI H-mode plasmas and in the region of both electron and ion ITBs. This work was jointly funded by the UK Engineering and Physical Sciences Research Council and EURATOM.

OV/2-5 · Overview of Alcator C-Mod Research Program

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Abstract: Research on the Alcator C-Mod tokamak has emphasized RF heating, self- generated flows, momentum transport, scrape-off layer turbulence and transport and the physics of transport barrier transitions, stability and control. Auxiliary heating is by ICRF alone, up to 6 MW in the H minority regime, enabling experiments in which power, particle, momentum and current sources are decoupled. As a high plasma density device ion and electrons are strongly coupled in most regimes. Further, as a high power density machine with all metallic walls, C-Mod is prototypical of reactor scale devices with respect to plasma wall interactions as well. Studies of rotation and momentum transport in source-free plasmas were extended through analysis of the transient response of rotation profiles. Momentum was clearly observed to diffuse and convect from the plasma edge to the core. The correlation of self-generated plasma flows and topology has led to a novel hypothesis for the dependence of the H-mode power threshold on the grad-B drift direction in which the SOL flows play a dominant role. Research into internal transport barriers has focused on control of the barrier strength and location, while studies of edge barriers have emphasized regimes with small or no ELMS.

OV/3-1 · Recent Advances in Indirect Drive ICF Target Physics

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Abstract: The Lawrence Livermore National Laboratory's Inertial Confinement Fusion Program, in collaboration with Los Alamos National Laboratory (LANL), Sandia National Laboratory (SNL), the Laboratory for Laser Energetics (LLE), General Atomics (GA), and the Commissariat a l'Energie Atomique (CEA), is working to refine ignition target designs, develop and activate experimental capabilities, and fabricate and test cryogenic targets, in preparation for ignition experiments on NIF. This paper will review NIF progress, the first NIF experiments, and summarize advances in these areas. This work was performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract No. W-7405-Eng-48, and by Los Alamos National Laboratory under Contract No. W-7405-Eng-36

OV/3-2 · Laser Fusion research with GEKKO XII and PW laser system at Osaka

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Abstract: Fast heating of the compressed core plasma up to 500eV has been successfully demonstrated by injecting a 400J/0.6ps PW laser into a compressed CD shell through a hollow gold cone. According to this result, we started the FIREX (Fast Ignition Realization Experiment) project toward demonstrating the ignition of the highly compressed DT fuel by the high energy PW laser heating. A new heating laser LFEX (Laser for Fast Ignition Experiment) is under construction. In this paper the progresses in the experimental studies on scientific issues related to fast ignition and the integrated code development toward the FIREX will be reported. Research results on implosion hydrodynamics, Rayleigh-Taylor instability growth and a new stabilization mechanism are also reported.

OV/3-3 · Direct-Drive Inertial Confinement Fusion Research at the Laboratory for Laser Energetics: Charting the Path to Thermonuclear Ignition

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Abstract: Direct-drive inertial confinement fusion (ICF) offers the potential for higher- gain than x-ray drive and is a leading candidate for an inertial fusion energy power plant. LLE's direct-drive ICF ignition target designs for the National Ignition Facility (NIF) are based on hot-spot ignition; a cryogenic target enclosed by a thin plastic shell with a spherical DT-ice layer or a DT-ice layer within a foam matrix is directly irradiated with $\sim$1.5 MJ of laser energy. Cryogenic and plastic/foam (surrogate-cryogenic) targets that are hydrodynamically scaled from ignition target designs have been imploded on the OMEGA 60-beam, 30-kJ, UV laser system to investigate energy coupling, hydrodynamic instabilities, and implosion symmetry. Deuterium-ice-layer finishes approaching the 1-mm NIF requirement have been demonstrated. A high-energy petawatt capability is being built at LLE next to OMEGA. The OMEGA EP (extended performance) laser will add two short- pulse, 2- to 3-PW, 2.6-kJ beams to the OMEGA laser system to

study fast-ignition physics. Prospects for direct-drive ignition on the NIF while it is configured for x-ray-drive irradiation are extremely favorable with polar direct drive (PDD). This work was supported by the U.S. Department of Energy Office of Inertial Confinement Fusion under Cooperative Agreement No. DE-FC03-92SF19460, the University of Rochester, and the New York State Energy Research and Development Authority. The support of DOE does not constitute an endorsement by DOE of the views expressed in this article.

OV/3-4 · Acceleration Technology and Power Plant Design for Fast Ignition Heavy Ion Inertial Fusion Energy

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Abstract: This talk gives an update on the progress in Heavy Ion Beam IFE experimental and theoretical activities conducted under the auspices of the Ministry of Atomic Energy of Russian Federation under contract No. 6.25.19.19.03/996. The considerations of heavy ion fusion power plant concept are based on the fast ignition principle for fusion targets [1]. The cylindrical target is irradiated subsequently by a hollow beam in compression phase and by powerful ignition beam for initiation of the burning phase. The ignition is provided by the high energy 100 GeV Pt ions of different masses accelerated in RF-linac. The efficiency of the driver is taken ∼25%. The main beam delivers ∼5 MJ energy and the ignition beam ∼0.4 MJ to the target. Cylindrical DT filled target provides ∼600 MJ fusion yield, of which 180 MJ appears in X-rays and ionized debris and 420 MJ in neutrons. The repetition rate is taken as 2 Hz per reactor chamber. The first wall of the reactor chamber employs "liquid wall" approach, particularly the wetted porous design. The lithium-lead eutectic is used as a coolant, with initial surface temperature of 550oC. Computation of neutronics results in blanket energy deposition with maximum density of the order of 10E8 J/m3. The heat conversion system consisting of three coolant loops provides the net efficiency of the power plant of ∼35%. The Heavy Ion IFE experimental program is focused on a major upgrade of the ITEP accelerator complex for acceleration and accumulation of high current beams - the TeraWatt Accumulator project (ITEP-TWAC). Commissioning of the whole acceleration/accumulation beam gymnastic scheme with stacking of ∼10E10 C6+ and fast extraction to the experimental area has been done in 2003. The ion bunch is being compressed from 1 mks to ∼ 170 ns and focused down to a spot ∼ 1 mm. Current experiment efforts are aiming at measurements of ionization degree, charge state distribution, conductivity, plasma pressure, ion and electron plasma temperatures, and density with temporal resolution on the order of 10 - 100 ns.

. S.Medin et al. Fusion Science and Technology (2003) v.43, 3, 437-446.

OV/3-5Ra · Progress on Z-Pinch Inertial Fusion Energy

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Abstract: The goal of z-pinch IFE is to extend the single-shot z-pinch ICF results on Z to a repetitive-shot z-pinch power plant concept for energy. Z produces up to 1.8 MJ of x-rays at powers as high as 230 TW. Results of recent target experiments on Z are discussed, including capsule implosion convergence ratios of 14-21 with a double-pinch driven target, and DD neutron yields up to 8×10^{10} with a dynamic hohlraum target. For z-pinch IFE, a power plant concept is discussed that uses high-yield targets (3 GJ) with a low rep-rate per chamber (0.1 Hz). The concept includes a repetitive driver at 0.1 Hz, a Recyclable Transmission Line (RTL) to connect the driver to the target, high-yield targets, and a thick-liquid wall chamber. Recent results on z-pinch IFE research on RTLs, repetitive drivers, shock mitigation, full RTL cycle planned experiments, high-yield IFE targets, and z-pinch power plant technologies are discussed.

OV/3-5Rb · Wire Array Z pinch precursors, implosions and stagnation

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Abstract: Recent measurements of the precursor ablation velocity show that the velocity is reduced significantly when the gap to core size ratio is less than π, implying a higher mass ablation ate and a change in the precursor density profile at the time of implosion. This implies an optimal inter-wire gap for the shortest rise time of the x-ray pulse. 2-D kinetic modelling of the precursor plasma shows how long mean-free-path ions can lead to the accumulation of a central dense, radiating column. Precursor interaction with a foam cylinder is also modelled with material mixing allowed. Two effects are being

studied that can increase the final x-ray radiation. One is the effect of later implosion of trailing mass, diagnosed by both laser probing and x-radiography, and modelled by 2-D simulations. The second is ion viscous heating arising from saturated non-linear, short wavelength MHD m=0 instabilities. Experimental evidence at 20MA on the z-accelerator shows Doppler broadened spectra at stagnation with ion temperature in the 100-300 keV range. Neutron time of flight measurements from deuterated plastic targets also show high in temperatures.

OV/3-5Rc · The Research of Radiating Z Pinches for the Purposes of ICF

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Abstract: The results of experiences on current implosion of multiwire arrays on installation "ANGARA-5-1" are given at a current up to 5 MA with the purpose of study of the phenomena responsible for formation powerful radiating Z pinches and parameters of radiating pulse of x-ray emission with output power up to 5 TW. The application of active x-ray probing and data passive radiography of Z-pinch plasma together with tiny magnetic probes provides a reliable basis for recording as details of fine spatial - temporary structures of the pinch formation, and to receive the quantitative information on value and distribution of a magnetic field. The data on distribution of ion density and magnetic fields are used for check of results of 2-D numerical RMHD simulation paying in view the pattern of the phenomenon prolonged plasma production in (r-z) and (r-j) coordinates. The defining influence of the phenomena of spatially non-uniform, prolonged in time plasma production on dynamics of formation and parameters of x-ray emission of radiating Z pinch is reconfirmed. Are construed the basic principles of construction of the powerful generator of a current of "Baikal" developed for application as the driver for experiments on ignition of targets for ICF with use radiating Z pinches with a current at a level 50 MA. The stages of creation and results of research of installation "MOL" staging in real scale the basic elements of the module of the generator of "Baikal" are debated.

OV/4-1 · Overview of Transport, Fast Particle and Heating and Current Drive Physics using Tritium in JET plasmas

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Abstract: The JET Trace Tritium Experimental (TTE) campaign used trace tritium (T) plasmas (nT/nD<5%) to investigate thermal fuel-ion transport, fast particle dynamics and heating and current drive physics. Precisely timed tritium gas puffs, and short pulse (<500 ms) Tritium Neutral Beam Injection (TNBI) were used. Transport coefficients were derived by fitting 14 MeV and 2.5 MeV neutron profiles' evolution. Rho*, nu*, beta and q95 were varied in pairs of sawtoothing ELMy H-Mode discharges, comparing fuel-particle and energy transport coefficient scalings. Particle transport data are consistent with gyro-Bohm scaling, similar to energy transport. Particle confinement increases with nu* and declines with beta, but energy confinement degrades weakly with nu* and is independent of beta. High edge diffusion during ELMs clearly affects the neutron emission. TTE experiments compared three fuelling methods in otherwise similar Internal Transport Barrier (ITB) plasmas: T gas-puffs; T wall-recycling; and TNBI. T gas-puff data showed reduced triton diffusion at all ITB locations. Thermalised TNBI ions deposited inside the ITB radius, demonstrated ITB effects on central particle confinement. In hybrid plasmas (qmin$\sim$1, low positive shear, no sawteeth), T particle confinement time (τ_{pT}*) improved by $\sim$50% as triangularity was varied from 0.2-0.46 at constant energy confinement ($H\beta_N/q95^2 \sim 0.42$). At low delta, τ_{pT}* scaled with Ip. Fast-ion confinement in Current-Hole (CH) plasmas, with minimal central toroidal current density was tested by injection of TNBI into JET CH plasmas. Gamma-rays from nuclear reactions between fusion alphas (generated by TNBI) and beryllium impurities were measured in majority-deuterium plasmas after TNBI turn-off. Gamma-ray emission rate decay times measured the fast fusion-alpha population evolution. These decay times for JET plasmas are consistent with classical alpha slowing down times for high plasma currents (Ip >2MA) and monotonic q-profiles. In CH discharges the gamma-ray emission decay times are a factor 5 below classical values based on total Ip, indicating alpha confinement degrades due to orbit losses as predicted by a 3-D Fokker Planck numerical code. In TTE, ICRF at 23 MHz was coupled to T-minority ions, producing energetic tritons with tail temperatures 80-120keV, derived from neutron energy spectra.

OV/4-2 · Overview of Results in the MST Reversed Field Pinch Experiment

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Abstract: Confinement in the reversed field pinch (RFP) has been shown to increase strongly with current profile control. Auxiliary current drive and heating is now critical to RFP research. In the MST RFP, progress has accrued in several techniques, including lower hybrid and electron Bernstein wave injection, ac helicity injection current drive, pellet injection, and neutral beam injection. With inductive current profile control, MST can operate in two regimes: the standard regime with a naturally occurring current density profile, robust reconnection and magnetic self-organization; and the improved confinement regime with strong reduction in reconnection, self-organization, and transport. New results in standard plasmas include observation of a strong two-fluid Hall effect in reconnection and dynamo, determination that the m = 0 edge resonant mode is nonlinearly driven, and determination that tearing modes lock to the wall via eddy current in the shell. New results in improved confinement plasmas include observation that such plasmas are essentially dynamo-free, contain several isolated magnetic islands (as opposed to a stochastic field), and contain reduced high frequency turbulence.

OV/4-3 · Overview of TJ-II experiments

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Abstract: Global confinement studies in the TJ-II stellarator are in agreement with ISS95 predictions for metal wall conditions, but deviate for bornized wall conditions. These deviations maybe related to improved confinement regimes and/or plasma-wall interactions. An improved confinement regime was induced by limiter biasing, leading to an increase in the edge velocity shear and turbulent velocity fluctuations. Two different global confinement regimes were found to be associated with different ion confinement times, compatible with different ambipolar electric fields. Model calculations suggest non-local transport may be fundamental to its understanding. Particle and impurity transport was found to require a pinch term. However, edge cooling experiments provided evidence for non-diffusive transport mechanisms. A model for such non-local non-diffusive transport was developed, based on the use of Levy distributions. The influence of the magnetic topology on plasma profiles and turbulence was investigated. Internal barriers were formed depending on the relative position of ECH power deposition and rationals. Electric fields were measured locally, confirming the relation of the ambipolar electric field and the e-ITB's. Island divertor discharges were created and studied. First Neutral Beam Injection heated plasmas are reported, and the density limit was studied for these discharges.

OV/4-4 · Summary of Experimental Core Turbulence Characteristics in OH and ECRH T-10 Tokamak Plasmas.

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Abstract: This report summarizes the results of experimental turbulence investigation, carried out at T-10 more then ten years. The turbulence characteristics were investigated by means of correlation reflectometry (CR), multipin Langmuir probes (MLP) and Heavy Ion Beam Probe diagnostics (HIBP). The OH and ECRH discharges show the distinct transition from the core turbulence, having complex spectral structure, to the unstructured one at the periphery. The core turbulence includes the "Broad Band" (BB), "Quasi-Coherent" (QC) features, arising due to the excitation of rational surfaces with high poloidal m-numbers, "Low Frequency" near zero, and the special oscillations at 20 – 30 kHz. All experimentally measured properties of LF and HF QC are in good agreement with behavior of the linear increments of ITG/DTEM instabilities. Significant local decrease of the turbulence amplitude and coherency were observed near q=1 radius 5 – 15 ms. after ECRH switch off. These evidenced the ITB formation due to turbulence stabilization near rational q=1 surface with low magnetic shear.

OV/4-5 · Progress in the understanding and the performance of ECH and plasma shaping on TCV

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Abstract: Powerful ECH with an adaptable launching geometry and plasma shaping capability are exploited on TCV to create and control high performance regimes, with 3MW of 2nd and 1.5MW of 3rd harmonic (X3) and real time optimisation of the absorption by mirror feedback. Full X3 absorption with launching parallel to the resonant surface was obtained. Electron temperature profile stiffness was measured as a function of the shape up to large normalised temperature gradients and confirmed that the diffusivity is lower at negative triangularity and high elongation. The link between shear and transport was verified by interleaved modulation of co and counter ECCD. ECCD efficiency and fast electron generation and transport measurements demonstrate the role of transport on the driven current profile. Stationary electron ITBs were created with a confinement enhancement over the RLW scaling of 5. The role of the current profile was clarified by improving or destroying the barrier with a small induced electric field. The MHD activity indicates that NTM and global kinks limit the performance. Sawtooth stabilisation at high positive or negative triangularity is observed and modelled by internal kinks.

OV/4-6 · Overview of the FTU Results

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Abstract: Main results from FTU (R=0.93m, a=0.3m, Bt up to 8T, Ip up to 1.6 MA) are presented. Advanced Tokamak (AT) scenarios with improved confinement at ITER relevant magnetic field and densities have been achieved with combined LHCD (8GHz) and ECRF (140 GHz) with mild negative magnetic shear: Electron temperature in excess of 6 keV at ITER relevant densities, H89 up to 1.6 have been achieved. Analysis on the ion transport has shown that the 40% increase in ion temperature and the substantial increase of neutron yield is a result of electron-ion transfer. Synergy between LHCD and ECRF is promising to extend further the AT studies at higher Bt. The first experimental test of an LHCD launcher compatible with ITER demands, the passive active multijunction (PAM), has been successfully tested with an equivalent power density 1.5 times higher than the value matching the ITER request showing similar current drive efficiencies than a conventional launcher. Some supporting Physics experiments such as transport, ablation studies for vertical pellet injection (0.5km/s at mid radius on the high field side), Ion Bernstein Wave experiments (433MHz) and high frequency MHD spectroscopy are also described.

OV/5-1Ra · Recent Advances in the HL-2A Tokamak Experiments

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Abstract: The first experiment campaign was conducted on HL-2A tokamak[1] in 2003 after its first plasma was obtained at the end of 2002. Progresses in many aspects have been made, especially in the divertor discharge and feedback control of plasma configuration. After the effective wall conditioning, the device is operated with good discharge reproducibility at Ip = 168 kA, Bt = 1.4 T and discharge duration $T_d = 920$ ms in the first physics campaign. The feedback control system of plasma current and horizontal position has been realized successfully. With the feedback control system, single null (SN) divertor configuration has been obtained. The HL-2A SN divertor configuration is simulated with the MHD equilibrium code SWEQU. Two modes of power supply for the multipole coils and the radial field (RF) coils have been used for HL-2A SN divertor discharges according to the code results. When the divertor configuration is formed, the impurity radiation in main plasma decreases obviously and the radiation power near X-point rapidly rises.

OV/5-1Rb · Overview of the last HT-7 experiments

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Abstract: The operational scenarios of H-mode, negative reversed shear and high l_i were investigated for quasi steady-state high performance. HT-7 has produced a variety of discharges with the normalized performance $\beta_N H_{89} > 1 \sim 3$ for a duration of several hundred energy confinement times in the weak negative reversed shear. The maximum fraction of the non-inductive current was > 90% Ip. Increase of total injection power up to 1 MW did not degrade the plasma confinement in the RS operational scenario.

An ITB was formed at the footprint of the minimum q. Plasma performance and duration was limited by two kinds of MHD instabilities and the recycling. The high l_i plasma was created by fast plasma current ramp down at a rate of –0.8MA/s and sustained by central LHCD and IBW heating for >1 s with the strongly peaked electron temperature profile. The highest central electron temperature up to 4.5 keV has been obtained. A stationary improved confinement has been observed in the high l_i plasma. The reproducible long pulse discharge with $Te(0) \sim 0.5keV$ and central electron density $> 0.8 \times 10^{19}m^{-3}$ has been obtained with a duration of > 60 seconds. A new operation mode without ohmic current was demonstrated and sustained for 28s. The main limitation for the pulse length was due to the recycling, which caused un-controllable rise of the electron density. The MHD stabilization by LHW and IBW power modulation were successfully demonstrated. The poloidal large-scale $E \times B$ time-varying flows, electrostatic and magnetic Reynolds stress were directly measured in the boundary plasma of the HT-7 tokamak.

OV/5-2 · overview of steady-state tokamak operation in TRIAM-1M

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Abstract: With the goal of addressing the critical issues of steady-state tokamak operation (SSTO) of future fusion devices, experiments aiming at 'day long operation at high performance' have been carried out in a limiter configuration of TRIAM-1M plasma ($R_0 = 0.84m$, $a \sim 0.12m$ and $B = 6 - 7T$). The record value of the discharge duration was updated to 5 h and 16 min by localizing the particle recycling profile at a limiter with high heat removal capability and reducing the surface temperatures of plasma facing components PFCs. Real time measurements of metal deposition on PFCs and ex-situ study of hydrogen co-deposition have been progressed in order to understand temporal change in wall pumping rate. Formation and sustainment of an internal transport barrier ITB in enhanced current drive mode (ECD) has been investigated by controlling the lower hybrid driven current profile by changing the phase spectrum. An ITER relevant remote steering antenna for electron cyclotron wave ECW injection was installed and the synergetic effects of ECW and lower hybrid wave LHW on the plasma performance were found. In this paper, recent SSTO experimental results in TRAIM-1M since last IAEA conference are presented.

OV/P4-9 · New developments in JET Neutron, Alpha Particle and Fuel Mixture Diagnostics with potential relevance to ITER.

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Abstract: JET recent experimental programme, with its high power and Trace Tritium Experimental (TTE) campaigns, is progressively more focused on producing a plasma of reactor relevance. Major progress made in several measurement techniques, in particular in the fields of neutron, alpha particle and isotopic composition diagnostics, provides not only essential new physical information but will also contribute to the design of ITER systems. The Magnetic Proton Recoil spectrometer is unique and has been refined recently to provide for the first time an independent and absolute measurement of the total 14 MeV neutron yield, which is in very good agreement with the historical calibration based on the delayed emission of suitable samples. The availability of two cameras, one vertical and one horizontal, is essential to interpret the spatial neutron emissivity in many JET experiments, which quite often show differences between the neutron profiles and the magnetic topology. Another very important line of research involves the improvement of compact detectors like NE213 liquid scintillators and CVD diamond diodes. Since a new technique to generate α particles using 3rd harmonic heating NBI injected ^4He has become available, more attention has been devoted to the confinement and slowing down of the alphas. The method based on x-rays from the $^9Be\ (\alpha, x)^{12}C$ nuclear reaction is now commonly used to determine the spatial distribution of energetic ions in the plasma. During TTE the slowing down of fusion born alphas was measured for the first time with this approach. Classical estimates of α confinement were confirmed but some advanced scenarios with a current hole show significantly reduced slowing down time of the fusion αs. From the 2-D imaging of the γ radiation, the spatial distribution of both the alphas and the fast deuterons (from the $^{12}C(d, x)^{13}C$ nuclear reaction) is now routinely derived together with the evolution of runaway electrons. The Tritium content was measured for the first time during TTE using a Neutral Particle Analyser, explicitly designed for the isotopic composition determination by detecting simultaneously the neutral fluxes of all hydrogen isotopes leaving the plasma. This diagnostic is particularly effective for neutrals born in the external part of the plasma, where it can complement the results of the neutron cameras.

OV/P6-15 · Joint Research Using Small Tokamaks

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Abstract: Small tokamaks have an important role in fusion research. More than 40 small tokamaks are operational. Research on small tokamaks has created a scientific basis for the scaling-up to larger tokamaks. Well-known scientific and engineering schools, which are now determining the main directions of fusion science and technology, have been established through research on small tokamaks. Combined efforts within a network of small and medium size tokamaks will further enhance the contribution of small tokamaks. A new concept of interactive co-ordinated research using small tokamaks in the mainstream fusion science areas, in testing of new diagnostics, materials and technologies as well as in education, training and broadening of the geography of fusion research will be presented. Besides the presentation of the scientific results achieved at the participating laboratories, information will be provided about the organisation of the co-ordinated research project.

EX

Magnetic Confinement Experiments

EX/1-1 · Development on JET of Advanced Tokamak operations for ITER

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Abstract: Recent progress in the study of Advanced Tokamak regime at JET are reported. ITBs with core density close to Greenwald value are obtained with plasma target preformed by opportune timing of pellet injection and LHCD. ITBs start with toroidal rotation 4 times lower than the standard ones. Wide ITBs, R∼3.7m, are formed at ITER relevant triangularity, with ne/nG∼70% and Vl=0.05V that coexist with the H-mode when ELMs are moderated by Ne injection. Full CD is achieved in reversed shear ITBs at 3T/1.8 MA, by using 10MW NBI, 5MW ICRH and 3MW LH. In these conditions ITBs located at R=3.6m, without impurity accumulation and type-III ELMs edge, are sustained for a time close to neo-classical resistive time. These pulses have been extended to the maximum duration allowed by JET subsystems (20s) with the JET record injected energy: E∼330 MJ. Integrated control of pressure and current profiles is an essential feature used in these discharges. Finally ion ITBs are obtained with low torque injection, by ICRH 3He minority heating of ions, on pure LHCD electron ITBs. Hybrid Scenarios are also been developed at ρ^* as low as $\rho^* \sim 4.5 \times 10^{-3}$. The development of this regime with dominant electron heating has also started at JET in the RF dominated campaign at beginning of 2004.

EX/1-2 · 100 Percent Noninductive Operation at High Beta Using Off-axis ECCD

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Abstract: The Advanced Tokamak (AT) program on DIII-D is aimed at developing the scientific basis for steady state, high performance operation in future devices. The key element of the program is to demonstrate sustainment of 100% noninductive current for several seconds at high beta. Guided by integrated modeling, recent experiments using 2.5 MW of off-axis ECCD and 15 MW NBI with $q_{95} = 5$ have sustained ≈100% of the plasma current noninductively for 1 s at high beta ($\beta = 3.6\%$, $\beta_N = 3.4$, above the no-wall limit) with $q_{min} > 1.5$ and good confinement ($H_{89P} = 2.3$). Modeling validated with the experiment indicates that a full noninductive discharge for >3 s (>current replacement time) can be achieved with a longer ECCD pulse and resistive wall mode feedback presently available. These experiments have achieved the parameters required for the ITER $Q = 5$ steady-state scenarios, and the same modeling tools are already being applied to ITER simulations.
* Work supported by US DOE under DE-AC05-00OR22725, DE-FC02-04ER54698, W-7405-ENG-48, and DE-FG02-89ER53297.

EX/1-3 · Steady State High betaN Discharges and Real-Time Control of Current Profile in JT-60U

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Abstract: The period with high β_N has been extended remarkably in JT-60U. $\beta_N = 3$ was sustained for 6.2s and $\beta_N = 2.1$ for 20 s, only interrupted by the limit of neutral beam heating system, with low safety factor (q95=2.2 and 3.2, respectively). In the case of $\beta_N = 2.1$, in particular, the current profile reached a steady state. In this low safety factor regime, the resonant surfaces with q=3/2 and 2/1 were shifted outside the steep pressure gradient layer, stabilizing the neoclassical tearing mode (NTM) above the previous β_N-limit imposed by NTMs. To establish the feedback control of the safety factor profile $q(\rho)$, a real-time $q(\rho)$ control system has been developed using the motional Stark effect (MSE) diagnostic as detector and the lower hybrid current drive (LHCD) as actuator. For the first time, real-time feedback control has been demonstrated based on local pitch angle measurement and control of parallel refractive index of LH waves adjusting the LHCD location.

EX/1-4 · Studies of the Quiescent H-mode regime in ASDEX Upgrade and JET

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Abstract: The "Quiescent H-mode" (QH-mode) regime, originally discovered in the DIII-D tokamak,is studied in ASDEX Upgrade and JET in campaigns with reversed plasma current for neutral beam injection in opposite direction to the plasma current. In ASDEX Upgrade, long stationary discharges without Edge Localised Modes (ELMs) are obtained, exhibiting high central ion temperatures (up to 10 keV), low to medium density (around 40% of the Greenwald density) and, after fresh boronisation, the lowest values of

$Z_{eff} = 2.5$ in QH-mode so far. QH-modes occur with all four neutral beam source geometries. Compared to ELMy H-mode, the plasma density decreases after transition to QH-mode with more tangential and increases with more radial counter-injection. So far, QH-mode has not occured with co-injection. The "Edge Harmonic Oscillation", continuous MHD activity which replaces ELMs, is clearly observed with a fundamental (toroidal mode number n=1) and harmonics up to n=11. ECE, Soft X-ray and reflectometer measurements show that the EHO is always resonant in the edge barrier region, is not poloidally localised, and is not a magnetic island. Reconstruction of the edge current in the pedestal region from magnetic measurements of the pick-up coils near the X-point suggests that the edge current in QH-mode is normally not smaller than in ELMy H-mode. Doppler reflectometry measurements indicate that the radial electrical field and its shear are significantly larger in QH-mode than in ELMy H-mode. The EHO appears in conjunction with a magnetic oscillation at about 10-20 times higher frequency ("High-frequency oscillation", HFO), which occurs in bursts that have a fixed phase relationship with the EHO cycles. In JET, a dedicated experiment was conducted to find QH-mode in plasmas with high clearance to the wall and various combinations of plasma current and toroidal field. ELM-free phases with constant density and radiation level with up to 1.5 s duration are found, mostly in pulses conducted shortly after a fresh evaporation of beryllium onto the vessel wall. A characteristic MHD oscillation, similar to the EHO, with n=1 and up to seven harmonics is observed in magnetic measurements and X-mode reflectometry with a cut-off layer in the H-mode barrier region.

EX/1-5 · Confinement Study of Net-Current Free Toroidal Plasmas Based on the Extended International Stellarator Database

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Abstract: A global database in stellarators has been updated by incorporating new experiments as part of an international collaboration. The present study examines the expanded database involving data from 9 devices for confirmation and examination of new trends in confinement performance of stellarators. A simple regression analysis with the set of parameters of major and minor radii, magnetic field, density, heating power and rotational transform is useful for unified data description but its application is limited to the available data set alone. Acceptance of a systematic difference in different magnetic configurations is a prerequisite for derivation of a useful unified scaling law. A deterministic parameter characterizing the magnetic configuration has not been clarified yet, but certainly involves the details of the helically corrugated magnetic fields, so an enhancement factor on the ISS95 scaling is used for description of the magnetic configuration effect. This renormalization of data results in a scaling expression with gyro-Bohm nature and no clear dependence on beta and collisionality. The underlying physics of the enhancement factor and potential for reactor assessment are discussed.

EX/2-1 · Energy loss for grassy ELMs and effect of plasma rotation on the ELM characteristics in JT-60U

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Abstract: The establishment of the H-mode plasma with a small ELM is an important subject to extend the lifetime of the divertor target plate in ITER. In JT-60U, an attractive operational mode in the low collisionality region less than 0.15, such as the grassy ELM regime, has been found. In this paper, the following new findings in ELM characteristics are reported. In the grassy ELM, the frequency of periodic spike in divertor D-alpha signal is proportional to the power through the separatrix. The smaller energy loss at each ELM can be explained by the narrow affected area in T(e) profile and by the higher ELM frequency. A good capability to change the plasma rotation actively using a combination of the tangential and perpendicular NBIs in JT-60U has clarified the effect of the counter rotation on the ELM characteristics systematically. In both grassy ELM and type I ELM regimes, sufficiently large counter rotation lead to no-ELM phase. In less counter rotation case for grassy ELM regime, on the other hand, usual type I ELM appeared even in the same plasma shape, edge pressure and poloidal beta as the target plasma.

EX/2-2 · H-mode pedestal, ELM, and power threshold studies in NSTX

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Abstract: H-mode pedestal, ELM, and power threshold studies in NSTX H-mode operation plays a crucial role in NSTX operational scenarios, allowing access to higher beta limits due to reduced plasma pressure peaking and to long pulse operation due to high bootstrap current fraction. Here new results will be presented in the areas of L-H turbulence and power threshold studies, edge localized modes (ELMs), and H-mode pedestal physics. We find via a gas puff imaging diagnostic that ohmic and L-mode discharges have turbulent structures in the edge emission, and that H-modes are generally more quiescent. In addition, many high performance discharges in NSTX have tiny ELMs, which have some differences as compared with ELM types in the published literature, and the regime itself is distinct from the QH-mode on DIII-D and the EDA mode in Alcator C-MOD. ELMS of several other types (as observed in conventional aspect ratio tokamaks) are often observed: 1) type I ELMs, 2) type III ELMs, and 3) giant ELMs which can reduce stored energy by 30% in certain conditions. High field side gas fueling is still required for reliable H-mode access, and leads to a lower apparent power threshold. Edge rotation measurements indicate that the edge toroidal rotation is higher with high-field side fueling, consistent with a recent neoclassical theory revision. The H-mode pedestal typically contains between 25-33% of the total stored energy, and that the NSTX pedestal energy agrees with a recent international multi-machine scaling. Finally the edge plasma parameters just before the L-H transition were compared with theories of transition; it was found that while the theories can separate well developed L- and H-mode data, they have little predictive value.

EX/2-3 · The structure of ELMs and the distribution of transient power loads in MAST

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Abstract: The spatial distribution of energy released from the core plasma during edge localised modes (ELMs) and disruptions is a key area of study for ITER, where the resultant power loadings, both inside and outside the divertor region, have an important impact on the choice of plasma facing materials. The excellent diagnostic visibility afforded by the spherical geometry of the MAST tokamak, remote first wall and extensive divertor and edge diagnostics make it an ideal device for the study of transient power loads. An important aspect of this work is the poloidal and toroidal localization of the energy deposited during the transient event and in particular the fraction that reaches the divertor compared to the first wall. The observations presented in this paper provide strong evidence for the ELMs examined on MAST being a filament like structure, which is extended along a field line, is generated on a 100 μ s timescale,erupts from the outboard side and connects back into the plasma. This filament is observed on several diagnostics to extend beyond the separatrix and ELM effluxes are observed radially up to 20 cm from the plasma edge. For the first time a high-speed picture has been obtained of an ELM showing this filament like structure. Divertor target power loadings during locked mode disruptions have been derived from IR camera measurements. The loss of thermal energy from the core plasma was derived from Thomson scattering temperature and density profiles. In the disruptions analysed, three well defined phases are observed. At the beginning of the thermal quench around 50 % of the thermal energy is released from the core. Target power loads rise but there is no significant broadening of the heat flux width. Around the time of the current redistribution, the remaining thermal energy is released in a rather short period of a few hundred μ s. The heat flux width in this phase broadens by a factor 10 but the peak heat flux still rises. In the third phase, the plasma current begins to decay and magnetic energy is released from the core. Although the heat flux width for most of the disruption period does appear to broaden, there can be phases of the disruption during which significant energy is released without broadening.

EX/2-4Ra · Power Exhaust on JET: an Overview of Dedicated Experiments

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Abstract: Over the past years, a series of dedicated experiments were performed on JET to study both inter-ELM and ELM energy transport. Three types of scans were employed: a) mass/charge of plasma ions: D, He, H, b) magnetic field and plasma current: $B_{tor} \sim$ 1-3 T, $q_{95} \sim$ 2.6-3.8, B×∇B direction ↓ & ↑, c) heating power (4-18 MW) and fuelling rate ($n/n_{GW} \sim$ 0.3-1). This contribution summarizes what has been learned about steady-state and transient power exhaust from analysis of these experiments. In type-I H-mode, the majority (70-90 %) of the core heating power crosses the separatrix during the inter-ELM phase. The peak power flux on the divertor plates is then determined by competition between energy transport in ||, ∧ and ⊥ directions within one-to-two radial e-folding lengths of the separatrix, i.e. in the so-called near-SOL. Analysis of measured divertor power profiles reveals || energy transport in the near-SOL to be largely classical (with possible kinetic effects), ∧ energy transport to be governed by classical drifts (primarily E×B) and ⊥ energy transport to differ strongly between electrons and ions.

Based on a comparison of both the scaling and the magnitude of the divertor power profiles with predictions of published theories of SOL energy transport, it was found that ⊥ electron energy transport is highly anomalous, most likely governed by electrostatic turbulence driven by interchange MHD and drift-wave instabilities, while ⊥ ion energy transport is dominated by (neo-)classical ion conduction. During the ELM crash, the ELM filament propagates outwards with an average velocity of $\langle v_\perp \rangle \sim 0.5$ km/s, in agreement with the lower bound of the || sheath-limited model, in which polarisation and curvature drifts acting within the ELM plasma filament provide the ⊥ driving mechanism ∧2. As expected, much more ion than electron energy reaches the outboard limiters during the ELM burst. The evolution of both temperatures can be effectively modelled by 1-D kinetic equations in the frame of reference moving radially with the ELM. Extrapolating to ITER, we expect 4±1 % of the electron and 13±3 % of the ion ELM energy to reach the beryllium limiter, 5 cm beyond the separatrix.

EX/2-4Rb · Wall and divertor heat load during ELMy H-mode and Disruptions in ASDEX Upgrade

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Abstract: Experiences from the divertor tokamak ASDEX Upgrade reveals that non- divertor components receives up to 10% of the heating energy during plasma discharges. Whereas this outside-divertor load is no problem in ASDEX Upgrade it might cause serious problems for experiments with large volume and energy content. IR cameras, Langmuir probes and bolometry was used to characterize heat deposition to the upper and lower divertor, and to all other significant in vessel components, such as the central column, the ICRH, and the protection limiters. The improved bolometry evaluation algorithm allows together with the high resolution IR camera to get power balances for ELMs and disruption. This paper will present results on 3 selected topics with emphasis on ITER relevance. (i) the heat load to non-divertor components during ELMs, including energy balance, as well as the implication of fast particle and charge exchange loss for localised heat loads to limiters during current flat top. (ii) ELM structures and localized heat deposition remote from the separatrix in the upper divertor, and (iii) heat load pattern during disruptions.

EX/2-5Ra · Suppression of Large Edge Localized Modes With a Resonant Magnetic Perturbation in High Confinement DIII-D Plasmas

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Abstract: Large Type-I ELMs are suppressed with resonant magnetic perturbations from small dc currents driven in a simple magnetic perturbation coil (the DIII-D I-coil). The perturbation field from the coil resonates strongly with magnetic flux surfaces across most of the pedestal region creating remnant islands surrounded by weakly stochastic field lines. The current required to eliminate all but a few isolated ELM-like impulses is less than 0.4% of I_p. The stored energy, β_N and H factor are unaffected by the perturbation field. The electron pressure profile, radial electric field and poloidal rotation across the pedestal are also unaltered along with the H-mode transport barrier. Since large Type-I ELMs represent a severe constraint on the survivability of the divertor target plates in ITER, a proven method of eliminating these impulses is a critical issue. Results presented in this paper suggest that a relatively simple set of coils may provide a promising option for eliminating ELMs in ITER.
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EX/2-5Rb · Structure, Stability and ELM Dynamics of the H-mode Pedestal in DIII-D

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Abstract: Experiments are described that have increased understanding of the transport and stability physics that set the H-mode edge pedestal width and height, determine the onset of Type-I ELMs, and produce the nonlinear dynamics of the pedestal and SOL ELM perturbation. Predictive theories now exist for the n_e pedestal profile and the p_e height at the onset of Type-I ELMs, and progress has been made toward predictive models of the T_e pedestal profile and nonlinear ELM evolution. Similarity experiments between DIII-D and JET suggested that neutral penetration physics dominates in the relationship between the width and height of the ne pedestal while plasma physics dominates in setting the T_e pedestal width.

Measured pedestal conditions including edge current at ELM onset agree with intermediate-n peeling-ballooning (P-B) stability predictions. Midplane ELM dynamics data show the predicted (P-B) structure at ELM onset, large rapid variations of the SOL parameters, and fast radial propagation in later phases, similar to features in nonlinear ELM simulations.
*Work supported by US DOE W-7405-ENG-48, DE-FC02- 04ER54698, DE-AC05-00OR22725, DE-FG03-95ER54294, DE-FG03-95ER54309, & DE- FG03-01ER54615.

EX/2-6 · Integrated exhaust scenarios with actively controlled ELMs

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Abstract: A high performance scenario for simultaneous feedback control of the divertor neutral flux and temperature using argon injection has been developed at ASDEX Upgrade. In the vicinity of the H-L threshold, those radiative feedback scenarios in the type-I ELMy regime often encounters run-away situations. With ongoing radiation increase the ELM frequency drops until a short H-L transition occurs, quickly cured by the feedback. More stationary conditions were achived by active ELM frequency control. External control was obtained by injection of small cryogenic D pellets. Pace making was found able to break the correlation between plasma edge parameters and ELM frequency, now becoming a free parameter. Enhancing the frequency results in amelioration of the ELMs while keeping particle and energy confinement high. No significant difference could be observed with respect to power load features or dynamic behaviour of intrinsic and triggered ELMs occurring at indentical frequencies.

EX/3-1Ra · Control of the Resistive Wall Mode With Internal Coils in the DIII-D Tokamak

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Abstract: New coils have been installed inside the vacuum vessel on the DIII-D device, for producing non-axisymmetric magnetic fields. These coils are predicted to stabilize the resistive wall mode (RWM) with beta close to the ideal wall limit. Feedback using the new internal coils was found to be more effective compared with previous results using external control coils, because the location inside the vessel allows faster response and better geometrical coupling to the mode structure. A proper choice of feedback gain increased the plasma beta above the no-wall stability limit by up to 70% of the amount predicted for an ideally conducting wall. Feedback-driven dynamic error field correction also stabilizes the RWM. The critical rotation frequency for rotational stabilization is about 2.8% of the Alfven frequency, close to the value predicted with theory, implying severe constraint on high beta operations in ITER and the need of an auxiliary momentum input system or direct magnetic feedback.
*Work supported by U.S. DOE under DE-AC02-76CH03073, DE-AC02-76CH03073, DE-FC02-04ER54698, W-7405-ENG-48, DE-FG02-89ER53297, DE0FG02-03ER83657, and DE-FG03-99ER82791.

EX/3-1Rb · Active Measurement of Resistive Wall Mode Stability in Rotating High Beta Plasmas

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Abstract: We present an absolute measurement of the damping rate and the rotation frequency of the stable resistive wall mode (RWM) in DIII-D plasmas above the conventional pressure limit. In these plasmas toroidal plasma rotation of the order of a few percent of the Alfvén velocity is sufficient to stabilize the $n = 1$ RWM. A reliable extrapolation of the stabilizing effect of plasma rotation in a future experiment, however, requires a complete understanding of the underlying dissipative process. The stability of these high beta plasmas has been probed by extending the technique of active MHD spectroscopy, previously used at frequencies above 10 kHz, to frequencies of a few Hertz. The measured spectrum is in good agreement with a single mode model yielding a damping rate and a rotation frequency, which are compared with numerical calculation using the MARS code in order to test the proposed dissipation mechanisms. A first comparison suggests that the sound wave damping model describes the damping rate of the RWM but overestimates its rotation frequency and hence the dissipation in the resistive wall.
*Work supported by U.S. DOE under DE-FG02-89ER53297, DE-AC02-76CH03073, and DE-FC02-04ER54698.

EX/3-2 · Wall Stabilized Operation in High Beta NSTX Plasmas

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Abstract: The National Spherical Torus Experiment, NSTX, has reached high toroidal beta of 38% through boundary and profile optimization and normalized beta of 7 utilizing moderate current profile modification. The resistive wall mode (RWM) has been observed and violation of the n = 1 ideal no-wall stability limit from the DCON stability code is a necessary condition for mode destabilization. Plasma rotation and dissipation from ion Landau damping have been linked to RWM stabilization. However, drift-kinetic theory indicates that trapped particle effects strongly reduce ion Landau damping and increase the Pfirsch-Schluter toroidal inertia enhancement at RWM frequencies. The relative importance of inertia over dissipation is consistent with the observed RWM stabilization at increased q. Analysis of experimental plasmas shows that the critical rotation frequency of Bondeson and Chu, $\Omega_c = \Omega_A/4q^2$ describes well the RWM stability when applied over the entire rotation profile. Recent experiments have utilized a new internal RWM sensor array capable of detecting n up to 3. Sustained control of global instabilities at high beta is planned using an active control system presently being installed. The VALEN 3D code has been used as a physics design and performance assessment tool for this system. Rotation damping in plasmas below the no-wall beta limit is well described by electromagnetic drag due to islands and associated viscous plasma coupling. The large rotation damping enhancement and global rotation collapse observed in plasmas exceeding the no-wall limit can be described by drag due to neoclassical toroidal viscosity (NTV) in the helically perturbed field of an ideal displacement. Initial NTV calculations show reasonable quantitative agreement between theory and experiment. NSTX EFIT reconstructions now include measured ion pressure and toroidal rotation profiles. The profiles are input in real space and the full solution to the Bernoulli equation is used. Peak pressure shifts of 8% of the minor radius from the magnetic axis have been reconstructed. Constraint of the flux surface position using electron temperature, which will decrease the uncertainty in q, is presently being tested in reconstructions including rotation. Aspect ratio and rotation effects will be examined in recent experiments with $\omega_{rot}/\omega_A = 0.48$ and $\epsilon - \beta$ poloidal = 1.3.

EX/3-3 · Effects of global MHD instability on operational high-beta regime in LHD

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Abstract: In tokamaks, it is well known that the operational beta limits are quite consistent with theoretical predictions of ideal linear MHD theory. On the contrary, in helical plasmas, there were a few experimental studies about MHD instability effects on the operational beta range, where the MHD effects were not clear, and the effects of the local pressure gradient and the toroidal current profile were not treated. In order to clarify the MHD instability effects on the operational beta range, it is necessary to take the effects of the local pressure gradient and the change of MHD equilibrium due to the toroidal current into account, because the stability of interchange modes, which is quite important in helical plasmas, strongly depends on them. In the Large Helical Device (LHD) the increased heating capability and a new magnetic configuration with a high aspect ratio enable exploration of the operational beta range up to 4%. Powerful profile measurement systems of LHD enable detailed profile analysis of the global MHD instability in a wide beta range. In this paper, in order to evaluate effects of global ideal MHD modes on the operational beta range in helical plasmas, we compare the observed pressure gradients at typical resonant rational surfaces with the theoretical prediction, and study the global confinement properties in the stationary states. In the core, a suppression of the pressure gradients due to the global ideal MHD instability has been observed in the intermediate beta range. The suppression leads to around 5% degradation of the global confinement, which is not serious. In the edge, a clear saturation of the pressure gradients due to the ideal MHD instability has not been observed up to the high beta regime over 3%, where global ideal MHD modes are predicted to be unstable. A degradation of global transport in the high beta regime has not been observed. Here the beta value is based on the diamagnetic flux measurements. Although the experimental pressure gradients are in the nonlinear saturation phase, our approach (evaluating the experimentally achievable pressure gradients by the linear growth rate and/or Mercier parameter) would still be useful, because it could be a reference for more complicated nonlinear analyses, and a criterion for a reactor design.

EX/3-4 · Equilibrium and Stability of High-Beta Plasmas in Wendelstein 7-AS

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Abstract: Quasi-stationary, MHD-quiescent discharges with volume averaged beta-values up to 3.4% were sustained in the W7-AS for more than 100 energy confinement times. A stability limit was not observed. The achieved beta appears to be limited by confinement, but is sensitive to the magnetic configuration. The decrease in beta for vacuum rotational transform < 0.5 is consistent with an equilibrium beta-limit given by a Shafranov axis-shift of one-half the minor radius. Free-boundary equilibria calculated by PIES indicate that the beta-limit and its variation may be due to deterioration of the flux surfaces and generation of magnetic stochasticity. Low-frequency n =1 and 2 MHD activity is often observed at intermediate beta-values, but does not impede access to higher-beta. Linear ideal-MHD free-boundary stability calculations indicate that the mode should be unstable for beta $< 2.5\%$, and thus severely underestimates the achievable beta-limit. At low electron temperature Te $\sim$ 200eV, fast MHD bursts sharply reduce the plasma confinement and limit the plasma beta. The onset conditions approximately agree with the stability threshold for resistive ballooning modes.

EX/4-1 · Development, Physics Basis, and Performance Projections for Hybrid Scenario Operation in ITER on DIII-D

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Abstract: Experiments in the DIII-D tokamak have demonstrated the ability to sustain ELMing H-mode discharges with high beta and good confinement quality under stationary conditions ($t_{dur} > 35\tau_E > 3\tau_R$). In recent experiments, the range in q_{95} over which this performance could be maintained has been expanded with only small reductions in the sustainable β_N and H_{89P} observed. In all cases, the achieved normalized fusion performance (in terms of $\beta_N H_{89P}/q_{95} \wedge 2$) is at or above the value of this parameter projected for $Q_{fus} = 10$ in the International Thermonuclear Experimental Reactor (ITER) design. Projections using the standard ITER H-mode scaling laws based on these discharges indicate that $Q_{fus} = 5$ can be maintained for >1.5 hours in ITER at $q_{95} = 4.5$ while $Q_{fus} = 40$ can be obtained for $\sim$2400 s at $q_{95} = 3.2$. These projected performance levels further validate the ITER design and suggests long-pulse, high neutron fluence operation as well as very high fusion gain operation may be possible in ITER.
*Work supported by U.S. DOE under DE-AC05-00OR22725, DE- FC02-04ER54698, and W-7405-ENG-48.

EX/4-2 · The "hybrid" scenario in JET: towards its validation for ITER

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Abstract: In 2003, the hybrid scenario performance have been successfully verified in JET up to bN=2.8 at low toroidal field (1.7T), with plasma triangularity and normalised Larmor radius (rho*) corresponding to identical AUG discharges. Stationary conditions have been achieved with the figure of merit for fusion gain (H89.bN/q95) reaching 0.42 at q95=3.9. The JET experiments focused mainly on enhancing the existing database of hybrid scenario towards ITER parameters. The regime has also been validated at toroidal field 1.7T with in an ITER-like plasma shape with trace tritium injection to assess the transport properties of the fusion fuel in this regime. The experiments are now aiming at decreasing the normalised Larmor radius rho* towards projected ITER values. An intermediate step has already been successfully reached at a toroidal field 2.4T demonstrating the potential of the hybrid scenario. Other experiments have also produced this scenario at high field with dominant RF heating at BT=3.1T. The maximisation of confinement and stability properties together provides to this scenario a good probability for achieving discharges in excess of 1000s in ITER.

EX/4-3 · Stationary high confinement plasmas with large bootstrap current fraction in JT-60U

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Abstract: Recent experiments on the JT-60U tokamak have achieved long sustainment of a large non-inductive current drive fraction (>90%) with a large bootstrap current fraction ($\sim$75%) for 7.4 s (>current relaxation time) in a reversed shear plasma with an ELMy H-mode edge. The high confinement enhancement factor of ∼3.0 (HH∼1.7) was sustained, and the profiles of current and pressure approached the stationary condition after the current relaxation time. The large bootstrap current and the off-axis beam driven current sustained this reversed shear q- profile. This duration was limited only by the duration of the neutral beam injection.

EX/4-4 · Comparison of plasma performance and transport between tangential co- and counter-NBI heated MAST discharges.

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Abstract: Counter Neutral Beam injection is of interest due primarily to the discovery of QH-mode, a means by which high performance ELM free operation can be achieved. On MAST, experiments and modelling indicate that only ∼30-50% of the fast ion energy is absorbed using 40keV Deuterium counter-NBI. The angular momentum delivered to the plasma, however, is predicted to be ∼70% larger with counter-NBI due to the rapid loss of co-moving NBI ions. In terms of plasma performance, it is perhaps surprising that the resulting plasmas exhibit stored energies comparable to those for co-injection, in line with similar results from ASDEX and JFT-2M. Counter-NBI ELMy and ELM free H-mode are readily accessed with acceptably low Z_{eff} (typically ∼2-3), resulting in H_H factors ∼2.0 (with respect to ITER H-mode scaling IPB98[y,2]), dropping slightly with the onset of large ELMs and tearing activity. Confinement is thus double that for typical co-NBI heated quiescent H-mode discharges using >2MW of NBI. Equally striking, electron density profiles are routinely much more peaked (with little or no evidence of edge density "ears"), and temperature profiles are much broader than for co-injection. In addition, core plasma rotation is of order ∼340km/s (using charge exchange recombination spectroscopy), greatly exceeding the carbon thermal velocity and approaching the plasma sound speed. This results in a strong inboard-outboard asymmetry in the electron density and Z_{eff} radial profiles, consistent with measured rotation and fully stripped carbon as the dominant impurity. In this paper we present the first particle and energy transport analyses of counter NBI heated ST plasmas. Initial estimates suggest that the narrower density and broader temperature profiles are due to the increased Ware pinch (and the associated "outward" heat flux), augmented by a beam driven pinch (due to friction between the NBI ions and thermal electrons, acting in the same direction as the Ware pinch for counter injection). In addition, it is possible that the increased contribution to the ExB flow shear from the large NBI driven toroidal flow (which reinforces that due to the pressure gradient for counter-NBI) results in a reduction in turbulence growth rates.

EX/4-5 · The Improved H-Mode at ASDEX Upgrade: a Candidate for an ITER Hybrid Scenario

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Abstract: Since 1998 ASDEX Upgrade has developed a stationary regime of operation with improved core confinement for both electrons and ions in combination with a H-mode edge, called "Improved H-Mode". This scenario is characterized by a stationary q profile with central q close to 1, low magnetic shear in the centre, no sawteeth, a confinement factor with respect to H-mode of up to 1.4, and a normalized beta of around 3. The qualification of the improved H-mode as a "hybrid scenario" for ITER is a central part of the ASDEX Upgrade programme. These studies focus on confinement and MHD stability in different parameter scans ($q - edge, rho_*, density$), the role of sawteeth and NTMs, impurity transport, reactor relevant issues like strong electron heating and high edge densities with low amplitude ELMs, as well as the comparison experiments with other tokamaks like JET and DIII-D. ASDEX Upgrade with its flexibility in plasma shaping, high NBI power, ICRH, ECRH, and extended set of diagnostics plays a key role in the development and characterization of this scenario which potentially offers for ITER long pulses at reduced plasma current or higher Q values at the nominal plasma current.

EX/4-6Ra · Studies of HRS H-mode plasma in the JFT-2M tokamak

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Abstract: An attractive operational regime without large ELMs, namely "High Recycling Steady" (HRS) H-mode, was found on JFT-2M during Y2000 campaign after boronization. We have newly extended the

HRS operational regime toward higher energy confinement utilizing the capability of both high triangularity ($\delta > 0.7$) and elongation ($\kappa > 1.6$) operation under the double-null (DN) configuration. The energy confinement time enhancement factor (H89P) in the HRS plasma under the DN configuration increased up to ~2 at around ne/nGW~0.4, which was close to that of the standard ELMy H-mode regime having large ELMs under the single-null (SN) configuration ($\delta \sim 0.4/\kappa \sim 1.4$). A series of parameter scan was also performed in the q95-DELTA space under the limiter (LIM) configuration. The HRS H-mode was easily reproduced in the D-shaped plasmas, while the large ELMs appeared in the circular cross-sectional shape, indicating an important role of plasma shaping to access the HRS regime. A key feature of the HRS H-mode is a reduction of the transient heat load to the divertor target due to large ELMs. It was estimated to be about $0.3MW/m^2$ in the typical HRS H-mode plasmas, which was significantly small in comparison with the transient heat load of ELMy H-mode discharge. The edge MHD activities in the frequency range of the order of 10-100 kHz were recognized to be important for an enhanced particle transport and steady H-mode edge condition in the absence of large ELMs.

EX/4-6Rb · Electrostatic fluctuation and fluctuation-induced particle flux during formation of the edge transport barrier in the JFT-2M tokamak

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Abstract: The potential and density fluctuations are measured in the edge region by a Heavy Ion Beam Probe (HIBP) in the JFT-2M tokamak. The particle flux induced by fluctuations is estimated with the HIBP, and seems to dominate the particle transport. It decreases at L-H transition because the potential and density fluctuation are suppressed. In the Ohmic and L-mode plasma, the particle flux shows intermittent behavior. At the same time, the coherent potential fluctuation appears. The frequency of the coherent mode is about 15 kHz and the poloidal mode number is less than or equal 8. On the other hand, no such coherent mode in the density fluctuation is observed. The characters may have a similarity to those of Geodesic Acoustic Mode (GAM) which is a kind of zonal flow. The period of the intermittent particle flux is comparable to that of the coherent mode, so the intermittent behavior of particle flux may relate to the coherent mode.

EX/5-1 · Energetic Particle Driven Modes in Advanced Tokamak Regimes on JET, DIII-D, Alcator C-MOD and TFTR

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Abstract: An outstanding issue for fusion science is the possible effect of collective instabilities in enhancing fast ion diffusion in Advanced Tokamak regimes. Understanding such instabilities, both their linear properties and saturation level, is essential for predicting collective alpha particle phenomena in a future burning plasma experiment. Magnetic and interferometer measurements of high frequency oscillations in the core of JET, Alcator C- MOD and DIII-D AT plasmas reveal similar characteristics, consistent with the theoretical properties expected for Alfvén Cascades. These modes are predicted to occur in plasmas with weak and reversed magnetic shear, relevant to Advanced Tokamak regimes, and are found to dominate the high frequency spectrum up to the TAE range of frequency in plasmas with sufficient fast ion population. Linear stability analysis using the NOVA-K and CASTOR codes, together with fast particle orbit analysis using the ORBIT code, address the excitation and damping mechanism of these modes, their observed amplitudes, and possible impact on fast ion confinement. Issues relevant to burning plasmas are discussed, together with directions for future investigation of these instabilities.

EX/5-2Ra · Experimental studies of instabilities and confinement of energetic particles on JET and on MAST

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Abstract: In preparation for next step burning plasma such as ITER, which will have several different groups of fast ions: alpha-particles, NBI in the MeV range, and ICRH- accelerated ions, experimental studies of instabilities and confinement of energetic ions were performed on JET and on MAST, with innovative diagnostic techniques, in conventional and shear-reversed plasmas, exploring a wide range of effects. Trace tritium experiments and alpha tail production using third harmonic ICRH of 4He beam

ions have been employed on JET for dedicated fast particle studies in DT plasma and in a 'neutron-free' environment. Measurements of DT neutron profiles in the presence of instabilities excited by NBI with velocity close to the Alfvén speed on JET showed a departure of the tritium beam transport from classical collisional relaxation. The evolution of ICRH-accelerated ions of 4He, 3He, D and H was assessed by measuring profiles and energy spectra of these energetic ions with time-resolved diagnostics of nuclear gamma-rays born by the fast ions colliding with thermal ions and with Be and C impurities. The gamma-diagnostics allowed changes in the fast ion distribution function to be assessed during sawteeth, TAEs, EAEs and Alfvén Cascades. Coupling between modes of different types and the fast ions is investigated based on the measured profiles. Simultaneous measurements of spatial profiles of fast 4He and fast D ions relevant to ITER were found to be in agreement with the theory of fast ion orbits in positive and strongly reversed magnetic shear discharges. MHD spectroscopy based on measurements of TAEs and Alfvén Cascades excited with fast ions has become a routine technique used on JET for developing plasmas with Internal Transport Barriers. In parallel, instabilities excited by super-Alfvénic fast ions were investigated in NBI-heated plasmas on the spherical tokamak MAST. Due to higher values of beta and a higher proportion of fast ions, a wider variety of modes and nonlinear regimes for the instabilities (such as hole-clumps and chirping modes) were observed. The MAST data showed that TAE and chirping modes decrease both in their mode amplitudes and in the number of unstable modes with increasing beta. Combining the JET and MAST results allows key parameter tests to be made and an integrated understanding of behaviour of energetic particles to be developed.

EX/5-2Rb · Energetic Ion Transport by Alfvén Eigenmode Induced by Negative-Ion- Based Neutral Beam Injection in the JT-60U Reversed Shear and Weak Shear Plasmas

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Abstract: The relation between energetic ion transport induced by Alfvén eigenmodes (AEs) and their mode structure has been systematically investigated from the measurement of neutron emission profile for the first time in the JT-60U. A bursting mode called Abrupt Large-amplitude Event (ALE) in weak magnetic shear (WS) plasmas redistributes energetic ions from the core region to the outer region of the plasma. The newly installed diamond detector indicates that the neutral particle flux in a limited energy range (100 $\sim$ 370 keV) is enhanced by ALEs. The energy range of enhanced neutral particle flux is consistent with the resonance condition between the mode and the energetic ions. On the other hand, AEs induced in reversed shear (RS) plasmas expel energetic ions from the plasma. This difference in energetic ion transport is consistent with the difference in the mode structure of AEs in each magnetic configuration; energetic particle mode in a WS plasma and a global mode in a RS plasma.

EX/5-3 · Study of aspect ratio effects on MHD instabilities

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Abstract: Aspect ratio affects kinetic instabilities in many ways. In this paper we describe explicit aspect ratio scaling studies of kinetic instabilities using the NSTX and DIII- D devices and introduce new theoretical work on the general kinetic physics of KBM, TAE, CAE with applications on NSTX. The National Spherical Torus Experiment (NSTX) and the DIII-D tokamak are nearly ideal for an Alfvén mode similarity experiment, with similar neutral beams, fast-ion to Alfvén speed, fast-ion pressure, and shape of the plasma, but the major radius differs by a factor of two. A similarity study of the toroidal Alfvén mode (TAE) shows that the most unstable toroidal mode number scales as expected, supporting an expectation of a "sea" of unstable modes in a reactor. Alfvénic instabilities with frequencies that chirp rapidly are common in NSTX but rare in DIII-D. Efforts to understand this difference in terms of the hole-clump theory of Berk and Breizman are reported. Compressional Alfvén modes (CAE) on NSTX have the frequency scaling, polarization, dependence on the fast-ion distribution function, and low frequency limit qualitatively consistent with CAE theory. Experiments are planned to compare the stability limits on DIII- D with the NSTX stability limits, with the aim of determining if CAE will be excited by alphas in a reactor. The ballooning instability results from the release of free energy of non- uniform pressure that has a gradient in the same direction as the magnetic field curvature. We show that the combined kinetic effect of trapped electron dynamics and ion Larmor radii produces a large parallel electric field and hence a parallel current that greatly enhances the stabilizing effect of field line tension. We are grateful to the NSTX and DIII-D teams. This work supported by U.S. DoE Contracts DE-AC02-76CH03073, DE-AC03-99ER54463 and SC-G9034

EX/5-4Rb · Configuration Dependence of Energetic Ion Driven Alfvén Eigenmodes in the Large Helical Device

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Abstract: Energetic ion driven Alfven eigenmodes (AEs) such as toroidicity-induced AEs (TAEs) and helicity-induced AEs (HAEs), are observed in neutral beam injection (NBI) heated plasma of the Large Helical Device (LHD). It is important to clarify the configuration dependence of AEs because the existence and stability of them sensitively depend on the magnetic configurations, which are characterized by the magnetic axis position and plasma beta. These parameters have been scanned for the study of the configuration dependence of energetic ion driven AEs in LHD. We have studied the energetic ion driven AE in plasma obtained in the following three types of magnetic configuration, (i) Rax=3.6 m with high magnetic shear, (ii) Rax=3.5 m with moderate magnetic shear, and (iii) high beta (>2 %) of Rax=3.6 m plasma with weak magnetic shear. In order to identify the observed AEs, we have compared between the experimental data and the global mode analysis by CAS3D3. In the case of (i), two toroidicity-induced AEs(TAEs) with m∼2/n=1 (m, n: poloidal and toroidal mode number) having the character of core-localized mode and m∼3/n=2 are typically observed. In the case of (ii), of which magnetic shear is smaller than that of Rax =3.6 m (case (i)), a number of TAE with the n = 2∼5 are and an EAE with the n=5 are simultaneously observed. In the case of (iii), of which magnetic shear is further reduced from the core to the plasma peripheral region even in the configuration of Rax=3.6 m, a number of bursting TAE are observed. The TAE gaps are well aligned from the plasma core to the edge with fairly large gap width because of low magnetic shear and large Shafranov shift. The CAS3D3 analysis has demonstrated that the eigenfunction of TAE is widely extended from the core to the edge. Accordingly, the TAE is strongly excited having bursting characteristics because of high energetic ion pressure and less impact of continuum damping. From these studies of AE in these three magnetic configurations, continuum damping, of which damping rate depends on the magnetic shear, plays a key role in stabilizing AEs in LHD.

EX/5-5 · LHCD and Coupling Experiments with an ITER-like PAM launcher on the FTU tokamak

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Abstract: Succesful experimental tests on the PAM (passive active multijunction) antenna for the Lower Hybrid (LH) waves similar to that foreseen for ITER have been carried out for the first time on the FTU tokamak. They validate the main features indicated by the simulation codes, concerning the power handling and the coupling. More than 1.5 times the power requested for ITER and scaled for the different LH frequencies (8 GHz in ITER, 5 GHz in FTU) can be routinely managed for the maximum time allowed by the LH power plant, namely 0.9 s, without any fault in the transmission lines. The power level achieved in FTU is 250 kW against a design value of 170 kW, which correspond respectively to 50 and 33 MW/m^2 through the ITER active area of the antenna. The power reflection coefficient Rc is always $\leq 2\%$, once the PAM launcher has been properly conditioned, even with the grill mouth retracted 2 mm inside the port shadow, with density in front of the launcher very close to the cut-off value. The current drive efficiency is comparable to a conventional grill in similar conditions, provided the lower directivity is taken into account. The flexibility in the N—— spectrum is confirmed by the HXR spectrum, detected with a fast electron bremsstrahlung camera. Conditioning the PAM to operate at the ITER equivalent power level has required only one day of RF opeartion, without a previous baking at 200 °C of the waveguides.

EX/5-6 · ICRH experiments on the spherical tokamak Globus-M

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Abstract: The results of RF heating experiments obtained on the spherical tokamak Globus-M in ion cyclotron frequency range are reported. The experiments were performed in deuterium plasma with high amount (up to 50%) of hydrogen. The RF power was fed by a single-loop antenna at frequency 9 MHz, when resonance conditions for fundamental and second cyclotron harmonics were satisfied simultaneously in plasma cross-section. The temperature of both ion components was doubled (from 170 up to 300 eV). The characteristic times of ion temperature rise and decay corresponded to energy life time in the installation. The results of FMS wave absorption modelling by 3D-code are reported for the real Globus-M geometry.

EX/6-1 · Compatibility of advanced tokamak plasma with high density and high radiation loss operation in JT-60U

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Abstract: Compatibility of advanced tokamak plasma with high density and high radiation loss has been investigated in both high β_p H-mode plasma with a weak positive shear and reversed shear (RS) plasma. In the high β_p H-mode plasma, high confinement (HH_{y2}=0.96) is maintained at high density ($\bar{n}_e/n_{GW}$=0.92, where n_{GW} is Greenwald density) with high radiation loss fraction (f_{rad} ~0.9) by utilizing high-field-side (HFS) pellets and Ar injections. In the RS plasma, high confinement of HH_{y2}=1.3 is achieved at the high density above n_{GW} ($\bar{n}_e/n_{GW}$=1.1) even with NB fuelling only. In these plasmas, the high $\bar{n}_e/n_{GW}$ is obtained due to a peaked density profile inside the internal transport barrier (ITB). The pedestal β_p, defined as β_p-*ped*=*p- ped*/$(B_p^2/2\mu_0)$, where *p-ped* is the plasma pressure at the pedestal top, is almost proportional to the total β_p (β_p-*tot*) in the high β_p H-mode plasma with the HFS pellets and Ar injections, as well as without Ar injection. On the other hand, dependence of β_p-*ped* on β_p-*tot* is weak in the RS plasma. The radiation loss profile in the main plasma is peaked due to impurity accumulation in both plasmas. The impurity transport analyses indicate that core radiation loss from Ar impurity more accumulated by a factor of 2 than the electron, as observed in the high β_p H-mode plasma, can be compensated with slightly enhanced confinement in a fusion reactor.

EX/6-2 · Density Limit Studies in the Large Helical Device

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Abstract: In recent experiments in the Large Helical Device (LHD) , line-averaged densities of up to $1.6 \times 10^{20} m^{-3}$ have been sustained for more than 0.7 s by 11 MW neutral beam injection, which is 1.36 times Sudo scaling [1]. The density limit in LHD is investigated from two aspects. First of all the mechanism of the radiative collapse which limits the density is investigated through the observation of the evolution of various parameters including the temperature profiles, radiation and density. The onset of the thermal instability leading to collapse is quantitatively defined as the point when radiation is increasing as the third power of the density, while it increases in proportion to the density during the earlier, steady-state portion of the discharge. This critical point is associated with a certain edge electron temperature observed as 150 eV at the normalized minor radius (r/a) of 0.9 independent of the density and input power, which indicates that the edged electron temperature is playing a key role in triggering the thermal instability leading to the radiative collapse of the plasma. Rapid increase of first the oxygen impurity radiation at this point followed ~50 ms later by the carbon impurity radiation indicates that this temperature threshold is related to the onset of the radiative thermal instability in the light impurities as the temperature profile drops and the volume of the low temperature edge plasma where light impurities can radiate strongly is increasing rapidly. The density limit is also studied from the point of view of degradation of confinement at high density. This is done by observing the changes in the pressure profiles at high density. The pressure gradient scale length is observed to be rather constant as a function of density except at the plasma edge (r/a=0.9) where it is seen to decrease with density. When comparing the pressure profiles with those of a model case, which matches those of LHD plasma in the plateau regime, it is observed that the profile is degraded at the edge in the Pfirsh-Schlueter regime of collisionality. These results also indicate that the density limit in LHD is affected by the edge plasma properties.

S. Sudo et al., Nucl. Fusion 30 (1990) 11.

EX/6-3 · Scaling Study of ELMy H-Mode Global and Pedestal Confinement at high triangularity in JET

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Abstract: The ITER Q=10 inductively driven reference scenario requires H98=1 at high density (0.85 nG, nG is the Greenwald density). With high plasma triangularity, high density can be achieved whilst maintaining H98 sufficient to match the ITER requirement, but questions remain over the scaling of these results with machine size and other plasma parameters. For this reason, the scaling of pedestal and global confinement of high triangularity ELMy H-modes with density, plasma current, and edge safety factor was studied at JET in a wide range of parameters (Ip from 1 to 3.5MA and q95 from 3 to 4.5). The data appear to confirm the scaling of global and pedestal confinement with plasma current. The link between pedestal and core confinement will be discussed and pedestal scaling models will be tested. In JET, the ability to

increase the density up to nG whilst maintaining good confinement is associated with the H-mode entering the mixed Type I/II ELM regime at high density. This regime was obtained in a wide range of plasma current but not at the highest q95. Confinement and operational space of the Type I/II ELMy regime will be discussed. Finally, the data on ELM size will be summarised.

EX/6-4Ra · The Role of Rotation in the H-mode Transition in Different Magnetic Configurations and Anomalous Momentum Transport in Alcator C-Mod Plasmas with No Momentum In

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Abstract: Anomalous momentum transport has been observed in Alcator C-Mod tokamak plasmas. Following the L-mode to EDA H-mode transition in both Ohmic and ICRF heated discharges, the ensuing co-current toroidal rotation velocity, which is generated in the absence of any external momentum source, is observed to propagate in from the edge plasma to the core with a time scale of order of the observed energy confinement time, but much less than the neo-classical momentum confinement time. With the aim of understanding the higher H-mode power threshold observed in plasmas with an upper single null, the role of rotation in the L-H transition has been examined in discharges with different magnetic configurations. In L-mode plasmas, the ambient rotation in upper single null plasmas is found to be substantially more counter-current than in lower single null. With application of ICRF power, the rotation velocity increases in the co-current direction, in proportion to the stored energy increase. The H-mode transition occurs when the velocity reaches a characteristic value, which for the core is near 0 km/s. L-mode plasmas with stronger counter-current rotation consequently have a higher H-mode power threshold.

EX/6-4Rb · Plasma Rotation in Electron Cyclotron Heated H-modes in DIII-D

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Abstract: Rotation profiles are measured in DIII-D for ECH H-mode discharges, with no auxiliary momentum input. To our knowledge these are the first such measurements isolating the effect of ECH, with no additional neutral beam torque. The toroidal rotation in the interior ($\rho < 0.6$) is found to be in the counter-I_p direction, while outside it is co-directed. There is a reversal layer in toroidal velocity. This is in contrast to comparison measurements in Ohmic H-mode discharges in DIII-D that show co-rotation everywhere, as also measured in other tokamaks. Neoclassical theory is used to compute the bulk D ion velocity given the measured impurity C velocity and the toroidal velocity reversal remains for D. The region of counter-rotation coincides with that of ECH power deposition. The rotation profiles evolve in time with long ELM-free periods. The counter rotation decays, possibly with the equalization of T_e and T_i with rising density.
*Work supported by U.S. Department of Energy under Contracts DE-AC03-99ER54463, DE-AC05-00OR22725, and DE-AC02-76CH03073.

EX/6-5 · Effect of the Dynamic Ergodic Divertor in the TEXTOR Tokamak on MHD Stability, Plasma Rotation and Transport

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Abstract: In the TEXTOR tokamak a new system of helical magnetic field coils has been installed. The unique feature of this Dynamic Ergodic Divertor (DED) is the capability to apply rotating magnetic perturbation fields of up to 10 kHz. Both, the poloidal and toroidal components of the plasma velocity are modified by the DED. At low DED fields the toroidal velocity always increases and always in co-direction. This acceleration occurs both for static and co- and counter-rotating DED field. The latter case produces the strongest toroidal spin- up, which is opposite to the rotation of the perturbation field. When raising the perturbation amplitude, an m/n = 2/1 tearing mode is excited and the plasma rotation locks to the DED field. In static operation the threshold of this induced tearing mode depends on plasma rotation, plasma beta and collisionality. While beta and collisionality are stabilizing, plasma rotation is destabilizing, i.e. the induced tearing mode is excited at lower perturbation amplitude. Reflectometer fluctuation measurements, ECRH heat pulse and impurity injection experiments indicate a partial compensation of the confinement loss in the magnetic islands.

EX/6-6 · Particle and Energy Transport in Dedicated ρ^*, β and ν^* Scans in JET ELMy H-modes

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Abstract: To study particle and energy transport physics, a series of non-dimensional parameter scans were performed in JET ELMy H-modes by varying one of ρ^*, beta, and ν^* whilst fixing the other two. Particle transport is studied through the propagation of trace tritium, in discharges seeded by both gas puffing (1.2MW for 80ms) and a 100keV NBI source (1.2MW for 100ms), using 14MeV neutron cameras with 19 lines of sight. Empirical and physics based models are fitted to the data using a hybrid of the UTC, SANCO and TRANSP codes. Global results indicate that tritium particle transport is largely gyro-Bohm, in line with the IPB98(y,2) scaling for ITER predictions, but that it increases with increasing beta. Energy transport is studied using the TRANSP code. Global energy transport is found to be consistent with the gyro-Bohm-like IPB98(y,2), but with negligible beta dependence. Global ν^* scaling is somewhat stronger than IPB98(y,2), and, in conjunction with a C-Mod match, demonstrates that ν^* is a more relevant energy confinement scaling parameter than Greenwald density fraction. The implications of these results for ITER will be discussed. Many of the scans were performed in a low q (q95=2.8) scenario relevant to the ITER 17MA operating point.

EX/7-1 · Cross-machine NTM physics studies and implications for ITER

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Abstract: Control and avoidance of Neoclassical Tearing Modes (NTMs) will be necessary to ensure good performance in ITER. Recent joint ITPA/IEA experiments on ASDEX-U, DIII-D, and JET, are reported providing new insights into the transport effects, seeding, underlying physics, and threshold scaling of NTMs. Studies highlight the key role of sawteeth in triggering NTMs, with advances made in prediction and control at ITER-relevant fast particle densities. A range of trigger mechanisms are found in ELMy H and hybrid scenarios, with impact on NTM onset beta discussed and utility of existing NTM onset scalings assessed. A significant impact on performance is observed from three types of NTM, with detailed transport analysis made of the effects of 3/2 NTMs in Trace tritium experiments. Measurements of the underlying NTM physics indicates a scaling of NTM metastability beta thresholds with ρ^* across devices, suggesting increased sensitivity to NTMs in ITER and challenging requirements for ECRH control systems. These observations suggest that further development of both control and avoidance strategies will be prudent for various types of NTMs in both ELMy H mode and hybrid scenarios in ITER.

EX/7-2 · Active Control of MHD Instabilities by ECCD in ASDEX Upgrade

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Abstract: MHD activity in the core of a tokamak fusion plasma shows positive and negative influence on the overall performance in terms particle and energy confinement. Sawteeth and neoclassical tearing modes (NTMs) are the most prominent examples of core MHD. The external tailoring of these instabilities for increasing the plasma performance is of great interest. At ASDEX Upgrade the main tool to modify the stability of sawteeth and NTMs is the local electron cyclotron current drive (ECCD). For the modification of the sawtooth repetition rate and triggering the so called FIR-NTMs a variation of the local shear plays the major role. For the complete stabilisation of NTMs replacing the missing bootstrap current over the island plays the main role. The scenarios and schemes for modifying the core MHD behavior and its applicability for ITER will be discussed.

EX/7-3 · Onset and Suppression of 2/1 NTM in DIII-D

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Abstract: Complete suppression of the important $m = 2/n = 1$ neoclassical tearing mode (NTM) has been achieved recently in DIII-D using electron cyclotron current drive (ECCD) in the island O-point. High performance, hybrid-regime discharges were used in these experiments with $\beta_N \approx 2.8$ ($\beta \approx 3.5\%$), which equals 90% of the ideal kink no wall stability limit. These NTMs were shown to be classically destabilized by varying $d\beta/dt$ prior to onset and comparing the resulting island growth rate to a comprehensive model,

the latter which predicts maximum and minimum rates of heating for immediate NTM onset. Using five gyrotrons with 2.7 MW of unmodulated power to drive 40 kA of current at $\rho = 0.66$ (the location of the $q = 2$ surface), this NTM was stabilized resulting in a ≈30% improvement (recovery) of the energy confinement time and an increase in the toroidal rotation velocity. This demonstration of the complete suppression of the $m = 2/n = 1$ NTM using ECCD improves confidence that a control system to prevent confinement loss and disruptions arising from this mode can be developed in ITER.

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EX/7-4 · Stabilization of Neoclassical Tearing Mode by Electron Cyclotron Current Drive and Its Evolution Simulation on JT-60U Tokamak

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Abstract: Stabilization of an m=3/n=2 neoclassical tearing mode (NTM) by a local electron cyclotron current drive (ECCD) has been studied in JT-60U tokamak. In the stabilization experiment, the ECCD applied at the growth phase is found more effective than that at the saturation phase for the first time. The necessary power for the suppression is reduced and the mode onset is delayed. A numerical study has been made on the basis of the modified Rutherford equation coupled with the 1.5D transport code and the EC code. The simulation well reproduces the time evolution of the magnetic island at the growth and stabilizing phases. The peaked EC current profile with low intensity or the broad EC current profile with high intensity mitigates the sensitivity of the stabilization.

EX/8-1 · On the influence of the magnetic topology on transport and radial electric fields in the TJ-II stellarator

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Abstract: The influence of the magnetic topology on plasma confinement and turbulence has been investigated in ECH plasmas in TJ-II. A wide range of rotational transform can be attained in TJ-II and the rotational transform profile can be tailored driving inductive and non- inductive currents. In this way it is possible to study the effect on confinement of rational surfaces and magnetic shear. Electron internal heat transport barriers (e-ITBs), characterized by an increase in the core electron temperature and plasma potential, are triggered by positioning a low order rational surface close to the plasma core region while they disappear when the rational surface is positioned at outer positions. The measured radial electric field in the plasmas that present e-ITB is about 10-15 kV/m, three times the measured in the plasmas without barriers. Magnetic topology affects the radial transport of fast passing electrons and provokes fast changes in the emissivity profiles close to low order rational surfaces. These changes are correlated with the evolution of electron distribution function, as deduced from soft X-ray spectra. We have also obtained the first experimental evidence of the existence of significant radial gradients in the cross-correlation between parallel and radial fluctuating velocities close to rational surfaces, positioned near the LCFS. This result shows that the contribution of the interplay of radial and toroidal fluctuating fluxes to parallel momentum equation is at least comparable to the contribution of charge-exchange mechanisms. While these results support the importance of turbulence to understand the observed interplay between magnetic topology and transport in the edge, the time scales of the perturbation in density and plasma potential, measured by HIBP, and its localization within ECH deposition profile, support the dominant role of ECH convective fluxes in the formation of e-ITBs in the core. The influence of magnetic shear on confinement has also been studied by inducing positive and negative currents. It is observed that the electron density profile changes mainly in the gradient zone due to the current and it is also shown that the negative shear (provoked by negative current) tends to improve the confinement, while a non-monotonic behavior of plasma confinement versus positive shear is observed.

EX/8-2 · Turbulent Particle Transport in Tore Supra

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Abstract: Anomalous particle pinch has been unambiguously identified in Tore Supra [1]. Density profile remains peaked, in absence of the Ware pinch, over a time up to 6min30s. This anomalous particle pinch is predicted by turbulence theories [2]. Here, the parametric dependence of the anomalous pinch is studied

in order to descriminate among the two main theoretical predictions. One predicts a dependence to the temperature gradient, the other to the magnetic field curvature. The main finding of this work is that the inward pinch driven by the magnetic field gradient is dominant. Nevertheless, in the plasma core (r/a < 0.3), the direction of the pinch is sensitive to the temperature gradient. The results are consistent with micro-stability analyses, which indicate that TEM turbulence is dominant in the gradient region (0.3 < r/a < 0.6), while ITG turbulence takes place in the core.
Hoang, G.T., et al., Phys. Rev. Lett. 90, 155002 (2003) [2] Garbet, X. et al., Phys. Rev. Lett. 91, 035001 (2003)

EX/8-3 · Measurement and Modeling of Electrode Biased Discharges in the HSX Stellarator

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Abstract: Measurements of plasma flow damping have been conducted in the HSX stellarator. A fast switching biased electrode system is used to impulsively generate plasma flow. Measurements of plasma flow using Mach probes and floating potential measurements with Langmuir probes are used to characterize the plasma response to the electrode. We observe two time scales in the flow evolution, consistent with the notion that the two directions on a flux surface will have different flow damping rates. Experiments show that the damping is reduced in the quasi-symmetric configuration compared to configurations where the symmetry is intentionally broken. The results are compared to neoclassical modeling of the flow damping and radial conductivity. It is found that the time scale for the damping of flows across the direction of symmetry is consistent with neoclassical theory, but that flows in the direction of symmetry are damped more quickly than the neoclassical prediction. The measured radial conductivity also appears to be larger than the neoclassical prediction. These results are similar to the results in axisymmetric tokamaks, where anomalous transport of toroidal momentum is thought to occur.

EX/8-4Ra · Turbulent transport and plasma flow in the Reversed Field Pinch

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Abstract: The results of an extensive investigation of electrostatic and magnetic turbulence in the edge region of two Reversed Field Pinch (RFP) experiments EXTRAP-T2R and RFX are reported . In both experiments particle transport is mostly driven by electrostatic turbulence and a highly sheared ExB flow takes place. Recent results have shown that almost 50% of the particle losses is due to coherent structures. These structures have features reminiscent of monopolar or dipolar vortices and their relative population and preferred vorticity is determined by the local ExB shear. It has been demonstrated that the plasma diffusivity D can be separated in two comparable terms: one due to coherent structures and another one due to background turbulence. The ExB shear results to affect both terms, modifying the populations of vortices and the phase of background density and velocity fluctuations . This effect has been proved by several experiments of transport control based on modification of the ExB shear by insertion of active electrodes or by applying transient modifications of the magnetic field. In order to investigate the process responsible for the observed ExB flow profile, the momentum balance has been recently addressed and it has been found that also transport of momentum is anomalous as experimental kinematic viscosity results consistent with anomalous diffusion. Moreover the balance reveals that plasma flow profile is regulated by turbulence and in particular by the electrostatic component of the Reynolds Stress (RS). These results prove the existence of a dynamic interplay between turbulence properties, anomalous transport and mean profiles. All results are discussed highlighting the similarities with other magnetic configurations.

EX/8-4Rb · Generation of Sheared Poloidal Flows by Electrostatic and Magnetic Reynolds Stress in the Boundary Plasma of HT-7 Tokamak

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Abstract: The radial profiles of electrostatic and magnetic Reynolds stress have been measured in the plasma boundary region of the HT-7 tokamak using two types of triple-tip- array Langmuir probes and an insertable magnetic probe. Experimental results show that the radial gradient of electrostatic Reynolds stress changes sign across the last closed flux surface (LCFS). Detailed measurements of all quantities in the poloidal momentum balance indicate that the neoclassical flow damping and the damping due to charge exchange processes are balanced by the radial gradient of electrostatic Reynolds stress, which sustains

the equilibrium sheared flow structure in a steady state. To find the relationship between the turbulence propagation and the radial structure of electrostatic Reynolds stress, the radial propagation properties of boundary turbulence are first investigated in detail. Conclusive evidences have been obtained that both poloidal wavenumber and radial wavenumber of turbulence reverse across the LCFS, which means that in the scrape off layer turbulence propagates outward and in the ion diamagnetic direction, while at the plasma edge it propagates inward and in the electron direction. And it is demonstrated that the changes of propagation directions and the radial variation of the potential fluctuation level are responsible for the reversion of the electrostatic Reynolds stress gradient across the LCFS. On the other hand, the magnetic Reynolds stress component (in poloidal direction) of turbulence is first measured in a tokamak. A radial gradient of magnetic Reynolds stress is observed close to the shear layer location, however, as a result of small magnetic fluctuation level in low beta plasma, its contribution to the poloidal flows is small compared with the electrostatic component. In conclusion, the new results presented here suggest that the electrostatic turbulence-induced Reynolds stress might be the dominant mechanism to generate the poloidal flow shear at the plasma edge. The potential importance of this paper is that it presents a complete physical picture regarding the shear flow generation by turbulence in the plasma boundary region, and we hope that it would improve our understanding of the shear suppression physics and facilitate establishing the turbulence self-regulation dynamics.

EX/8-5Ra · Edge and Internal Transport Barrier Formations in CHS

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Abstract: Edge transport barrier (ETB) formation is found in a small low-aspect-ratio stellarator CHS (R = 1 m, B = 0.95 T) with the neutral beam injection (NBI) heating. A sharp drop (~1 msec) of H alpha emission is observed at the transition and the clear back transition appears when the heating power is decreased. The local density measurements show the increase of edge density at the transition although the edge temperature does not increase much. The heating power threshold exists (0.7 MW for $n_e = 2.7 \times 10^{19} m^{-3}$), which is about two times larger than the tokamak H-mode scaling. When the heating power is increased, the internal transport barrier (ITB) is also created at the transition of ETB formation with NBI heating. The electron temperature in the plasma core region increases (without ECH) with the temperature gradient foot point at $(r/a) \cong 0.5$. The line averaged density for this ITB formation is more than 5 times higher compared to the previous ITB experiments in CHS with the combination heating of NBI and ECH.

EX/8-5Rb · Experimental Studies of Zonal Flows in CHS and JIPPT-IIU

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Abstract: The crucial role of zonal flows in determining the turbulence level and resultant transport in plasmas is widely recognized to enhance the prospect of plasma burning in the ITER. Direct nonlinear simulations confirm the appearance of two branches of zonal flows in toroidal plasmas; a branch of nearly zero frequency, and the other of higher frequency called geodesic acoustic modes (GAM). Heavy ion beam probe is a powerful candidate to identify the zonal flows experimentally since the zonal flows manifest themselves in electric field fluctuation constant on magnetic flux surfaces with finite radial wave numbers. The simultaneous observation of electric field at two distant toroidal locations (~1.5m apart), using dual heavy ion beam probes on Compact Helical System (CHS), has succeeded to identify the existence of the zonal flow of nearly zero frequency branch. The presented results are fluctuation spectrum of electric field, spatio- temporal structure of the zonal flow (characteristic radial length of ~1.5 cm and life time of ~ 1.5 ms), the long-range correlation over the magnetic flux surface, and difference in the zonal flow amplitude with and without a transport barrier. The paper describes the first experimental identification of zonal flows and their characteristics on CHS, together with the detection of oscillatory mode presumed to be GAM in JIPPT-IIU.

EX/9-1 · Effect of Plasma Shape on Electron Heat Transport in the Presence of extreme Temperature Gradients in TCV

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Abstract: Electron heat transport is investigated as a function of plasma shape under unprecedented intense electron cyclotron heating power in a regime of high electron versus ion temperature ratio. Varying the high off-axis ECH power density distribution allows extending the range of the normalised electron temperature gradient by a factor four compared to the highest value obtained on other tokamaks. The focus is placed on the influence of plasma shape on various electron heat transport parameters such as the stiffness of the electron temperature profile, the heat diffusivity and diffusivity threshold with normalised electron temperature gradient. The heat conductivity is lowest at lowest triangularity, consistent with the energy confinement time in Ohmic or centrally heated L-mode, which is maximum at the most negative triangularity (and highest elongation). For a given power, off- axis power deposition led to the formation of electron internal transport barriers (eITB) at low triangularity, more easily than at higher triangularity. These findings suggest ways of facilitating (or hindering) access to eITBs using plasma shaping: using low triangularity, and possibly also high elongation.

EX/9-2 · Confinement Studies of Helical-axis Heliotron Plasmas

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Abstract: Recent 70-GHz, 0.4-MW ECH experiments in Heliotron J have revealed the existence of the two edge iota windows for the occurrence of H-mode ($0.54< \iota <0.56$ in separatrix discharge and $0.62< \iota <0.63$ in partial wall-limiter discharge) at line-averaged densities higher than a certain threshold value. It is noted that the boundaries of these two edge iota windows are near the natural resonances such as (i) the intrinsic n=4, m=7 (fourfold symmetry of Heliotron J) and (ii) n=8, m=13 and 15, and (iii) n=12, m=19, where m and n are the poloidal and toroidal mode number, respectively. With regard to the H-mode transition, two types of transitions were observed: a single-step L-H transition and repetitive L-H-L transitions. The latter type is, in fact, more common. Furthermore, a strong dependence of the quality of the H-mode on the edge topology conditions was revealed. The increase in plasma energy content in the H-mode phase reached about 70% only in separatrix discharge and the estimated energy confinement time was found to be enhanced beyond the normal ISS95 scaling, being 50% higher than that in the "before transition" phase. A candidate cause of the observed saturation and decrease in the transient energy confinement time is due to the edge activities. The termination event of the H-mode seems to fall into two groups: one is the lower density termination and the other is the higher density termination. The lower density termination is indicated by a rapid recovery of H-alpha back to its pre-H-mode level (or much higher level) and the close correspondence of the subsequent density decay. The higher density termination appears to be triggered by a back-transition of H-alpha, but in many cases leading to a radiation collapse. A candidate mechanism for the latter type of termination is the ECH cut-off. The H-mode window characteristics are discussed on the basis of the calculated geometrical poloidal viscous damping rate in the plasma boundary. The calculation indicates that the behavior of the viscous damping rate coefficient alone could not fully explain the observed characteristics. It is expected that the recently discovered H-mode discharges in NBI operation could provide further information about the H-mode physics in Heliotron J.

EX/9-3 · LH Transition by a Biased Hot Cathode in the Tohoku University Heliac

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Abstract: The electron injection into a plasma by a biased hot cathode can trigger an improved confinement mode in Tohoku University Heliac (TU-Heliac). The biased electrode induces a strong negative radial electric field and a radial electric field shear. After formation of the electric field, the energy confinement time increases by a factor of 2, the line density increases by a factor of $2 \sim 3$, the electron density profile steepens, the potential and density fluctuation levels are suppressed and the normalized impurity light emission decreases. The bifurcation phenomena (a negative resistance feature in the electrode characteristics and hysteresis in the stored energy), which associate with this L-H transition, are observed. The ion viscous damping force is estimated from the J x B driving force for the poloidal rotation and the effect of the local maxima on the ion viscosity is investigated. The measured damping forces have a local maximum

and agree well with the neoclassical prediction. It is also found that the plasma shows the negative resistance characteristics in the region where the neoclassical ion viscous damping force has a local maximum. The local maxima in the ion viscosity have the key role in the L-H transition in stellarators.

EX/9-4 · Limiter and Emissive Electrode Biasing Experiments on the Tokamak ISTTOK

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Abstract: In this contribution, the detailed behavior of the plasma, under both limiter and emissive electrode biasing will be compared. We have observed that large currents (>15 A) can be drawn at negative applied voltage by both limiter and emissive electrode bias, leading to significant modifications in the edge plasma potential profile and to an improvement in particle confinement. However, compared with the limiter, the electrode has the advantage of perturbing significantly less the plasma column. Furthermore, its use leads to the formation of stronger radial electric fields and to a much larger improvement in confinement. The main result of these experiments is the discovery of improved confinement events, which are characterized by a further increase in particle confinement during short periods. These events are observed for both negative and positive bias, provided that the bias current is larger than a threshold value, —Ibias— = 20 A. As the polarization voltage is increased, the bias current starts to increase linearly with the radial electric field and reaches a saturation value, after which it decreases. This is qualitative agreement with the Stringer expression for the radial current density [T.E. Stringer, Nucl. Fusion 33, 1249 (1993)].

EX/9-5 · Improved Operation and Modeling of the SSPX Spheromak

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Abstract: Progress in understanding both magnetic field generation and confinement is enabling the production of high magnetic field spheromaks with plasma core electron temperatures (Te) >200eV in the Sustained Spheromak Physics Experiment (SSPX). The highest measured Te occurs when the edge magnetic fluctuation amplitude is lowest. Improvements over previous results were produced with higher formation bank current, longer discharges, and better matching of edge current and bias flux to minimize magnetic fluctuations. New experiments show for the first time that the field energy of the spheromak can be increased in a step-wise manner using repetitive current pulses. These multi-pulse discharges produce the strongest magnetic fields yet in SSPX (0.7 T at the geometric axis), and an important scaling of magnetic field with current has been exceeded. 3D resistive MHD (NIMROD) simulations for similar conditions show an increment to the magnetic energy and an increase of closed flux volume, consistent also with Te measurements in the experiment. NIMROD and analytic theory are also used to investigate the role of ideal helical central-column instabilities in a "pillbox" configuration (uniform axial field bounded by conducting end plates) and other geometries. Other advances in understanding spheromak field generation and confinement include new measurements in the injector region that show for the first time that the magnetic flux exits the gun non-uniformly and results from deuterium fueled experiments that show little difference (compared to hydrogen fueled discharges) in spheromak formation dynamics or characteristic MHD behavior.

EX/9-6Ra · Progress in the Study of Plasma Heating, Stability, and Confinement on HANBIT Mirror Device

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Abstract: HANBIT is a magnetic mirror confinement device, which consists of a simple mirror-type central cell, an anchor, a plug, and two end tanks. HANBIT started its physics experimental campaign in 2001, following the development phase of basic heating and diagnostic systems over 1996 to 2000. Initial physics experiments were concentrated on the basic physics studies of RF-heating, stability, and confinement in a simple mirror configuration of the central-cell, trying to identify the discharge characteristics and stable plasma operation modes. A main result from this initial study was that a stable, high-density plasma mode exists in the slow wave regime, where the RF frequency is smaller than the ion cyclotron freueuncy. This result in HANBIT is somewhat different from those from other mirror devices where the stable modes were mostly observed in the fast wave regime with the RF frequency lager than the ion cyclitron frequency. Since the last IAEA conference, a significant effort has been made to clarify the physics origin of this new

stable mode. Also, an attempt has been made to produce the high-density mode in the fast wave regime, as observed in the other mirror devices. In addition, a study is on the way to increase the plasma beta and ion temperature, mainly through the beach wave heating by the DHT antenna system. Here, we present a brief overview of main results from these recent studies in HANBIT.

EX/9-6Rb · Influence of radio frequency waves on the interchange stability in HANBIT mirror plasmas

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Abstract: It has been known that radio frequency (RF) waves in the range of ion cyclotron frequency have influence upon the interchange stability in mirror plasmas by the generation of a ponderomotive force and/or the process of nonlinear sideband wave coupling. In the present work, an investigation is made of the influence of high frequency RF waves upon low frequency interchange modes in HANBIT mirror plasmas. An emphasis is put on the interchange stability near the resonance region, $\omega_0 = \Omega_i$, where ω_0 is the angular frequency of the applied RF wave and Ω_i is the ion cyclotron frequency. Recent HANBIT experimental observations have shown the existence of interchange-stable operation window which is favorable to $\omega_0/\Omega_i \leq 1$ region with its sensitivities on the applied RF power and ion-neutral collisions. A strong nonlinear interaction between the RF wave and the interchange mode has been observed with the generation of sideband waves. Onset characteristics of the unstable interchange mode and the sideband waves are reported. A theoretical analysis including both the ponderomotive force and the nonlinear sideband wave coupling has been developed and applied to the interpretation of the experimental results, resulting in a good agreement. From the study, it is concluded that the nonlinear wave-wave coupling process dominates the RF effect on the interchange stability in HANBIT mirror plasmas operating near the resonance region.

EX/9-6Rc · Heating and Confinement of Ions at Multimirror Trap GOL-3

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Abstract: New experimental results from multimirror trap GOL-3 are presented. Recently magnetic system of the facility was converted into completely multimirror one. This results in further improvement of energy confinement time of plasma with ion temperature ∼1 keV. Plasma is confined in 12-meter-long solenoid which consists of 55 cells with Bmax/Bmin=5.2/3.2 T. Collective plasma heating by ∼120 kJ (∼8 us) relativistic electron beam results in Te∼2 keV at $\sim 10^{21} m^{-3}$density. High Te exists for ∼10us. To this time Ti reaches ∼1 keV. Then electron temperature rapidly decreases to below 100 eV. Ion temperature keeps at the high level during ∼1 ms. In a mode with full corrugation and increased beam energy density the energy confinement time is sufficiently increased and a value of $nt \sim 10^{18} m^{-3} s$ corresponds to theoretical confinement time in multimirror trap in optimum conditions. Dense hot plasma in GOL-3 trap is a source of D-D neutrons which lasts for ∼1 ms. New physical mechanism of effective heating of plasma ions, substantially dependent on the corrugation of the magnetic field, is discussed. Experiments with complete multimirror configuration of the GOL-3 facility have shown significant increase of energy confined time comparing with configurations with simple solenoid or short multimirror sections.

EX/9-6Rd · Advances in Potential Formation and Findings in Sheared Radial Electric- Field Effects on Turbulence and Loss Suppression in GAMMA 10

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Abstract: Following the Lyon IAEA Conference, (1) a factor of two progress in the formation of ion-confining potential heights in comparison to those attained 1992-2002 is achieved for tandem-mirror plasmas in the hot-ion mode with ion temperatures of several keV. For producing such high potentials, merely a 20% increase in plug ECH powers for the confining potential formation is applied as compared before. (2) The advance in the potential formation gives bases for a finding of the remarkable effects of radially produced shear of electric fields, dE(r)/dr, on the suppression of not only coherent drift waves but turbulence- like fluctuations for the first time in GAMMA 10. Such a shear [dE(r)/dr] effect on the central-cell plasmas is highlighted visually by x-ray tomography diagnostics; that is, spatially and temporally fluctuated vortex-like structures are clearly observed in plasmas produced by ICH alone [having a quite weak dE(r)/dr]. However, during the application of plug ECH into the ICH plasmas, an associated potential rise produces a stronger dE(r)/dr [=several $10 kV/m^2$]. The disappearance of the turbulent vortices on the basis of such a high-potential formation due to ECH is of remarkable importance for plasma confinement improvement. In fact, the associated temperature rise and nonambipolar loss suppression are observed. From the viewpoints of both (i) a conventional idea of higher and better potential confinement in the axial direction [i.e. E(z) effects] and (ii) the present new finding of a turbulent vortex disappearance due to a strong radial electric shear [i.e. dE(r)/dr effects] in the transverse direction, simultaneously, such a high potential formation is one of the most essential key issues. (3) For the physics interpretations and control of such potential [or the associated E(r) shear] formation, the validity of our proposed theory of the potential formation is extendedly tested under the conditions with auxiliary heatings. The data described above are well plotted on the extended surfaces calculated from our consolidated theory of the strong ECH theory (plateau formation) with Pastukhov's theory on energy confinement. The validity of the extension of our proposed physics mechanism encourages the future extendable scalability of potential formation having prospective simultaneous E(z) and E(r) shear effects on confinement improvements.

EX/10-1 · Overview of recent work on material erosion, migration and long-term fuel retention in the EU-fusion programme and conclusions for ITER

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Abstract: Material erosion from first wall components, short and long range material transport and the resulting codeposition with the plasma fuel is one of the most critical questions for fusion devices intended to operate long pulses at high fusion yield. A co-ordinated effort is underway in the EU fusion programme to understand the material erosion and transport, develop adequate modelling tools and control and mitigation scenarios for the codeposition of carbon with the tritium fuel in ITER. This contribution summarises the data on the global carbon deposition rates and the resulting fuel retention during different JET divertor campaigns, AUG ,TEXTOR and Tore-Supra and from a new shot resolved deposition monitor in the inner divertor of JET. Modelling of the measured carbon deposition patterns will be discussed showing an enhanced C- transport on plasma wetted areas due to a high chemical re-erosion of redeposited carbon while redeposition in remote areas is mainly in line of sight. Extrapolation of the present data to erosion, deposition and material transport in ITER and the associated fuel codeposition will be attempted.

EX/10-2 · Deuterium retention in Tore Supra long discharges

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Abstract: Tritium retention is a crucial point to investigate for next step machines using carbon plasma facing components. In this paper, possible mechanisms to explain the large deuterium retention observed in Tore Supra long pulses (up to 50% of the injected fuel), such as implantation of energetic particles and codeposition of deuterium with eroded carbon, are reviewed. During the pulse, the retention rate is first seen to decrease (phase 1, $\sim$ 100s), then remains constant (phase 2), showing no sign of wall saturation after more than 6 minutes of discharge. Phase 1 could be interpreted as the progressive saturation of carbon surfaces by particle implantation, combined with codeposition of deuterium with carbon. Phase 2, with a constant wall retention rate, would correspond to codeposition alone, once the implanted surfaces are saturated. Estimates of wall loading from plasma wall interaction modelling are presented. They show a complex pattern of particle implantation, with saturation time constants ranging from 1s to 100's s, consistent with the experimental behavior in phase 1. Estimates of carbon chemical erosion, as the source term for codeposition, are also in rough agreement with the retention rate in phase 2. The carbon deposition rate obtained from these calculations is compared with the deposited layers actually observed in the machine. With its ability to pursue long discharges with actively cooled components, Tore Supra offers a unique opportunity to distinguish between the processes at stake in deuterium retention over time scales relevant to plasma wall interactions in next step machines.

EX/10-3 · Impact of nearly-saturated divertor plates on particle control in long and high-power heated discharges in JT-60U

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Abstract: In order to understand plasma-wall interactions in a long time scale, the discharge pulse length has been extended from 15 s to 65 s, with the NB-heating duration extended to 30 s. Nearly-saturation of the divertor plates was observed in the latter half of long pulse ELMy H-mode discharges. Particle sink into

the divertor plates decreased, and subsequently, particle release from them increased significantly. This saturation resulted in a rise of the main plasma density without any auxiliary particle supply besides NB with the divertor exhausting. Even when the total injected energy reached up to ~ 360 MJ in a discharge (~ 12 MW x 30 s), neither sudden increase of carbon generation such as "carbon bloom" nor increase of the dilution of the main plasma was observed.

EX/10-4 · Edge Plasma Control by Local Island Divertor in LHD

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Abstract: In the Large Helical Device (LHD) program, one of the key research issues is to enhance helical plasma performance through the edge plasma control. For the first time in the LHD program, this was performed with a local island divertor (LID) that is a closed divertor, utilizing an m/n = 1/1 island generated externally. Since the particle recycling is toroidally localized, the technical ease of hydrogen pumping is the advantage of the LID over a closed full helical divertor. The LID experiment has begun since 2002, and fundamental LID functions were demonstrated experimentally. The effect of the LID on plasma performance was studied using hydrogen puffing NBI discharges. The plasma parameters change significantly with the LID. For example, the line-averaged electron density is reduced typically by a factor of ~2, compared with discharges without the LID. The outward heat and particle fluxes crossing the island separatrix were found to flow along the field lines to the backside of a divertor head, where carbon plates are placed to receive heat and particle loads. The electron temperature, T(e), profile with the LID rises steeply from the inner island separatrix, and the central T(e) is as high as that without the LID, although the low-T(e) plasma flows along the outer island separatrix. The low-temperature and low-density plasma outside the outer island separatrix is scraped off, which exists even outside the last closed flux surface without the LID. Thus, since almost no plasma exists between the outer island separatrix and the vacuum vessel, the recycling of particles is localized only near the LID head, located inside the pumping duct. This was confirmed by the H(alpha) emission measurement. High efficient pumping is the key in realizing high temperature divertor operation, resulting in an improvement of energy confinement. In the present experiment, a factor of ~1.2 improvement of the energy confinement time, tau(E), was observed over the international stellarator scaling. The connection length of the magnetic field from near the inner island separatrix on the equatorial plane to the divertor head is ~120 m, and will be shortened by increasing the perturbation coil currents further. Thus, the density along the outer island separatrix will decrease significantly, and hence, further improvement of $\tau(E)$ is expected.

EX/10-5 · Tungsten: An option for divertor and main chamber plasma facing components in future fusion devices

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Abstract: All major design studies of future fusion research and reactor devices reconsider the use of high-Z plasma facing components (PFCs). A careful investigation on the implications from the use of tungsten for the plasma behavior is pursued in ASDEX Upgrade and 65% of all PFCs, including the upper divertor, have been W coated. Only the lower divertor strike zones and some low field side guard limiters are left uncoated. Diagnostic has been implemented in order to quantify closely the W influx and the W content. Most of the experimental programme of ASDEX Upgrade could be performed without serious limitations resulting from excessive tungsten contamination and the W concentration usually stayed below some 10^{-5}. The central impurity content could be controlled successfully by central wave heating and ELM pace making. There is no strong difference evident, comparing discharges run with the upper W-divertor to ones with the lower C-divertor, even during a continuous transition. W influx from the low field side guard limiter is only detectable if the separatrix is close enough to the W surface to allow its contact with hot ions.

EX/10-6Ra · Disruption Thermal Quench Mitigation by Noble Gas Jet Injection in DIII-D

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Abstract: Mitigating the potential first-wall damage associated with the thermal quench phase of disruptions may be achieved by radiating the plasma thermal energy as uniformly as possible over all plasma-facing components. By using fast bolometry, it is shown that high- pressure noble gas injection in DIII-D

is very successful at providing thermal quench mitigation, with around 90% of the plasma thermal energy lost to radiation and only 10% conducted into the vessel walls. The measured propagation of the impurity gas jet into the plasma is found to be consistent with an initial penetration through the plasma edge by self- shielded neutrals, followed by a slower diffusion of impurity ions into the pedestal, followed by a rapid mixing of impurity ions into the core plasma by MHD turbulence. The resulting total thermal quench timescale is found to be a complicated function of gas jet pressure, jet species, and jet geometry. Future plans using optimized gas nozzles and liquid jets will be discussed, as well as the relevance of these results to ITER.
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EX/10-6Rb · Disruption Mitigation Experiments in the JT-60U Tokamak

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Abstract: Experiments in the JT-60U tokamak have shown that disruption deleterious effects on plasma facing components of a tokamak reactor can be greatly reduced or avoided by simultaneous puffing small amounts of high-Z noble gases, particularly, krypton and large amounts of hydrogen gas. A high electron density caused by the intense hydrogen gas puffing amplified the radiation of High-Z species. In turn the stored energy was radiated and plasma was terminated quickly. The high electron density and high effective charge generated by the high-Z species prevent the runaway electron formation. It was also found that injecting neon ice pellets during the post-disruption runaway plateau can enhance the runaway electron losses.

EX/10-6Rc · Disruption Mitigation on Tore Supra

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Abstract: During disruptions, the plasma energy is lost on the first wall within a milliseconds. Forces of several hundred tons are thus applied to the structures and kiloamps of electrons are accelerated up to 50 MeV (runaway electrons). Already sources of concern in present day tokamaks, extrapolation to ITER shows the necessity of mitigation procedures, to avoid serious damages to in-vessel components. Massive gas injection was proposed, and encouraging tests have been done on Textor and DIII-D. Additional experiments where performed on Tore Supra, with the goal to validate their effect on runaway electrons, observed during the majority of disruptions. Up to 1/10 of mole of helium was injected within 5 ms in ohmic plasmas, up to 1.2 MA, either stable, or in a pre-disruptive phase (disruption by argon puffing). Beneficial effects were obtained: reduction of the current fall rate and eddy currents, easy recovery for the next pulse, without noticeable helium pollution of following plasmas, and most of all, a total disappearance of runaway electrons. Analysis of the 4 ms period between injection and disruption indicates that to reach these goals, one needs to inject enough helium to keep it only partially ionised. This threshold corresponds to a pressure of 5 Pa for Tore Supra, and extrapolates to hundreds of grams for ITER (1000 Pa). Effect of such a sudden rise in pressure on the pumps and diagnostics has still to be assessed.

EX/P1-2 · Impurity-seeded ELMy H-modes in JET, with high density and sustainable heat load.

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Abstract: The paper reviews the experimental and modelling work that has been done at JET during the past two years to develop an H-mode scenario with tolerable ELMs and tolerable steady state power load in the divertor using impurity radiation at the plasma edge. With nitrogen seeding, the heat flux density is reduced to about 2-5 MW/m2 in a Type III ELMy H-mode regime with partially detached divertor conditions and reduced confinement. The feasibility of an integrated scenario with Type-III ELMs, and q95=2.6 to compensate for the low H factor, has been demonstrated on JET [1]. This scenario can be proposed for ITER Q=10 operation at Ip=17MA. In argon seeding experiments, the argon level is carefully adjusted to maintain a Type I ELMy H-mode with high confinement (H98 $\sim$ 1). A high density (ne/nGW=1.1) and a radiation fraction up to $\sim$65% are obtained in quasi steady-state. Infrared thermography and thermocouple measurements indicate a reduced heat load to the inner target plate. The global power balance of these discharges is analysed in detail in the paper. The issue of possible radiative dissipation (radiation buffering) of Type I ELMs with DWdia up to 500kJ is also addressed. It is found experimentally that only a small fraction of the ELM energy ($\sim$20% at most) is radiated before reaching the

target plates [2]. Finally, EDGE2D/NIMBUS and the time dependent SOLPS5.0 modelling are presented. These modelling aim at assessing the potential of different impurity species for mitigation of ELM heat fluxes through radiation, and for maximizing edge radiation while maintaining low core contamination. The results for ELM buffering are consistent with what is found experimentally, suggesting that large ELM buffering by radiation is not to be expected in ITER except for very small ELMs.
References: [1] J. Rapp et al., Nucl. Fusion 44 (2004) 312-319. [2] P. Monier-Garbet et al., 30th EPS Conf., St Petersburg, Russia, 7-11 July 2003.

EX/P1-3 · ELMs, strike point jumps and SOL currents

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Abstract: Strike point jumps in JET plasmas were first reported in 1995 [1]. New, more detailed measurements, are contrasted with the supposition that a layer of plasma is peeled off after an ELM. The initial model in this study is that the ELM is due to local loss of solution of the Grad-Shafranov equation at a critical point [2], possibly due to a separatrix instability. A complete layer of previously closed field lines would open. Particles and energy would flow along these newly opened field lines, dominantly in the public scrape-off layer. Since before the ELM the plasma pedestal has large pressure, it must necessarily have large net toroidal current. In the new configuration the plasma would have less toroidal current in closed field lines. The loss of a co-current carrying plasma layer from inside the separatrix results in the formation of a new, smaller separatrix, which would have displaced X and strike points. Here we present experimental confirmation of these expectations, which are not due to simple plasma motion or deformation. A linearised model of the plasma equilibrium [3] is used to calculate the perturbation to plasma boundary and strike points caused by a large ELM, showing that peeling of a current-carrying plasma layer is a plausible explanation of observed strike point jumps. Typical strike displacements of 5-8cm were observed. As the time resolution of the diagnostics improved, it appears that in fact the ELM phenomenology can be more complex. Often the sudden (100 ms) upward shift in strike position (large!) is followed by an equally sudden downshift to an intermediate position. We are presently considering a more detailed model of the ELM, in which the toroidal current density near the X-point in the pre-ELM state is transferred to the newly formed private region, thereby transiently increasing the potential upward strike jumps. When those charges arrive at the target, the current in the private region will be dissipated and the strike points would move downwards, to a position determined by the balance of current between peeled plasma and the divertor coils. [1] J. Lingertat, B. Alper, S. Ali-Arshad et al., 22nd EPS Conf. on Contr. Fus. and Plas. Phys., Bournemouth, 3rd-7th July 1995, Part 3. p.281. [2] Emilia R Solano, Plasma Phys. Control. Fusion 46 L7-L13 (2004) [3] R. Albanese, F. Villone, Nucl. Fus. 38, 723- 738 (1998)

EX/P1-4 · Small ELM regimes with good confinement on JET and comparison to those on ASDEX Upgrade, Alcator C-mod, and JT-60U

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Abstract: Since it is still uncertain if ITER operation is compatible with type-I ELMs,the study of alternative H-mode pedestals is an urgent issue. The paper reports on experiments on JET aiming to find scenarios with small ELMs and good confinement as the type-II ELMs in ASDEX Upgrade, the enhanced D-alpha H-mode at Alcator C-mod or the grassy ELMs on JT-60U. The study includes shape variations, especially the closeness to a double-null configuration, variations of q95, density and beta poloidal. Steady-state H-mode pedestals without type-I ELMs have been observed only at lowest currents ($\leq$ 1.2 MA), showing similarities to the observations in the devices mentioned above. These will be discussed in detail on the basis of edge fluctuation and pedestal analysis. For higher currents, only the mixed type-I/II scenario is observed. Although the increased inter-ELM transport reduces the type-I ELM frequency, a single type-I ELM is not reduced in size. Differences in the edge behaviour as compared to the staedy-state low-current cases is discussed as well. Obviously, these results do question the existence of such small ELM scenarios on ITER, except perhaps the high beta-poloidal scenario at hig q95, which could not be tested at higher currents at JET due to limitations in heating power.

EX/P2-1 · JET RF dominated scenarios and Ion ITB experiments with no external momentum input

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Abstract: Advanced Tokamak scenarios include two different regimes: "steady state" (characterized by the presence Internal Transport Barrier: ITB) and the "hybrid" scenario. So far both the regimes have always been obtained in presence of strong injection of external momentum by Neutrals Beam Injection (NBI) heating. By using Lower Hybrid Current Drive (LHCD) to sustain the central q slightly above one and with a large plasma region having the magnetic shear close to zero, an "hybrid scenario" has been established, for the first time, in discharges with dominant Ion Cyclotron Resonance Heating (ICRH) and with a normalized beta close to two. By starting from a configuration with reversed magnetic shear (sustained only by LHCD) and with a well established ITB on the electron specie, an ITB also on the ions specie has been obtained by using ICRH in an ion heating scheme, (3He)D. No external momentum input was provided by the NBI, except for the charge-exchange and the MSE beams. In these discharges the evaluated ExB shearing rate was always very small and lower than analytical evaluations of the turbulence growth rate.

EX/P2-5 · Development of Integrated Real-Time Control of Internal Transport Barriers in Advanced Operation Scenarios on JET

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Abstract: An important experimental programme is in progress on JET to investigate plasma control schemes which could enable advanced tokamak operation scenarios to eventually provide steady state burning plasmas in ITER. In particular, we have recently developed a multi-variable model-based technique for the simultaneous control of the current and pressure profiles in discharges with internal transport barriers (ITB), using lower hybrid current drive (LHCD) together with neutral beam injection (NBI) and ion cyclotron resonance heating (ICRH). The proposed control scheme relies on the experimental identification of a linearized integral model operator and retains the intrinsic couplings between the plasma parameter profiles, as well as their distributed nature. A first set of experiments was performed in the low-density/low-power LH-driven phase of the JET advanced scenarios, using only one actuator (LHCD) and a lumped-parameter version of the algorithm. Several requested steady state magnetic equilibria (defined by the values of the safety factor at 5 specified radii) were thus obtained and sustained for about 7s, up to full relaxation of the ohmic current throughout the plasma. Then, more interestingly in view of high power operation, a second set of experiments was dedicated to the control of the q-profile during the intense heating phase of advanced scenarios. The safety factor profile was also shown to approach a requested profile (again defined by its values at 5 radii) within about 5s. The achieved plasma equilibrium state was close to steady state. Finally, during the recent high power experimental campaign, experiments have been conducted in a 3T/1.7MA plasma, achieving for the first time the simultaneous control of the current density and electron temperature profiles in ITB plasmas. Here, the distributed-parameter version of the algorithm was used, with 3 actuators (LHCD, NBI and ICRH), and 8 output parameters [the profiles are projected upon 5 cubic-spline basis functions for the inverse safety factor, and 3 piecewise- linear functions for the normalized electron temperature gradient profile]. Real-time control was applied during 7s, and allowed to reach successfully two different target q-profiles - a monotonic and a reversed-shear one, respectively - and different ITB strengths quantified by their normalized electron temperature gradient.

EX/P2-7 · Stationary, High Bootstrap Fraction Plasmas in DIII-D Without Inductive Current Control

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Abstract: Operation at high β_N and β_p without inductive current drive is necessary for steady-state tokamak reactors. The constraint of self-consistency reduces the possibility of external optimization and raises questions of current control, pressure control, and stability limits. To address these questions, DIII-D experiments with stationary plasmas but without transformer induction have reached $\beta_N \approx \beta_p \approx 3.0$ with $f_{bs} > 75\%$. These conditions have been maintained for >2.2 seconds, with $I_p \approx 0.65\,\mathrm{MA}$ and $\beta \approx 1.5\%$. There is intermittent ITB formation in all channels (n_e, T_e, T_I, V_ϕ). Improved confinement and higher beta associated with the ITBs leads to current overdrive (>50 kA/s). The self-consistent plasma state has a broad current profile, with low internal inductance ($\ell_i \leq 0.6$) and weakly inverted q. The amplitudes of the current and pressure variations increase as beta is raised; at $beta_N \sim 2.8$, the rms current and

energy variations are ∼1.5% and 6%, respectively. As such excursions may be unacceptable in a reactor, transformer control of the current may be a necessity.
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EX/P2-8 · Improved electron confinement in negative magnetic shear NSTX plasmas

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Abstract: Electron transport is dominant in most beam heated plasmas in the National Spherical Torus Experiment (NSTX), while ion transport approaches neoclassical levels. A regime of decreased electron transport is nevertheless observed in low density plasmas heated by early beam injection, where the core chi-e decreases several times. Analysis of the ultrasoft X-ray and magnetic fluctuations, as well as TRANSP modeling indicates a reversed q-profile in this regime. The micro-stability analysis of these plasmas with the GS2 code predicts suppression of the Electron Temperature Gradient (ETG) instability by negative magnetic shear, suggesting that ETG activity might be causing strong electron transport in NSTX. Concurrent with the chi-e reduction, the core chi-i and impurity diffusivity increase a few times. The GS2 calculations nevertheless predict that long wavelength instabilities should still be suppressed in this regime. A proposed explanation for the increased ion transport may thus be the persistent MHD activity in the TAE frequency range observed in the core of these discharges. Despite the increased ion transport, the large reduction in the dominant electron channel increases the energy confinement time to values significantly exceeding the L-mode scaling.

EX/P2-11 · Core Heat Transport in the MAST Spherical Tokamak

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Abstract: The spherical tokamak (ST) provides a unique environment for the study of core transport mechanisms in high-beta, low aspect ratio plasmas, where it is predicted that anomalous transport might easily be suppressed to the underlying neo-classical level by the high intrinsic ExB flow shear. The availability of state-of-the-art imaging and profile diagnostics is allowing analyses of core transport in MAST. In quasi-stationary, H-mode plasmas at the mid-radius region, where transport is not perturbed by sawteeth or ELMs, both the ion and electron heat diffusivities are found to be within a factor 1-3 of the ion neo-classical level. Work is progressing towards a physics-based understanding of the transport in these plasmas using the gyro-kinetic code GS2. Mixing length estimates based on calculated linear growth rates of ITG modes are comparable with the observed ion thermal diffusivity but for ETG modes these are too small to account for the electron thermal transport. Challenging non-linear calculations are also underway to investigate possible mechanisms of anomalous electron transport. ITBs can be formed in MAST by early NBI heating of low-density plasmas during a current ramp to produce weak or reversed magnetic shear and strong driven toroidal rotation. With co-NBI ion thermal transport is suppressed to the neo-classical level at r/a ∼ 0.4, the location of the maximum ExB flow shear and minimum q. Electron thermal transport is also reduced in this region. With counter-NBI heating the ExB flow shear is higher than with co-NBI heating and the region of low magnetic shear is broadened. Under these conditions a strong electron ITB is produced at r/a ∼ 0.7, where the electron thermal transport is reduced to the ion neo-classical level. Results from these experiments have contributed to a multi-machine assessment of various ITB existence criteria, where it is found that these are not universally applicable. Further detailed studies of the core heat transport in MAST are planned using diagnostics with higher spatial and temporal resolution, including its scaling with engineering and dimensionless parameters. Detailed transport analysis of these experiments and comparison with results of micro-stability calculations will help elucidate the processes and dynamics of core transport in the ST.

EX/P2-12 · Comparison of Transient Electron Heat Transport in LHD Helical and JT- 60U Tokamak Plasmas

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Abstract: Transient transport experiments are performed in plasmas with and without Internal Transport Barrier (ITB) on LHD and JT-60U. In the no-ITB plasmas, two different types of non-linearity of the electron heat transport are observed from cold pulse propagation. The heat diffusivity depends weakly on both of electron temperature and temperature-gradient in LHD, while the heat diffusivity depends strongly on the temperature-gradient in JT-60U. In the ITB plasmas, a cold pulse growing driven by the negative temperature dependence of heat diffusivity is commonly observed both in LHD and JT-60U.

EX/P2-14 · Neutral Beam Injection Heating in the TUMAN-3M

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Abstract: The neutral beam injection heating (NBI) is aimed on increasing experimental resources of the TUMAN-3M tokamak. The NBI setup allows tangential launch of H or D beam with power up to 500 kW and beam energy up to 25 keV. Main purposes of the NBI experiment are exploration of high beta plasma stability, study of improved confinement regimes formation under conditions of rotational momentum input and comparison of transport barrier formation in plasmas with various Te/Ti. In order to model the effect of NBI heating on plasma parameters the transport simulations with ASTRA code were performed. The simulations of absorbed, shine-through and orbit loss powers prove that up to 95% of the injected power can be absorbed by the plasma with the average density of $4 \times 10^{19} m^{-3}$. The NBI power exceeds by a factor of 2.5 the ohmic power in a typical discharge and is by an order of magnitude higher then the L-H transition threshold power estimated according to the ITER scaling. This gives a confidence that the H- mode can be achieved and studied in the TUMAN-3M with the NBI heating. The first NBI campaign is planned for spring 2004. The campaign will be focused on beam absorption and ion temperature measurements at various target plasma densities. This will allow validation of the transport simulations. Power balance with different Te/Ti ratios will be studied as well.

EX/P2-15 · Investigation of the Dynamics of Accelerated Compact Toroid Injected into the JFT-2M Tokamak Plasmas

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Abstract: Accelerated compact toroid (CT) injection has been developed as one of the advanced particle refuelling methods for fusion plasmas. In recent CT injection experiments in the JFT-2M tokamak (R=1.31 m, a=0.35 m) using the improved CT injector, rapid increase in core electron density within about 0.06 ms was clearly observed. The fuelling efficiency is estimated to be about 25 %, which is deduced from the increment of the line averaged electron density, $n_e \sim 0.2 \times 10^{19} m^{-3}$and the CT particle inventory of $\sim 1.5 \times 10^{19}$. This is the first observation of the density increase on such a fast time scale, which is an encouraging demonstration for deep fueling of a medium size tokamak discharge by CT injection. Two types of MHD fluctuations have been observed after CT injection. The first fluctuation lasts for 0.02-0.04 ms which agrees well with the time scale of CT injection. Time-frequency analysis shows that the magnetic fluctuation induced by the CT has the maximum spectral peak of around 280 kHz. The estimation of Alfvén frequency based on the observed chirping indicates the excitation of Alfvén wave by CT injection. The latter fluctuation with larger amplitude but lower frequency occurs ∼1 ms after CT injection and is dominant by the m=2/n=1 tearing mode. We have observed that the fluctuation of the ion saturation current is clearly reduced by CT injection, and correspondingly the D_α spectral line intensity decreases, suggesting that the confinement may be improved because the CT may assist NBI with L-H transition.

EX/P2-16 · Influence Of Electrode Biasing On Plasma Parameters In The Tcabr Tokamak

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Abstract: Confinement improvement was obtained in the TCABR tokamak applying radial electric fields using a polarized movable electrode. The experiment was performed with plasma current of 90 kA (q = 3.1), electron density of 1.0×10^{19} per cubic meter and hydrogen gas injection necessary to keep the density constant. Time profiles of loop voltage, plasma current, Halpha and poloidal beta were measured with the electrode inserted 2 cm inside the plasma and biased to 470 volts for 20 ms, and zero volts. The comparison of the profiles shows an increase of the density, by a factor 2.7, while Halpha stays even bellow the values with no biasing. The plasma current changed only 3% and the loop voltage, 7%. The ion saturation current and turbulence, measured behind the limiter, decreased strongly with a quenching time of about 50 microseconds. The profile of poloidal beta show an increase by a factor of 3, compatible

with the density increase. Spectroscopic measurements of the 464.74 nm spectral line intensity of CIII did not show any abnormal intensity increase. MHD activities were detected using a set of 22 Mirnov coil, poloidaly displaced, and the analysis, using FFT, indicated that the predominant modes are m=2 and 3 with an increase in the intensity of mode m = 2 and a decrease of 15% in the frequencies, under biasing. Taking the energy balance in the stationary regime it can be shown that the confinement increases by a factor corresponding to the ratio of the densities. The dispersion equation for the electron drift frequency could explain the decrease of the frequency of the MHD modes admitting an increase of a factor 2 in the density gradient, supporting the hypothesis of the creation of a transport barrier. Another indication of better confinement in TCABR with biasing is the strong decrease of the turbulence in SOL, detected with a Langmuir probe and the calculation of the particle flux induced by electrostatic turbulent fluctuations, which showed a decrease in the total transport, by a factor of 6.9. The density begins to increase 2.8 ms after the bias triggering, but the decrease in recycling and the increase in the confined energy respond faster. The turbulence quenching in about 50 microseconds is in agreement with theoretical predictions that the transition from L to H mode occurs in less than 100 microseconds.

EX/P2-17 · Characteristics of the TPE Reversed-Field Pinch Plasmas in Conventional and Improved Confinement Regimes

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Abstract: We present the characteristics and experimental scaling laws of reversed-field pinch (RFP) plasmas, which are obtained from the recently established toroidal pinch experiment (TPE) database. The database contains information for approximately 1500 discharges, which cover two decades of RFP experiments at National Institute of Advanced Industrial Science and Technology under conventional operating conditions, consistently selected from TPE-1RM (R/a = 0.5/0.09 m), TPE-1RM15(R/a = 0.70/0.135 m), TPE-1RM20 (R/a = 0.75/0.192 m), and TPE-RX (R/a = 1.72/0.45 m, the present device), where R and a are the major and minor radii of plasma, respectively. We present the results of our study on increased confinement improvement and the physics in pulsed poloidal current drive (PPCD) discharges in the TPE-RX RFP device. A comparison of the improved energy confinement time, tau(E), with the reference scaling law of the TPE database is attempted. The result shows that tau(E) in PPCD agrees well with the TPE scaling because of the strong pinch parameter dependence in the TPE scaling law of tau(E). A potential improved confinement mode in the quasi-single helicity (QSH) state is also investigated in TPE-RX regarding the operation conditions where the QSH spontaneously appears, with a typical island structure observed in soft X-ray tomography.

EX/P2-20 · Experimental and theoretical studies of active control of resistive wall mode growth in the EXTRAP T2R reversed-field pinch

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Abstract: Experimental and theoretical studies of active feedback control of resistive wall modes (RWMs) have been carried out. Control of RWMs has been demonstrated on the EXTRAP T2R reversed-field pinch experiment. The T2R device has a resistive wall and has previously been used for studies of the stability of RWMs under different equilibrium conditions. The focus of the programme has now turned to development of methods for active feedback control of the RWMs. In the closed feedback loop that has been tested, the sensor is a 4x32 array of magnetic coils (mode harmonics) and the actuator is a 4x16 saddle coil array (control harmonics). Both an analogue controller (intelligent shell) and a digital controller (real time Fourier transform, feedback law, inverse Fourier transform, distribution of saddle coil current) have been tested. The growth of the dominant RWMs can be suppressed by feedback using both the intelligent shell and digital control systems. However there are a number of important factors that affect the feedback. Field errors can affect the marginally stable region of the RWM spectrum and also the feedback control of the modes. Also, because the number of toroidal positions in the saddle coil array is 16, it is predicted by theory and also shown experimentally that the control harmonic spectrum has sidebands. Therefore not all relevant modes can be controlled separately. Theoretical models based on linear theory and on DEBS numerical code simulations have been devloped for the feedback stabilisation of the system including the sideband mode coupling effect. Experimental studies are presented that demonstrate the control of RWM growth, the effect of field errors on the RWM spectrum and the mode coupling effect due sidebands.

EX/P2-21 · Optimizing the Beta Limit in DIII-D Advanced Tokamak Discharges

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Abstract: Results are presented from comparisons of modeling and experiment in studies to assess the best choice of discharge shape, q profile and pressure profile for high beta, steady-state, noninductive advanced tokamak operation. This is motivated by the need for high safety factor (q) and normalized beta (β_N) to maximize the bootstrap current, the requirement that the q profile, pressure profile and bootstrap current profile be self- consistent, and the close coupling between the discharge shape and the geometry of the divertor cryopumps used to reduce density for maximum EC-driven current. In the experiment, increases in achievable β_N are obtained through broadening of the pressure profile and use of symmetric double-null divertor shape. The general trend is for β_N to decrease as the minimum q value (q_{min}) increases, but with broad pressure $\beta_N = 4$ is obtained at q_{min} near 2 and $\beta_N * q_{min}$ increases with q_{min}. Modeling of equilibria with near 100% bootstrap current indicates that operation with $beta_N$ near 5 should be possible with a sufficiently broad pressure profile.
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EX/P2-22 · Resistive Wall Mode Studies in JET

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Abstract: In advanced tokamak operating scenarios, such as those foreseen for ITER and compatible with the steady-state operation of a power plant, the ultimate performance limit is set by resistive wall modes (RWMs). The nature of the plasma damping term governing RWM stability is not unambiguously established. A damping model based on ion Landau damping represented through a parallel viscosity term has been used extensively, but recently a more accurate 'kinetic' model, based on drift-kinetic theory, has been implemented in the MARS stability code. The damping of stable RWMs may be determined experimentally by measuring the response to n=1 helical magnetic perturbations from coils external to the plasma - in JET, saddle coil systems both internal and external to the vacuum vessel are available for such studies. The resonant field amplification (RFA) has been measured for both DC and AC applied magnetic perturbations. RFA is observed in JET as beta increases, particularly beyond the no-wall limit and good agreement with MARS is found for either the kinetic damping model or for strong ion Landau damping. The occurrence of a critical flow velocity below which the RWM becomes unstable can also be compared with modelling. Magnetic braking is used to slow the plasma until a naturally unstable mode occurs. Comparison of the critical velocity with MARS modelling again shows reasonable agreement with the kinetic damping model or strong ion Landau damping. The results presented provide a very important experimental validation of RWM damping models - it should be noted that the kinetic model involves no free parameters and so can unambiguously be applied to make predictions. For ITER it is found that the observed strong damping leads to a requirement for a flow of ∼2 to 3% of the Alfven velocity at the plasma centre to stabilise the RWM. This work was performed under EFDA and partly funded by Euratom and the UK EPSRC.

EX/P2-26 · The role of flow and q profile in internal kink saturation in NSTX

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Abstract: NSTX has achieved both high toroidal beta up to 35%, and high normalized beta approaching 7. In the highest toroidal beta discharges obtained at high $I_p/aB_T > 6MA/mT$, the central safety factor is typically near 1 and the limiting modes are most often n=1 pressure- driven internal kink instabilities. However, in the highest-beta discharges of NSTX with normalized $\beta = 5-6$, these instabilities can saturate or even decay in amplitude for beta values well above the no-wall limit at mode onset. An important element in this saturation appears to be the high toroidal rotation velocity and strong rotational shear in NSTX plasmas with neutral beam heating. Stability analysis with the MARS and M3D codes indicates that growth times as long as 5ms are possible, and this relatively slow growth rate may contribute to reducing the deleterious impact of the mode on confinement. However, M3D simulations carried into the non-linear phase find that the flow profile typically flattens during the reconnection process allowing

complete magnetic reconnection to occur. Similar flattening is observed in the experiment, yet for the highest-beta cases in NSTX, complete reconnection is apparently avoided.

EX/P2-27 · Study of runaway electron generation process during major disruptions in JET

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Abstract: Analysis of large number of JET disruptions provided further data on the trends of the disruption induced runaway process in large tokamaks. The role of primary runaway electrons generated at the thermal quench has been examined to assess their influence on secondary avalanching, which is recognized as a main source of large runaway currents created during disruptions. The inverse reconstruction of soft X-ray emission during the thermal quench has made possible the observation of the magnetic flux geometry evolution and to locate the most probable zones for generation and confinement of the primary runaway electrons. Runaway currents have been found to increase with toroidal magnetic field and pre- disruption plasma current values. The average conversion efficiency is approximately 40- 45% in a wide range of the plasma currents. This agrees well with results of numerical simulations, which predict similar conversion rate at assumed post-disruption plasma electron temperature 10 eV. The experimental trends and numerical simulations show that runaway electrons might be an issue for ITER and, therefore, it remains prudent to develop mitigation methods, which suppress runaway generation.

EX/P2-28 · Nonthermal electrons and small-scale plasma perturbations during density limit disruptions in the T-10 tokamak

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Abstract: Small-scale quasi-coherent oscillations of the non-thermal x-ray emissivity are identified in high density plasmas in the T-10 tokamak with the use of a toroidal view x-ray imaging system and in-vessel hard x-ray detector. Small-scale x-ray oscillations are associated with forward bremsstrahlung emission of the suprathermal electrons (25 - 100 keV) from the plasma region where the m=2,n=1 magnetohydrodynamic (MHD) mode is localized. The oscillations are typically represented by a solitary high frequency (15 - 85 kHz) harmonic in the spectrum of perturbations accompanying MHD modes during density limit disruptions. During a disruption energy quench, the small-scale oscillations are often transformed to repetitive bursts of the nonthermal x-ray radiation localized at the outer part of the plasma. Analysis indicates a possible connection of the oscillations with nonthermal electron beams induced during magnetic reconnection around X-points of the m=2,n=1 magnetic islands. It is argued that the small-scale oscillations can be produced due to modulation of the electron beams moving though an equilibrium magnetic field with ripples during rotation of the m=2,n=1 mode.

EX/P2-32 · MHD instabilities leading to disruption in JT-60U reversed shear plasmas

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Abstract: High performance reversed shear (RS) discharges with strong internal transport barrier (ITB) disrupt frequently even at low beta. It is found that this low beta disruption is triggered by mode coupling of the surface MHD instability driven by peripheral large plasma current and the internal mode in the RS region at the rational surface whose safety factor correspond to the mode number of the surface mode. Most of observed disruptions can be categorized into two processes. One is that the internal rational surface is changed discretely from in the region of small pressure gradient to the region of large pressure gradient by change of the correspondence surface mode number due to change of plasma current. This process occurs when surface safety factor is integer during current ramp up. The other is that the pressure gradient of the internal rational surface in the RS region is changed continuously.

EX/P2-33 · Compatibility of Reduced Activation Ferritic Steel Wall with High Performance Plasma on JFT-2M

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Abstract: In JFT-2M, applicability of reduced activation ferritic steel to a fusion demonstration reactor has been demonstrated mainly concerning with ferromagnetic effect on plasma production, control, stability, and confinement. No adverse effects had been reported in the previous IAEA conference at relatively far wall position. As a progress of this study, the compatibility of closer ferritic wall with high-normalized beta plasma has been mainly investigated because the demo-reactor will utilize wall stabilization effect. Due to modifications of operation scenario and limiter configuration, the region showing good compatibility was extended for closer wall position (normalized wall position$\sim$1.4) and higher normalized beta ($\sim$3.5). Reduction of growth rate of the instability was also observed in the close wall case, which presumably corresponds to wall stabilization effect, similar to resistive wall without ferromagnetism. Behavior of low beta tearing mode and L/H transition power were also investigated, showing no adverse effect related to ferromagnetism.

EX/P3-2 · Plasma Performance Improvement with Neon Gas Puffing in HT-7

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Abstract: The Neon gas puffing for the production of a radiative layer near the plasma edge with the energy and particle confinement improved has been investigated in HT-7 during the 2003 campaign. From the peaked electron temperature and broadened density profiles formed in these discharges with combined LHCD and IBW heating, it can be seen that the central electron temperature was increased by nearly 50%, compared to those discharges with the same plasma parameters and injected power without the Neon gas puffing. Discharges with Neon gas puffing in HT-7 exhibit a drop in edge electron temperature immediately after the initial neon gas puffing. This latter phase is characteristic with best operation to date in sawtoothing discharges. Soft-X array and Hard-X ray measurements show the profile of X-ray intensity in these discharges. Neon impurity concentrations in these discharges were of order 1%, producing only a moderate increase in Zeff (0). During discharges with the neon gas puffing, Quasi- steady state plasma with $b_{NH89} > 1$ has been obtained in HT-7. These discharges also exhibited a relatively higher normalized inductance compared to reference discharges without the Neon Gas puffing. For long pulse higher performance discharges in HT-7, it is characterized by a large radiation emission from the injected impurities: up to 70% of the total input power can be radiated, creating a cold radiating mantle at the plasma boundary, thus strongly alleviating the heat load problem on the first wall materials. Furthermore, the impurity profiles, fluctuation behavior, and transport analysis with Neon gas puffing in HT-7 will be presented. Future experiments are planned to extend operation to higher densities ($ne/nGW \sim 1$), including extending RI–mode to higher normalized parameters, will also be discussed.

EX/P3-3 · Rapid eITB formation during magnetic shear reversal in fully non-inductive TCV discharges

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Abstract: TCV's high power ECRH system with its flexible launchers is used to tailor the current profile and to generate hollow current profiles with off-axis co-electron cyclotron current drive (ECCD). In this manner, steady-state electron internal transport barriers (eITB) have been obtained where the confinement is strongly enhanced over L-mode scaling. eITBs may be parametrised by the strength of the barrier, i.e. the depth of the hollow current profile, and by its volume, i.e. the region enclosed within the barrier. The formation of the eITB has been found to occur on an even faster time scale than the energy confinement time ($\approx 2ms$). The barrier is formed off-axis and causes the confinement to improve throughout the core of the plasma. The fast increase in confinement occurs without any change in external parameters and without any additional heating deposited in the core. A simplified model of the current density profile evolution from a peaked to hollow profile that correlates the barrier formation with the inversion of current profile, along with the evidence that the barrier is formed in a very localized zone (both temporal and spatial), will be discussed in detail. After the barrier is formed, its strength can be controlled by adjusting the depth of the central current trough. For example, small amounts of ohmic power ($\sim$3 kW) are used to drive up to $\sim$50% counter current (relative to the total current), resulting in further confinement enhancement of up to 50%. Conversely, ohmic co-current in the center has been shown to weaken and even destroy

the barrier. These experiments demonstrate that the experimentally observed confinement enhancement associated with the eITB strongly depends on the depth and width of the current trough in current density profile.

EX/P3-4 · Characterisation of the H-mode Edge Barrier at ASDEX Upgrade

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Abstract: The scaling of the edge transport barrier (ETB) that sustains H-mode plasmas is crucial for the performance of next step tokamaks. At ASDEX Upgrade, the suite of edge diagnostics has been significantly improved so as to address this issue. High spatial resolution profiles of most of the key edge plasma parameters necessary to determine the MHD stability are now available. New high temporal resolution measurements give clear indications of the nonlinear evolution of the ELM crash. Profiles of the edge turbulence are correlated with the radial electric field shear using a new Doppler reflectometer system. The measured pressure gradient in the ETB is found to be consistent with ideal MHD stability limits, both for Type I and II ELMs. In addition, the edge electron temperature and density gradient lengths are found to be strongly correlated, leaving only the ETB width as a free parameter. Using the experimentally observed fact that the ETB width varies little across the ASDEX Upgrade operating range, a model is developed which reproduces the observed ETB size.

EX/P3-8 · H-mode transition physics close to DN on MAST and its applications to other tokamaks

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Abstract: ELMy H-mode is the base-line operating scenario for the next step fusion device ITER. To improve active and passive pedestal control a deeper understanding of H- mode physics is desirable. MAST contributes towards this understanding with good edge diagnostics, and by accessing extreme parameter regimes. The first inter-machine comparisons with respect to the influence of the magnetic topology on the power threshold with ASDEX-Upgrade and NSTX reveal a reduction of the power threshold in true double null (C-DN) configuration opening new operation regimes in both devices. The 30% reduction in threshold power close to C-DN observed on ASDEX-Upgrade, though significant, is less than the factor of two or more observed in both large spherical tokamaks, MAST and NSTX. This points towards the importance of field line curvature for this effect. The power thresholds measured in C-DN on MAST and NSTX are very similar. Despite this strong effect on the power threshold, changes in most edge parameters in L-mode due to the different magnetic configurations are small. However, significant changes are seen in the toroidal impurity flow velocity, related to the radial electric field, and in the scrape-off-layer temperature decay length at the high field side. The statistical comparison of MAST data with various H-mode theories suggests that different instabilities need to be stabilised at different spatial positions in the region where the pedestal forms to access H-mode. Pedestal temperatures observed on MAST are two to five times lower than in MAST equivalent discharges at ASDEX-Upgrade. However, the pedestal densities are similar. The differences in L-mode are less significant. The usual DN operating regime with co current NBI in MAST has been extended to include single null (SN) configurations, to provide more direct comparisons with conventional tokamaks. The plasma edge in SN on MAST is more stable to ELMs and the typical type-III ELMs, often observed in C-DN, are absent, despite input powers close to the H-mode threshold power. In this respect, the role of the suppression of the eigenfunction of ideal MHD modes in the vicinity of an X-point, which leads to a strong localisation of the mode on the bad curvature, low field side in C-DN will be discussed. Furthermore, first counter current NBI H-modes in STs have been developed in anticipation of QH-mode.

EX/P3-10 · Understanding of the density profile shape, electron heat transport and internal transport barriers observed in ASDEX Upgrade

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Abstract: Density profiles: peaking is observed to increase with decreasing collisionality, and explained through the effect of collisions on the trapped electron response of the (combined) Ion temperature Gradient (ITG) and trapped electron mode (TEM). Density pump out through electron heating is observed only in the case in which the TEM is dominant, and is due to the reversal of the sign of the temperature

gradient driven flux with the reversal of the mode frequency. This leads to the prediction of a peaked density profile in a reactor. Electron heat dominated plasmas: Linear GS2 calculations show that the TEM is dominant, and sensitive to collisionality. At higher density (collisionality) a transition to an ITG dominated case occurs, in which the incremental electron heat transport is reduced, in agreement with observations. Transport barriers: The stabilisation of the ITG through a uniform electric field is shown to vanish when this field is connected with a toroidal rotation, i.e. it does not explain the barrier. Global simulations show that zero magnetic shear does not lead to any stabilisation. The ExB shearing rates do not always explain the existence of the barrier.

EX/P3-11 · Development of Internal Transport Barrier scenarios at ITER-relevant high triangularity in JET

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Abstract: The development of ITB scenarios in high triangularity discharges is of particular interest for ITER AT operation. Previous JET experiments have shown that high triangularity favours ELM-free or type I ELMs, which inhibit long lasting ITBs. The recent experiments reported here concentrate on integrated optimisation of edge and core conditions. Edge pedestal was controlled using gas injection, Deuterium or light impurities, and plasma current ramps. Both methods produced ITB-friendly type III ELMs. In parallel, the conditions for triggering and sustaining a wide ITB were optimised. This plasmas have deeply reversed target current profiles with qmin $\sim$ 3. Wide ITBs are triggered when the input power exceeds 20-22 MW. The best results, in terms of sustained high performance, have been obtained with Neon injection: a wide ITB is triggered during the L-mode and survives into H-mode for about 2s at H89bN $\sim$ 3.5 and $\sim$ 60% of the Greenwald density limit. In summary, a high triangularity scenario has been developed, which combines the desirable characteristics of controlled edge, long lasting wide ITBs and high performance at density higher than the low triangularity JET scenarios.

EX/P3-12 · Transition Phenomena and Thermal Transport Property in LHD Plasmas with an Electron Internal Transport Barrier

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Abstract: Two kinds of improved core confinement were observed during centrally focused Electron Cyclotron Heating (ECH) into plasmas sustained by Counter (CNTR) and Co Neutral Beam Injections (NBI) in LHD. One shows transition phenomena to the high- electron-temperature state and has an electron Internal Transport Barrier (eITB) in CNTR NBI plasma. Another has no clear transition and no power threshold, but shows the improved core confinement with additional ECH in Co NBI plasma. The electron heat transport characteristics of these plasmas were directly investigated by using the heat pulse propagation excited by Modulated ECH (MECH). The difference of the features could be caused by the existence of the m/n=2/1 rational surface determined by the direction of NBI beam-driven current.

EX/P3-14 · Edge Stability and Performance of the ELM-Free Quiescent H-Mode and the Quiescent Double Barrier Mode on DIII–D

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Abstract: The quiescent H (QH) mode, an ELM-free, high-confinement mode, combines well with an internal transport barrier to form quiescent double barrier (QDB) stationary state, high performance plasmas. The QDB achieves performance of $\beta_N H_{89} \sim 7$ in quasi-stationary conditions for a duration of 10 τ_E, limited by hardware. Recently we demonstrated stationary state QDB discharges with no change in kinetic and q profiles ($q_0 > 1$) for 2 s, comparable to that of ELMing "hybrid scenarios", yet without the debilitating effects of ELMs. At high triangularity, QH mode discharges operate at ITER level values of β_N and $\nu*$. In this paper we will report on the progress made on DIII-D in edge profile and stability analysis in QH mode discharges, suggesting that ELM suppression results from a reduction of the edge bootstrap current compared to ELMing phases. We will also report on the advances in performance of the QDB discharges, including demonstrations of plasma profile control.

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EX/P4-1 · Experiments on tokamak ADITYA

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Abstract: It is well known that the Greenwald limit is in reality a limit on edge particle confinement that leads to loss of edge thermal equilibrium. While the radiative collapse is relatively well understood, questions remain about the exact dynamics of convectively driven collapse. More specifically, the scaling of edge temperature with edge density needs to be better understood. We report experimental results from ADITYA to study such a scaling under various operating conditions. Specifically we have actively modified the edge profiles by heating / cooling / gas puffing / impurity injection etc. as well as wall conditioning to examine in detail the density limit scaling. In another set of experiments, we look at thermal quenches and find evidence for two different types of thermal quenches. In one kind saw tooth oscillations disappear while the overriding oscillation (m=1) continues in the SXR plasma temperature data at the time of the thermal quench which occurs in a few milliseconds and the quench recovers in similar time scale to the earlier level of SXR signal. In another type of discharge, it is observed that the signal does not recover till the end of the plasma discharge. To understand the difference between the two types, we have correlated data from different diagnostics. On the basis of that we speculate that in the first type energy ejected out of the m=1 surface redistributes within the m=2 surface. Island size is still within the LCFS in these cases. In the second type, on the other hand, the island size and plasma position are such that the surfaces are destroyed and the energy is leaked out and hence the quench does not recover. Slight increase in the loop voltage signal is observed in this case. Linear mode analysis shows a much faster growth rate for the quench data in type II case. Finally we also observe good correlation between data obtained from soft X-ray and Mirnov coils. Details of these experimental findings with theoretical calculations and interpretations will be presented in this paper.

EX/P4-3 · Stiffness of Central Current and Temperature Profiles in JT-60U Current Hole Plasmas

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Abstract: In an extreme situation of the hollow current profile, a region with nearly zero plasma current ("current hole") is formed and the temperature profile becomes nearly flat in the central part of tokamak plasma. Our concern is whether these profiles, namely small current density and temperature gradients, are the results of small driving force (current drive and heating) or are maintained by some damping mechanism against the driving force. To address this issue, we applied local current drive by EC waves and inductive electric field and applied local heating by EC waves in the current hole to enhance the driving force, and investigated responses of profiles in JT-60U. We found that no toroidal current was driven in the current hole, both in co- and counter-directions to the plasma current, in spite of high electric conductivity. On the other hand, an electron temperature gradient was formed with heating localized on the low-field-side equatorial plane. These results, namely stiff current profile and non-stiff temperature profile, suggest that there are finite poloidal fields to confine electrons and a mechanism to clamp the current density nearly zero in the current hole.

EX/P4-4 · Pellet Ablation in FTU discharges

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Abstract: Fuelling the central plasma region is a major issue in future tokamak experiments. The use of high field side (HFS) pellet injection is considered to be one of the main method for fuelling the central plasma region and obtaining a peaked density profile. Experiments has been carried out on FTU with its vertical injection system A penetration of the particles closer to the plasma center than the 14 cm injection vertical position seems to indicate the presence of some radial drift of the ablated material. Dependence on Te seems to confirm that the radial drift is more effective at higher temperature, a feature that will be confirmed in future FTU experiments.

EX/P4-5 · On the momentum re-distribution via turbulence in fusion plasmas: experiments in JET and TJ-II

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Abstract: The mechanisms underlying the generation of plasma flows play a crucial role in understanding transport in magnetically confined plasmas. The amplitude of parallel flow measured in the scrape-off layer (SOL) including the effects of diamagnetic, $E \times B$ and $B \times \nabla B$ drifts is significantly larger than those resulting from simulations. Recent experiments have pointed out the possible influence of turbulence in explaining a component of the anomalous flows observed in the plasma boundary region. In the plasma core region, evidence of anomalous toroidal momentum transport has been reported in different tokamak devices. Different mechanisms have been proposed to explain these results, including neoclassical effects, turbulence driven models and, in the case of ICRF heating, fast particle effects. Spontaneous toroidal flow not driven by neutral beams has also been observed in stellarator devices. The flow reversal observed in the CHS stellarator can be explained by the spontaneous flow driven by large radial electric fields. The Reynolds stress term, $d < v_r M_{\|} > /dr$, provides a mechanism for parallel momentum redistribution, allowing the turbulent fluctuations to be converted into mean parallel flow. The first experimental evidence of significant radial gradients in the cross-correlation between parallel and radial fluctuating velocities near the LCFS in JET tokamak and in the plasma boundary region of the TJ-II stellarator is reported here. These gradients are mainly due to radial variations in the level of poloidal electric field fluctuations and in the cross-phase coherence. The observations in JET tokamak and TJ-II stellarator have shown that the contribution of the Reynolds stress in the plasma edge region can be larger than charge-exchange damping mechanisms and comparable to the damping due to effective parallel viscosities (dominated by the anomalous perpendicular viscosity). The comparison between the driving and damping terms of the momentum shows that in the plasma edge, the Reynolds stress seems capable of sustaining a significant parallel velocity. The results may be consistent with recent observations in Alcator C-Mod showing that the toroidal momentum propagates in from the plasma edge, without any external source involvement, and highlight the possible role of turbulence in the generation of toroidal momentum in fusion plasmas.

EX/P4-6 · Control of the radial electric field shear by modification of the magnetic field configuration in LHD

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Abstract: Control of the radial electric field, Er, is considered to be important in helical plasmas, because the radial electric field and its shear are expected to reduce neoclassical and anomalous transport, respectively. In general, positive or negative electric field have been obtained by decreasing or increasing the electron density. Although the sign of the radial electric field can be controlled by changing the collisionality, modification of the magnetic field is required to achieve further control of the radial electric field, especially producing a strong radial electric field shear. In the Large Helical Device (LHD) the radial electric field profiles are shown to be controlled by the modification of the magnetic field by 1) changing the radial profile of the helical ripples, 2) creating a magnetic island with an external perturbation field coil and 3) changing the local island divertor coil current. Strong shear of the radial electric field is demonstrated to be produced inside the plasma by controlling the helical ripple, magnetic topology (magnetic island and LID configuration) to trigger an internal transport barrier and reduce impurity influx.

EX/P4-8 · Investigation of the Specific Plasma Potential Oscillations with Frequencies near 20 KHz by Heavy Ion Beam Probing in T-10

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Abstract: Investigations of the specific oscillations with frequencies 15-30 kHz on the T- 10 tokamak (R = 150 cm, a = 30 cm) with Heavy Ion Beam Probe (HIBP) diagnostics were conducted in regimes with off-axis ECRH (B = 2.33 T, I = 220 kA, $r_ECRH = 12cm$). Previous experiments in OH regimes have shown that "20 kHz" modes are mainly the potential fluctuations. These oscillations are seen on the signals of HIBP, Langmuir probes and reflectometry. They should cause the fluctuations of the poloidal rotation, i.e. the torsional plasma oscillations with m=0, called as the zonal flows. The HIBP sample volume was localized at r = 24 - 29 cm. It was observed that both in OH and ECRH regimes, the power spectrum of potential oscillations has a form of solitary quasi-monochromatic peak with the contrast range of 3-5. The

frequency of "20 kHz" mode is varied in the region of observations; it diminishes to the plasma edge from 20 kHz at 22-24 cm till 13-15 kHz at 28- 29 cm. Increase of the frequency was observed, when ECRH was turn-on. The frequency change is correlated with change of the electron temperature Te, measured by 2nd ECE harmonic on the nearest chord 24 cm. Analysis have shown that frequency of the "20 kHz" mode is varied with local Te : $f_{20} \sim Te^{0.5}$, which is similar to a theoretically predicted dependency for Geodisic Acoustic Modes: $f_{GAM} \sim c_s/R \sim Te^{0.5}$, where c_s is a sound speed. The absolute frequencies are close to GAM values within a factor of unity $(1.3 - 1.5)$.

EX/P4-10 · Experimental Test of Neoclassical Theory of Poloidal Rotation in Tokamaks

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Abstract: Rotation plays an important role in the suppression of turbulence and the formation of internal transport barriers through $E \times B$ shear. It is also involved in the stabilization of both resistive wall modes and neoclassical tearing modes. However, momentum confinement remains a poorly understood topic in fusion plasmas. We present a detailed evaluation of the neoclassical theory of rotation, through comparisons of measured poloidal rotation profiles with predictions from NCLASS. A new analysis technique has been devised for the interpretation of charge exchange recombination (CER) measurements, which are affected by systematic errors caused by atomic physics effects. We use a specialized set of CER viewing chords to self-consistently determine the atomic physics corrections directly from the measurements. The poloidal velocity is found to differ by more than an order of magnitude from the neoclassical prediction in a quiescent H-mode discharge. Two reasons for this deviation are investigated, namely the effects of fast ions and core turbulence.
*Work supported by U.S. DOE under DE-AC02-76CH03073, DE-FC02-04ER54698, DE-0AC05-00OR22725, and DE-FG03-01ER54615.

EX/P4-11 · High-ion temperature experiments with negative-ion-based NBI in LHD

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Abstract: High-Z plasmas have been produced with Ar- and/or Ne-gas fueling to increase the ion temperature in the LHD plasmas heated with the high-energy negative-ion-based NBI, where the electron heating is dominant. In the high-Z discharges the ion-density-normalized ion-heating power is much enhanced by a factor of around 5, compared with the hydrogen discharges, by both an increase in the beam absorption rate and a reduction of the ion number. As a result, the ion temperature increases with an increase in the density-normalized ion- heating power, and reaches 10 keV. A large toroidal rotation is observed corresponding to an increase in the ion temperature, and a toroidal rotation velocity of 40 km/s is achieved in a plasma with an ion temperature of 7 keV, which is about 30 % of the Ar-thermal velocity. When the centrally focused ECRH is superposed on the high-Z NBI plasma, the ion temperature is much increased by a factor of 1.5 as well as the electron temperature, suggesting an improvement of the ion transport. Two scenarios for increasing the ion temperature, i.e., to increase the direct ion heating power and to improve the ion transport, are experimentally demonstrated with high-Z plasmas in LHD.

EX/P4-12 · Transport Phenomena In The Edge Of Alcator C-Mod Plasmas

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Abstract: Investigations into particle transport in the edge and Scrape-Off-Layer of Alcator C-Mod have found: 1) in plasmas with Enhanced D_α H-mode confinement, the transport appears to be regulated by the presence of a quasi-coherent (QC) density and potential fluctuation localized within the edge pedestal; and 2) in the SOL, the cross-field transport is convective, dominates the parallel transport, and occurs predominantly as intermittent events that propagate both radially and poloidally. Study of the QC fluctuation shows that it exists in a narrow radial region on the outboard, bad-curvature side, is not present at the inboard midplane, and has a poloidal wavenumber of $1 - 2cm^{-1}$ at the outboard midplane. It is located within the edge pedestal and moves with it. Movies of the edge/SOL turbulence show generation of the intermittent events ("blobs") at and outside of the steep edge gradient region, with clear radial and poloidal propagation. Near the density limit the region of generation is seen to move with

the steep gradient to inside the separatrix. Radial tracking of the events shows that the largest amplitude fluctuations move with radial speeds of $\sim 0.5 - 2km/s$, typically accelerating with radius.

EX/P4-13 · The Greenwald density limit in the Reverse Field Pinch

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Abstract: In the Reversed Field Pinch RFX the density operating space exhibits an upper boundary that is well described by the Greenwald limit. The basic findings are described and similarities and differences with the Tokamak case are analysed. The high-density limit occurs as a non-disruptive limit. Fast terminations have been observed but in a wide area of the operating space and therefore are unlikely to be related to the density limit. Radiation induced thermal collapse can also be ruled out being radiation losses always a relatively small fraction of the ohmic input power. The occurrence of plasma detachment while approaching the density limit can be excluded. The absence of additional heating makes it difficult to distinguish with certainty between a power balance effect and an intrinsic physical limit but the overlapping of the Greenwald limit with the upper boundary of the density operating space is however remarkable, especially in the case of He plasmas . While in Tokamaks one of the plausible causes for the limit is a transport induced edge thermal instability, in RFX, as density increases the density profile becomes hollow, inside the last closed magnetic surface particle diffusion decreases and the global energy confinement time improves. The normalised density fluctuations measured by Langmuir probes and by the outermost chords of an interferometer do not increase. Simulations of the hydrogen discharges with the RITM code confirm the importance of recycling in determining the edge density gradient and the minor role of radiation losses. Differences between H and He cases are analysed in terms of particle penetration capability and in terms of edge ExB shear. A statistical analysis of the edge density fluctuations is presented, looking for differences arising as density is increased. Finally, the possible role of pressure driven modes is discussed.

EX/P4-14 · 1 GJ long pulse control on Tore Supra

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Abstract: With the goal of addressing the critical issue of long pulse steady-state operation of next fusion devices, the experimental programme of the Tore Supra has been devoted in 2003 to study simultaneously current profile control, heat removal capability and particle exhaust. This required both advanced technology integration and steady-state real time plasma control, obtained on Tore Supra with a complete set of measurements and actuators built around a shared memory network. Plasma position control was improved within a few millimetres range, taking in account small changes in the pick up coils temperature. Fully non inductive current drive, in a MHD stable regime, was maintained during more than 6 minutes, with the capability to recover from transient failures of the RF power (arcs in the wave-guides). Infrared imaging of the first wall, analysed in real time, shows a stable surface temperature, as more than 98% of the injected power was recovered in the cooling water loop. A world record of injected-extracted energy, exceeding 1 GJ, was obtained in this new powerful regime.

EX/P4-15 · Investigation of plasma performance in high l_i scenario in HT-7

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Abstract: Fast plasma current ramp down have been used to create different high l_i target plasmas at a rate of –$(0.6 - 1)MA/s$ in HT-7. The LHW pulse was applied before or just before the plasma current is ramped down. The IBW heating was applied to further increase the plasma beta and improve the plasma confinement. With such a scenario, the steady state value of $l_i > 1.5$ was obtained for a duration of several current diffusion times, which is nearly quasi steady state. The global electron heating was observed, but the electron temperature profile was strongly peaked. Highest central electron temperature up to 4.5 keV at line-averaged density of $2.2 \times 10^{19} m^{-3}$ has been obtained by applying the 400 kW LHW at $N_{\parallel} = 2.3$ and 200 kW just before Ip is ramped down from 200 kA to a 120 kA plateau at a rate of –$0.8MA/s$. The ion temperature in such discharge was 1.5 keV. The fraction of the non- inductive current was about 80% of Ip. A stationary improved confinement has been observed in such a high l_i plasma. No impurity accumulation was observed during the improved confinement phase. The global confinement time at lower PLHW and lower density is close to the ITER-89P scaling, but higher than the ITER-89P scaling at higher PLHW and density. The energy confinement time is increased to the level above the ITER-89P scaling when IBW

was applied. The current profile effect on the global confinement has been investigated through changed of the plasma internal inductance, l_i. An increase of the energy confinement time with l_i in the range of 1.2-1.7 is observed at the constant line averaged electron density and the injected power. No MHD activity was detected in the stationary phase, and small sawteeth only existed during the ramp down phase. The high l_i mode provides an alternative operational scenario for the high performance plasma under steady state condition.

EX/P4-17 · Third-harmonic, top-launch, ECRH experiments on TCV Tokamak

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Abstract: In the moderate magnetic field of TCV (1.5T), the recently installed X3 system (3 gyrotrons @118GHz, 0.45MW/each, 2s) broadens the operational space with the possibility of heating plasmas at high density, well above the cutoff density of the X2 system (X2 cutoff at $ne = 4.2 \times 10^{19} m^{-3}$). To compensate the significantly weaker absorption coefficient compared to the absorption of X2, the top-launch injection allows to maximize the ray path along the resonance layer thus maximizing the optical depth. An extensive experimental study has been devoted to the validation of the absorption predicted by the ray- tracing code TORAY-GA. Comparisons of the absorption calculated with the TORAY-GA ray-tracing code and the beam-tracing code, ECWGB, which includes diffraction effects, will be presented. With a target plasma density of $ne = 4 \times 10^{19} m - 3$ (L-mode, ohmic phase) and 1.35MW of X3, nearly full single pass-absorption has been obtained, with the total plasma energy increased by a factor of 2.5. To maintain the maximum absorption in plasma discharges with a dynamic variation of both density (refraction) and temperature (relativistic shift) a real time control system on the injection angle has been developped.

EX/P4-19 · Experimental study of a lower hybrid wave multi-junction coupler in the HT- 7 tokamak

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Abstract: A phase-controlled lower hybrid wave (LHW) multi-junction (MJ) coupler has been developed in the HT-7 tokamak. Studies of the plasma and wave coupling experiment, performed by scanning plasma density and phase-difference between the subwaveguides, show that good coupling efficiencies with low reflection coefficients (RC) below 10% are obtained. The coupling experiments performed by moving the plasma horizontal displacement in one discharge show that the movement of plasma displacement has little effect on the wave-plasma coupling and that during the whole process the reflection remains very low (∼5%). The low reflections suggest that the plasma density in front the grill mouth satisfies the coupling condition. The little effect of plasma displacement on reflection is possibly ascribed to the non-shifted last closed magnetic flux surface (LCFS) determined by the fixed poloidal limiter in our experiments. The investigation of the capability of current drive of the MJ antenna by scanning phase-difference between the subwaveguides and plasma density show that there is an optimized density region where good drive efficiency is obtained, which is mainly interpreted by the wave accessibility and the electron components that interact with the LHW. For a certain density, the drive efficiency for the MJ antenna is higher than that for the traditional one, indicating that the MJ grill is more effective to drive current than the traditional one. Analysis shows that one of the possible candidates for the difference drive efficiencies is that the MJ coupler drives a smaller negative/counter-direction current fraction, hence enhancing the total non-inductive current. Studies suggest that that the newly developed MJ coupler in the HT-7 tokamak is effective to couple the wave power into plasma, to drive current, and to modify current profile, thereby improve plasma confinement. The achievement of stationary plasma longer than 60s suggests that the LHW system is capable of operating at a steady state scenario.

EX/P4-21 · Modelling of Profile Control with LH Wave Injection in the HL-2A Single- null Divertor Plasma

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Abstract: In the HL-2A tokamak a single-null divertor (SND) configuration has been established in recent Ohmic discharges. A fast algorithm is elaborated for identifying the diverted plasma boundary from measurements performed with the pick-up coils and magnetic flux loops. Separatrix of the SND plasma was identified using the algorithm, and the determined striking area on the target plate is in agreement with the measurements of electric probe array. Higher LH power (1.5MW) is injected to the SND plasma with a nearly symmetric spectrum. Since parallel refractive index of the radiated LH power spectrum is rather high, the injected LH wave with lower phase velocity is absorbed in outer region (with peak of the absorbed power at x∼0.65). Plasma heating by electron Landau interaction results in operation scenarios of preferentially dominant electron heating in a low- density plasma. Due to off-axis driven current, an optimized q-profile, of which the magnetic shear is weak in the central region and negative in the mid-plasma region (x = 0.5-0.65), is formed. A steep gradient of the electron temperature is produced around the power deposition region, and the normalized gradient, R/L_T at the steepest gradient (where x = 0.65, and Te = 0.86keV) is 18, which exceeds largely the critical gradient value for temperature profile stiffness. The electron thermal conductivity is reduced significantly at the steep temperature gradient and in the region inside it, showing that an electron-ITB is developed. With higher LH power injecting into a target plasma that is heated by NBI (0.5MW), not only the electron temperature has a large increment, but the ion temperature increases significantly as well in the LH injection phase. The temperature profiles indicate that an enhanced core confinement is established with both ion-ITB and electron-ITB developed, and the analysis of power balance in the ion channel shows that the large increment of Ti within the transport barrier is mainly attributed to the reduction of ion heat transport.

EX/P4-22 · Synergy between Electron Cyclotron and Lower Hybrid current drive on Tore Supra

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Abstract: An improvement of the Current Drive (CD) efficiency of Electron Cyclotron (EC) waves in the presence Lower Hybrid (LH) waves has been predicted by kinetic calculations since the '80s, but not yet observed in stationary conditions. This effect is potentially important because a LH+EC system could combine the much higher efficiency of LHCD with the good localisation of ECCD, to control the profile of the non-inductive current. In order to provide the first experimental demonstration of this synergy effect, dedicated experiments have been performed on Tore Supra. A multiple feedback strategy has been used to realise discharges with a 20 s phase at zero loop voltage, constant plasma current and density. During this phase, a 10 s ECCD pulse has been applied. The additional current driven by the EC waves is determined from the drop of the LH power. This technique has allowed a clear demonstration of the synergy effect in steady state conditions: for instance, 500 kW of LHCD have been replaced by 700 kW of ECCD only. ECCD efficiency improvements of up to a factor of 4 have been measured, in good agreement with kinetic calculations performed by means of a 3-D Fokker-Planck code.

EX/P4-23 · Upper Hybrid Resonance backscattering Enhanced Doppler Effect and plasma rotation diagnostics at FT-2 tokamak

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Abstract: Observations of giant Doppler frequency shift effect of the highly localized microwave BS in the Upper Hybrid Resonance (UHR) are reported. The experiment is performed at FT-2 tokamak, where a new steerable focusing antenna set, allowing off equatorial plane plasma X-mode probing from high magnetic field side, was installed. A separate line less than 1.5 MHz wide and shifted by up to 2 MHz is routinely observed in the BS spectrum under condition of accessible UHR. The enhanced frequency shift is explained by the growth of poloidal wave number of the probing wave in the UHR. The new scheme for local diagnostics of plasma poloidal rotation based on this effect is proposed and benchmarked against Doppler reflectometry, spectroscopy and probe diagnostics data and applied to detailed local study of plasma poloidal rotation profiles in FT-2 ohmic discharges at different plasma density and current and during LH heating.

EX/P4-24 · Plasma Heating and Fuelling in the Globus-M Spherical Tokamak

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Abstract: The results of last two years plasma investigations in Globus-M are presented. Described are improvements helped to achieve high performance OH plasmas, which is used as the target for auxiliary heating and fuelling experiments. Specific features such plasmas exhibited, such as increased energy content, high beta poloidal, good confinement are discussed. Upgrade of neutral beam injector (NBI) is outlined and test NB parameters are presented. Experiments on NBI plasma heating in the wide range of plasma parameters are presented and analyzed. Results of experiments on RF plasma heating in the frequency range of fundamental ion cyclotron harmonics are given. Such experiments are performed for the first time in spherical tokamaks and demonstrated unusual results. Noticeable ion heating was recorded at low launched power and high concentration of hydrogen minority in deuterium plasma. Plasma fuelling experiments with the help of significantly modified double stage coaxial gun are described. Modification increased plasma jet velocity, total number of injected particles and reduced impurity content.

EX/P4-25 · Transport barrier formation and its maintenance by LHCD on TRIAM-1M

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Abstract: Internal transport barrier (ITB) can be obtained in full lower hybrid current driven (LHCD) plasmas on TRIMA-1M ($R = 0.84m$, $a \times b = 0.12m \times 0.18m$, BT<8T). This discharge is classified with the enhanced current drive (ECD) mode, because the current drive efficiency is spontaneously increasing during the discharge. The formation of ITB is caused by the reduction of the lower hybrid (LH) power deposited at the center of plasma. The plasma with ITB can be maintained by the LH power deposited around the foot point of ITB up to 12 sec, which corresponds to more than 50 times of current diffusion time, $\tau_{L/R}$. This indicates that the formation of ITB has the relation to the current density at the foot point of ITB.

EX/P4-26 · Expanding the operating space of ICRF on JET with a view to ITER

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Abstract: The size and capability of JET to confine very energetic particles, together with its versatile ICRF system, provide a unique environment to develop ICRF techniques relevant to the Next Step. During the 2003-2004 experimental campaigns, in addition to further development of ICRF as a tool for the JET experimental programme, several heating and current drive scenarios have been investigated, contributing to the physics understanding and operational expertise required for successful use of ICRF on ITER: heating and current drive in hydrogen plasmas for the low activation phase of ITER, minority heating of tritium, experimental demonstration of Larmor radius effects in the second harmonic heating scenario, direct electron heating and current drive, new results on ICRF heating efficiency. The first part of the paper reports on these advances. The second part presents studies of a more technical nature: studies of ELMs using fast RF measurements, and the proof of principle of a new ELM-tolerant antenna matching scheme. Finally, the last part of the paper reviews the technical enhancements planned on the JET ICRF system, the major component being the new ITER-like antenna.

EX/P4-27 · Formation of Spherical Tokamak Equilibria by ECH in the LATE Device

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Abstract: Main objective of the LATE (Low Aspect ratio Torus Experiment) device is to demonstrate formation of Spherical Tokamak plasmas by ECH alone without center solenoid. By injecting a 2.45 GHz microwave pulse up to 10 kW for four seconds, a plasma current has been initiated and ramped up to Ip=4 kA under the external vertical field controlled for equilibrium of the plasma loop. Magnetic measurements show that an ST equilibrium having the last closed flux surface with an aspect ratio of $R_0/a \simeq 23cm/17cm \simeq 1.35$ and an eleongation of K=1.34, in coincidence with the soft X-ray computer tomography image of the plasma cross section, has been produced and maintained.

EX/P4-28 · ITER Relevant Coupling of Lower Hybrid Waves in JET

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Abstract: Lower Hybrid (LH) is a highly desirable tool for ITER, particularly for the steady state and hybrid scenarios that rely on forming and maintaining a specific q profile. However, coupling of the LH waves in ITER represents a challenge because the launcher, flush with the outer wall, will be at least 12cm away from the last closed flux surface (LCFS), where the electron density (n_e) is predicted to be below the

cut-off density. Also, the plasma will be in H-mode with ELMs that could trigger trips of the protection systems, resulting in reduction of the averaged power. To study solutions to improve coupling of LH waves in these conditions, experiments were performed in JET using plasmas with Internal Transport Barrier (ITB) and H-mode, with distance between the LCFS and the limiter, DPL, up to 10 cm. To increase n(e) in front of the launcher, $n_{e,grill}$), CD_4 or D_2 is puffed from a specially designed pipe, recently modified to optimise the gas flow. When no gas is puffed, the LH coupling is poor during the high power phase, indicating that $n_{e,grill}$ is below the cut-off density. With CD_4 puffing, the coupling improves dramatically, and 2.5 MW of LH power is coupled with DPL= 9cm and type I ELMs. Measurements with a reciprocating Langmuir probe show that $n_{e,grill}$) is at, or slightly above, the cut-off density, in this case. With D_2 puffing, preferable to CD_4 for ITER because of concerns about T co-deposition, 3MW was coupled successfully with DPL up to 10cm. The ELMs are smaller with D_2, but the improvement in coupling can not be attributed only to this. The use of D_2 or CD_4 was explored further in other plasmas with ITB and H-mode. In all the scenarios, the coupling is better with D_2 than with CD_4. Moreover, the ITB existence and performance are not affected by the D_2. To help understand the processes leading to the increase in $n_{e,grill}$), C_2H_6 and C_3H_8 were compared to CD_4 and D_2 in the same scenario. n_e scales with the ionisation cross-section in the case of the hydro-carbide gases, but is higher than expected for the D_2, possibly because of higher D_2 recycling from the walls. This paper also presents results from experiments investigating LH counter current drive in JET, in plasmas with reversed magnetic field and plasma current. Preliminary simulations indicate significant current drive efficiency.

EX/P4-29 · ICR Heating at the Fundamental Frequency of T-11M Hydrogen Plasma

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Abstract: The ion - cyclotron heating of plasma, used at the tokamak T-11M, allows to execute different scenarios. From the possible scenarios of RF plasma heating in experiments the heating of hydrogen plasma at the basic cyclotron frequency was taken as the main scenario. The experiments, carried out at the installation T-11M, have shown, that RF heating of hydrogen plasma at fundamental frequency is effective. Temperature of hydrogen atoms in these regimes increased in 2-2,5 times. In a number of regimes the increasing of electron temperature was registered. In regimes with relative concentration of deuterium less than 2 % the straight hydrogen heating at fundamental frequency implemented, to what the essential deviation of energy spectrum of hydrogen atoms from Maxwellian testifies. The experiments on RF heating of plasma at fundamental frequency will be continued after boronization of the tokamak chamber, and their results will be reflected in the paper.

EX/P4-32 · Mode Conversion, Current Drive and Flow Drive with High Power ICRF Waves in Alcator C-Mod: Experimental Measurements and Modeling

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Abstract: Recent results from high power ion cyclotron heating experiments (ICRF) in the Alcator C-Mod tokamak are reported. In particular, mode converted waves in the vicinity of the ion-ion hybrid layer have been measured by means of Phase Contrast Imaging techniques (PCI). The measured k-spectrum and spatial location of the waves is in agreement with theoretical predictions as well as with the full-wave TORIC code modeling results. For appropriate ion species (i.e., H-He3 in C-Mod, which is equivalent to D-T in ITER), in tokamak geometry mode-conversion of the fast magnetosonic wave into the electromagnetic ion cyclotron waves (ICW) dominates over ion Bernstein waves (IBW). The waves propagate back toward the antenna on the low field side, in contrast to what is expected for IBW wave propagation. Experimental verification of such code predictions is essential for reliable predictive capability in future burning plasma experiments (ITER). In agreement with experimental observations, the TORIC results verify that the RF power is mainly converted to the ICW rather than IBW. In the C-Mod experiments no evidence of mode conversion into IBW was observed. Associated with the electron Landau absorption of the mode converted waves, we are predicting mode conversion current drive of the order of 100 kA at 3 MW RF power, using the new improved 4-strap antenna at 90 degree phasing, centered at $N_{\|} = 7$. These experiments are in progress. In addition, flow drive experiments are being carried out with a relative 180 degree phasing of the 4-strap antenna elements. The aim of the shear flow-drive experiments is to test theoretical predictions of flow stabilization of turbulence and thereby control of ITBs. Initial estimates indicate observable effects with the injection of 3 MW of RF power. The flow will be measured at 3 spatial locations with Doppler broadening of high resolution X-ray spectra (HIREX).

EX/P4-34 · Development of a Completely CS-less Tokamak Operation in JT-60U

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Abstract: Elimination of the center solenoid (CS) has a large impact on the economic competitiveness of a tokamak fusion reactor. CS-less start-up favors the EC fundamental resonance to be placed near the center of the vacuum vessel, and lower neutral densities than normal start-up with the CS. The EC power necessary for full utilization of inductive flux is about 1 MW in JT-60U. A completely CS-less start-up to a plasma current of 80 kA was also achieved with the outboard vertical field coil alone. A large reduction in the recirculating power can be realized by increasing the bootstrap current fraction. If it were possible to achieve a bootstrap current fraction of greater than 1 (bootstrap overdrive), this can be used for plasma current ramp-up. An indication of bootstrap overdrive was obtained by predominantly perpendicular neutral beam (NB) heating. The negative loop voltage and the positive slope of the F coil current during the later part of the discharge indicate clearly that this plasma is overdriven non-inductively. Poloidal betas of up to 4 and normalized betas of up to 2.5 have been achieved in such plasmas. Overdrive was observed even when only counter and perpendicular beams were injected, in which case the beam driven current should be in the counter direction.

EX/P4-36 · Experiments of full non-inductive current drive on HT-7

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Abstract: Some experimental results of steady-state operation and full non-inductive current drive have been obtained on HT-7. Three types of experiment are used to study long pulse discharge, quasi-steady-state operation and full non-inductive current drive. The first one is OH plasma current together with current droved by LHW for long pulse discharge. The second is to keep the plasma current constant and the loop voltage come close to zero by means of adjusting of the LHW power for quasi-steady-state operation or quasi non-inductive current drive, the OH heating field still has a little feedback control effect when the current droved by LHW is instable. The third is the current droved by LHW larger than the plasma current kept by feedback control system of OH heating field for full non-inductive current drive. In the third type experiment there are two phases. Phase I: constant plasma current and negative loop voltage, the OH heating coil current is reducing; Phase II: the loop voltage and the current of OH heating coil are zero, so the plasma current is not controlled by OH heating field and which is larger than the plasma current in phase I set in the feedback control system of OH heating field. The phase II is known as full non-inductive current drive plasma and can be sustained tens of seconds. The experiments show that the plasma current in the full non-inductive drive case is instable due to no adjusting effect of OH heating field, when the waveguide tube discharge lead to the LHW power injecting tokamak plasma decrease. This instability of plasma current will increase the interaction of plasma with limiter and first surface and bring impurity. All discharges of full non-inductive current drive are terminated because of impurity spurting. The off-axis deposition of LHW power will drive fast electrons and generate a halo plasma current profile. In full non-inductive current drive plasma, the impurity profile is different from that in the plasma of type I and type II. The new experiments of the interaction of LHCD and IBW (unbalance momentum injecting) or ICRH (balance momentum injecting) in the full non-inductive current drive plasma phase will show in the meeting.

EX/P4-37 · Destabilisation of TAE modes using ICRH in ASDEX Upgrade

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Abstract: In ASDEX Upgrade, toroidicity induced Alfvén eigenmodes (TAEs) are destabilised by ICRH in conventional and advanced scenarios, at low density. Most unstable TAEs have toroidal modes numbers (n=3,4,5,6) and experiments with reversed current and magnetic field showed that the TAE propagate in the current direction. On one hand, the analysis of the unstable TAE in ASDEX-Upgrade shows that the data is consistent with the results from previous studies performed in other tokamaks. In particular, the measured TAE frequency, in the range (150-200kHz), is consistent with theoretical TAE frequency calculated for the parameters of the discharges performed in these experiments. On the other hand, some interesting new features have also been observed in the ASDEX-Upgrade data. TAE (n=-1) was observed, which propagates in the opposite direction to the plasma current. It was also concluded that plasma rotation is insufficient to explain the experimentally observed frequency differences between two adjacent

toroidal mode numbers. The measured relative fluctuation amplitude of the TAE eigenfunction in the soft X-rays channels increases towards the plasma edge. These results are consistent with the ideal MHD calculations and show that the TAE are of a global nature and not core localised TAE modes. These results are particularly important, because the radial extent of AE is a key factor in the redistribution of the energetic ions in the presence of unstable TAE.

EX/P4-41 · Studies of high energy ions in Heliotron J

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Abstract: For the sake of heating plasmas and studying of the high energy ion confinement in helical-axis Heliotron J, the NBI experiment has been newly started. The hydrogen neutral beam can be injected in a tangential direction with maximum power of 0.7 MW. In order to investigate the effect of the magnetic field components on the energetic particle confinement in the helical-axis heliotron configuration, the bumpiness (mirror ripple) control experiment was carried out in the NBI plasmas. In the experiment, the normalized bumpiness (B_{04}/B_{00}) was changed from 0.01 to 0.14 with keeping the magnetic axis position (1.2 m), plasma volume (70 m^3), edge iota (0.56) and the electron density ($0.5 \times 10^{19} m^{-3}$). In the case at B_{04}/B_{00} = 0.14, the 1/e decay time of the CX flux after the NB turn-off was much shorter (less than 1 ms) as compared with the slowing down time of the 7 keV proton (23 ms). It was found that the decay time was decreased with the bumpiness, which was interpreted by the change in the loss cone shape predicted by the non-collisional orbit calculation for the ion guiding center. The minority heating experiment with H-minority and D-majority was performed in order to study the ICRF heating mechanism and high energy ion behavior in Heliotron J, using a loop antenna. The ICRF power injected from the antenna on the low field side to the ECH target plasma was up to about 350 kW. The high energy proton up to 8 keV having the tail temperature of 0.7 keV was observed.

EX/P4-44 · Classical and non-classical confinement properties of energetic ions on LHD

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Abstract: The confinement properties of tangentially injected energetic particles are experimentally investigated on the Large Helical Device(LHD). The local confinement times of energetic particles are evaluated by experiments and have good correlation with pitch-angle scattering times in the core region and with charge exchange loss times in the edge regions. Thus, the classical process is considered to be the dominant process of the energetic particle confinement on LHD. In addition to these classical effects, the enhanced energetic particle transports induced by MHD-instabilities are observed on LHD. These effects become significant at low magnetic field (Bt$<\sim$0.75) configurations

EX/P4-45 · Experimental Studies of Alfvén Mode Stability in the JET Tokamak

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Abstract: Controlling the interaction between alpha's and modes in the Alfvén frequency range is a crucial issue for ITER operation in order to prevent alpha's losses. We study experimentally the dependence of the stability threshold of Alfvén Eigenmodes (AEs) with various toroidal mode numbers (n's) on different Ion Cyclotron Resonance Frequency (ICRF) heating schemes and background plasma parameters. Multi-frequency ICRF heating gives rise to a lower perpendicular fast ion temperature in the plasma core. By spreading the power deposition profile, this lowers the fast ion pressure gradient, hence leading to a reduced AE activity. The use of 2nd harmonic minority ICRF heating at higher plasma density reduces the fast ion temperature in the plasma core, and gives rise to a lower fast ion driven plasma rotation. A controlled radial redistribution of the alpha's can be beneficial for the stability of AEs with intermediate n's. This mechanism is experimentally simulated in JET by using the saddle coils to produce error fields that couple to the q=2 surface. This scheme gives rise to a significantly higher threshold for core localised AEs in the presence of error fields, which we attribute to a modification of the radial profile of the fast ion distribution function. The direction of the ion gradB-drift is an important parameter in determining the accessibility conditions for the H-mode regime, with multi-machine experimental evidence that a lower power is required for the L-H mode transition when the ion gradB-drift is directed towards the divertor. We find that the damping rate of n=1 TAEs with similar frequency and radial location is approximately a factor three higher for the case of ion gradB-drift directed away from the divertor. The present JET saddle coil system can only excite low-n AEs, whereas the n's that can be driven unstable in ITER by the alphas are expected to be in the range n∼5- 20. The mismatch between the modes driven by the saddle coils and those that are made unstable by the fast particles is already observed on JET. Building on the present system, a new set of antennas to drive AEs with mode numbers up to n∼10-15 is being designed for JET. Design principles and constraints will be presented along with the results of the coupling and engineering analysis, together with a discussion of the possible extrapolation of such a system to ITER.

EX/P4-47 · Magnetic Field Structure and Confinement of Energetic Particles in LHD

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Abstract: It is theoretically and experimentally shown that drift surfaces exist for highly energetic particles being extended over the last closed flux surface (LCFS) in LHD. Those particles are previously considered as lost particles due to the loss-cone in theories, where the analyses are limited inside the LCFS. The present theory predicts that LHD has no loss-cone and that highly energetic particles confined over the LCFS exist. These are consistent with the LHD experimental results in both the ion cyclotron resonance heating (ICH) experiments and the low magnetic field neutral beam injection (NBI) heating experiments. From particle orbit analyses and studies on the connection length of diverter field lines, it is also shown that plasma can exist in the chaotic field line region located outside the LCFS in LHD. The plasma in the chaotic field line region is clearly detected by CCD-cameras in the LHD experiment. This ambient plasma might be expected to have the role of a kind of impregnable barrier for the core plasma, which suppresses both the MHD instabilities and the cooling of the core plasma due to charge exchange processes.

EX/P5-1 · Observation of high-frequency secondary modes during strong tearing mode activity in FTU plasmas without fast ions

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Abstract: MHD spectroscopy in FTU has revealed high frequency (HF) oscillations between 30 and 80 kHz that accompany the development of m=-2, n=-1 islands in ohmic plasmas. The frequency range is one order of magnitude below the first toroidal gap in the Alfvén continuum and of the same order of the low frequency gap introduced by finite beta effects. The HF spectrum is organized in pairs of modes with toroidal numbers n=1 and n=-1 forming standing waves in the island rest frame. The poloidal structure of HF modes appears to vary from —m—=2 on the low field side to —m—>5 on the high field side. HF lines appear when the level of m=-2 poloidal field perturbation exceeds 0.2%. The absence of energetic ions in FTU ohmic plasmas, the strict correlation between HF and tearing modes and the existence of a threshold indicate that HF modes tap energy from the tearing mode by a non-linear mechanism.

EX/P5-3 · New observations concerning the origin and consequences of MHD activity in the MST Reversed Field Pinch

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Abstract: Reversed Field Pinch (RFP) plasmas exhibit a broad spectrum of MHD activity. The instability underlying much of this activity in the standard RFP discharge has been identified as resistive tearing and is well modeled by nonlinear, resistive MHD. In recent years two additional types of discharge have been produced: those with reduced tearing activity leading to improved confinement and those with one large mode termed Quasi-Single Helicity (QSH). Our understanding of MHD activity and its consequences in these plasmas is incomplete. In this paper, we present new data concerning (1) the origin of m=0 modes in standard plasmas, (2) the MHD dynamo in QSH plasmas, (3) the effect of MHD modes on plasma rotation, and (4) the achievement of a dynamo-free RFP plasma.

EX/P5-4 · Two-Fluid Hall Effect on Plasma Relaxation in a High-Temperature Plasma

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Abstract: For fast current profile relaxation (∼100 microseconds) in the high temperature (collisionless) MST Reversed Field Pinch, the inductive electric field is not completely balanced by the collision force thereby requiring a fluctuation-induced electromotive force. Measurements of the two-fluid Hall effect during plasma relaxation (magnetic reconnection) are made in the plasma core and edge. Three new results are described: (1) The Hall dynamo is found to be small for r/a>0.9a but substantially increases towards the interior at r/a=0.85 as measured by edge probes. (2) In the high-temperature plasma core (r/a=0.35), magnetic and current density fluctuations (and their spatial distribution) are directly measured using a high-speed laser Faraday rotation diagnostic. The Hall electromotive force is found to be significant, suppressing (flattening) the equilibrium core current near the resonant surface. (3) Quasilinear theoretical calculations show that the Hall dynamo can exceed the MHD dynamo under certain conditions, consistent with experimental observation. These experimental results imply that effects beyond single-fluid MHD are important.

EX/P5-5 · Correlation of Electron Super-thermal Acivities with Magnetic Turbulence and MHD Processes in SINP Tokamak

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Abstract: The nonthermal electron phenomena in SINP tokamak is being investigated by measuring the hard x-ray emissivity with the aim to find out the role of tearing MHD modes and magnetic turbulence in the generation of runaways. A set of large NaI(Tl) detectors kept at about 15 metres away from the limiter measures individul photons. The time evolution chart indicates, besides the runaways at the startup causing the emission of a bunch of photon, generation of superthermals in the middle part of the discharge, energy of which keeps growing till it reaches about 5MeV. A spectral analysis of the radiation gives a realistic measure for magnetic turbulence level and the thickness of the turbulence layer. To investigate the reason for generation of superthermals in the middle part of the discharge two detectors were placed within 0.5 metres from the tokamak. The data from the one that looks into the plasma tangentially through a thin aluminium window, indicates a periodic emission of large bunches of low energy hard x-rays. This regular emission appears with the appearance with m/n = 2/1 tearing mode and a strong correlation exists between the two oscillations. From the close similarity of the observed phenomenon with the quasi-coherent oscillation recently observed in T-10 tokamak prior to a disruption, a resonant transfer of energy of a nonthermal electron beam, formed by the strong electric field during magnetic reconnection, to radiation while the beam moves through the toroidal magnetic field modulated by ripples.

EX/P5-6 · Asymmetric-Field Mode Locking in Alcator C-Mod

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Abstract: Asymmetric field coils have been installed on Alcator C-Mod and used to establish the locked mode threshold and scaling for this compact size device. The magnitude and phase of the intrinsic field error has been identified and is found to be consistent with a comprehensive model of the sources of field errors. There proves to be no major difference between the fractional (m=2,n=1) field level (typically 1.e-4 of toroidal field) for mode locking on Alcator and that on JET (which has 4.3 times the linear dimensions) for typical densities on each machine. This result is a first direct experimental indication based on size scaling that locked modes should not be more problematic for ITER than for existing devices. Initial results are consistent with a linear scaling of the total asymmetric field locking threshold with density, as has been observed elsewhere, and within a factor of 2 of the scaling previously deduced from the toroidal field scaling on JET or DIII-D using Connor-Taylor constraints. Specifically chosen dimensionlessly similar plasmas in C-Mod and JET are under investigation to determine more definitively the field and size scalings.

EX/P5-7 · Ion Kinetic Effect on Bifurcated Relaxation to a Field-Reversed Configuration in TS-4 CT experiment

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Abstract: Ion kinetic effect on the bifurcated relaxations of merging spheromaks to a field-reversed configuration (FRC) and another spheromak were studied experimentally using varied S_* (averaged number of ion skin depth). Several ion species was used in the up-scaled TS-4 device to vary S_* widely from 1 to 7 during the bifurcated relaxations. The series of experiments indicates that the merging spheromaks with higher helicity relax into an FRC with zero helicity under lower S_* condition. Decrease in S_* promoted the

relaxation to an FRC, annihilating the magnetic helicity, in sharp contrast with the conventional Taylor relaxation. We used the poloidal eigen value $\lambda_p I/\Psi$ as a key parameter to measure the following relaxation, where I and Ψ are the poloidal current function and the poloidal flux respectively. The final state with $\lambda_p \approx 0$ represents the relaxation to an FRC with $B_t \approx 0$ and that with $\lambda_p \approx \lambda_{Taylor}$ indicates the relaxation to another spheromak. The bifurcated relaxations to an FRC ($\lambda_p \approx 0$) and another spheromak ($\lambda_p \approx \lambda_{Taylor}$) were clearly identified in the low- S_* cases. It was observed that the threshold value λ_0 for the relaxation to $\lambda_p \approx 0$) increased inversely with S_* value. The ion kinetic effect or the two-fluid effect is the most probable reason why low- S_* helps the merging spheromaks to relax into an FRC. This phenomenon revealed a non-MHD stability effect of high-beta/ high-flow equilibria useful for the future large-scale FRC and the high-beta ST experiments. The low-n mode suppression by the rotation shear of the toroidal modes is the most probable reason why the low- S_* condition promotes the relaxation into an FRC: the helicity annihilating process.

EX/P5-8 · Effect of Viscosity on Magnetohydrodynamics Behavior During Limiter Biasing in the CT-6B Tokamak

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Abstract: Effects of Viscosity on Magnetohydrodynamics Behavior During Limiter Biasing in the CT-6B Tokamak has been investigated. The results shown that subsequent to the application of a positive bias, a decrease followed by an increase in the frequency of magnetic field fluctuations was observed. With contribution of viscous force effects in the radial force balance equation for Limiter Biasing, in terms of the nonstationarity model, it allows us to identify the understanding physics responsible for change in the Mirnov oscillations that could be related to poloidal rotation velocity and radial electric field. It could be seen that the time scale of responses to biasing is important. the response of ∇p_i , decrease of poloidal rotation velocity, the edge electrostatics and magnetic fluctuations to external field have been investigated. Therefore, the results shown that momentum balance equation with considering viscous force term can be use for modeling of limiter biasing in the tokamak.

EX/P5-9 · Experimental Investigations of Plasma Periphery in the T-10 Tokamak.

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Abstract: Intensive experimental investigations of the plasma periphery physics are carrying out on the T-10 tokamak. It was shown that the perpendicular anomalous particle flux in scrape-off layer increases with an average plasma density rise. Probe measurements indicate that intermittent events can play a significant role in the cross-field transport. Intermittent behavior of the plasma parameters is associated with formation and propagation of the structures with high density which can be responsible for more than 50 % of the total radial turbulent particle flux. Improved confinement mode was achieved by inserting a positively biased electrode into the plasma edge in the T-10 tokamak in regimes with electron-cyclotron resonant heating. Probe measurements of the radial turbulent particle flux show that no strong changes of the flux are observed after biasing switch-on. It can be assumed that radial electric field affects mainly neoclassical transport. Locking of the tearing mode (m = 2, n = 1) by a controlled halo-current was observed. It was shown that tearing mode frequency and amplitude could be driven by non-disruptive halo-current at the flat-top period of the tokamak discharge. Electron temperature variations accompanied the changes in mode amplitude were observed at the plasma periphery. Investigation of plasma-surface interaction displayed two types of the re-deposited crystal carbon films were found and analyzed: smooth multi-layer films and globular films. Smooth films appear to be "soft" or "hard". The D/C ratio in the smooth films is 0.3-0.35 for "hard" films and 0.6-0.8 for "soft" films. In the globular films, the D/C ratio can be less than 0.01. The deuterium concentration in films most likely depends on the surface temperature.

EX/P5-11 · Sawtooth Physics and Plasma Shapes

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Abstract: DIII-D is conducting experiments to separate the roles of interchange stability and the internal kink in the sawtooth instability. The principal feature of the experiment is the use of the DIII-D shaping capability to change the magnetic well depth and separate the resistive Mercier stability criterion from the kink stability condition. Examining two plasma shapes, an oval and a bean, does this. In an oval, the

Mercier criterion will be violated at $q_0 < 1$ and in a bean it will be satisfied for $q_0 > 1$. The remaining discharge conditions are determined by diagnostic requirements; namely high quality MSE signals, ECE signals that span the midplane diameter, and ion temperature (CER) data with 300 ms time resolution. The result is that the oval exhibits a very mild sawtooth with miniscule change in a very flat q profile. The bean has a much harder crash with significant change in the q-profile and a loss of stored energy from the plasma. A new approach to equilibrium analysis is yielding the q-profile with unprecedented accuracy, allowing accurate stability analysis.

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EX/P5-12 · Snake Perturbations during Pellet injection and LHCD in the HL-1M Tokamak

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Abstract: Excitation of snake perturbations has been observed in the core region of pellet- fuelled HL-1M plasmas when the pellets cross surface with a q value of 1. It is observed that the snake oscillation has an m=1,n=1 helicity with quite a long lifetime.A detailed comparison has been made between the locations of q=1 surface and snake oscillation. Through measurements of plasma q-profile by means of multi-exposures with CCD camera during pellet ablation, and investigation on pellet ablation process, possible mechanisms for the formation of snake oscillation are discussed. In addition, a large, long-lived snake-like oscillation is frequently observed in lower hybrid current driven discharge in which the sawtooth has been stabilized early in the discharge. There is evidence that such a perturbation is due to impurity accumulation during sawtooth-stabilization, and good performance with peaking profiles after LHCD is limited by magnetohydrodynamic (MHD) instabilities including sawtooth and snake activities in HL-1M plasma.

EX/P5-13 · Dynamics and Control of Resistive Wall Modes with Magnetic Feedback Control Coils: Experiment and Theory

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Abstract: New observations, theory, and modeling of RWM dynamics and of active feedback control of the RWM are reported. RWM dynamics are studied by direct measurement of the plasma response to quasi-static and rapid "phase-flip" changes of resonant magnetic perturbations applied as a function of kink stability and plasma rotation. The dynamic plasma response is characterized by a paramagnetic amplitude and by a phase that are easily measured and evolve in time. For discharges near marginal stability of the RWM, we find the resonant field amplification to be weakly dependent on kink stability while the time-scale of the dynamical phase relaxation lengthens considerably at marginal stability. Analysis of the measured response from many discharges is consistent with single-mode RWM theory and determines the stability boundary, the viscous dissipation, and critical rotation rate associated with RWM stabilization. A remarkable feature of the theory is that little information is required about the plasma; much more must be known of the surrounding current-carrying structures. Feedback control of the RWM is illustrated using VALEN- optimized mode control techniques. The theory behind the VALEN code and the key factors allowing design of practical feedback systems that can provide robust control of the RWM are presented.

EX/P5-14 · Experiment of Magnetic Island Formation in LHD

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Abstract: Magnetic island formation is experimentally investigated in the Large Helical Device (LHD). An error field or the external field generates the seed island. The island shrinks as the electron temperature or the beta increases in the case of lower magnetic hill, but it grows in the case of higher magnetic hill. An evidence of (m,n)=(1,1) current in the magnetic island is obtained, where m and n are the poloidal and the toroidal mode numbers, respectively.

EX/P5-15 · Observation of Current Profile Evolution Associated with Magnetic Island Formation in Tearing Mode Discharges on JT-60U

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Abstract: Evolution of the current density profile associated with magnetic island formation during tearing modes was revealed in JT-60U for the first time. Equilibrium reconstruction faithful to a small change in the Motional Stark Effect (MSE) measurement showed decrease and flattening of the current density profile at the radial location of the magnetic islands during growth and saturation of the m/n = 2/1 tearing mode. As the island width shrank spontaneously, the flattened region in the current density profile became narrower, and it finally faded out after disappearance of the m/n = 2/1 tearing mode. For the m/n = 3/2 neoclassical tearing mode (NTM), in addition to such flattening in the current density profile, increase in current density due to electron cyclotron current drive (ECCD) at the mmagnetic islands and disappearance of the flattened region were observed when the m/n = 3/2 mode shrank and was completely suppressed. These measurements lead to advance in an understanding of the NTM physics suc as clarification of the mode onset condition and quantitative evaluation of the neoclassical effect in tearing modes.

EX/P5-16 · Observation and manipulation of mesoscale structures in TEXTOR

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Abstract: Hot, magnetically confined plasmas have a rich structure on the mesoscale with a significant impact on global plasma dynamics and performance. The major objectives of the FOM team at TEXTOR are the understanding and control of this structure. These objectives are to be achieved through high-resolution measurements and manipulation of this structure using hardware recently commissioned at TEXTOR: a 800 kW, 140 GHz, >3 s gyrotron for ECRH and various advanced diagnostics. First experiments with the new equipment have concentrated on studying the features of steady-state and transient internal transport barriers near q=1, the effects of ECRH/ECCD on sawtooth stabilization, and the combined action of modulated ECRH and the Dynamic Ergodic Divertor on m/n = 2/1 modes. In plasmas that are strongly perturbed by large 2/1 and 1/1 modes, sometimes fractal islands structures have been observed in the layer between the main islands. Moreover, it has been observed that the density fluctuations are large near the X-point of the large islands than near the O-point.

EX/P5-19 · Implications of Wall Recycling and Carbon Source Locations on Core Plasma Fueling and Impurity Content in DIII-D

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Abstract: Plasma recycling at the divertor and main chamber walls in DIII-D has been spectroscopically assessed by measuring the neutral deuterium and low-charge-state carbon line intensity distribution in the divertor and the high-field side scrape-off layer. The significantly stronger D_α emission from the inner divertor leg observed in low- density, lower-single null, L-mode discharges suggests a much colder, recombining inner divertor plasma. The calculated distribution of the ionization source from the UEDGE/DEGAS code package shows that 80% of all the neutrals penetrating the core are ionized around the divertor X-point. Measurement and modeling indicates that the core carbon content is determined by carbon sources at the divertor entrance and ion transport into the main chamber. Analogous conclusions can be drawn from emission profiles obtained in medium-density H-mode plasmas with Type-I ELMs. The effect on fueling and core carbon content of poloidal $E \times B$ and $B \times \nabla B$ drifts, magnetic configuration and divertor geometry is presented.
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EX/P5-20 · Comparison of Plasma Turbulence in the Low- and High-Field Scrape-off Layer in T-10.

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Abstract: Experimental investigation of the SOL turbulence and associated transport in high density regime in T-10 is presented. Comparison of the plasma parameters at low- (LFS) and high-field side (HFS) of tokamak shows that inboard turbulence essentially differs from outboard one. Turbulence at

LFS and HFS has intermittent character but fluctuation level of the inboard ion saturation current is about two times lower than at LFS. Radial turbulent particle flux and perpendicular effective diffusion coefficient measured at LFS are about 3-5 times higher than ones at HFS. At that values of the radial and poloidal velocities of the plasma fluctuations on the inboard and outboard sides are similar. Poloidal velocity is directed to an ion diamagnetic velocity. Radial velocity is directed towards the vacuum vessel wall. Measurements of a plasma density n and an electron temperature Te indicate that plasma pressure (nTe) is approximately constant along magnetic field lines in the SOL. Analysis of experimental data allows us to conclude that enhanced plasma turbulence most probably originates on the outboard of the tokamak and then plasma fluctuations propagate to the inboard. These experiments suggest that SOL parallel flow might be caused by an additional mechanism such as "ballooning" which generates a larger flux of particles from the core into the SOL at the outer mid-plane.

EX/P5-21 · Structure, phase analysis and component composition of multilayer films depositing in tokamak T-10

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Abstract: Two types of deuterocarbon films (homogeneous and globular) differing in their structure were found inside the T-10 tokamak vacuum chamber. Other films similar in microstructure to the homogeneous ones have formed on the smooth plasma facing surfaces of the 04Kh16N11MÇT stainless steel mirror-specimens located inside the upper stub pipe of the vacuum chamber. Both types of the homogeneous films have been studied using X-ray diffraction analysis, Rutherford backscattering spectroscopy, scanning electron microscopy and elastic recoil detection method. The films formed on the chamber walls were reddish- brown and yellow. Those on mirrors were thinner, semitransparent and had a play of green, yellow and pink colors. The surface of the films was smooth, without any signs of physical sputtering. The films had a multilayer structure. Few hundreds of thin (<100 nm) layers were detected in cross sections of reddish-brown, ∼0.1 mm thick, films, while ∼10 layers formed on mirrors during a campaign. X-ray diffraction analysis indicates that all the films have crystalline structure and represent compounds of C and D. Both types of films have close values of the Wulf-Bragg diffraction angles and of lattice constants. These facts suggest that the film formation in both cases has the same mechanism. Measurements of refractive indices of the films in the ultraviolet range suggest that the films contain compounds of D with fullerens. At the same time, the two film types essentially differ in D-content. Films formed on the chamber walls are "soft" with the D/C atomic ratio varying from ∼0.6 to 1.4. In the films on mirrors D/C = 0.2 - 0.35. D is distributed uniformly across the film thickness, except a narrow near-surface layer of films on mirrors where its concentration abruptly decreases. The difference in the amount of accumulated D in the films is determined by different conditions of the deuterocarbon molecules condensation. Simulation of the formation of hydrocarbon films having microstructure and internal texture similar to those in films formed in tokamaks was performed in the accelerator VITA and have shown that formation of the "soft" films is not observed at temperature of 673 K. Our studies suggest that, in principle, it is possible, to suppress formation of "soft" carbon films with high hydrogen isotope content in tokamaks, containing carbon plasma facing elements.

EX/P5-22 · Overview of gas balance in plasma fusion devices

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Abstract: The amount of tritium retained in ITER has to be strictly limited due to safety considerations. The evaluation of the hydrogenic retention in present tokamaks is of crucial importance in order to estimate what can be expected for the very long discharges foreseen in ITER. Most of our knowledge about long term fuel retention is from post mortem surface analysis of tiles. However, the data do not inform on dynamic aspects neither they are obtained in conditions which are completely ITER relevant. The gas balance in JET, ASDEX, TEXTOR and Tore Supra has been compared, taking account of magnetic configurations, plasma edge conditions, and plasma facing component cooling type. Based on a shot-to-shot analysis of short discharges (<30sec), this contribution shows that a large fraction of the fuel is pumped out from the vessel during and shortly after the pulse independent of the machine and the edge plasma conditions. However, for long discharges (>6min) in Tore Supra, steady state retention rates result in a constant increase of the long term vessel inventory. The implications of these new observations for ITER are discussed in this paper.

EX/P5-23 · Liquid Lithium Limiter Experiments In CDX-U

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Abstract: Recent experiments in the Current Drive experiment – Upgrade (CDX-U) provide a first-ever test of large area liquid lithium surfaces as a tokamak first wall, to gain engineering experience with a liquid metal first wall, and to investigate whether very low recycling plasma regimes can be accessed with lithium walls. The CDX-U is a compact (R=34 cm, a=22 cm, toroidal field = 2 kG, I_p =100 kA, $T_e(0) \sim 100$ eV, $n_e(0) \sim 5{\times}10^{19} m^{-3}$) spherical torus at the Princeton Plasma Physics Laboratory. A toroidal liquid lithium tray limiter with an area of 2000 cm^2 (half the total plasma limiting surface) has been installed in CDX-U. Tokamak discharges which used the liquid lithium limiter required a fourfold lower loop voltage to sustain the plasma current, and a factor of eight increase in gas fueling to achieve a comparable density, indicating that recycling is strongly reduced. Modeling of the discharges demonstrated that the lithium limited discharges are consistent with $Z_{effective}$ <1.2 (compared to 2.4 for the pre-lithium discharges), a broadened current channel, and a 25% increase in the core electron temperature. Spectroscopic measurements indicate that edge oxygen and carbon radiation are strongly reduced. Experimental results, modeling, and future work will be presented.

EX/P5-24 · Carbon deposition and deuterium inventory in ASDEX Upgrade

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Abstract: Carbon erosion and deposition in the ASDEX Upgrade divertor was investigated using a poloidal section of marked divertor tiles and samples below the divertor structure. The whole inner divertor and a fraction of the outer divertor are net deposition areas. More than 31 g carbon were redeposited in the divertor, while only 4 g carbon were eroded from the outer divertor baffle, showing clearly that redeposited carbon originates predominantly from the main chamber. About 2.7% of the total deuterium input were trapped in deposited layers on divertor tiles and below the divertor structure, while less than 0.1% are trapped in remote areas like pump ducts. The thickest layers were observed on the inner strike point. Redeposited layers in remote areas are mainly observed in line-of-sight to the strike points, indicating that layer formation is due to particles with high sticking coefficient originating from the strike points. This is confirmed by measurements of the sticking coefficient of hydrocarbon radicals with cavity probes, which give a surface loss probability > 98%. Time resolved measurements with quartz micro balances are used to study the mechanism of layer formation.

EX/P5-25 · Experiments with Lithium Limiter on T-11M Tokamak

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Abstract: The paper presents the results of two stages of the Li Capillary-Pore System (CPS) researches in T-11M. An ability of capillary forces to confine the liquid Li in (CPS) tokamak limiter during disruption was demonstrated. The Li erosion process and tokamak first wall sorption properties were also investigated. As a next step of program was the development of a new thin (0.6 mm) CPS limiter for a steady-state limiter mode achievement. The second stage of T-11M Li-program was experiments with a clean ($Z_{eff} = 1$) deuterium plasma in discharge duration up to 0.3 s. The temporal evolution of Li surface temperature was studied during discharge by IR radiometer as function of different initial limiter temperatures. The neutral Li line emission was measured for estimating the Li influx. The temperature increase of Li erosion was obtained. The radial distribution of radiation losses shown up to 70 % of radiation power from a thin (5 cm) plasma layer and only 30 % from a plasma core even at Li high influx. The Li emission oscillation and saw-tooth like oscillations of the limiter surface temperature have been detected on the highest level of Li limiter temperature (> 650 C).

EX/P5-27 · Variation in Particle Control With Changes in Divertor Geometry

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Abstract: Recent experiments on DIII-D show the importance of magnetic balance [i.e., the degree to which the divertor topology is biased toward single-null (SN) or double-null (DN)] in determining how efficiently deuterium particles can be removed with divertor pumps and, ultimately, in determining how

well core density can be controlled. In-vessel pumping at the inner and outer divertor targets was done from a single divertor at two poloidal locations. We find that the pumping rate at the outer divertor target is much less sensitive to changes in the magnetic balance than that at the inner target. This sensitivity results from interplay between the scrape-off layer (SOL) geometry and the poloidal distribution of particles entering the SOL. These pumping rates also depend strongly on edge density. UEDGE modeling suggests that ∇B- and $E \times B$-induced particle flows in the SOL are qualitatively consistent with observation. Even for a passive secondary divertor, as possible for ITER in an advanced tokamak scenario, density control may be only weakly dependent on magnetic balance.
*Work supported by U.S. DOE under DE-FC02-04ER54698, W-7405-ENG-48, DE-0AC05-00OR22725, and DE-AC04-94AL85000.

EX/P5-29 · Far SOL Transport and Main Wall Plasma Interaction in DIII-D

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Abstract: Far scrape-off layer (SOL) and near-wall plasma parameters in DIII-D depend strongly on the discharge parameters and confinement regime. In L-mode discharges cross- field transport increases with the average discharge density and flattens far SOL profiles, thus increasing plasma-wall contact. In H-mode between edge localized modes (ELMs), plasma- wall contact is generally weaker than in L-mode. During ELMs plasma fluxes to the wall increase to, or above the L-mode levels. Depending on the discharge conditions ELMs are responsible for 30%-90% of the total plasma flux to the outer chamber wall. Cross-field fluxes in far SOL are dominated by large amplitude intermittent transport events that may propagate all the way to the outer wall and cause sputtering. High levels of plasma interaction with the outer chamber wall are observed during disruptions. A divertor Material Evaluation System (DiMES) probe containing samples of several ITER-relevant materials was exposed to a series of upper single null (USN) discharges as a proxy to measure the first wall erosion.
*Work supported by U.S. DOE Grant DE-FG03-95ER54294, DE-FC02-04ER54698, W-7405-ENG-48, and DE-AC04-94AL85000.

EX/P5-30 · Toroidal Structure of Hydrogen Recycling in ultra-long discharges on TRIAM-1M

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Abstract: In the superconducting tokamak TRIAM-1M, the study and development of stable long pulse operation have actively been carried out. In the last year, a new record of a long duration discharge of 5 h 16 min was achieved using a movable limiter (ML) with good cooling capability. The plasma current was sustained by 2.45 GHz LHCD (Prf $\sim$ 6kW) and was almost constant of $\sim$15 kA during the discharge. The injected energy finally reached $\sim$0.11 GJ. The increase in the temperature of the vacuum chamber and the poloidal limiter (PL) could be successfully suppressed by $\sim$30% and $\sim$70% of those of the ultra-long discharge without ML, respectively. The heat loads to each plasma facing component of the discharge with and without ML are the following; the vacuum chamber 55% and 70%, the fixed limiters 10% and 30%, the movable limiter 35% and 0%, which are estimated using the calorimetry. The toroidal profiles of the H-alpha line intensity have been measured to study the structure of the hydrogen recycling in the steady state operation. The profiles are measured after a sufficiently long time from the plasma production. Namely, the effect of the external hydrogen supply is negligible small and the profile indicates the toroidal structure of hydrogen recycling itself. It is found that the structure of hydrogen recycling is changed at the parts of ML and PL due to the insert of ML, but at the parts of the main chamber it still remains the same structure as that without ML. The hydrogen recycling at PLs is relatively reduced and that at ML is increased. It seems to be caused by the reduction of the diffused hydrogen flux to PLs due to the insert of ML, since the LCFS is determined by ML. The broadening of the toroidal profile of the neutral density of hydrogen around the position of ML can be estimated to be about 30 cm from the toroidal profile of H-alpha. This characteristic length corresponds to a mean free path of hydrogen atom with the energy of 0.4 eV. The value of 0.4 eV is consistent with the energy estimated from the spectrum broadening of the H-alpha line.

EX/P5-32 · Retention of hydrogen isotopes (H, D, T) and carbon erosion/deposition in JT-60U

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Abstract: In JT-60U, quite recently, PMI and PFM studies have started under the joint research between Japanese universities and JAERI. The present paper summarizes recent results relating tritium issues of ITER, in particular retention of hydrogen isotopes (H, D, T) in plasma facing carbon materials and erosion/deposition including dust formation. In the operation periods from June 1997 to October 1998, about 4300 discharge (3600 D-D discharge) experiments were made with W-shaped divertor configurations with (or without) inner private flux pumping through a full toroidal inner slot. More than 300 shots were with neutral beam injection (NBI) heating with power of 14-23 MW. During this period, total amount of 18 GBq of tritium was produced and distributed in the vacuum vessel and/or exhausted. In December 1998, a poloidal set of CFC divertor tiles were sampled from P5 section. The first wall graphite sample tiles (exposed to plasma from Mar. 1991 to Oct. 1998) were also removed for the measurements of hydrogen isotopes retention and erosion/deposition measurements. Before air ventilation, 700 shots of hydrogen discharge were carried out to remove tritium from PFM. All removed tiles were analyzed by IP, TDS, SIMS, XPS and SEM to observe tritium retention, H and D retention, depth profiling of H and D, surface characterization, and erosion/deposition measurements, respectively. During summer break of JT-60U in 2003, the inside of the vacuum vessel and pumping ducts was inspected carefully to observe the deposition of carbon and other impurities on plasma shadowed area and dust/flakes were collected. Probably owing to higher temperature divertor operation, hydrogen retention (H,D,T) in redeposited layers in JT-60U was much less than those observed in JET. Moreover, no clear correlation between the redeposition and hydrogen retention was observed. Carbon deposition on the plasma shadowed area and remote area like pumping duct was very small and only small amount of dust was collected. All these results are promising for utilization of carbon as PFM but at rather high temperature. At present the results are still qualitative. More detailed and quantitative discussion is in preparation.

EX/P5-33 · Impact Of The Dynamic Ergodic Divertor On The Plasma Edge In The Tokamak TEXTOR

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Abstract: The Dynamic Ergodic Divertor (DED) has recently been installed in the Tokamak TEXTOR. It consists of 16 helical perturbation coils and 2 compensation coils at the inboard side of the machine which produce a resonant magnetic perturbation at the q=3 surface with mode numbers m/n=12/4, 6/2 or 3/1. In the 12/4 configuration an ergodic region is formed in the plasma edge with overlapping magnetic islands, characterised by very long connection lengths to the target, and a laminar zone with short connection lengths. The resulting helical divertor structure in the laminar zone has been investigated by thermography and CCD cameras with various interference filters. A characteristic four fold stripe-like pattern is observed, aligned to the DED coils. The pattern is smeared out when the DED is operated with AC. With increasing perturbation each of the stripes splits with a private flux region between the legs. Once such a flux tube with poloidal turn is formed, a reduction of the electron density measured by a thermal Li- beam diagnostic upstream at the low field side is observed. All these measurements are in good agreement with the magnetic field topology calculated by field line tracing and mapping techniques. In the 3/1 configuration, characterised by a much deeper penetration of the perturbation field, intrinsic 2/1 and 3/1 modes are triggered. Here, a recycling pattern well aligned to the respective mode structure is observed. The poloidal rotation in the edge is damped with the onset of the 2/1 mode and reverses sign with the onset of the 3/1 mode at a higher perturbation threshold. At this latter stage, a strong drop of the electron pressure at the low field side is observed. The experimental results are compared to modelling results with the EMC3 –Eirene code which has originally been developed to study edge transport in stellarators and has been adapted to the DED in TEXTOR.

EX/P5-34 · Microscopic Modification of Wall Surface by Glow Discharge Cleaning and its Impact on Vacuum Properties of LHD

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Abstract: Glow discharge cleaning (GDC) is a widely used technique for wall conditioning in fusion experimental devices. Though the cleaning effects of GDC are essentially related to the microscopic modification of the wall surface, there are few reports about it. In the present study, samples of the wall

material (SUS 316L) were exposed to GDC plasma of hydrogen, helium and neon in the Large Helical Device (LHD) by using the retractable material probe transfer system and examined microscopic modification by transmission electron microscopy (TEM) to understand the underlying atomistic mechanism of GDC. A special feature of He-GDC was very heavy damage at the subsurface region; formation of dense nano-size bubbles, dislocation loops and cracks connecting the bubbles, in spite of low incidence energy. Helium trapping in the defective sub-surface region often disturbed quick start of hydrogen plasma experiments by desorbing slowly even at room temperature. In contrast to the He-GDC, radiation induced defects were scarcely formed by H-GDC and Ne-GDC. Based on the microscopic observation of the damage, GDC of LHD was successfully improved. Gas impurities in the LHD could be efficiently reduced by a combination of Ne-GDC and H-GDC.

EX/P5-35 · Plasma Boundary Determination on HL-2A

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Abstract: In HL-2A, the current filament model is adopted to quickly determine the plasma parameters from external magnetic measurements, including the shape and position of the plasma column. The reconstructed plasma boundary is compared with the boundary of equilibrium calculation, and the error in radial direction between them is less than 8mm for limiter, dual null and single null configurations when 3-6 current filaments are selected in the center area of plasma. The computed results based on the magnetic measurements in HL-2A accord with those measured by other diagnostic methods for plasma boundary parameters and they show that the code can accurately identify the plasma diverter configuration.

EX/P6-12 · Shear modulation experiments with ECCD on TCV

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Abstract: Anomalous electron transport is determined by turbulence, which in turn is affected by magnetic shear. A novel application of ECCD, aiming at localized shear modulation, has been applied on TCV tokamak for experiments on shear-dependent electron transport. Pairs of EC beams, absorbed at the same radius but oriented for co and counter injection, are modulated out of phase in order to force a local modulation of current density at constant input power. Off-axis deposition (r/a=0.24) is performed for sawteeth control. A significant impact on local shear is achieved with IECCD=0.1IOH even if the modulation period is much shorter than current diffusion time. Although source (heat and particle) terms are constant, both electron density and temperature are modulated during alternated ECCD. Thomson Scattering is the diagnostic for local Te and ne measurement, in order to overcome suprathermal problems on ECE. Once equilibrium effects on TS measurements are taken into account, Te and electron pressure modulation, peaked on-axis, is confirmed at all radii internal to EC deposition. Best confinement (dnT=+12%) is for co-injection, when shear drops from about 0.5 to less than 0.2.

EX/P6-13 · Search for a Critical T_e Gradient in DIII-D L-Mode Discharges

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Abstract: Experiments have been performed to search for evidence of an electron heat pulse thermal diffusivity with a dependence on a critical electron temperature gradient scale length L_c. The technique utilized two different electron cyclotron heat pulse trains, each absorbed at slightly different radii and modulated out of phase to maximize the change produced in the local T_e gradient, attempting to change the local gradient from below to above L_c where a nonlinear jump in the heat pulse diffusivity is expected to produce a nonlinear change in T_e. Experiments were carried out at two spatial locations, normalized radii of 0.2-0.3 and 0.4-0.5, and at the inner location three heat flux conditions were tested with 0, 2.8 and 4.0 MW NBI. The maximum change in T_e gradient, over 100%, was produced in the Ohmic case, making it the most likely case to observe a nonlinear T_e response. However, no nonlinear T_e response was observed in any of the cases studied. These results remain at odds with results from other tokamaks where a model containing L_c has been successful in describing electron transport.
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EX/P6-14 · Impurity Transport and Control in ASDEX Upgrade

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Abstract: The temperature profiles in ASDEX Upgrade are generally observed to be self similar: variations of the heat deposition profiles have thus strong effects on the profile of the heat conductivity. In various experiments using on/off-axis NBI, ICRH or ECRH, the dependence of the impurity transport on the heat deposition was investigated, and a strong link of turbulent impurity and heat diffusion coefficient was found. The radial profile of the impurity transport coefficients was investigated in detail using silicon laser blow-off and puffing of neon in standard and improved H-mode plasmas, where the latter do not have sawteeth. For low central power deposition, the impurity transport is neoclassical for normalised radii less than about 1/4, while central heating increases the diffusion coefficient a factor of 2-10 above the neoclassical values depending on the heating method. Impurity profiles of carbon and tungsten were studied, central accumulation was never found for predominantly turbulent transport. For H-mode plasmas, impurity peaking can thus be controlled with central heating as was demonstrated for W being most critical with respect to neoclassical inward convection.

EX/P6-16 · Non linear electron temperature oscillations on Tore Supra : experimental observations and modelling by the CRONOS code

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Abstract: The recently discovered plasma regime characterised by stationary electron temperature oscillations [1] (O-regime), has been the object of further experimental investigation and of extensive modelling by the integrated plasma simulation code CRONOS. The experiments include the observation of the O-regime in discharges at exactly zero loop voltage, oscillations trigger by ECCD and the study of the interplay of the O-regime with MHD activity. The possibility of feedback control of the amplitude and frequency of the oscillations using the LH power as an actuator has been investigated. Simulations with the CRONOS code have revealed the nature of the temperature oscillations, which are originated by a non-linear coupling of temperature and current density profiles, of the predator-prey type. It has been shown that oscillating solutions of the coupled heat and current diffusion equations are possible only for a selected class of heat transport models. Therefore, this regime represents a sharp test for transport theories and for integrated plasma simulation codes.

G. Giruzzi et al., Phys. Rev. Lett. 91 (2003) 135001.

EX/P6-17 · Electron Heat Transport Studies Using Transient Phenomena In ASDEX Upgrade In ASDEX Upgrade

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Abstract: Experiments in tokamaks suggest that a T_e critical gradient length may cause the resilient behavior of Te profiles, in the absence of ITBs. This agrees in general with ITG/TEM turbulence physics. Experiments in ASDEX Upgrade using modulation techniques with ECH and/or cold pulses demonstrate the existence of a threshold in R/L_{Te} when $T_e > T_i$ and $T_e \leq T_i$. For $T_e > T_i$ linear stability analyses indicate that electron heat transport is dominated by TEM modes. They agree in the value of the threshold (both Te and ne) and for the electron heat transport increase above the threshold. The stabilization of TEM modes by collisions and the destabilization of ITG due to increased ion heating, yielded by gyrokinetic codes, which suggests a transition from TEM to ITG dominated transport at high collisionality, is experimentally demonstrated by comparing heat pulse and steady-state diffusivities. For the $T_e \approx T_i$ discharges above the threshold the resilience, normalized by $T_e \wedge 3/2$, is similar to that of the TEM dominated cases, despite very different conditions. The heat pinch predicted by fluid modeling of ITG/TEM turbulence is investigated by perturbative transport in off–axis ECH-heated discharges.

EX/P6-18 · Progress in understanding heat transport at JET

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Abstract: This paper will report on recent progress in understanding heat transport mechanisms either in conventional or advanced tokamak scenarios in JET. A key experimental tool has been the use of perturbative transport techniques, both by ICH power modulation and by edge cold pulses. The availability of such results has allowed careful comparison with theoretical modelling using either empirical and 1D fluid transport models, 3D fluid turbulence simulations or gyrokinetic stability analysis. In conventional L- and H- mode plasmas the issues of stiffness and non-locality have been addressed, while in plasmas characterized by Internal Transport Barriers (ITB), the properties of heat transport inside the ITB layer and the ITB formation mechanisms have been investigated. A threshold value R/LTe∼5 has been found for the onset of stiff electron transport, while the level of stiffness appears to vary strongly with plasma parameters, in particular the ratio of electron and ion heating. An increase in electron stiffness is observed with increasing ion heating, when the ITG instability becomes dominant over the TEM branch. Detailed predictive modelling of modulation results shows that the best agreement is presently obtained by the Weiland collisional model. The plasma current plays a major role in ITB formation, through the role of negative magnetic shear and of rational magnetic surfaces. Theoretical modelling of this evidence will be discussed. First results of perturbative transport in ITBs will be presented. The inner part of the ITB layer is highly stable and rapidly damps the propagation of heat waves due to the low thermal diffusivity. The outer part of the ITB can instead be easily destabilized by an edge cold pulse, resulting in an increase in cold pulse amplitude when it meets the ITB foot. Large cold pulses can lead to ITB loss. This evidence is consistent with a picture of the ITB as a narrow layer where the ITG/TEM turbulence growth rate is strongly reduced by an increase of the critical threshold driven by the favourable magnetic shear. Turbulence simulations show the same qualitative features.

EX/P6-19 · Study of an Anomalous Pinch Effect in the T-11M Tokamak

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Abstract: The peaked density profiles Ne(r) are typical for tokamaks with ohmic heating, while the sources of plasma – gas puffing and hydrogen recycling are localized in plasma periphery. Usually one tries to eliminate this contradiction assuming impurity accumulation, development of neoclassical Ware pinch or the turbulence-driven particle pinch. In the experiments on the T-11M (BT = 1 T, R/a = 0.7/0.2m, Jp = 100 kA, t = 300 ms) two first probabilities were excluded. In conditions of practically impurity-free discharge (Zeff = 1.1±0.1) and at rather high density $Ne(0) = 5\times10^{19} m^{-3}$ [1], which excluded neoclassical effects, a formation of peaked density profiles Ne(r) has been observed. Pulse hydrogen puffing was located in the limiter region while the main flux of hydrogen recycling and hydrogen release were close to a lithium limiter. The measurements of electron density were carried out with the help of five-channel phase jump-free Cotton-Mouton polarimeter. The profiles Ne(r) were reconstructed using Abel inversion. The profiles obtained were approximated by a set of parabolas: $Ne(r) = Ne(0)\ (1-r^2/a^2)^k$. Here the parameter k is an indicator of the profile factor. During the plasma shot it varies in the range from 0.3 to a quasi-stationary value ∼1.5. The analysis revealed that the sharp profile Ne(r) in T-11M could take place only as result of the strong plasma flux from periphery to the center with velocity about of 3m/s, which is 3-5 times higher than the velocity of neoclassical Ware pinch. This effect could not be explained by impurity accumulation since it must be accompanied by an increase of Zeff(0) much higher 1. The effect does not depend on the plasma purity. The effect obviously depends on the plasma density, increasing with the growth of Ne. Any assumptions concerning the influence of direct flux of neutral atoms on the creation of density profile Ne(r) thereby are eliminated. The formation of peaked profiles Ne(r) could be explained in terms of the development of convective transport processes from the periphery to the center.

. V.B.Lazarev et al. Proc. 30 EPS Conf. on Contr. Fus. and Plasma Phys. St. Petersburg 7-11July 2003 P-3.162.

EX/P6-20 · Density Profile Evolution During Dynamic Processes in ASDEX Upgrade

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Abstract: The impact of ELM activity and density limit disruptions on the plasma density as well as the use of reflectometry for plasma position and shape control are discussed here. Fast density measurements allow the definition of three characteristic phases of the ELM: precursor, collapse and recovery. At the precursor phase low frequency oscillations starting approximately 3 ms prior to the ELM collapse are observed in the density profiles. Fluctuation measurements show a continuous rise of the broadband fluctuations prior to the ELM collapse. In the collapse phase the crash of the density profile is observed. In the recovery phase, the density profile starts to recover its pre-ELM shape. A comparison between HFS and LFS density behaviour shows a delay on the density collapse, comparable to the ion parallel transport

time indicating a ballooning character of the ELM. For a set of ASDEX Upgrade ELMy H-modes, where plasma parameters such as plasma current, plasma density, triangularity and input power were varied, the radial velocity, ELM particle losses and affected depth, density pedestal width and edge density gradient are determined and its dependence with the plasma parameters determined. A comparison between intrinsic and pellet triggered ELMs shows no relevant differences suggesting that the triggering mechanism for both ELM type cannot be distinguished by the density profile dynamics or the density fluctuations. The changes observed on the density profiles during density limit disruptions support the observation that the erosion of the temperature is due to convection and not by stochastization. The information about the density profiles also provides a powerful tool for plasma position and shape control, as proposed for ITER to complement the magnetic measurements. Provided that the density is constant within the magnetic flux surfaces, a scaling factor between the line average density and the density at the separatrix can be used to estimate the density at the separatrix. The separatrix position can then be found from the measured density profiles. Software tools developed to quantify the errors involved in reflectometry gap evolution were used in ASDEX Upgrade ELMy H-mode discharges. The estimated position of the separatrix is found within 1 cm of accuracy when compared tosimilar magnetic data.

EX/P6-21 · Radial electric fields and improved confinement regimes in the TJ-II stellarator

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Abstract: The influence of plasma density, heating power and plasma configuration on the structure of radial electric fields and transition to improved confinement regimes have been investigated in the TJ-II stellarator. The development of the naturally occurring edge velocity shear layer requires a minimum plasma density. Near this critical density, the level of edge turbulent transport and the turbulent kinetic energy significantly increases in the plasma edge. The resulting shearing rate in the phase velocity of fluctuations is comparable to the one required to trigger a transition to biasing H-mode like regimes, suggesting that spontaneous sheared flows and fluctuations keep themselves near marginal stability. These findings provide the first experimental evidence of coupling between sheared flows development and increasing in the level of edge turbulence. Experimental results are consistent with the expectations of second-order transition models of turbulence driven sheared flows. During the transition to biasing H-mode like regimes edge sheared flows increase and turbulent transport is strongly reduced. The ratio between plasma density and particle recycling as well as the stored energy increase by up to a factor of two. The investigation of the relaxation of plasma potential and radial electric fields shows evidence of two different characteristic decay times: a fast scale, comparable with the correlation time of turbulence, and slow scale related with the TJ-II global confinement properties. The phenomelogy of TJ-II improved confinement regimes (a device designed for high beta operation but not optimized for neoclassical transport) looks similar to the H-mode regimes previously reported in stellarators. This similarity calls into question the leading role of neoclassical viscosity to access improved confinement regimes in stellarator devices.

EX/P6-22 · Amplitude correlation analysis of W7-AS Mirnov-coil array data and other transport relevant diagnostics

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Abstract: This work is based on the amplitude correlation analysis of the signals from a poloidal Mirnov-coil array on the Wendelstein 7 - Advanced Stellarator (W7-AS). The motivation behind this work is the early finding, that changes in the RMS amplitude of Mirnov-coil signals are correlated with the amplitude of small scale density turbulence measured by CO_2 Laser Scattering. Based on this and other measurements, the hypothesis was set, that some of the magnetic fluctuations is caused by transient MHD modes excited by large turbulent structures. The statistical dependencies between the power modulation of different eigenmodes can provide information about the statistics of these structures. Our amplitude correlation method is based on linear time-frequency representations of the signal: Short-Time Fourier Transformation (STFT) or Complex Continuous Wavelet Transformation (CWT) using Morlet-wavelets, depending on the time-frequency resolution required by the signal. Both transformations map the signal onto the time-frequency plane, as two dimensional power distributions. From these transforms we can recover the power modulation of different frequency bands. Provided the selection of the resolution of the transforms and the limits of the frequency bands were correct, the time series calculated this way resembles the original power fluctuation of the selected eigenmode. The only distortion introduced is a convolution smoothing by the time-window used in the transformation. In order to be able to select the right time-frequency resolution and meaningful frequency band limits, other types of analyses were carried out on the signals, for example reconstruction of poloidal phase response using cross-spectra, and correlation analysis of expected band powers between the Mirnov-coil array signals. In order to reveal the true structure and cause of magnetic fluctuations Mirnov-coil diagnostic signals were also compared with ECE, Lithium beam and CO_2 Laser Scattering measurements. In our analysis we have found, that there was a strong and systematic difference in the cross-correlations of power bands between different confinement states.

EX/P6-23 · Comparison of Broad Spectrum Turbulence Measurements ($0 - 40cm^{-1}$), Gyro-kinetic Code Predictions, and Transport Properties from the DIII-D Tokamak

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Abstract: The understanding of anomalous electron transport represents a significant experimental and theoretical challenge. Progress in this area will result in improved confidence in predictions for next-step fusion devices and, potentially, improved control of transport. Detailed comparison of transport properties and turbulence measurements - for a broad k range - to theoretical/simulation predictions is essential in developing this predictive capability. Relevant to this we report on comparison of broad wavenumber ($0 - 40cm^{-1}$) density turbulence measurements from a variety of DIII-D diagnostics to gyro-kinetic code predictions. High-k ($40cm^{-1}$) data from a new backscattering diagnostic show significant broadband fluctuation activity in an ETG relevant wavenumber range. Data from this system and from 1 and 15 cm^{-1} systems are found to be in qualitative agreement with linear gyro-kinetic code predictions. Nonlinear gyro-kinetic simulations are currently underway.
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EX/P6-24 · studies of confinement and turbulence in FTU high field high density plasmas

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Abstract: The Frascati tokamak Upgrade is an high field, high density device that allows for studying confinement and transport issues relevant for next generation tokamaks. Peaking of the electron density profile in pellet injected discharges has been found to enhance the energy confinement time up to values in agreement with the neo Alcator scaling of confinement (linear scaling with density) which is otherwise lost above a certain threshold density of the order of half the Greenwald density. Confinement times as high as 120 ms at densities of the order of those of ITER have been reached transiently, and we have foreseen scenarios in which we can further improve the above confinement time. The change in turbulence from linear to saturated ohmic confinement has been addressed through measurements from an improved reflectometry system. The scaling of density fluctuations with the toroidal magnetic field has been investigated scanning the field in the range 4 to 8 T.

EX/P6-25 · Measurements of density profile and density fluctuations in Tore Supra with reflectometry

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Abstract: Tore-Supra is now running with a unique set of reflectometers using three different techniques to measure density profile and density fluctuations which are a key element to qualify plasma performances. A 50-110 GHz X-mode fast scanning frequency set- up, which also routinely measures the density profile, a fixed frequency 105-155 GHz X- mode set-up and a 50-75 GHz O-mode Doppler set-up, with k-spectra determination, achieve complementary measurements from low to high field side of the discharge, and from MHD to micro-turbulence scale. They give details on central MHD modes with higher temporal resolution than using ECE or magnetic coils diagnostics. MHD modes are radially localised and coincide with q-profile reconstruction for various plasma configurations. Toroidal rotation of the plasma is also directly measured, in agreement with CXRS, as well as turbulence poloidal rotation profiles. The radial profile of small scale density fluctuations is obtained with a unique spatial resolution, showing a minimal level in the centre, and a strong increase toward the edge. Moreover, in the core, the fluctuation level in the high field side shows a tendency to be lower compared to the low field side.

EX/P6-26 · Turbulence suppression in discharges with off-axis ECR heating on T-10 tokamak device.

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Abstract: Transient electron temperature gradient has been observed in T-10 tokamak plasmas at $\rho = 0.25$ immediately after off-axis ECRH switch off. The turbulence characteristics were investigated in these discharges by means of correlation reflectometry. It was found that density fluctuations amplitude was 2 times lower than Ohmic level in narrow region near $\rho = 0.25$ after ECRH switch off. Poloidal coherency of fluctuations is also decreased in this region. Suppression of quasi-coherent oscillations has been observed in considered discharges during strong temperature gradient existence. Turbulence poloidal rotations measurements showed no any velocity shear after ECRH switch-off. Analysis of the linear growth rates of instabilities shows that ITG is unstable at $\rho \sim 0.25$ during all discharge. Possible explanation of the observed phenomena is the rational surface density decrease near q = 1 due to q profile transient flattering after off-axis ECRH switch off.

EX/P6-27 · Low-Frequency Structural Plasma Turbulence in the L-2M Stellarator

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Abstract: Steady-state low-frequency strong structural (LFSS) turbulence has been observed in magnetized plasma of the L-2M stellarator [1]. LFSS turbulence was observed in the L-2M stellarator throughout the plasma volume, although different mechanisms are responsible for the excitation of turbulence in different plasma regions because of the onset of various instabilities: drift-dissipative instability, MHD resistive ballooning instability, and instability driven by trapped electrons [2]. A key characteristic feature of LFSS turbulence is the presence of stochastic plasma structures. The nonlinear structures comprise a considerable fraction (from 10% to 30% in different plasma regions) of the turbulence energy. High wavelet coherence (up to 50% for frequencies below 150 kHz) between time samples of the magnitudes of density fluctuations in the central region and near the plasma edge has been observed in L-2M. Another characteristic feature of LFSS turbulence is that the PDF of fluctuations differs from a normal distribution by heavier tails and a larger peakedness. Stable non-Gaussian PDFs were measured for plasma density fluctuations in the central region and for the local turbulent flux in the edge plasma. Non-Gaussian probability densities of stochastic plasma processes point to non-Brownian character of the motion (diffusion) of particles. The role of rare events related to stochastic plasma processes with larger spatial and temporal scale becomes important. It is shown that the first-order differences of fluctuation samples are stochastic and their probability distribution is a mixture of Gaussians with different scales. Registered process can be successfully modeled by a combination of a finite number of a diffusive processes each of which corresponds to a certain diffusive mechanism related to the structural turbulence in a plasma transport process. Subordinated Lèvy process can be used to describe the turbulent transport process. This work was supported in part by the Russian Foundation for Basic Research (project nos. 03-02-17269, 04-02-16571).
G.M. Batanov, V. E. Bening, V. Yu. Korolev et al. JETP Lett., 78, 502 (2003). [2] G.M. Batanov, L.V. Kolik , A.E. Petrov et al. Plasma Physics Reports, 29, 363 (2003).

EX/P6-28 · Experimental study of particle transport and density fluctuation in LHD

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Abstract: A variety of electron density (Ne) profiles have been observed in Large Helical Device (LHD). The Ne profile changes from peaked one to hollow one with increase of heating power of NBI. These changes are due to the difference in transports since particle fueling from NBI is negligible and particle source from wall recycling, gas puffing are located outside of the last closed flux surface. The particle transport coefficients, i.e., diffusion coefficient (D) and convection velocity (V) are obtained from density modulation experiments under different heating power keeping averaged plasma density constant. The D is anomalously large and is proportional to the 2.1 and 1.2 power of electron temperature (Te) in the core and edge respectively. The existence of V indicates the occurrence of off-diagonal term of transport matrix. In the core, V changes direction from inward to outward as Te gradient increases suggesting that off-diagonal term associates with Te gradient. In edge, V is always directed inward signifying other origin for off-diagonal terms. Fluctuation spectrum changes significantly with heating power and qualitatively correlates with variation of particle transports.

EX/P6-29 · Observation of neoclassical ion pinch in the Electric Tokamak

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Abstract: A persistent inward ion pinch is observed in the Electric Tokamak (ET). As a result, particle confinement and the line-averaged density increase dramatically, and the Troyon limit $\beta_N = I/aB \sim 3$ is reached in Ohmic plasmas. The pinching is stimulated with gas puffing. The density trigger point is $n_e(0) > 10^{18}m^{-3}$ in Ohmic discharges and $3{\times}10^{18}m^{-3}$ in ICRF heated discharges. The ramp-up time of the density is typically 1 second. The ramps are terminated by internal disruptions due to internal beta collapse without any significant radiation loss, mostly above the density limit $n_{DL}(0) = 10^{20}B/R = 5 \times 10^{18}m^{-3}$ with $B = 0.25T$ and $R = 5m$. The loop voltage remains low during the ramp. In ICRF-heated discharges the ramps terminate at lower densities. We observe no reduction of fluctuations at the onset. An analysis of the radial particle flux from equilibrium plasma profiles is presented. It is found that the neoclassical "viscous" pinch and the Ware pinch both contribute to the spontaneous density ramps and dominate neoclassical and anomalous (neo-Alcator like) particle diffusion in the core plasma.

EX/P6-30 · Energy and Particle Confinement in MAST

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Abstract: Quasi-stationary H-mode plasmas have been produced on MAST in order to provide the data for heat and particle transport at low aspect ratio. A dataset of thermal energy confinement data has been submitted to the ITPA H-mode confinement database. The MAST data are expanding the database by a factor of 2.2 in inverse aspect ratio a/R and by a factor of 2.5 in toroidal beta. The MAST energy confinement times broadly agree with the IPBH98y2 scaling law. Merging with the published database shows that MAST supports a/R-dependence consistent with the subset of conventional cross-section tokamaks. MAST data show that the energy confinement time normalised to IPBH98y2 scaling has a systematic dependence on collisionality indicating that the dependence on this parameter is stronger than in the IPBH98y2 scaling. A dedicated factor of 1.7 beta-scan has been identified on MAST complementing experiments on DIII-D and JET. In addition a factor of 2 aspect ratio scan has been identified between MAST and DIII-D. The MAST confinement database is being expanded towards low collisionality and lower safety factor in order to approach the dimensionless parameters expected in next step spherical tokamaks. To assess the potential of L-mode, a direct comparison with H-mode discharges having the same engineering parameters has been carried out. The total energy confinement time is about 80% of the value in the H-mode discharges. Particle transport studies are also being carried out. The effective particle diffusivity is typically a similar fraction of the electron heat diffusivity to that measured in JET. These deuterium plasmas have a relatively large particle flux due to the pure NBI heating and collisional transport has to be considered. In addition, H-mode plasmas with pellet injection provide data on particle transport in a region of steep positive density gradient, a situation expected in ITER plasmas fuelled by pellets.

EX/P6-31 · Anomalous particle and impurity transport in JET

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Abstract: Results from wide survey of particle and impurity transport experiments in JET are reported. Experiments with LHCD, some of which are at zero loop voltage, show for the first time that in source-free MHD-quiescent L-modes with decoupled electrons (Te>2Ti) the curvature pinch associated with trapped electrons is the dominant convective process. Consistently, the peakedness of the density profile in these plasmas is observed to increase with the peaking of the current profile. There is no significant evidence for a dependence on temperature gradient length, nor on collisionality. By contrast, in H- modes density peaking decreases as collisionality increases. In both H- and L-modes the peaking of the intrinsic carbon impurity density profile measured using CXS is lower than the peaking of the electron density profile, in qualitative agreement with predictions by the Weiland model. Both this "anomalous screening" effect of light impurities and the moderate peaking of the density profiles at low collisionality are favourable for reactor performance. Convective transport of Ni injected by laser ablation is neoclassical, but diffusive transport is found to be anomalous.

EX/P6-32 · Impurity transport and confinement in the TJ-II Stellarator

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Abstract: We address in this paper the study of impurity transport, injected by laser ablation, in a broad range of experimental situations in electron cyclotron heated plasmas (ECRH): density scan, magnetic configuration scan, power scan and its dependence with power deposition profile. In addition, a comparison of impurity transport between ECRH plasmas and NBI heated plasma will be performed. Data analysis was performed by means of 1-D impurity transport code STRAHL. The code fits the localized temporal evolution of perturbed radiation signatures in tomographycally reconstructed global radiation signals in order to obtain local transport radial dependences in this range of experimental conditions. The results obtained will be discussed in the context of stellarator physics.

TH

Magnetic Confinement Theory and Modelling

TH/1-1 · Paleoclassical Electron Heat Transport

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Abstract: Most "anomalous" radial electron heat diffusion in low collisionality toroidal plasmas is shown to result from a combination of parallel free-streaming and paleoclassical Coulomb collision processes (magnetic field diffusion and parallel electron heat conduction). The key new physical point is that as magnetic field lines diffuse radially they carry with them electron heat equilibrated over a long parallel length L, which is the minimum of the electron collision length and a maximum effective field line length. Since L is much longer than the poloidal periodicity length l_p, the radial electron heat diffusivity is $M = L/l_p$ times the magnetic field diffusivity. The multiple M is typically of order 10 to 100, except near low order rational surfaces (short field lines) and just inside a magnetic separatrix (divergent q) where it is of order unity. This new model of radial electron heat diffusivity is in reasonable agreement with the following experimental results: magnitude (in tokamaks, STs and quiescent RFPs), radial profile, collisionality regime, Alcator scaling in "collisional" plasmas, and smaller heat diffusivity on low order rational surfaces and just inside a separatrix.

TH/1-2 · Non-diffusive transport in 3-D pressure driven plasma turbulence

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Abstract: Recent experimental and theoretical evidence suggests that transport in fusion plasmas deviates from the standard diffusion paradigm. Typical examples include the confinement time scaling in L-mode plasmas, perturbative experiments, and the non- Gaussianity of turbulence fluctuations. The standard diffusive paradigm breaks down in these cases because it relies, through Fick's law, on locality, it neglects memory effects (i.e., it assumes Markovian transport), and assumes the existence of a well-defined microscopic scale where particles follow an un-correlated, Gaussian stochastic process (random walk). To overcome these limitations, we propose a class of transport models using fractional derivative operators in space and time that incorporate in a natural way the non-locality, non- Gaussianity, and non-diffusive scaling believed to be present in fusion plasmas. As a concrete case study we consider tracer particles transport in three-dimensional pressure- gradient-driven plasma turbulence. In this system, tracers exhibit anomalous, super-diffusion. The fractional transport model describes the evolution of the probability density function of particles. In quantitative agreement with the turbulence simulations, the model exhibits non- Gaussian (Levy) behavior, algebraic decaying tails indicative of space-time non-locality, and non-diffusive confinement time scaling with system size.

TH/1-3Ra · Scaling Intermittent Cross-Field Particle Flux to ITER

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Abstract: Intermittent transport with long range ballistic propagation in the SOL have been the matter of recent experimental and theoretical investigation. This transport appears to be a key issue to determine the particle flux to the main chamber in ITER that will govern the wall erosion and tritium trapping. Extrapolation of present data requires to scale this transport with device size. The theoretical analysis of this scaling and its implications to ITER is achieved both numerically and analytically in the case of density transport generated by the interchange instability in the SOL. One finds in the simulations that the SOL e-folding length increases with device size with a power 0.64. This is significantly smaller than would result from ballistic transport at fixed velocity. The mean velocity of the ballistic transport is thus found to decrease with device size. This trend is recovered in the analytical work (scaling with a power 5 / 8). As a result one finds that the particle flow to the wall in ITER will only be slightly larger than expected from diffusive laws. It is shown that the departure from a ballistic transport scaling is governed by shearing effects generated by the turbulence on the turbulence scale.

TH/1-3Rb · Nonlinear Dynamics of Transport Barrier Relaxations in Fusion Plasmas

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Abstract: Relaxation oscillations of transport barriers are studied by three dimensional turbulence simulations. Barriers generated by an imposed $E \times B$ shear flow are found to relax intermittently on confinement time scales, even when fluctuations of the $E \times B$ shear flow are suppressed. The characteristic frequency increases with heating power. A relaxation event has a complex dynamical behavior, characterized by the intermittent growth of a mode at the barrier center. An analytical study reveals that this dynamics is governed by the $E \times B$ velocity shear. The system stays close to the threshold of the underlying linear instability. However, when crossing this threshold, the dynamics gets highly nonlinear. A crucial ingredient therein is a time delay for effective $E \times B$ velocity shear stabilization. As the dynamics bears similarities to edge localized modes (ELMs), this suggests that such effect of $E \times B$ shear flow might be included into the ELM description.

TH/1-3Rc · Non-linear Heat Transport Modelling with Edge Localized Modes and Plasma Edge Control in Tokamaks.

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Abstract: The present paper presents a new approach for the modelling of the pedestal energy transport in the presence of Type I ELMs based on the ideal linear MHD code MISHKA coupled with non-linear up to the second order of perturbations energy transport 2D code TELM in a realistic tokamak geometry. The main mechanism of increased transport through the External Transport Barrier (ETB) in this model of ELMs is the increased convective flux due to the MHD velocity perturbation and an additional conductive flux due the radial perturbation of the magnetic field leading to a flattening of the pressure profile in the unstable zone. Typical Type I ELM time-cycle including the destabilisation of the ballooning modes leading to the fast (few Alfvén times)collapse of the pedestal pressure followed by the edge pressure profile re-building on a diffusive transport time scale was reproduced numerically for JET and DIII-D parameters. The possible mechanisms of Type I ELMs control using stochastic plasma boundary created by the external coils is discussed in the paper. In stochastic layer the transverse transport is effectively increased by the magnetic field lines diffusion. The modelling results for DIII-D and JET demonstrated the possibility to optimise the ergodic magnetic field in order to decrease the edge pressure gradient just under the ideal ballooning limit, leading to the high confinement regime without Type I ELMs.

TH/1-3Rd · Impact of zonal flows on turbulent transport in tokamaks

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Abstract: Zonal flows (ZFs) are known to play a prominent role in the self-regulation of turbulence in tokamak plasmas. Fluid simulations are presented which investigate their dependence on the plasma and geometric parameters, and their effect on turbulent transport. Core Ion Temperature Gradient turbulence is studied with a 3D global code in which ZFs are damped by collisions. Improved confinement is observed with decreasing collisionality. This is found to result from the non-linear upshift of the effective threshold for ITG instability, mediated by an increase of the ZF shearing rate. It is also shown that the threshold for the turbulence onset, and therefore the effective conductivity, depends on input power. Interchange turbulence is studied in the SOL assuming flute modes. When the timescale of the parallel losses is large, transport is controlled by ZFs: its level strongly increases when ZFs are artificially suppressed. At sufficiently shorter timescale ZFs eventually no longer regulate the turbulence. In this case, the sheath response prevents local interactions in wavenumber space to drive ZFs and it clamps the electric potential to the floating potential.

TH/1-4 · Gyrokinetic Studies of Turbulence in Steep Gradient Region:Role of Turbulence Spreading and $E \times B$ Shear

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Abstract: An integrated program of gyrokinetic particle simulation and theory has been developed to investigate several outstanding issues in both turbulence and neoclassical physics which need to be addressed

for the success of future burning plasma experiments including ITER: i) An energy conserving set of a fully electromagnetic nonlinear gyrokinetic Vlasov equation and Maxwell's equations, which is applicable to edge turbulence, has been derived via the phase-space action variational Lie perturbation method. Our generalized ordering consists of the ion poloidal gyro radius of the order of the radial electric field gradient length. ii) Previous global gyrokinetic particle simulation of ion temperature gradient (ITG) turbulence spreading using the GTC code and its related dynamical model have been extended to the strong turbulence regime with radially increasing ion temperature gradient, to study the inward spreading of edge turbulence toward the core and its impact on transport scaling. iii) GTC-Neo has been used for accurate calculation of the key physical quantities for the neoclassical physics. A calculation of ion poloidal rotation in the presence of sharp density and toroidal angular rotation frequency gradients shows that there exists significant differences between GTC-Neo particle simulation results and conventional neoclassical theory predictions.

TH/1-5 · Density effects on tokamak edge turbulence and transport with magnetic X- points

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Abstract: Results are presented from the 3D electromagnetic turbulence code BOUT, the 2D transport code UEDGE, and theoretical analysis for studies of boundary turbulence and transport in a real divertor-plasma geometry and its relationship to the density limit. Key results include: (1) a transition of the boundary turbulence from resistive X-point to resistive-ballooning as a critical plasma density is exceeded; (2) formation of an X-point MARFE in 2D UEDGE transport simulations with increasingly large radial outboard transport, such as found by BOUT for increasing density; (3) identification of convective transport by localized plasma "blobs" in the SOL of 3D BOUT simulations at high density; (4) decorrelation of turbulence between the midplane and the divertor leg due to strong X-point magnetic shear; (5) formation of a density pedestal inside the separatrix in L-mode, even though the calculated plasma diffusion coefficients are almost radially constant and no temperature pedestal is formed.

TH/1-6 · Profile Formation and Sustainment of Autonomous Tokamak Plasma with Current Hole Configuration

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Abstract: The tokamak plasma with the current hole (CH) configuration is considered an attractive candidate for the steady-state reactor due to its high confinement and high bootstrap current fraction. We have investigated the profile formation and sustainment of tokamak plasma with the CH by using 1.5D time-dependent transport simulations. A model of the current limit inside the CH on the basis of the Axisymmetric Tri-Magnetic-Islands (ATMI) equilibrium is introduced into the transport simulation. We found that a transport model with the abrupt suppression of anomalous transport in the reversed-shear (RS) region can reproduce the time evolution of profiles observed in JT-60U experiments. The transport becomes neoclassical-level in the RS region, which results in the formation of profiles with internal transport barrier (ITB) and CH in RS plasmas. The CH plasma has an autonomous property because of the strong interaction between a pressure profile and a current profile through the large bootstrap current fraction. The energy confinement inside the ITB determined by the neoclassical-level transport also agrees well with the energy confinement scaling based on the JT-60U data. The scaling means the autonomous limitation of stored energy in the CH plasma. It is found from this scaling that the plasma with larger CH has higher normalized beta value and higher bootstrap current fraction. The plasma with the large CH is sustained with the full current drive by the bootstrap current. In this plasma, the CH size and the normalized beta value are self-determined. The plasma with the small CH and the small bootstrap current fraction shrinks due to the penetration of inductive current. This shrink is prevented by adding an appropriate external current drive.

TH/2-1 · Feedback and Rotational Stabilization of Resistive Wall Modes in ITER

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Abstract: In order for advanced tokamaks to reach sufficiently high beta to be of interest for a reactor, and support sufficient bootstrap current for steady state operation, low-n resistive wall modes (RWM) must be stabilized. Wall stabilization can be exploited by plasma rotation or active feedback. The toroidal stability code MARS has been extended to study the physics of RWM and methods to stabilize these modes. we have shown that feedback stabilization method works best when sensors for the poloidal field are placed inside the first wall. Recently MARS has been used to study active control of RWM for the present ITER design. It was established that a single PID controller (from sensor flux to control current) works well even for pressures quite close to the ideal-wall limit. With stabilization of RWM by plasma rotation, an important part of the physical mechanism is the "damping" or "drag" on the RWM by the rotating plasma. This comes both from Alfvén continuum damping and wave-particle interaction with the motion along the field lines. More recently, we have implemented a semi-kinetic damping model in MARS code, and computed the critical plasma rotation speeds which are in good agreement with the experimental measurements from both DIII-D and JET. Using the semi-kinetic model to model an advanced scenario for ITER, we found that the central critical rotation for wall stabilization is at about 2% of the Alfvén speed, which is close to the rotation speed predicted by transport calculations for ITER. A prudent strategy is therefore not to rely on rotational stabilization but develop active feedback.

TH/2-2 · Halo Current and Resistive Wall Simulations of ITER

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Abstract: A number of ITER relevant problems in resistive MHD concern the effects of a resistive wall: vertical displacement events, halo currents caused by disruptions, and resistive wall modes. Simulations of these events have been carried out using the M3D and the newly developed AEGIS (Adaptive Eigenfunction Independent Solution) linear full MHD code. Toroidal peaking factors of halo currents are calculated with M3D for kink and resistive wall mode disruptions. The growth rate of vertical displacement events is enhanced during disruptions. Kink mode disruptions tend to have much larger halo current toroidal asymmetry than resistive wall modes. The AEGIS code uses an adaptive mesh which resolves thin inertial layers responsible for stabilization of resistive wall modes by plasma rotation. Stabilization of resistive wall modes by rotation is examined for realistic thick walls. Thick blankets are examined for equilibria relevant to ITER and reactors including low mach number plasma rotation.

TH/2-3 · Non-disruptive MHD Dynamics in Inward-shifted LHD Configurations

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Abstract: Recent experiments of the Large Helical Device (LHD) show good confinement in linearly MHD unstable configurations. Aiming to understand the mechanism of the good confinement, two kinds of nonlinear simulations, the three-field reduced magnetohydrodynamics (MHD) simulations and direct numerical simulations of 3D compressible MHD are conducted. The two kinds of simulations show that the pressure- driven MHD instabilities with low toroidal mode numbers can be saturated in the early stage of their development. It indicates the MHD nonlinearity of the plasma can keep the system non-disruptive in LHD. By using sequnece technique for beta increase, the reduced MHD simulations show that plasmas can be stabilized through local flattening of pressure profile around rational surfaces. The direct simulation is good for examining the precise dynamics by including all essences of MHD such as geometry effects, toroidal flows and compressibility into the simulations. While the direct simulations show pressure deformations similar to those observed by the reduced simulations, they also reveal importance of compressibility and toroidal flows. It is shown that about 1/2 of the kinetic energy is occupied by the toroidal components of the velocity. It is also shown that compressibility contributes to reducing the growth of pressure-driven instability.

TH/3-1Ra · Nonperturbative effects of energetic ions on Alfvén eigenmodes

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Abstract: Nonperturbative effects of energetic ions on Alfvén eigenmodes both in the linear property and in the nonlinear evolution were investigated using a particle- magnetohydrodynamic hybrid simulation code . For a JT-60U experiment, where the fast frequency sweeping (FS) mode was observed, a new kind of energetic particle mode (EPM) was found near the plasma center at frequency close to the central frequency of the fast FS mode. Frequency and spatial profile of the new EPM depend on spatial profile of energetic ions. This is a nonperturbative effect of energetic ions on the linear property. We also found nonperturbative effects of energetic ions on the nonlinear evolution. Two types of nonlinear evolution take place depending on the initial energetic ion pressure. When reduced distributions of energetic ions are considered for the initial condition, frequency shifts both upward and downward at a rate of frequency sweeping close to that of the fast FS mode. The change in the EPM spatial profile is small. The frequency shifts are comparable to the linear growth rate of the instability. This suggests that the saturation mechanism of the instability is the particle trapping which gives rise to the frequency shifts of the bounce frequency. The time evolution can be explained by the perturbative approach where the EPM spatial profile is assumed fixed. On the other hand, when a classical distribution is taken for the initial condition, a large redistribution of the energetic ions occurs, leading to frequency downshift, an appreciable change in the EPM spatial profile, and breakdown of the perturbative approach. In this paper, we report the nonperturbative effects of energetic ions on the EPM in JT-60U. Numerical analysis results of toroidal Alfvén eigenmodes (TAE) in LHD are also presented.

TH/4-1 · Mechanisms for ITB Formation and Control in Alcator C-Mod Identified through Gyrokinetic Simulations of TEM Turbulence

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Abstract: Mechanisms for transport barrier control are elucidated via nonlinear gyrokinetic turbulence simulations of trapped electron mode (TEM) turbulence in internal particle transport barriers in Alcator C-Mod, produced with off-axis RF heating. The simulations reveal new nonlinear physics of TEM turbulence and explain the observed transport barrier (density profile) control with on-axis RF heating. The critical density gradient for onset of TEM turbulent transport is nonlinearly up-shifted. Upon exceeding this nonlinear critical gradient, the turbulent particle diffusivity from GS2 gyrokinetic simulations matches the particle diffusivity from transport analysis, within experimental error bars. A stable equilibrium is established with the TEM turbulent diffusion balancing the Ware pinch in the ITB. This equilibrium is sensitive to temperature through gyroBohm scaling of the turbulent transport, which allows control of the density profile with on-axis RF heating. With no core particle source and ∼1 mm resolution density diagnostics, the C-Mod experiments provide a nearly ideal test bed for particle transport studies.

TH/5-1 · Transition from weak to strong energetic ion transport in burning plasmas

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Abstract: The change in non-linear EPM dynamics that accompanies the transition from weak to strong energetic ion transport is discussed in the present work. It is demonstrated that the non-linear threshold in fast ion energy density for the onset of strong convective transport occurring in avalanches is close to the linear EPM excitation threshold. This phenomenology is strictly related with the resonant character of the modes, which tend to be radially localized where the drive is strongest. After the convective loss phase, during which non-linear EPM mode structure is displaced outwards, fast ion transport continues due to diffusive processes. Theoretical analyses, presented here, are the starting point for consistency analyses of operation scenarios in proposed burning plasma experiments. Comparisons between theoretical predictions and both simulation and experimental results are also briefly discussed.

TH/5-2Ra · Theoretical Studies of Alfvén Wave - Energetic Particle Interactions

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Abstract: Several studies of the interaction of Alfvén waves with energetic particles will be reported: (1) We show how the slow frequency-sweeping spectrum of Cascade modes is modified by resonant interaction with the background plasma. (2) We study theoretical predictions for phase space structures to emerge in JET and MAST experiments where rapid frequency sweeping is observed. The observed frequency shifts may determine the absolute amplitudes of internal fields. (3) In a burning plasma we assess destabilization due to 1 MeV negative-ion NBI; alpha particle redistribution due to TAE instability; and damping from mode conversion to kinetic Alfvén waves. (4) We report new Alfvénic instabilities in the MHD second stability regime: an energetic particle mode and a so-called aTAE mode (here a measures the pressure gradient) which has a low instability threshold.

TH/5-2Rb · Fast ion effects on fishbones and n=1 kinks in JET simulated by a non- perturbative NOVA-KN code

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Abstract: A new global nonperturbative hybrid code, NOVA-KN, is employed to simulate resonant MHD type modes in JET plasmas. The NOVA-KN code employs the ideal MHD description for the background plasma and treats non-perturbatively the fast particle kinetic response, which includes the fast ion finite orbit width (FOW) effect. In particular, the n=1 fishbone mode, which is in precession drift resonance with fast ions, is simulated. The NOVA-KN code is applied to model a new type of n=1 MHD activity observed in JET low density plasma discharges with high fast ion (H-minority) energy content obtained during the ICRH. The new n=1 MHD activity is interpreted as the n=1 fishbone mode. The effect of the n=1 fishbone on fast ion transport and sawteeth activity is investigated.

TH/5-3 · Internal kink mode stability in the presence of ICRH driven fast ions populations

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Abstract: The internal kink mode is commonly observed in tokamak experiments and is often responsible for sawtooth instabilities. The presence of fast ions in the plasma, as those generated by auxiliary heating, may not only change sawtooth behaviour but also cause fishbones oscillations to be observed, which are caused by a different branch of the internal kink dispersion relation. Mainly, two different approaches to analyse the internal kink mode stability in the presence of fast ions can be used. One method is based on a perturbative approach. In this case, the fast ions energy functional is taken as a perturbation of the ideal MHD functional and the use of numerical codes like the new version of the CASTOR-K code allow an accurate calculation of this effect. The second method is based on a variational formulation where the full dispersion equation including diamagnetic effects is explicitly solved using simplified expressions for fast ions energy functional and fast ions distribution function. The marginal stability equation allows then the determination of the regions of stability for each of the branches in the parameter space. In this paper both methods were described and applied to explain phenomena observed in the experiments.

TH/5-4 · Theory and Theory-based Models for the Pedestal, Edge Stability and ELMs in Tokamaks

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Abstract: Models are developed for use within integrated modeling codes with the objective of predicting the height, width and shape of the pedestal at the edge of H-mode plasmas in tokamaks, as well as the onset and frequency of Edge Localized Modes (ELMs). The model for relaxed plasma states with flow, which uses two-fluid Hall-MHD equations, predicts that the natural scale length of the pedestal is the ion skin depth and the pedestal width is larger than the ion poloidal gyro-radius, in agreement with experimental observations. Computations with the GS2 code are used to identify micro-instabilities, such as electron drift waves, that survive the strong flow shear, diamagnetic flows, and magnetic shear that are characteristic of the pedestal. Other instabilities on the pedestal and gyro-radius scale, such as the Kelvin-Helmholtz instability, are also investigated. Time-dependent integrated modeling simulations are used to follow the transition from L-mode to H-mode and the subsequent evolution of ELMs as the heating power is increased. The flow shear stabilization that produces the transport barrier at the edge of the plasma reduces different modes of anomalous transport and, consequently, different channels of transport at different rates. ELM crashes are triggered in the model by pressure-driven ballooning modes or by current-driven peeling modes. The model that is used for the ballooning mode stability criterion includes possible access to the second stability region at sufficiently low values of magnetic shear. When an ELM crash is triggered in the model, the temperature, density, and current density profiles are reduced near the edge of the plasma instantaneously on the transport time scale. The profiles are then allowed to rebuild between ELM crashes. The height of the temperature pedestal just before each ELM crash and the ELM crash frequency are both observed to increase with increasing auxiliary heating power in the simulations. The scalings of the pedestal height and ELM crash frequency are studied as a function of heating power, plasma density, isotopic mass, plasma current, and toroidal magnetic field strength.

TH/5-5 · The stability of internal transport barriers to MHD ballooning modes and drift waves: a formalism for low magnetic shear and for velocity shear

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Abstract: Internal transport barriers (ITBs) provide improved confinement for tokamaks, so it is important to understand their stability properties. The plasma profiles near an ITB are characterised by low, or even reversed, magnetic shear and strong velocity shear, both features which lead to a breakdown of the ballooning transformation, which is the conventional approach to stability calculations for short wavelength modes in a torus. At low magnetic shear one finds radially extended mode structures are inhibited, being replaced by a narrower spectrum of 'Fourier modes' localised about adjacent resonant surfaces. A complementary approach, based on solving the recurrence relation satisfied by the amplitudes of these 'Fourier modes', has been developed and applied to the stability of MHD ballooning modes and those micro-instabilities responsible for anomalous transport (i.e. electron drift waves and ion temperature gradient modes, ITGs) at low shear or near a minimum in q. In the case of the MHD modes, inclusion of favourable average curvature allows the existence of stable barriers with high pressure gradients, provided the ITB is sufficiently narrow; the effect of the bootstrap current has also been addressed. In the case of micro-instabilities one finds that, for modes of sufficiently long wavelength, only isolated Fourier modes persist near an ITB; thus extended ballooning modes cannot penetrate to the ITB, which thus acts as a 'barrier' to them and suppresses their contribution to transport. This more analytic work has been complemented by a computational study of the effects of low magnetic shear and velocity shear on MHD ballooning modes. This work shows the transition from conventional ballooning modes driven by unfavourable curvature, to more stable modes in which only the average curvature drive survives, as the velocity shear increases; this transition appears at lower velocity shears at low magnetic shear. However, sufficiently high velocity shears can drive additional instabilities. These computational results on stability and corresponding mode structures provide valuable insights into the role of velocity shear on ballooning modes, guiding an analytic interpretation.

TH/5-6 · Boundary modulation effects on MHD instabilities in Heliotrons

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Abstract: In three-dimensional configurations, the confinement region is surrounded by the stochastic magnetic field lines related to the separatrix or magnetic islands, leading to the fact that the plasma-vacuum boundary is not so definite compared with tokamaks that the various modulations of the plasma-vacuum boundary will be induced around the stochastic region by a large Shafranov shift of the whole plasma, in especially high-beta operations. To examine the modulation effects of the plasma boundary on MHD instabilities, high-beta plasmas allowing a large Shafranov shift are considered in the inward-shifted LHD configurations with the vacuum magnetic axis of 3.6m, where pressure-driven modes are theoretically more unstable compared with experimental observations. The boundary modulation expressed by eliminating the bumpy components has not only significant stabilizing effects on ideal MHD instabilities, but also characteristics consistent to experimental observations, which gives us useful information on reconstructing an MHD equilibrium in Heliotron configurations.

TH/6-1 · Effect of sheared flows on neoclassical tearing modes

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Abstract: The influence of shear flows on the evolution of neoclassical tearing modes (NTMs) is an important issue for long pulse tokamak experiments and future reactor devices due to their role in limiting the achievable plasma pressure. In this paper we study certain aspects of this problem using both analytic and numerical simulation approaches. In our analytic study we examine the influence of sheared poloidal flows on the evolution of a single helicity NTM within the framework of an extended Rutherford model that takes account of inertial effects due to the flow as well as deviations of shape symmetry of the equilibrium magnetic islands. We carry out a systematic parametric analysis of the island evolution equation to assess the influence of these additional contributions to the threshold and saturation amplitudes of the islands. In general we find that pure poloidal flow has a strong destabilizing influence on the mode except in a narrow band of island rotation frequencies. The effect of the flow shear in combination with the island asymmetry brings about a reduction of the saturated island width. We also carry out a detailed numerical

investigation of the evolution of NTMs in an arbitrary sheared flow profile by using an initial value code called NEAR. This code solves a set of reduced generalized MHD equations in which we have incorporated some recently proposed heuristic neoclassical closures. In the linear regime we examine the influence of various flow parameters and profiles on the evolution of classical tearing modes with particular attention to toroidal coupling effects, flow induced equilibrium modifications etc. Since classical tearing modes are suspected to often act as triggers for the NTMs these detailed results can provide useful inputs for such trigger models. In the nonlinear regime we retain neoclassical contributions and examine the island evolution and saturation levels over a wide range of parameter regimes. A detailed quantitative measure of the dependence of the growth rate and the saturated island width on the flow shear parameter is obtained for comparison with experimental observations.

TH/6-2 · Kinetic Calculations of the NTM Polarisation Current: Reduction for Small Island Widths and Sign Reversal Near the Diamagnetic Frequency

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Abstract: The polarisation current associated with a Neoclassical Tearing Mode is studied by means of drift kinetic delta-f simulations. This current has been invoked as a possible explanation for both the observed threshold for the minimum island size that can grow unstable and the scaling of the plasma pressure at the mode onset with the normalised gyroradius. The numerical approach presented in this paper does not require assumptions on the island size or the island rotation frequency, which are in contrast necessary in the analytic theory. The calculations are performed in toroidal geometry in the presence of a helical perturbation. In the case of an island width comparable to the ion banana width (typical for the early phase of a NTM) it turns out that the polarisation current decays linearly with decreasing island width. Moreover it is found that the sign of the polarisation current can flip for rotation frequencies close to the diamagnetic frequency. The kinetic effects mentioned above are not included in the present theory and must be considered in order to determine both sign and size of the polarisation-current contribution to the NTM evolution.

TH/6-3 · A Possible Mechanism for the Seed Island Formation

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Abstract: The theory of neoclassical tearing modes (NTM), supported by experimental evidences, requires existence of large enough seed island for the NTM onset. The seed island must be induced by some MHD event. Usually sawteeth, fishbones, or edge localized modes (ELMs) are considered as possible candidates, however, the mechanism of the seed island formation has not been resolved experimentally and remains unclear. Here we discuss a possible role of the error fields in destabilizing the NTM. The error fields are an inherent property of any device, therefore, they can serve as a destabilizing factor when all other destabilizing mechanisms are absent. It is shown that the error field amplification near the plasma stability boundary can result in a sufficiently large magnetic perturbation. This process is essentially dynamic. The time necessary for the growth of the perturbation to the critical level of NTM excitation is estimated. Within the model, the NTM onset must be preceded by slowly growing perturbation similar to that observed in DIII-D experiments on resistive wall modes.

TH/7-1 · The Confluence of Edge and Core Turbulence and Zonal Flows in Tokamaks

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Abstract: We report on gyrofluid and gyrokinetic numerical studies of edge and core turbulence in tokamak geometry, with emphasis on the interaction between spatial regions of differing physical character: core and edge regimes, and the edge/SOL interface. The core/edge transition is parametric, following the parallel/perp scale ratio. For high performance plasmas the core is significantly modified by finite beta effects, with microtearing entering in the electron channel. The edge/SOL transition occurs as field line connection is broken in the SOL, with flute mode character resulting. Geodesic curvature is found to couple zonal flows to the global Alfven mode, whose resistive dissipation acts as an overall sink, preventing self generated transport barrier formation in the edge. The edge shear layer is provided by the sheath dissipation in the SOL, and acts to suppress the smaller scales but drive the larger ones. Pedestal modelling must therefore consider the equilibrium and turbulence as a unified, electromagnetic dynamical system.

On the gyrokinetic level, a new numerical scheme shows highly accurate representation of the kinetic shear Alfven frequency and damping rate. Rapid progress is being made at present time towards a full-f gyrokinetic code applicable to the edge region, able to treat arbitrary levels of nonlinearity and parameter inhomogeneity.

TH/8-1 · Global Gyrokinetic Simulations of Toroidal Electron Temperature Gradient Driven Mode in Reversed Shear Tokamaks

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Abstract: The electron temperature gradient driven (ETG) turbulence is considered as one of experimentally relevant electron transport mechanisms in tokamak plasmas. In recent local flux tube toroidal ETG simulations [2], it has been shown that in the moderate normal shear configuration, an enhanced electron thermal diffusivity χ_e, which is order of magnitude larger than the mixing length estimate, may be caused by radially elongated structures or streamers. On the other hand, in our previous global slab ETG simulations [1], it was found that in the reversed shear configuration, χ_e may be suppressed by ETG driven zonal flows in weak magnetic shear regions near the q_{min} surface. In order to understand these qualitatively different turbulent structures, our previous work is extended including toroidal effects which may significantly affect on the turbulent structure through toroidal driving effects and zonal flow damping effects. In the present work, using a global gyrokinetic toroidal particle code GT3D [3], the toroidal ETG turbulence is first studied in reversed shear tokamaks. From the simulation results, it is found that turbulent structures in the positive and negative shear regions show qualitatively different features. In the negative shear region, the ETG driven zonal flows, which suppress χ_e, are sustained even in the presence of collisionless zonal flow damping effects. In the simulation, zonal flow generation processes are similar to those observed in slab ETG simulations [1]. On the other hand, the positive shear region is characterized by radially elongated structures or streamers. The results suggest a correlation between streamers and (linear) toroidal driving effects which depends on the sign of the magnetic shear through the magnetic drift frequency. According to the simulation results, at least for the ETG turbulence, transport suppression by zonal flows could be expected in the reversed shear configuration. [1]Y. Idomura et al., Phys. Plasmas 7, 3551 (2000). [2]F. Jenko et al., Phys. Rev. Lett. 89, 225001 (2002). [3]Y. Idomura et al., Nucl. Fusion 43, 234 (2003).

TH/8-2 · Advances in Comprehensive Gyrokinetic Simulations of Transport in Tokamaks

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Abstract: This paper illustrates recent advances in gyrokinetic simulations of tokamak transport with the GYRO code: a comprehensive nonlinear continuum (Eulerian) gyrokinetic code which can treat either (i) gyroBohm scaled flux tubes at vanishing $\rho_\star$ or (ii) full-radius core profiles at small but finite $\rho_\star$. It contains the physics needed for physically realistic simulations of the tokamak core: toroidal ITG physics, trapped and passing electrons, electron-ion pitch angle collisions, electromagnetic effects, real geometry, $E \times B$ and magnetic flutter transport, $E \times B$ and diamagnetic rotational shear stabilization which can effectively break gyroBohm scaling, as well as parallel rotational shear drive. Bohm-scaling of DIII-D L-mode $\rho_\star$ dimensionally similar pairs is illustrated. Simulated and experimental power flows match after 10% reductions in the ion temperature gradients. Recent additions include impurity dynamics, ion-ion collisions, neoclassical drivers with neoclassical flows embedded in turbulence, dynamos, and transport feedback algorithms tuning profiles to match experimental flows.
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TH/8-3Ra · Intermittent Transport and Relaxation Oscillations of Nonlinear Reduced Models for Fusion Plasmas

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Abstract: In order to understand the common nature of relaxation oscillations and associated non-local/non-diffusive anomalous transport that are widely observed in fusion plasmas, we have studied nonlinear evolutions of drift-wave and magnetohydrodynamic (MHD) instabilities, using various nonlinear reduced models describing them. One of the new and most notable results we have obtained in this study is the clarification of the minimum mode structures that are necessary to cause relaxation oscillations for

ion temperature gradient (ITG) driven instabilities. To study this, we have constructed a low degree-of-freedom model for ITG modes, which is based on straightforward Fourier expansion of the toroidal ITG equations in the x (radial) and y (poloidal) directions. Using this model, we have examined how "low" the degrees of freedom can be in order for the model to exhibit relaxation oscillations that are at least qualitatively similar to those observed in actual fusion plasmas. Characteristics of anomalous transport (such as anomalous heat transport scaling) obtained from the low degree-of-freedom model are also compared with those obtained from the full- mode model equations. Another nonlinear instability we examined using numerical simulations of reduced equations is the resistive drift mode, which is generally considered to account for particle losses in the edge plasma region. For this problem, we solved time evolution of the governing nonlinear partial differential equations (PDE), known as Hasegawa-Wakatani equations, directly in the three dimensional space. In the presence of a sufficiently large density gradient, we have observed relaxation oscillations, which result in bursty density fluxes. As in the case of ITG turbulence, resistive drift turbulence generates sheared flows, which suppress the fluctuations and are slowly weakened by viscosity. We have also obtained scalings for the anomalous particle flux as functions of density gradient and resistivity in the presence of intermittent transport.

TH/8-3Rb · Velocity-Space Structures of Distribution Function in Toroidal Ion Temperature Gradient Turbulence

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Abstract: Detailed velocity-space structures of ion distribution function associated with ion temperature gradient (ITG) turbulent transport are investigated by means of a newly developed toroidal gyrokinetic-Vlasov simulation code with high velocity-space resolution. It has been confirmed by kinetic simulations of a quasisteady state of the collisionless slab ITG turbulence that micro velocity-scale structures of the distribution function, which are continuously generated by the phase mixing, lead to monotonic increase of an entropy variable defined by a square-integral of the perturbed distribution function. A balance equation for the entropy variable and the transport flux provides a good measure of judging whether the micro velocity-space structures consistent with the turbulent transport are correctly resolved or not. The present toroidal gyrokinetic-Vlasov simulations on the zonal flow and the geodesic acoustic mode (GAM), which accurately satisfy the entropy balance, successfully reproduce the neoclassical polarization of trapped ions as well as the parallel phase mixing due to passing particles. A mean profile of the distribution function resulted from the numerical simulation agrees with a bounce-averaged analytical solution. During the collisionless damping of GAM, finer-scale structures of the distribution function in the velocity space continue to develop due to the phase mixing while preserving an invariant defined by a sum of the entropy variable and the potential energy. Thus, the collisionless dynamics of the zonal flow and GAM are quantitatively understood in terms of a transfer process of the entropy variable from macro to micro velocity-scales. The obtained results suggest necessity of simulating the toroidal ITG turbulence with high velocity-space resolution, in order to accurately reproduce the velocity-space structures of the distribution function and evaluate reliable turbulent transport coefficients. Simulation results of the anomalous transport in the toroidal ITG turbulence, in attention to the velocity-space structures of the distribution function and the entropy balance, as well as a new kinetic-fluid closure model for the zonal flow components are also presented.

TH/8-4 · Electron Thermal Transport in Tokamak: ETG or TEM Turbulences?

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Abstract: The renewed interest in electron temperature gradient (ETG) modes comes from numerical simulations of ETG turbulence using the flux-tube geometry. These simulations find that radially elongated turbulence eddies, or streamers, can drive an electron transport level much higher than the mixing length estimates. However, the radial scale length of the ETG streamers is comparable to the radial box size of the flux-tube simulation. This contradicts the fundamental assumption of the flux-tube simulation with a radially periodic boundary condition. In our studies, a massively parallel, global gyrokinetic toroidal code (GTC) has been utilized to simulate electrostatic the ETG turbulences. We use the "cyclone" parameters of a representative DIII-D H-mode plasma. The size of the tokamak is roughly that of the DIII-D with the minor radius a=8000ρ_e. The simulation geometry is an annulus of r/a=[0.4,0.6] with a fixed radial boundary condition and with the whole flux-surface. We find that the electron thermal conductivity is only about three times of that predicted by the mixing length rule, even though the radial scale length of ETG streamers is comparable to our simulation radial box size of up to $1600\rho_e$. The transport level is well bellow experimental values and is an order of magnitude smaller than that reported by flux-tube simulations. We perform a machine size scan and find that the transport scaling is gyro-Bohm for a>$2000\rho_e$ although the radial length of ETG streamers scales with the machine size. Our GTC results are closer to those from a global fluid simulation of ETG modes with a small tokamak size of a=$100\rho_e$, and in the sheared slab geometry. We further find that the ETG transport level is much smaller than that driven by the trapped electron mode (TEM) turbulence for the same plasma parameters. The TEM driven electron thermal conductivity is found to be at a level of experimental relevance. Our studies cast doubt on the validity of the flux-tube simulations of the ETG turbulence, which claim that the ETG-driven electron transport is of experimental relevance.

TH/8-5Ra · Dynamics of large-scale structure and electron transport in tokamak microturbulence simulations

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Abstract: It is well-known that zonal flows can suppress turbulent fluctuations and relevant anomalous transport in fusion plasmas, while streamers may enhance the heat loss. An important question whether the zonal flow or streamer is preferentially formed in plasma turbulence with electron gyro-radius scale is studied based on general modulation stability analysis and gyrofluid simulations. Magnetic shear is found to play a key role in controlling the formation of zonal flow or streamer in electrostatic slab electron temperature gradient (ETG) driven turbulence. The underlying mechanism may result from the shear dependence of the turbulent fluctuation structures, which determines the generation of different large-scale structures. Three dimensional gyrofluid simulations of slab ETG turbulence show that in weak shear plasmas, ETG-generated zonal flows are enhanced so that the electron transport is strongly suppressed. Contrarily, radially elongated streamers are observed in stronger shear ETG turbulence. In a toroidal plasma, streamers are formed in bad curvature region and high electron transport is dominated by a few modes. The generation of streamer in toroidal ETG turbulence seems to relate to the saturation mechanisms of ETG turbulence. Further, we present significant evidences to show that the enhanced zonal flows in weak shear ETG turbulence may be damped by a Kelvin-Helmholtz (KH) instability. Also, results on the complicated mutual interaction among ETG turbulence, zonal flows and KH mode in finite- Beta toroidal ETG turbulence will be addressed.

TH/8-5Rb · Study of drift wave-zonal mode system based on global electromagnetic Landau-fluid ITG simulation in toroidal plasmas

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Abstract: Using a global Landau fluid code in toroidal geometry, an electromagnetic ion temperature gradient (ITG) driven turbulence-zonal mode system is investigated. Two different types of zonal flows, i.e. stationary flows in a low q (safety factor) region and oscillatory ones coupling with the geodesic acoustic mode (GAM) in a high q region are found to be simultaneously excited in a torus. The stationary flows efficiently suppress turbulent transport, while the oscillatory ones weakly affect the turbulence due to their time varying nature. The result indicates that the safety factor q is important in the zonal flow behavior and transport. It is identified that coupling of the zonal flows and poloidally asymmetric pressure perturbations plays a role in transferring zonal flow energy back to the turbulence.

TH/P1-1 · Mechanism of stabilization of ballooning modes by toroidal rotation shear in tokamaks

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Abstract: Ideal magnetohydrodynamic (MHD) ballooning mode is one of the candidates for edge localized activities in tokamaks. The plasma at the edge region often rotates. It was found numerically that toroidal rotation shear stabilizes ideal MHD ballooning modes; a damping phase alternates with an exponentially growing phase in the time evolution of the perturbation energy of a ballooning mode and the mode is stabilized when the damping is strong enough. However, it has not been clarified how the toroidal rotation shear causes the damping. It was tried to explore the stabilization mechanism through an expansion of a ballooning perturbation in a rotating plasma by eigenfunctions of the ballooning equation in a static

plasma. However, such an attempt has not been succeeded because of difficulty in treating the continuous spectrum. In the present paper, we resolve this difficulty by expanding the ballooning perturbation by eigenfunctions of an eigenvalue problem associated with ballooning modes in a static plasma: especially a weight function is chosen such that the eigenvalue problem has only the discrete spectrum. The eigenvalues evolve in time owing to toroidal rotation shear, resulting in countably infinite number of crossings among the eigenvalues. The crossings cause energy transfer from an unstable mode to the infinite number of stable modes. Such transfer works as the stabilization mechanism of the ballooning mode. The knowledge as well as the methodology acquired here will lead to an advanced experimental data analysis including sheared plasma rotation.

TH/P1-5 · Theory of Plasma Eruptions

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Abstract: This paper describes an extension of the non-linear ballooning mode theory to fusion plasmas with toroidal geometry, which introduces new physics, as we describe. We find that one can consider two regions along a magnetic field line. In the first region, in the vicinity of the peak of the mode amplitude, inertia can be treated as a perturbative correction; we call this the "ideal" region. However, far along the field line inertial terms are not perturbative and we call this the "inertial" region. The dominant contribution to the inertia depends on the shaping of plasma via the Mercier index DM. If DM<-3/4 the region far along the field line is not important and line-tied boundary conditions can be used. However, if DM>-3/4, the region far along the field line provides the dominant contribution to the inertia and then the theory must be revisited, which is the purpose of this paper. We first describe the solution in the ideal region, characterised by negligible inertia, but strong non-linearities. From this we identify a parameter delta-prime, which is a matching parameter between the two regions. This is expressed in terms of the plasma displacement and involves non-linearities. In the inertial region the non-linearities are not important (provided DM<0) and can be neglected. The resulting linear system can be solved by Laplace transforming in time, and the delta-prime parameter can again be identified, providing a second expression for it. Matching the results for delta-prime from the two regions provides the final non-linear evolution equation for the plasma displacement. Analysis of this equation shows that this mode develops into a filament-like structure, which is extended along the field line, but localised in the directions perpendicular to it. As a singular time is approached the growth rate increases dramatically, the filament broadens in the radial direction and it narrows in the other direction perpendicular to the field line. A picture of the ELM is proposed as being a consequence of a hot plasma filament pushing out from the core into the colder scrape-off layer (SOL) plasma. The filament connects back into the hot core far along the field line. It therefore provides a route for hot core plasma to escape rapidly into the SOL, and hence to the divertor target plates.

TH/P2-2 · Sheared flow layer formation in tokamak plasmas with reversed magnetic shear

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Abstract: Internal transport barriers (ITBs) in advanced tokamak (AT) discharges are formed due to turbulent transport suppression by sheared flow layers (SFLs) in accordance with anomalous transport theory. Conditions, especially radial positions, for triggering SFL and ITB formation, have been intensively investigated in recent years. However, dynamics for SFL and ITB formation is not yet well understood. The reduced dissipative 2D MHD equations with electron viscosity included are solved as an initial value problem for a single harmonic of perturbations. Nonlinear development of double tearing mode in reversed shear (RS) configurations is simulated. Time evolutions of the mode growth rate, kinetic and magnetic energy, as well as profiles of induced velocity, velocity shear and radial magnetic field are presented. Converting of the magnetic energy into kinetic energy and creation of sizable poloidal SFLs with magnitude of poloidal Alfvén velocity at boundaries of magnetic islands are demonstrated. It is shown that the nature and magnitude of the flows make the mode a strong candidate for the triggering of ITBs in RS tokamak plasmas. Possible correlation with experimental observations is discussed.

TH/P2-3 · Advanced Transport Modeling of Toroidal Plasmas with Transport Barriers

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Abstract: Transport modeling of toroidal plasmas is the most important issue to predict time evolution of burning plasmas and to develop control schemes in reactor plasmas. In order to describe the plasma rotation and rapid transition self-consistently, we have developed an advanced scheme of transport modeling based on dynamical transport equation and applied it to the analysis of transport barrier formation. First we propose a new transport model and examine its behavior by the use of conventional diffusive transport equation. This model includes the electrostatic toroidal ITG mode and the electromagnetic ballooning mode and successfully describes the formation of internal transport barriers. Then the dynamical transport equation is introduced to describe the plasma rotation and the radial electric field self-consistently. The formation of edge transport barriers is systematically studied and compared with experimental observations. The possibility of kinetic transport modeling in velocity space is also examined. Finally the modular structure of integrated modeling code for tokamaks and helical systems is discussed.

TH/P2-4 · Advanced ST Plasma Scenario Simulations for NSTX

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Abstract: Integrated scenario simulations are done for NSTX that address four primary milestones for developing advanced ST configurations: high beta and high betaN inductive discharges to study all aspects of ST physics in the high beta regime; non-inductively sustained discharges for flattop times greater than the skin time to study the various current drive techniques (NB, HHFW, and EBW); non-inductively sustained discharges at high beta for flattop times much greater than an skin time which provides the integrated advanced ST target for NSTX; and non-solenoidal startup and plasma current rampup. The simulations done here use the Tokamak Simulation Code (TSC) and are based on a discharge 109070. TRANSP analysis of the discharge provided the thermal diffusivities for electrons and ions, the beam deposition profile and other characteristics. CURRAY is used to calculate the HHFW heating depositions and current drive. GENRAY/CQL3D is used to establish the heating and CD deposition profiles for electron Bernstein waves. Analysis of the ideal MHD stability is done with JSOLVER, BALMSC, and PEST2. The simulations indicate that the integrated advanced ST plasma is reachable, obtaining a stable plasma with beta of 41.3% at betaN of 8.85, Ip = 1.0 MA and BT = 0.36 T. The plasma is 100% non-inductive and is quasi-stationary for 4 skin times. The resulting global energy confinement corresponds to a multiplier of H98 = 1.5. The simulations have demonstrated the importance of HHFW heating and CD, EBW off-axis CD, strong plasma shaping, and density control for producing and optimizing these plasma configurations.
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TH/P2-9 · Progress in Transport Modelling of Internal Transport Barrier and Hybrid Scenario Plasmas in JET

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Abstract: This paper will report on the recent progress in transport modelling of advanced tokamak scenarios in JET, including both the Internal Transport Barrier (ITB) plasmas and hybrid scenarios. Several issues will be covered, such as fully predictive transport modelling of ITBs, micro-stability analyses of ITBs, predictive closed-loop transport simulations of the q-profile and ITBs with real-time control, and understanding of the transport mechanisms leading to the improved core confinement in hybrid scenarios. With the semi-empirical Bohm/GyroBohm model, the agreement with respect to the radial location and the dynamics of the ITB between the experiments and transport simulations is good in JET. In JT-60U, the model performs as well as in JET, but in DIII-D the agreement is worse, suggesting different transport mechanisms that lead to ITB formation. The model predicts the alpha-stabilisation playing a crucial and more important role in DIII-D than in JET and JT-60U. The first principle transport models, Weiland and GLF23, have much worse agreement with the experiments than the Bohm/GyroBohm model. For the first time fully predictive transport simulations with a non-linear plasma model have been used in closed-loop simulations to control the q-profile and the strength and location of the ITB. Five transport equations are solved and the power levels of LHCD, NBI and ICRH are feedback controlled by the difference between the set-point and simulated values of q. Closed-loop simulations with JETTO real-time control algorithms

are able to approach and sustain various set-point q- profiles from monotonic to deeply reversed ones. The major difference between the experimental and simulated results is the time for the control to reach the set-point values of q – the time in the simulations being often a factor of 2 larger. This suggests that there are some ingredients in the experimental current diffusion that are faster than predicted by neo-classical theory or some discrepancies between the actual and simulated LH driven current. In the hybrid scenario, predictive transport modelling aims to clarify at least the roles of low magnetic shear, ELM behaviour, H-mode pedestal and toroidal rotation in creating the improved transport properties of the hybrid scenario.

TH/P2-10 · Pair Vortices Formation near magnetic Axis as an Explanation of the "Current Hole"

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Abstract: Some negative currents should be driven in the central region of the tokamak by bootstrap current and off-axis current drive, when the amplitude of driven current is large enough. Once a surface with zero poloidal magnetic field appears, however, a toroidal equilibrium is lost and a static state cannot exist. A pair of vortices with counter rotation grows in this case. Once the vortex rotation grows enough, the plasma current profile is kept flat by this convective motion. We investigate the growth of this convective motion and find the appearance of the flat current profile, the formation of a current hole, by resistive MHD simulations. After the current hole is formed, additional current drive to the central region is also inhibited by the plasma flow.

TH/P2-13 · Steady State Solutions to Neoclassical Transport Equations and Absence of Bifurcate Solutions for the Ambipolar Electric Field

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Abstract: Steady state solutions to the transport equations for stellarator plasmas in low collisional regime are derived and their stability is analyzed in terms of the purely neoclassical theory (without allowance for anomalous losses). It is shown that there exists a unique steady state continuous solution for the ambipolar electric field, i.e., in the case in question, the bifurcation of plasma states do not exist.

TH/P2-18 · Finite-Orbit-Width Effect and the Radial Electric Field in Neoclassical Transport Phenomena

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Abstract: In neoclassical (NC) transport theory, it is usually assumed that the orbit width Δ_b is negligibly small compared to the typical scale length of background plasma L. This small-orbit-width approximation, however, is sometimes not valid in realistic fusion plasma, for example at the internal transport barrier or around the magnetic axis where wide-width potato orbit appears. To evaluate neoclassical transport level in such cases, the finite-orbit-width (FOW) effect should be considered. Another recent interest in NC transport theory is the evolution of the radial electric field in tokamaks. If the higher order terms in $O(\Delta_b/L)$ are retained, intrinsic ambipolarity in tokamaks breaks and the radial electric field develops so as to satisfy the ambipolar relation $\Gamma_i(E_r) = \Gamma_e(E_r)$. Sheared $E \times B$ flow or the zonal flow, which is formed as a consequence of non-linear saturation of microinstabilities, is considered to reduce anomalous transport level. The relation between NC transport and microinstabilities on the evolution of the electric field has not been discussed well, but the formation of E_r field from NC transport mechanics is expected to affect the reduction of anomalous transport. We have investigated NC transport phenomena including the FOW effect with two approaches. One is the new formulation of a transport theory for near-axis region, in which the FOW effect of potato orbits around the axis is correctly evaluated by using Lagrangian description of the drift-kinetic equation. The NC heat conductivity around the axis is shown to reduce as found in recent experiments. The other is a more direct way by the δf method of the Monte-Carlo transport simulation. In the simulation, we calculate NC fluxes and evolution of the E_r profile simultaneously. We will show the recent results from the numerical simulation such as the global simulation of the geodesic acoustic mode (GAM) and the FOW effect on it, and the extension of the δf code to non-axisymmetric configurations. Comparison of the GAM and radial electric field profiles between NC transport simulation and the gyrokinetic simulations are also shown to discuss the relation of these two transport mechanics.

TH/P2-19 · Transport in a small aspect ratio torus

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Abstract: Calculations of collisional thermal and particle diffusivities in toroidal magnetic plasma confinement devices order the toroidal gyroradius to be small relative to the poloidal gyro-radius, i.e $\rho_\phi << \rho_\theta where \rho_\phi = mv/qB_\phi and \rho_\theta = mv/qB_\theta$. This ordering is central to what is usually referred to as neoclassical transport theory. This ordering is incorrect at low aspect ratio. This means that excursions of a particle from its nominal flux surface are much larger than estimated in neoclassical theory, with a consequent increase in radial transport. We calculate the correction to the particle and thermal diffusivities at low aspect ratio by comparing the diffusivities as determined by a full orbit code [R. B. White, L. Chen, Z. Lin, Phys. Plasmas 9 1890 (2002)] (which we refer to as omniclassical diffusion) with those from a gyro-averaged orbit code [R. B. White and M. S. Chance Phys. Fluids 27, 2455 (1984)] (neoclassical diffusion). In typical low aspect ratio devices such as the National Spherical Torus Experiment [Y-K. M. Peng and D. J. Strickler, Nucl. Fusion 26, 769 (1986), M. Ono, S. M. Kaye, Y. -K. M. Peng, et al Nucl. Fusion 40, 557 (2000)] the omniclassical diffusion can be up to 2.5 times the calculated neoclassical value. Analytical expressions are obtained which are in good agreement with numerical simulation. We also verify numerically that the bootstrap current is correctly given by the guiding center approximation, even when the diffusion is much larger than neoclassical. We discuss the implications of this work for low aspect ratio magnetic confinement experiments.

TH/P2-23 · Instability Suppression by Sheared Flow in Dense Z-Pinch Devices

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Abstract: The study of local MHD instabilities with sheared equilibrium flows, in cylindrical geometry, is ideal for the understanding of this kind of problems, since the symmetry allows simpler calculations. The feasibility of stabilizing a Z-pinch by means of an axial shear flow has been studied theoretically, numerically and experimentally (T.D. Arber and D.F. Howell, Phys. Plasmas 3, 554 (1996), U. Shumlak et al., Phys. Plasmas 10, 1683 (2003)). Since the results of the MHD stability analysis can depend on the equilibria that are being considered, we start by analysing families of self-consistent equilibria with sheared flow, which could represent feasible starting points for the stability analysis. Proposing binomial shapes for the current density radial profiles, we find families of equilibria, which are qualitatively different. In the case when only axial sheared flows are considered, the equilibrium pressure profiles are independent of the flow, but when azimuthal flows are included, the pressure surfaces depart form the magnetic field surfaces as the flow increases. The electric field is determined by the generalised Ohm's law. This leads to small deviations from local quasi-neutrality, as is found from Poisson's equation. The self consistency of the flow is related to such deviations, which can be proposed as a function of the plasma density. It is found that for certain flows, self consistent equilibria can only be possible if the plasma is non-neutral. Both the cases when finite Larmor radius effects are ignored or taken into account are considered, and the relevant parameters necessary for experimental study are established. The study of m=0 and m=1 instabilities is done for some of the equilibria which show significant sheared flows, and the reduction in their growth rates is classified. A comparison is made with the results of previous works, where usually no self consistency with the equilibrium sheared flows was considered. We also study the properties of a nozzle design that ejects a converging supersonic jet on the axis, which could be used as a target for the current sheath in plasma foci. This converging flow results in the formation of a reflected shock beyond the exit of the nozzle, in which the density is enhanced. A numerical simulation of this design is carried out, using an adaptive grid code that solves the gas-dynamics equations.

TH/P2-24 · Influence of pressure-gradient and shear on ballooning stability in stellarators

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Abstract: The sensitivity of ballooning stability boundaries to profile variations is addressed for stellarator equilibria. A semi-analytic method for calculating the ballooning growth rate as functions of the pressure gradient and averaged magnetic shear is introduced. The simplicity of the expressions allows for interpretation of the important physical effects at work in determining instability thresholds. The analysis determines whether or not a given stellarator configuration will possess a region of second stability and the strength of this second stable region. Whether such regimes can be accessed in a given device depends upon the interaction of the pressure gradient with curvature and the local shear. This interaction is cleanly accounted for using the method of profile variation whereby self-consistent local 3- D equilibria are calculated for arbitrary changes in the pressure and rotational transform gradient. The theory is applied to various configurations of experimental interest.

TH/P2-25 · Simulations of the Disruption of a DIII-D Plasma with the NIMROD Code

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Abstract: Understanding the onset and nonlinear dynamics of disruptions is crucial for preventing or mitigating them in next-step devices. Initial-value simulations with the NIMROD code of an ideal-MHD unstable plasma based on DIII-D discharge #87009 allows for detailed studies of the dynamical mechanisms of the disruption and the resultant heat flux distribution on the wall. The ideal mode grows and causes 2/1 magnetic islands as a result of forced reconnection at the two 2/1 surfaces. The mode amplitudes continue to grow until the magnetic islands overlap and the magnetic field is stochastic over a large part of the plasma domain. The rapid stochastization of the field allows the plasma to lose two thirds of its internal energy in approximately 200 microseconds in qualitative agreement with the experiment. The deposition of thermal energy on the wall is localized poloidally and toroidally on the wall due to helically-localized temperature increases and parallel heat flux carrying this increased heat flux to the wall. Understanding the heat flux localization requires a detailed understanding of the three-dimensional field line structure as the plasma undergoes changes in topology.

TH/P2-29 · Toroidal Momentum Confinement in Tokamaks and Magnetic Reconnection

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Abstract: Toroidal momentum confinement in tokamaks is anomalous. We present mechanisms that can affect it. (1) Both turbulent fluctuations and magnetohydrodynamic (MHD) activity degrade momentum confinement. We find that there is a convective term in the fluctuation-induced toroidal stress in a quasi-linear theory consistent with observations. The ratio of the diffusion to the convective term depends on the frequency spectrum. Because the magnetic surfaces are distorted in the presence of MHD activity, the —B— on the surface is toroidally asymmetric. This leads to enhanced toroidal momentum losses. (2) The connection between the "spontaneous rotation" phenomenon and the transport processes that are at the basis of the accretion theory of this phenomenon is found to be consistent with all the main relevant experimental observations. In particular, toroidal travelling modes whose source of excitation is at the edge of the plasma column are considered, which can transport net angular momentum toward the center. Thermal energy transport and magnetic reconnection are shown to be intrinsically related by the theory of the drift-tearing for which characteristic non-linear effects are pointed out.

TH/P2-30 · Two-fluid limits on stellarator performance: Explanation of three stellarator puzzles and comparison to axisymmetric plasmas

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Abstract: Experimental studies of the stability and operational limits of stellarators have generated a number of puzzling observations that resist explanation by MHD. We show that basic nonlinear two-fluid effects, those required beyond MHD to generate the self-consistent diamagnetic drifts, relax the major MHD limits on stellarator performance in a way that can explain these puzzles. Numerical simulation with the initial value code M3D shows that two-fluid effects effectively stabilize ideal MHD ballooning modes above a certain moderate-to-high mode number, as well as resistive MHD ballooning and interchange modes at their most unstable moderate-to-high mode numbers. The results suggest an intrinsic "soft" beta limit for stellarators at high electron beta, where two-fluid magnetic islands grow large enough due to the electron pressure gradient in Ohm's law, to seriously reduce plasma confinement and prevent further plasma heating, rather than a "hard" limit caused by instability. This picture more closely matches experiment, where ballooning and interchange modes, although often MHD unstable, rarely appear. Differences in helical and axisymmetric configurations are examined, with emphasis on the NCSX stellarator.

TH/P2-31 · Theoretical considerations of doublet-like configuration in stellarators

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Abstract: In order to understand the basic physics of the separatrix structure inside the plasma to the magnetohydrodynamics (MHD) equilibrium, stability and particles confinements, the doublet-like configuration of LHD are studied using the HINT code. The doublet-like configuration in stellarators has an eight-figured separatrix and two axes split by the external quadrupole field inside the plasma. For the doublet-like configuration in stellarators, since magnetic axes are split by the external field, vertical and horizontal split axes configurations exist. The rotational transform becomes zero at the separatrix and the magnetic shear is reversed. From equilibria obtained from HINT, finite beta effects to the separatrix structure are estimated. For finite beta equilibria of vertical split configuration, though the horizontal shift of two axes toward the outside of the torus is very small, the vertical shift from the equatorial plane is very large. Since two flows of Pfirsch-Schulter current is appeared by finite beta effects, the eight-figured separatrix is increased and sustained. On the other hand, the structure of magnetic field lines outside the separatrix is ergodized by finite beta effects. The rotational transform on axes are greater than one and the magnetic well becomes deeply for high beta equilibria. In order to investigate the influence of the separatrix structure to the particle confinement, the drift orbit of high-energy particle is studied. Since the safety factor is very large near the separatrix, the poloidal drift width is changed broadly. In such regions, the local transport theory is unsuitable. The influence of the bootstrap current is also studied. Since the direction of the rotational transform on two axes is same direction, two flows of the bootstrap current are appeared. The eight-figured structure is also changed and sustained by the bootstrap current.

TH/P3-1 · The evolution of the transport coefficients for the transient process after the ECRH switching on/off in tokamak T-10

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Abstract: The analysis of experiments with ECRH requires a good knowledge of ECRH power profile. This profile is reconstructed by the analysis of the transient process after on- axis ECRH switch on/off in special experiments with suppressing of sawtooth oscillations in T-10 tokamak. The calculations show that the absorbed ECRH power, determined by the change of time derivative of the electron temperature at the region of ECRH power input and the absorbed ECRH power, determined by the magnetic measurements are several times different. The analysis of various explanations of this effect shows that the adequate description of the transient process demands the introduction of the ballistic jump of the total heat flux just after the on-axis ECRH switching on. One of the possible explanations can be the appearance of the particle and heat convection velocity just after on-axis ECRH switch on. In fact, in all ECRH heated experiments the density reacts to electron heating, the density profile become flat with central ECRH. This effect is known as "density pump out". Therefore the heat convection outflows the heat from the heating region to the plasma edge. The careful analysis of the measurement of the temperature, the density and soft X-ray emission was done. It was shown that the density changes over the whole plasma cross-section just after the on-axis ECRH switch on/off. This experimental data confirms the appearance of the additional heat convection after the on-axis ECRH switch on/off throughout the cross-section.

TH/P3-5 · Electron Thermal Transport in Tore Supra and NSTX

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Abstract: We perform transport analysis of Tore Supra and NSTX discharges with centrally deposited fast wave electron heating, to derive the electron thermal fluxes from power balance. Measurements of the electron temperature and density profiles, combined with ray tracing computation of the power deposition, allow detailed interpretation of the thermal flux versus temperature gradient. We find strong evidence in Tore Supra for electromagnetic electron temperature gradient (ETG) turbulent transport: (i) critical temperature gradient and its dependence on magnetic shear, (2) small-scale fluctuations in electron density and magnetic field, and (3) thermal flux scaling with density and electron temperature that is consistent with collisionless skin depth mixing length formulas for electromagnetic ETG turbulence. Similar power balance analysis is underway for a High Harmonic Fast Wave heated NSTX discharge with Te0 = 4 keV, showing electron transport is explained by the ETG model. Theoretically, we explain how the

reversed magnetic shear produced in optimized NSTX/HHFW discharges can partially block the transport of electron guiding centers through formation of a shearless invariant curve in the corresponding drift wave transport model. Weakly reversed magnetic shear discharges in NSTX have shown global electron transport reduced in half, with an associated increase of impurity ions due to an inward electric field.

TH/P3-6 · Self-consistent modelling of L-H transition and H-mode pedestal characteristics

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Abstract: The 1-D transport code RITM [1] has been extended for a self-consistent modelling of plasmas strongly influenced by impurities. It was previously applied for modelling of L-mode plasmas in several tokamak and, in particular, of the transition to the RI- mode in TEXTOR. Recently RITM has been amended by including a model for anomalous transport contribution driven by drift-Alfven (DA) turbulence [2]. This allows to simulate H- mode plasmas in a self-consistent way, from the plasma axis to the separatrix including the edge transport barrier. It is demonstrated that the formation of edge pedestal requires suppression both DA and ion temperature gradient (ITG) [3] unstable modes. The former occurs due to decreasing plasma collisionality and increasing beta with growing heating power. The stabilisation of ITG turbulence is caused essentially by strong density gradients, which develops at the edge due to ionisation of incoming neutrals when DA transport is reduced. On the one hand, this explains the proportionality of the radial pedestal width to the penetration depth of charge-exchange neutrals, which scales as the square root of the ion temperature. On the other hand, the temperature gradient, which drives ITG instability, is controlled in the barrier by the neoclassical heat conductivity and, therefore, the pedestal width scales inversely with the plasma current. All together this leads to a proportionality of the barrier width to the poloidal Larmor radius, widely observed in experiments. The seeding of argon can lead to a significant modification of the edge transport, in particular, to a widening of the edge barrier and an increase of the plasma density by preserving a good confinement quality. However, a too intensive edge radiation from impurities, which cools the plasma edge too strongly, results in a resumption of DA activity and can trigger a H to L back transition.

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TH/P3-7 · Particle simulation of plasma turbulence and neoclassical Er at tokamak plasma edge

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Abstract: Numerical simulation of tokamak plasmas is very complex especially at the edge, where the gradients are steep and interaction with divertor and wall structure causes more complications. Simulations of various phenomena using five dimensional Monte Carlo guiding centre orbit following code ASCOT, which simulates neoclassical physics, and its gyrokinetic upgrade ELMFIRE, which takes into account also electrostatic turbulence, are presented.

TH/P3-9 · Recent Advances in the Theory and Simulation of Pellet Ablation and Fast Fuel Relocation in Tokamaks

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Abstract: This paper presents new theory and simulation of pellet ablation, and the rapid cross-field relocation ionized pellet substance following HFS pellet injection in tokamaks. A unique time-dependent 2-D Eulerian code CAP, was developed that is the first to self- consistently treat the key B-field effects: (1) Pellet deformation into a "pancake" shape, driven by the anisotropic surface ablation pressure, can cut pellet lifetimes by almost ∼3x (2) $J \times B$ funneling of the flow into a field-aligned cigar-shaped structure enhances shielding. Near-pellet cloud parameters from CAP are critical inputs for PRL and AMR codes that model fast advection of the "detached" clouds accelerated by the ∇B effect. PRL contains new geometrical effects of toridicity, magnetic shear, and curvature drifts by parallel flows. Consequently, the calculated fuel deposition is in better accord with density measurements on DIII-D, providing improved predictive capability for ITER. A new 3-D MHD simulation code AMR can provide the required fine-scale

mesh size needed for accurate modeling of strongly localized pellet clouds.
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TH/P4-2 · Effects of Steep Gradients and Stochasticity on the Rotation Dynamics of Collisional Tokamak Edge Layer

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Abstract: The poloidal and toroidal rotations in high collisionality regime at the toroidal edge plasma with steep thermal and density gradients were investigated in several papers (e.g. by Claassen, et al., Czech. J. Phys. 49, Suppl. S3, 69 (1999) and Claassen et al, Phys. Plasmas, 7 (9) 3699 (2000)) using modified stress tensors by finite Larmor-radius effects and corrections due to Mikhailowsky and Tsypin. We further improve their formalism, considering fast and slow time scale derivatives, to determine plasma rotations and their stability for given initial conditions with random temporal and spatial perturbations on radial temperature and density profiles. We apply the method of multiple scales as well as numerical methods to the solution of the equations for velocity components. Perturbation of regular profiles by such random components can, under specific circumstances, lead to a chaotic behaviour of rotation speeds, when the coupled equations for the velocities act like stochasticity amplifiers or even as stochasticity generators.

TH/P4-7 · Modeling and measurement of eddy currents in the ETE spherical tokamak

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Abstract: A series of calculations and measurements have been made to evaluate the effect of eddy currents in the Brazilian spherical tokamak ETE. Eddy currents occur both in the central column of the conventional copper toroidal field magnet, around which the ohmic heating solenoid was tightly wound, and in the continuous wall of the vacuum vessel manufactured from inconel alloy. During the startup phase, the currents circulating in the central column strongly affect the performance of the ohmic heating system, while the currents flowing in the vacuum vessel introduce large error fields that must be compensated for successful plasma breakdown and must also be taken into account in the magnetic reconstruction procedure. Elaborate analytical models have been developed to represent both systems of currents. The results of these models are compared with experimental values of the ohmic solenoid impedance, with the operation of the ohmic heating system, and with the distribution of eddy currents in sections of the vacuum vessel.

TH/P4-18 · Importance of Electron Cyclotron Wave Energy Transport in ITER

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Abstract: In order to describe energy transport in a steady-state reactor-grade tokamak plasma, characterised mainly by a good energy confinement and high peak plasma temperatures, not only a reliable model for plasma heat transport by conduction and convection is required, but also transport by electron cyclotron (EC) waves must be modelled satisfactorily. The non-local character of this transport mechanism, due to wave reabsorption and wave reflection, is taken into account by coupling the CYTRAN routine, which provides a reasonable approximation to the exact treatment of EC waves, to the 1.5D ASTRA transport code. For ITER steady-state operation conditions, the main result is that the net EC wave emission in the plasma core effectively may provide the most important cooling mechanism for electrons. Sensitivity studies of the energy transport in the plasma core to both the efficiency of electron heating and/or cooling and the wall reflectivity are performed, and the electron temperature profiles resulting from a local approach to EC wave emission are compared to those obtained from the non-local formalism. The effect of the EC wave energy transport on the plasma temperature is also analysed for analogous conditions of other reactor-grade tokamak plasmas.

TH/P4-20 · Theoretical Studies of non Inductive Current Drive by Oscillating Magnetic Fields, Neutral Beams and Helicity Injection in High Beta Plasmas

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Abstract: Recent advances on theoretical studies of various non inductive current drive methods, neutral beam injection (NBI), oscillating magnetic fields and helicity injection (HI), which apply to compact toroids and RFPs are presented. A Monte Carlo code is employed to study current drive by NBI in moderate flux Field Reversed Configurations (FRC) sustained by Rotating Magnetic Fields (RMF). The dimensions and parameters of the FRC are similar to those obtained in the Translation, Confinement and Sustainment (TCS) experiment (Univ. of Washington, USA). The current drive efficiency is much lower for injection through the ends, at a small angle to the FRC axis, than for perpendicular injection along the separatrix. The RMF degrades beam confinement but relatively high efficiencies can be obtained provided the RMF does not penetrate too deeply into the plasma. The conditions required for efficient current drive in a weakly resistive, infinitely long, plasma column subject to external, time dependent, helical magnetic fields and a uniform steady axial field are determined. A non linear Ohm's law is employed for the electrons and the ions are considered fixed. The efficiency is low when a large external axial field is present (as in tokamaks) but there is a range of small external fields (relevant for RFPs) where significant efficiencies result. A two fluid model, with finite electron inertia, is employed to study the possibility of using two counter rotating magnetic fields to drive electrons and ions in opposite senses in a FRC. Analytical calculations show the existence of steady-state solutions where equal and opposite torques are applied to electrons and ions and a time dependent code has been developed to test the stability and accessibility of these solutions. The principle of minimum rate of energy dissipation is employed to calculate relaxed states for a flux core spheromak sustained by HI. The values of external parameters (voltage, flux, current) where a transition to helical states occurs are determined.

TH/P4-30 · A Global Simulation of ICRF Heating in a 3D Magnetic Configuration

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Abstract: In a three-dimensional (3D) magnetic configuration, complicated drift motions of trapped particles would play an important role in the confinement of the energetic ions and the ICRF heating process. Additionally, since the wavelength of the ICRF heating is typically comparable to the plasma scale length and the 3D geometry effect on the RF wave field would be also important in a 3D magnetic configuration. Therefore a global simulation of ICRF heating is necessary for the accurate modeling of the plasma heating process in a 3D magnetic configuration. In this paper we study the ICRF heating in a 3D magnetic configuration combining two global simulation codes; a drift kinetic equation solver GNET and a wave field solver TASK/WM. We apply the simulation code to the LHD configuration as an example. We make clear the characteristics of energetic ions distribution in the phase space, and also show the confinement property of LHD configurations by comparing the simulation and experimentally observed results.

TH/P4-31 · On Electron-Cyclotron Waves In Relativistic Non- Thermal Tokamak Plasmas

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Abstract: A non-thermal feature which is frequently encountered in tokamak plasmas is the presence of a low-density population of superthermal particles. By adopting a model distribution function which possess the main envisaged features of the real distribution one can obtain an insight into the wave propagation and absorption properties in such media. As is well known, the problem of wave propagation in electron-cyclotron frequency range should require a relativistic approach, even for low plasma temperatures in the case of (quasi) perpendicular wave propagation. Much work has been devoted to this subject, however due to the complexity of the fully relativistic dielectric tensor, the results are mainly obtained by using various approximations (e.g. weakly relativistic thermal plasma, inclusion of relativistic effect in ad hoc manner, etc.). In order to investigate properties of ordinary, extraordinary and quasilongitudinal (electron Bernstein) modes in EC frequency range and their coupling in the case of nonthermal plasmas, for arbitrary wave propagation angle, we have developed a numerical code by using expressions for the fully relativistic dielectric tensor which contains products of Bessel functions instead of an infinite sum over harmonics. The perpendicular component (to the magnetic field) of the wave refractive index is treated as a complex quantity and the electron distribution function can be, in general, a non- Maxwellian, anisotropic electron distribution function. As an example of the solutions of the fully relativistic dispersion equation, taking into account non-thermal features of tokamak plasmas, we have examined relativistic bi-Maxwellian plasma. It is found that the location of the cut-offs depends on the weight and tail temperatures, however, the displacement to higher density values is relatively small for low population of superthermal electrons. Furthermore, the presence of superthermal particles induces an extended tail in the absorption profile and changes the location of the wave absorption region in large toroidal devices. This shift may be of consequence in experiments in which localized absorption or current drive is present. Finally, the superthermal tail may effect the mode interaction by decoupling or coupling the extraordinary and quasilongitudinal modes.

TH/P4-33 · Current Drive by Electron Bernstein Waves in Spherical Tori

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Abstract: The high-beta operating regime of spherical tokamaks (ST), such as in NSTX and MAST, make them attractive fusion devices. For access to high beta regimes, it is necessary to heat and to drive currents in ST plasmas. While such plasmas are overdense to conventional electron cyclotron waves, electron Bernstein waves (EBW) offer an attractive means toward this purpose. Besides providing better confinement, EBW driven current could also help suppress neoclassical tearing modes. This paper examines the characteristic features of EBW current drive. The propagation and damping of EBWs are described by the complete relativistic dispersion tensor, and the induced current is studied using the relativistic Fokker-Planck equation. The latter includes a relativistic, quasilinear description of the interaction of EBWs with electrons, and the effect of trapped electrons on the distribution function. Studies relevant to the implementation of EBW current drive on NSTX will be discussed. Along with previous results on coupling and excitation of EBWs, these results will provide a general basis for defining the role of EBWs in present-day ST experiments and future ST power plants.

TH/P4-35 · Nonthermal Particle and Full-Wave Diffraction Effects on Heating and Current Drive in the ICRF and LHRF Regimes

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Abstract: Fast waves (FW) are a primary technique for heating and current drive (CD) on the proposed burning plasma device, ITER, and lower hybrid (LH) waves are a candidate for edge current profile control. The models used to simulate these two waves rely on assumptions of Maxwellian populations that allow efficient analytic implementations of the plasma response, and in the case of the LH wave, the ray tracing models used are able to follow the very small wavelengths in a continuum manner without requiring a fine computational grid. Recent advances in algorithms and parallel computational methods have allowed these assumptions to be tested, permitting more accurate estimates of heating deposition and CD efficiencies in a burning plasma. Absorption by energetic particles for both waves can be significant, reducing electron heating and associated CD. Wave propagation and absorption is dependent on the velocity space distribution of particles in the plasma and on the geometric effects of focusing and diffraction. Fusion born alpha particles and neutral beam ions may interact with these waves in a manner that cannot be accurately modeled by Maxwellian distributions. The AORSA2D code has been modified to use a generalized non-Maxwellian conductivity, and has been applied to ITER reference scenarios. Preliminary analysis for ITER suggests that alpha absorption may be limited to a few tens of percent, and thus, allow reasonable CD efficiencies, assuming that the RF does not significantly alter the alpha slowing-down distribution. We also discuss the interaction of an energetic Tritium tail with FW in ITER. In addition, the effects of diffraction on LH waves in toroidal geometry are not well understood because computational limits have prohibited full-wave simulations at those small wavelengths. Simulations of LH waves have been restricted to WKB ray tracing techniques and 1D full-wave in the past, but the availability of massively parallel architectures have made full-wave calculations using an electromagnetic field solver tractable. The TORIC code has been adapted to run on parallel architectures making it possible to resolve the slow electrostatic LH wave. We present full-wave simulations of LH slow and fast waves in toroidal geometry for Alcator C-Mod at values of $(\omega_{pe}/\omega_{ce})^2$ comparable to those expected in the ITER device.

TH/P4-38 · Global Hybrid Simulations of Energetic Particle-driven Modes in Toroidal Plasmas

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Abstract: Global hybrid simulations of energetic particle-driven MHD modes have been carried out for tokamaks, spherical tokamaks and stellarators using the M3D hybrid code. The Results for NSTX show that both TAEs and Alfven Cascade Modes are excited by beam ions with their frequencies consistent with the experimental observations. Nonlinear simulations indicate that the n=2 mode frequency chirps down as the mode moves out radially. For ITER, it is shown that the internal kink mode can be stabilized by alpha particles when central safety factor q(0) is sufficiently close to unity. For FIRE, it is shown that toroidal Alfven modes are stable for the case considered. Simulations of beam ion-driven modes in LHD are being carried out and results will be reported.

TH/P4-39 · Runaway electron generation in tokamak disruptions

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Abstract: One serious concern for ITER is that high-energy "runaway" electrons accelerated during disruptions may seriously damage the first wall. Until now, there has been little quantitative understanding of runaway production and behavior in tokamak disruptions. Runaway electron theory has traditionally focused on the physical mechanisms producing the runaways, but relatively little effort has been devoted to calculating what actually happens in a tokamak disruption. This is the topic of the present work, where the post-disruption runaway current profile is calculated from pre-disruption plasma parameters. The evolution of the runaway electron population and the toroidal electric field is treated self-consistently. The dynamics is complex since the runaways modify the electric field responsible for their own creation and the electric field diffuses through the plasma during this process. The calculation is done by Monte Carlo simulation of the electron kinetics and by a simplified analytical treatment. The time scale and efficiency of runaway generation are found agree with JET experiments if the post-disruption temperature is taken to be around 10 eV. About half the pre-disruption current is converted to runaways in JET, and a higher conversion is predicted for ITER. It is also found that the current profile evolves dramatically as the runaways are generated. The post-disruption runaway current profile typically becomes more peaked on the magnetic axis than the pre-disruption current. This could have consequences for the stability of the post-disruption plasma. Finally, it is found that the runaway current is likely to become filamented in the radial direction because of the sensitivity of runaway generation to plasma parameters. This could explain why the X-rays produced when the runaway electron beam strikes the wall tends to be emitted in a series of sharp bursts. We also consider the production mechanism itself of runaway electrons, studying the effect of the finite cooling rate of the plasma on Dreicer runaway generation. Electrons in the tail of the pre-disruption Maxwellian lose energy relatively slowly. Just after the thermal quench, an enhanced population of fast electrons is therefore present. These electrons are easily accelerated, which can lead to efficient runaway electron generation if the thermal quench is sufficiently rapid.

TH/P4-40 · Non-Linear Study of Fast Particle Excitation of Global Alfvén Eigenmodes During ICRH

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Abstract: Fast ions created by ICRH have been proposed for simulating alpha particle heating. In order to extrapolate results regarding excitation of global Alfvén eigenmodes to that of thermonuclear alpha particles it is important to understand the differences between excitation by ICRH and by thermonuclear alpha particles. ICRH does not only produce strong anisotropic distribution functions of the resonant ion species compared to the nearly isotropic one of thermonuclear alpha particles, but also decorrelates the interactions between the high-energy ions and the global Alfvén eigenmode. In absence of decorrelation the resonant ions will make superadiabatic oscillations in energy. The decorrelation caused by collisions and RF interactions leads to an effective broadening of the MHD resonant region hence increasing the extent of the energy transport region in phase space. The decorrelations also affect the growth rate and the amplitude of the saturation level. ICRH decorrelates the MHD interactions and pushes ions in and out of resonance with the Alfvén wave leading to enhanced excitation or damping of the mode. The decorrelation by Coulomb collisions decreases with energy and is more important for particles with low energy, whereas the decorrelation by ICRH becomes more important for high-energy particles. A method to calculate the distribution function of the resonant ions and amplitude of the global Alfvén wave self-consistently during

ICRH has been developed and implemented in the SELFO- code. The SELFO code consists of the orbit averaged Monte Carlo code FIDO for calculating the distribution function of the heated ions and the global wave code LION for calculating the wave field for ICRF heating. Self-consistent calculations of the ICRF wave field and distribution function is carried out by solving the wave field in LION with a dielectric tensor calculated from the global distribution function obtained with the FIDO code. The wave field of the global Alfvén waves are either modelled or calculated with the LION code. The non- linear evolution of the mode amplitude and the distribution function is calculated in the FIDO code.

TH/P4-42 · Non-conventional fishbone instabilities

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Abstract: New instabilities of fishbone type are predicted. They differ from conventional fishbones by the mode structure and can have the mode numbers, m and n, not equal to unity. Two cases are considered: (i) the magnetic field is weak, so that the precession frequency of the energetic ions is not small as compared to the frequency of the corresponding Alfven continuum at r=0 [the case relevant to spherical tori (ST)]; (ii) the safety factor, q(r), is a non-monotonic (which occurs in both STs and tokamaks). Regimes with $q(0) < 1$ and $q(0) > 1$ are considered. In particular, it is shown that when q(r) is monotonic and $q(0) < 1$, the mode structure of the m=n=1 instability in the NSTX spherical torus, in contrast to the mode structure of the conventional fishbones in tokamaks, has nothing to do with the rigid kink displacement. An unusual instability, which we refer to as "doublet" fishbones, is predicted when q(r) is a non-monotonic. This instability is driven by circulating ions and characterized by two frequencies and two growth rates, although it is relevant to the same double kink mode. The ratio of the frequencies is about the ratio of the shears at two rational surfaces where the instability is localized. The instability can occur when the radial profile of the energetic ions has an off-axis maximum inside the region of the mode localization.

TH/P4-43 · Influence Of Anomalous Transport Phenomena On Onset Of Neoclassical Tearing Modes In Tokamaks

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Abstract: Influence of anomalous perpendicular heat transport and anomalous ion perpendicular viscosity on conditions of Neoclassical Tearing Mode (NTM) onset is studied theoretically. Importance of the two-fluid description of the perturbed plasma for evolution of the neoclassical islands is demonstrated. Series of different parallel transport mechanisms competitive to anomalous cross-island heat transport in formation of the perturbed electron and ion temperature profiles within the island are considered. The perturbed electron temperature profile is established in competition between anomalous perpendicular electron heat conductivity and parallel electron heat convection or heat conductivity. While in formation of the ion perturbed temperature profile perpendicular ion heat conductivity is balanced by the parallel transports associated with ion inertia for an island rotating with subsonic frequency or with island rotation with respect to plasma for supersonic islands. Analytical solutions to all four above heat balance equations were found and perturbed electron and ion temperatures profiles were calculated rigorously. The partial contributions from the plasma electron and ion temperature perturbations in the bootstrap drive of the mode and magnetic curvature effect were then accounted in construction of a generalized transport threshold model of NTMs. Our calculations demonstrate that taking into account the curvature effect weakening in the generalized transport threshold model predicts notable improvement of NTM stability. The anomalous perpendicular ion viscosity was shown to modify collisionality dependence of polarization current effect reducing it to the low collisionality limit. In its turn the bootstrap drive of NTM in the presence of this viscosity was found to be dependent on the island rotation frequency and direction. For island rotating in direction of the electron diamagnetic drift viscosity effect was shown to be stabilizing. The role of viscosity effect grows rapidly with rise of the plasma ion temperature. We found that generalized transport threshold model, which describes weakening the bootstrap drive with allowance for viscous corrections and weakening the curvature effect at small island width, looks more relevant to interpretation of the experimental data than the model based on the stabilizing polarization current effect.

TH/P4-46 · External mode analysis in a tokamak by the Newcomb equation

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Abstract: It is well known that the Newcomb equation, the inertia free linear ideal magnetohydrodynamic equation, plays fundamental roles in the MHD stability theory. We have developed a solution method of the Newcomb equation in a tokamak and proposed an associated eigenvalue problem defined such that it does not have continuous spectra[1]. In the present work, we extend the theory of the Newcomb equation for the analysis of low n or high n external modes (n: toroidal mode number). Since resistive wall modes are described by the quadratic form[2] that contains the changes of the potential energy, induced by plasma displacement, in the plasma and the vacuum regions, it is essential to compute the changes of the plasma potential energy in a matrix form on the surface values of the modes. The matrix A in the form is called the stability matrix. Since the plasma motion is assumed to be much slower than the ideal MHD Alfvén motion, the plasma kinetic energy can be neglected and the motion is described by the Newcomb equation. We can construct the matrix A by using the basis functions of external modes[3] that are also the solutions of the Newcomb equation. The matrix A represents the response of the plasma to external modes even if the ideal kink modes are stable. Benchmark tests between the present formulation and the ERATO code have been performed for n = 1 ideal kink modes and confirmed that agreement between the codes is excellent. The formulation also has an advantage that it can clarify the effects of stable internal modes on the stability of external modes since the eigenvalue problem associated with the Newcomb equation identifies stable internal ideal MHD perturbations as eigen-states. For examples, it is found that internal modes close to the marginal stability change the structure of an external mode from a surface mode structure to a global mode structure. The method of the Newcomb equation also enables us to analyze other important external modes, the peeling (high-n kink) modes, which will be reported.

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TH/P4-48 · Confinement Relevant Alfven Instabilities in Wendelstein 7-AS

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Abstract: An important feature of fast-ion-driven Alfven instabilities (AI) observed in W7-AS is that they can result in thermal crashes (the temperature can drop by up to 50%). The purposes of this work are to explain the mentioned phenomenon and develop further the theory of AI in stellarators. Possible mechanisms of the formation of magnetic islands / the stochastization of magnetic field lines are analysed. It is shown that the process affects mainly electrons but not the bulk plasma ions. The behaviour of the Alfven continuum near a point where two gaps cross is studied, and the phenomenon of the gap annihilation at the crossing point is found. The developed theory is applied to the W7-AS shot # 34723. The Alfven continuum and eigenmodes are calculated. The growth rate of the eigenmodes is determined with taking into account the continuum damping and the collisional damping. The effect of the instability on the thermal plasma is evaluated and compared with experimental results.

TH/P4-49 · Confinement of Charged Fusion Products in Reversed Shear Tokamak Plasmas

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Abstract: Recent tokamak studies indicate the attraction of operational scenarios with internal transport barriers (ITBs) that provide improved energy confinement with reversed shear (RS) in the plasma core. Whereas the presence of ITBs is beneficial to the energy confinement of the bulk plasma, RS is expected to deteriorate the confinement of fusion alphas (FA) in tokamaks with moderate plasma current, ~2-3MA, due to enhanced first orbit and collisional loss. Experimentally, the influence of RS on the relaxation of the FA distribution function after NBI tritium blips into deuterium plasma has been observed recently in Trace Tritium Experiments on JET. In discharges with relatively high monotonic currents (>2MA) the observed FA density decay, was consistent with classical slowing down, while in 2.5MA strong RS discharges with a current hole ~1/3 of the plasma radius the measured decay time was much shorter than the classical slowing down time, indicating a FA confinement degradation similar to that seen at 1MA current. Axisymmetric 3D Fokker-Planck modelling results presented confirm the confinement deterioration and the decay time decrease of FA distribution observed in RS JET discharges.

TH/P5-2 · Mode coupling effects on the triggering of neoclassical tearing modes and plasma momentum braking

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Abstract: A good understanding of the plasma dynamics and stability of tokamak confinement devices is essential for the progress towards a controlled, clean, exploitation of fusion power. Plasma energy confinement and performance may be deteriorated by the onset and growth of neoclassical tearing modes (NTMs) and by the penetration and amplification of locked modes induced by the intrinsic error-field of the machine. In this paper, we investigate the impact of electrodynamic mode coupling on some crucial details regarding these two instabilities. In particular, we show that mode coupling can effectively trigger some NTMs, despite a finite frequency mismatch between the mode (as soon as it is observed in the diagnostics) and coupling driven frequencies. In fact, as soon as the driven island width overcomes the bifurcation threshold (smaller than the metastable NTM threshold and possibly below diagnostic resolution) the angular mode frequency deviates from that imposed by coupling towards its' natural value connected to equilibrium bulk toroidal plasma rotation. In addition, we show that when mode coupling effects (either due to toroidicity or three-wave resonance coupling) are considered, the toroidal plasma rotation braking following the penetration and amplification of error-field driven (m=2,n=1) locked modes is distributed over all interacting rational surfaces and not only on the dominant q=2. Accounting also for the finite plasma elongation, the m=2 mode can also drive through coupling a significant m=0 perturbation. Retaining the neoclassical toroidal viscous force (proportional to the plasma angular rotation) in the plasma momentum balance equation (vanishing rigorously in toroidal axisymmetry) and taking into account this m=0 magnetic component, a radially distributed force results, favoring a self similar braking of plasma rotation.

TH/P5-10 · Development Of Theory Of Reversed-Shear Alfven Eigenmodes

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Abstract: New magnetohydrodynamic (MHD) phenomena with upward frequency- sweeping named Alfvén Cascades (ACs) were revealed recently on JT-60U and JET in discharges with non-monotonic safety factor (reversed magnetic shear, (RS)) and significant population of the hot ions. The first theoretical description of ACs was given in paper [Berk et al., Phys. Rev. Lett. 87, 185002 (2001)], where radial localization of Alfvén Eigenmode (AE) was provided by the non-resonant response of hot ion population. Another mode localizing toroidal MHD effect was considered by Breizman et al. [Phys. Plasmas 10, 3649 (2003)]. In the present report we extend the theory of AE in RS tokamak plasmas (RSAE) by incorporating the effect of thermal plasma density gradient taken from theory of cylindrical Global Alfvén Eigenmodes (GAE) and kinetic (finite ion Larmor radius) effects from theory of Kinetic Toroidicity induced Alfvén Eigenmodes (KTAE). It is shown that the localization effect of thermal plasma density gradient on AC mode can be stronger than the toroidal MHD effect as squared aspect ratio. Thus, the Alfvén Cascade modes can be theoretically demonstrated in cylindrical geometry approximation. Then the role of thermal plasma density gradient can be dominant if the localization effect of density gradient of large orbit hot ions is sufficiently weak. The mode localization effect considered in the present report can be essential only if the thermal plasma density gradient is not too small. It is localizing for the mode numbers satisfying the condition q(min)>m/n. Then ACs correspond to the "sub-Alfvén" modes. In the opposite condition, q(min) < m/n, this effect is delocalizing. Then to provide AC eigenmodes existence other localizing mechanisms should be considered. The shift of the localization region of the eigenmodes and the eigenfrequency shift caused by the thermal plasma density gradient were found to be sufficiently small. Taking into account the finite ion Larmor radius (kinetic) effects in Alfvén mode equation allows us to predict a new branches of these modes called the Kinetic Reversed-Shear Alfvén Eigenmodes (KRSAEs). These modes are shown to posses the features of Alfvén Cascades even for homogeneous thermal plasma density in cylindrical geometry approximation.

TH/P5-17 · Nonlinear Simulation of Tearing Mode and m=1 Kink Mode Based on Kinetic RMHD Model

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Abstract: In this paper, we investigate dynamics of sawtooth oscillation and neoclassical tearing modes based on kinetic RMHD model to elucidate the mechanism of nonlinear excitation of them, putting an emphasis on interaction with microscopic and transport processes. Various routes of excitation of MHD modes by turbulence are demonstrated. It is shown that research on the hierarchical interaction between turbulence and MHD is important in Tokamak physics.

TH/P5-18 · Integrated modelling of material migration and target plate power handling at JET

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Abstract: The complexity of the tokamak edge and SOL region is such that extrapolation to ITER requires modelling to be pursued through the integration of a number of edge codes, each of which must be thoroughly tested against results from present day machines. This contribution demonstrates how the edge modelling effort at JET is focused on such an approach by considering two examples, target power loading and material erosion and migration, the understanding of which are crucial issues for ITER.

TH/P5-26 · Propagation and stability of perturbations in radiative plasmas.

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Abstract: Abstract The influence of thermal force and finite relaxation time of impurity distribution over ionization states on wave propagation and instability growth rates in radiative plasmas are investigated. These effects are important for Multi-Faceted Asymmetric Radiation From the Edge (MARFE), detached plasma regimes in tokamaks, propagation of perturbations in edge plasmas produced by ELMs etc. The new characteristic frequency equal to the usual sound frequensy multiplied by the impurity charge square appears. The thermal force influence decreases the growth rate of radiative-condensation instability sometimes twice. The mode is not more purely aperiodic. The thermal force causes the strong relative motion of ion species, and additional damping of periodic perturbations. It is caused by the difference of thermal force for different species. The relative motion of species also exists in any steady-state with temperature gradient. Calculations of simultaneous influence of thermal force and finite time of impurity distribution over ionization states on radiative-condensation mode stability, sound speed and stability in carbon seeded hydrogen plasmas are performed.

TH/P5-31 · Radiative Improved Mode in a Tokamak: a theoretical model

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Abstract: It is well known from impurity seeding experiments in several limiter tokamaks that the plasma may bifurcate into an improved confinement mode, the so called RI mode. In this mode, the confinement improvement is associated with density and temperature peaking and stronger velocity shear. In this paper we propose a novel model for the RI mode. It is demonstrated that radiative effects from impurities distributed in a poloidally asymmetric manner lead to significant density and temperature perturbations on magnetic surfaces. These, in turn, interact with theta dependent toroidal field variations to produce a mean divergence of the stress tensor driving strong toroidal flows. The resulting enhanced toroidal velocity shear on the outer radiative layers produces a stabilizing effect on the instabilities like the drift resistive ballooning mode, drift trapped electron mode and the ion temperature gradient mode. By an investigation of the turbulent particle flux as a function of the density gradient for various values of the radiation asymmetry parameter, it is shown that the plasma can undergo a bifurcation into a better confined peaked density state.

TH/P6-1 · Complex nonlinear Lagrangian for the Hasegawa-Mima equation

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Abstract: The Hasegawa-Mima equation is the simplest nonlinear single-field model equation that captures the essence of drift wave dynamics, such as turbulent cascade and zonal flow generation. Like the Schr¨¨¨¨¨¨¨ödinger equation it is first order in time, but its coefficients are real so if the potential phi is initially real it remains real. However, by embedding phi in the space of complex functions a simple Lagrangian is found from which the Hasegawa-Mima equation may be derived from Hamilton's Principle.

TH/P6-2 · Convective Transport in Tokamaks

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Abstract: Convection of coherent objects (blobs, ELMs and pellets) is an important phenomenon in fusion plasmas and has a common physical basis. SOL convection of blobs may have important consequences for fusion experiments, e.g. "short-circuiting" the divertor in some parameter regimes. This paper summarizes recent advances in the theory of blob transport, its comparison with 2D and 3D computer simulations, and qualitative comparisons with experiments. It is shown that the 2D and 3D cases involve different physics parallel to the magnetic field lines and lead to different scalings of the radial velocity. A new theory of 3D blobs is discussed in which X-point enhanced cross-field currents disconnect the blobs from the divertor sheaths. This results in an increase in the outwards convective flux that can lead to edge cooling and suggests a mechanism for understanding the density limit. A unified picture will be presented of 2D and 3D dynamics of blobs with and without internal temperature variations. The relation of this work to both 3D turbulence simulations and experiments will be discussed.

TH/P6-3 · Forces on Zonal Flows in Tokamak Core Turbulence

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Abstract: The saturation of stationary zonal flows (ZF) in the core of a tokamak has been analyzed in numerical fluid turbulence computer studies. The model was chosen to properly represent the kinetic global plasma flows, i.e., undamped stationary toroidal or poloidal flows and Landau damped geodesic acoustic modes; reasonable agreement with kinetic simulations in terms of magnitude of transport, occurance of Dimits shift was verified. Contrary to common perception, in the final saturated state of turbulence and ZFs, the customary perpendicular Reynolds stress continues to drive the ZFs. The force balance is established by the - essentially quasilinear - parallel Reynolds stress acting on the parallel return flows required by incompressibility. To aid analytical theory, both stresses have been characterized by empirical formulas.

TH/P6-4 · On depenence of thermal transport on the safety factor q in tokamaks

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Abstract: A fully kinetic, electromagnetic ballooning integral equation code has been developed and used to analyze ion temperature gradient (ITG) and electron temperature gradient (ETG) modes in tokamaks. Simple mixing length estimate based on the growth rates of the ITG and ETG modes has revealed that the ion and electron thermal diffusivities are proportional to q, the safety factor. The strong dependence of the ion and electron thermal diffusivities on the safety factor q originates from the coupling to the ion acoustic mode in the case of the ITG mode, and to the electron skin mode in the case of the ETG mode. Charge neutrality does not hold in the case of ETG mode and a mixing length estimate does yield an electron thermal diffusivity large enough to be relevant to tokamaks.

TH/P6-5 · Multiscale Studies of ETG and Drift Wave Turbulence and Transport Bifurcation Dynamics

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Abstract: Transport barriers are of fundamental interest in fusion research because of the improved confinement they provide, and because they can uncover intrinsic electron transport physics, such as electron temperature gradient (ETG) turbulence. We present recent developments on the generation and stability of extended structures in ETG turbulence. These structures allow the electron thermal transport to exceed the ETG mixing length level, which is too small to be physically relevant. We also present work relevant to the case in which the ion gyroradius scale turbulence has not been completely suppressed. First, results from self-consistent theory of interactions between ETG and drift-ion temperature gradient (DITG) turbulence are presented. In addition, recent work in the areas of turbulence propagation and transport barrier dynamics is discussed. In particular, we discuss novel results on propagation, bifurcation hysteresis, and relaxation events.

TH/P6-6 · Statistical Analysis of ITG Turbulence

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Abstract: The fluctuations computed in plasma microturbulence codes are best viewed as a particular realization from an underlying turbulent ensemble. Benchmarking between codes (or even between different runs of the same code) is achieved by comparing realization- independent characterizations of the turbulent fluctuations, such as the fluctuation spectra, correlation functions, and thermal transport coefficients. Results from the global particle code, GTC, and continuum code, GYRO, are compared with each other and with results from the flux-tube particle code, PG3EQ, and the continuum code GS2 in order to benchmark the gyrokinetic codes within the Plasma Microturbulence Project (PMP). We examine parameter scans about the Cyclone test case in which a suite of gyrokinetic particle and gyrofluid simulation codes undertook the same three-dimensional simulation of electrostatic ion- temperature-gradient (ITG) turbulence in a tokamak plasma using physical parameters matching those in an General Atomics Doublet III-D (DIII-D) high-confinement (H-mode) shot #81499. We find significant but not full agreement among the PMP codes.

TH/P6-7 · Thermodynamical properties of gyrokinetic simulations in magnetized plasmas

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Abstract: Appropriate simulations of turbulent transport in ITER must include kinetic effects. The present work emphasizes the relationship between gradients and fluxes in non- linear kinetic simulations. A preliminary comparison to the fluid limit is also discussed. Conservation laws govern the thermodynamics of gyrokinetic codes. Semi-Lagrangian schemes allow for good conservation properties. In a simplified model for trapped ion turbulence, energy conservation within the percent target requires a very fine mesh. However, the transport level and dynamics are weakly modified even when energy conservation is somewhat worse. In the non-linear phase, asymmetric convective cells develop. When they are radially elongated, the gradient is nearly clamped to the threshold whatever the drive. In this case flux and gradient are decoupled, leading to a departure from a Fourier transport law. A similar gyrokinetic model is compared to its fluid counterpart, assuming a collisional closure. Both approaches are using the same numerical scheme, and have similar linear characteristics. When sheared flows are over damped, elongated radial cells dominate the non-linear spectrum, and the heat flux in the fluid case is still found to exceed that in the gyrokinetic case by about one order of magnitude. In a 4D model for slab ion temperature gradient turbulence, the system can relax preferentially either via heat transport or via mean sheared flows, depending on whether the initial density profile is flat or shaped, respectively. This comes from the fact that, at lowest order, the magnitude of the Reynolds stress is proportional to the curvature of the density profile. In that respect, a strong density gradient appears to be stabilizing both linearly, by increasing the instability threshold, and non linearly, by activating sheared flows. Such a mechanism provides a way to trigger pressure transport barriers.

TH/P6-8 · Numerical simulation of the electromagnetic plasma at the edge of tokamak.

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Abstract: The behavior of turbulent fluxes in the vicinity of the resonant point $m/n = q(xres)$ in a plane plasma layer near tokamaks wall is studied numerically. The nonlinear two-fluid MHD equations in a five-field (f,n,Te,Ti,A) electromagnetic model are used . The simulation was aimed to reveal the dependence of the turbulent particle flux on the electron temperature Tbe at boundary between the layer and the background plasma. Simulation shows that this dependence is non-monotonic. When Tbe increases, the turbulent particle flux reduces while the perturbations are purely electrostatic. Then the flux reaches its minimum value and begins to rise after the transition to electromagnetic regime. Such a behavior is found to result from the stabilizing effect of the shear of electron drift velocity, $v_{ey} \sim dp_e/dx$, which comes to the equation for the longitudinal component of the magnetic potential. The influence of this velocity on the magnetic fluctuations is similar to the well known effect of now the poloidal sheared velocity $U_{y0}(t, x)$ acts on the convective cells breaking them into smaller ones. Analogously, the poloidal electron velocity initiates the breaking of the tearing-mode magnetic islands into smaller islands. As a consequence, dissipation power increases, and turbulent particle flux reduces. It looks like the similar effect was founded in the numerical calculations in by B. Scott /Plasma Phys. And Control.Fusion,45 (2003)A385-389/ (see Fig.6). It was also

shown that, as the electron temperature increases, the frequency oscillations of the zonal velocity $U_{y0}(t,x)$ increases too. As a result, the stabilizing impact of this velocity diminishes, and the turbulent particle flux increases.

TH/P6-9 · A Comprehensive Spectral Theory of Zonal-Mode Dynamics in Trapped Electron Mode Turbulence

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Abstract: A comprehensive, self-consistent theory for spectral dynamics in trapped electron mode (TEM) turbulence offers critical new understanding and insights into zonal- mode physics. This theory shows that 1) zonal mode structure, anisotropy, excitation, and temporal behavior are nonlinear manifestations of linear wave properties; 2) waves induce a marked spectral energy-transfer anisotropy that preferentially drives zonal modes relative to non zonal modes; 3) triplet correlations involving density (as opposed to those involving only flow) mediate the dominant energy transfer at long wavelengths; 4) energy transfer becomes inverse in the presence of wave anisotropy, where otherwise it is forward; 5) zonal-mode excitation is accompanied by excitation of a spectrum of damped eigenmodes, making zonal modes nonlinearly damped; and 6) the combination of anisotropic transfer to zonal modes and their nonlinear damping make this the dominant saturation mechanism for TEM turbulence. This accounts for the reduction of turbulence level by zonal modes, not zonal-flow $E\times B$ shearing.

TH/P6-10 · Transport up the gradient and probabilistic transport models for fusion

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Abstract: A generalization of diffusive (Fickian) transport is considered, in which particle motion is described by Levy distributions. We show how this simple model exhibits various interesting characteristics that might provide a framework for the description of a range of unusual transport phenomena observed in fusion plasmas. The model produces power degradation and profile consistency, as well as a scaling of the confinement time with system size reminiscent of the gyro-Bohm/Bohm scalings observed in fusion plasmas, and rapid propagation of perturbances. In the present work we show how this model may also produce on-axis peaking of the profiles with off-axis fueling. If Levy distributions are indeed important for transport as suggested here, this may have profound consequences for transport studies.

TH/P6-11 · Synergistic effects of magnetic and velocity shear on electromagnetic drift modes in tokamaks

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Abstract: In the fluid model, electromagnetic drift mode in low beta tokamak plasmas is studied from a set of new-derived eigenequations, including magnetic shear, perpendicular and parallel velocity shears, perpendicular and parallel current density and their shears, and finite beta. It is found that there exists a threshold of perpendicular velocity shear, at which the growth rate tends to zero. The threshold increases with decrease of magnetic shear. On the other hand, the increase of beta reduces the growth rate but increases the velocity shear threshold. In addition, we study the effects of parallel velocity shear on the instability and find that it enhances the instability. Furthermore, the preliminary calculations show that the perpendicular current density shear suppresses the instability while the parallel current density shear strengthens it.

TH/P6-13 · Experimental Observations Related To The Thermodynamic Properties Of The Tokamak Plasmas

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Abstract: The enormous complexity of the description of the plasma in reactor relevant conditions suggests that insights coming from global approaches complementary to the microscopic dynamic description are useful as well, as much as their basic assumptions are physically meaningful and widely applicable. In this frame, the coarse-grained tokamak plasma description derived from the magnetic entropy concept presents appealing features as it involves a simple mathematics and it identifies a limited set of characteristic

parameters of the macroscopic equilibrium. The aim of this paper is to provide a comprehensive review of the work done in order to check the reliability of the SME predictions against experimental data collected from different tokamaks (FTU, JET, TS, AUG), plasma regimes (L and H modes, advanced scenarios) and heating methods (Ohmic, ECH, ICRH, NBI). The SME analysis so far performed provides a satisfactory description of the current density and derived quantities (safety factor, electron temperature if the Ohmic relaxation can be assumed) in all the machines and in L and H confinement modes. Results preliminary obtained in advanced tokamak scenario indicate a similar capability. The restrictions on the pressure profile provided by the SME theory are consistent with the experiments, showing that the normalised experimental pressure can be reasonably reproduced assuming its zero order moment only. The electron heat flux calculated with the SME shows a good agreement with the experimental data for L mode plasmas both in terms of radial profile and in terms of electron temperature gradient. The situation is more difficult in H mode and in the advanced scenarios, where the assumption of Ohmic relaxation is little or not at all verified. In these cases the calculated heat flux profile is still comparable with the experimental data, but the calculated electron temperature gradient is generally not satisfactory.

TH/P6-39 · Monte Carlo Particle Simulation of Neoclassical Edge Pedestal Formation Dynamics and Pedestal Scaling Law

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Abstract: After its introduction at the FEC 2002 conference, a massively parallel neoclassical Monte Carlo ion guiding-center code (XGC) has now been completed for the study of the neoclassical formation dynamics, pedestal scaling, and the related physics. XGC is a full f, Hamiltonian guiding-center code in 5 dimensions (3 in real and 2 in velocity spaces). XGC also includes a 2D Monte Carlo neutral atom transport routine. Realistic flux surface and first-wall geometries are read in from the EFIT output. The main plasma profile evolves in time following the time evolution of the marker ion profile (density, temperature, and flow profiles). The Coulomb pitch-angle and energy scatterings conserve energy and momentum. Plasma and neutral particles experience charge exchange, ionization, and elastic collisions with each other self-consistently. Er is evaluated dynamically from the flux-surface averaged Poisson's equation, consistently with the ion polarization currents, in the time- evolving plasma pedestal profile. It is found that the neutral penetration can buildup the edge pedestal from a mild L-mode shape in approximately 10 msec time order. Higher neutral ionization raises the density pedestal height and the heat outflow from the core raises the pedestal temperature. A fully grown neoclassical pedestal shape obtained numerically fits well with a Tanh curve. The particle source from neutral ionization is balanced by the strong convective particle loss around the X-point to give the distinctive pedestal shape, working together with the orbit squeezing effect in the self-consistent Er shear. The pedestal width shows an offset linear behavior with the square-root of the ion pedestal temperature. However, it does not show the $1/B_{poloidal}$ dependence. It rather shows a strong 1/B dependence. It is found that there is a positive toroidal momentum source at the separatrix edge with a strong plasma pedestal. It is also found that the ion temperature can be highly anisotropic at the foot of the pedestal.

TH/P6-55 · Simulation of Internal Transport Barriers by the Canonical Profiles Transport Model

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Abstract: To describe the energy balance in the L-mode, the model containing the critical gradient is widely used. This so-called "first" critical gradient can be found in particular from the canonical profile for the temperature. To describe the regimes with transport barriers (TB), we use an idea about the "second" critical gradient. If the pressure gradient exceeds the second critical gradient inside some plasma region then bifurcation to a new state is happened in this region with the TB formation. This idea is realized in the modified canonical profiles model, suitable for simulation of the energy and particle balance in tokamaks with arbitrary aspect ratio and plasma cross section. To choose the value of the second critical gradient, we compare the modeling results for many shots with experimental data. Connection of this gradient with the magnetic shear is found. We obtained the criterion of the TB formation, which is close to the experimental one obtained in JET. The constructed model was used for simulation of the internal TBs in TFTR, MAST, DIII-D and JT-60U. The possibility of the internal TB formation in T-10 is discussed. The possible dependence of the second critical gradient on plasma parameters is discussed also.

TH/P6-56 · Influence of Suprathermal Electrons Kinetics on Cyclotron Radiation Transport in Hot Toroidal Plasmas

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Abstract: Numerical studies of the contribution of suprathermal electrons to electron cyclotron radiation (ECR) transport in hot (Te > 10 keV) plasmas confined by a strong toroidal magnetic field (B > 5 T) are reported. The respective code [1] which, for maxwellian electron velocity distribution (EVD) with inhomogeneous temperature/density, has been tested against well-known numerical and semi-analytical codes by S. Tamor, is now applied to solving the following two problems for ITER-like conditions. (1) An analysis of spatial profile of the net radiated power density, $W_{ECR}(r)$, reveals strong sensitivity of $W_{ECR}(r)$ to suprathermal electrons and enables us to evaluate allowable limits for space-averaged values, and spatial profiles, of the effective temperature and density of suprathermal electrons (in terms of bi-maxwellian EVD). (2) Self-consistent treatment of (i) ECR transport and (ii) evolution of EVD for suprathermal electrons allows to evaluate kinetic effects (namely, spatial profile of deviations from maxwellian EVD) caused by the transport of an intense ECR in non-equilibrium hot plasmas.

Kukushkin A.B., Proc. 14th IAEA Conf. PPCF, Wuerzburg, 1992, v.2, p.35-45.

IT

ITER Activities

IT/1-1 · Progress in physics basis and its impact on ITER

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Abstract: Recently, a number of important results have been obtained in experiments, theory and modeling, which have given enhanced confidence in ITER achieving its goals. This paper summarises recent progress in the physics basis and its impact on the expected performance of ITER. Significant progress has been made in the development of hybrid and steady state operation scenarios. Experiments show that in hybrid operation with a combination of inductive and non-inductive current drive, tailoring the current profile can improve confinement over the standard H-mode and allow an increase in beta up to the no- wall limit at safety factors ~ 4. Extrapolation to ITER suggests that at the reduced plasma current of ~ 12 MA, high $Q > 10$ and long pulse (>1000 s) operation is possible with benign ELMs. Analysis of disruption scenarios has been performed based on guidelines on current quench rates and halo currents, derived from the experimental database. The estimated electromagnetic forces on the in- vessel components are below the design target values, confirming the robustness of the ITER design against disruption forces.

IT/1-2 · Dimensionless identity experiments in JT-60U and JET

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Abstract: Results of dimensionless identity experiments in JT-60U and JET, comparing the plasma pedestal characteristics and ELM behaviour in the two devices are presented. The similar size of JET and JT-60U results in dimensionless matched plasmas very similar in dimensional parameters, apart from the aspect ratios (kappa), differing by 15%. With positive NB injection (PNB) heating, the pedestal pressure in JT-60U is up to a factor of 2 lower than in JET, and the pedestal ne-Te obtained in JT-60U with Type I ELMs are accessible in JET only with Type III ELMs. MHD analysis shows that differences in kappa are not sufficient to explain these results. The effect on pedestal and ELMs of ripple-induced fast ion losses (larger in JT-60U that in JET) are discussed, since they may affect magnitude and direction of edge toroidal rotation velocity (VT). Differences in the current profile (broader in JT-60U) could also affect edge kink-peeling stability. First results indicate that H-modes in JT-60U with negative ion NB injection (low ripple losses) obtain pedestal pressure matching JET values. VT and current profile effects on pedestal and ELMs, as well as pedestal profile scaling and MHD stability will be compared and discussed.

IT/1-3 · ITER Licensing

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Abstract: ITER was fortunate to have four countries interested in ITER siting to the point where informal licensing discussions were initiated. This experience uncovered the challenges of licensing a first of a kind, fusion machine under different licensing regimes and helped prepare the way for the site specific licensing process now underway. These initial steps in licensing ITER have allowed for refining the safety case and provide confidence that the design and safety approach will be licensable. With site-specific licensing underway, the necessary regulatory submissions have been defined and are well on the way to being completed. Of course, there is still work to be done and details to be sorted out. However, the informal international discussions to bring both the proponent and regulator up to a common level of understanding have laid the foundation for a licensing process that should proceed smoothly. This paper provides observations from the perspective of the International Team.

IT/1-4 · Design of the ITER Magnets to Provide Plasma Operational Flexibility

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Abstract: The ITER magnets have been optimised and refined since the ITER FDR in 2001. Multiple design options have been eliminated and there is improved ability to drive a wide range of plasma configurations. Design iterations on the TF out of plane supports have eliminated stress concentrations in the inner keyways and led to the choice of a so called friction-joint on the outside. Selection of compact joints for the CS has enabled the peak field and cyclic stress levels in the conductor to be reduced while maintaining the flux capability. The uncertainty in the nuclear heat levels in the inner legs of the TF coils, and the need to operate with plasma nuclear powers from 360 to 700MW, lead to a thermal screen on the

inside of the case with variable cooling capability. The RWM stabilisation provided by the side CC has been extended, leading to higher voltages and heating from AC losses. R&D results from the model coils have led to adjustments in the design margins of the Nb3Sn conductors, offsetting the impact by adopting the latest advances in strand performance. Preparation for procurement is underway with considerations on technically acceptable ways of splitting the magnet supply.

IT/1-5 · Convergence of Design and Fabrication Methods for ITER Vacuum Vessel and In-vessel Components

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Abstract: Design and R&D activities of the vacuum vessel and in-vessel components have been proceeding interactively. R&D items are categorized into three levels: 1 to demonstrate basic feasibility of design solutions, 2 to demonstrate feasibility of selected fabrication methods, 3 to improve fabrication methods or to reduce cost. R&D of 1 and 2 must be completed consistently with the reference design before the start of the procurement. One of the key R&D areas is the development of the vacuum vessel fabrication methods to achieve the required tolerances. The tolerance of the vacuum vessel sector is defined based on the physics requirements and the machine parameters, to be ±10mm for the internal and external surface deviation from the reference geometry. Basic feasibility of this challenging tolerance was demonstrated in the EDA R&D program for the original vessel design. The current vessel design is more complex including keys and support housings for the blanket modules. A new R&D task has been launched to confirm its feasibility and to select fabrication methods. Based on interaction between the design activities and the R&D programs, most of the vessel and in-vessel component designs have already converged to their final form through the cooperative efforts of the International Team and Participant Teams.

IT/P3-16 · Transport and Deposition of Hydrocarbons in the Plasma Generator PSI-2 : Experiment and Modelling

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Abstract: Carbon based materials have good characteristics for their use as plasma facing materials in fusion devices. However, they have a major drawback for ITER, caused by the formation of hydrocarbons and the associated Tritium retention. This retention is the result of many complex processes which include hydrocarbon formation, transport and deposition whose understanding must be improved in order to provide reliable estimates of Tritium retention in ITER. For this purpose, a series of experiments dedicated to the analysis of transport and deposition of hydrocarbon molecules has been performed at the plasma generator PSI-2. As a source of hydrocarbons, defined amounts of CH4 and C2H4 were blown into the stationary plasma. The thickness of deposited layers collector surfaces (outside the plasma column) was measured in situ for various plasma conditions, as a function of the collector temperature. The experiments have been analysed with the 3-D Monte Carlo code ERO, adapted for the PSI-2 geometry and conditions, which is one of the tools used to predict Tritium retention in ITER. In general, ERO predicts much lower deposition rates (factors 4-5) than observed in experiment, particularly for low density conditions. On the other hand, the observed shift of the deposition maximum away from the centre of the collector plate could be predicted by the modelling and is caused by the ExB-drift of the generated ions, indicating that the some of the physical processes involved in hydrocarbon transport are properly included in the code. The possible reasons for this discrepancy and the associated implications will be discussed.

IT/P3-17 · Validated Design of the ITER Main Vacuum Pumping Systems

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Abstract: FZK is developing the ITER high vacuum cryogenic pumping systems (torus, cryostat, NBI) as well as appropriate enabling technology for tritium compatible mechanical roughing pump trains. Control of the gas throughput, especially the helium ash produced by D-T fusion reactions, is one of the key issues affecting the performance of a fusion reactor. All cryopumps incorporate similar design of charcoal coated cryopanels cooled to 5 K. NBI and torus primary pumps are characterised by an integral inlet valve. A model of the torus exhaust cryopump (scale 1:2) is being investigated in the TIMO testbed at FZK. This paper gives typical results of that (pumping, regeneration, cryogenic, safety) and outlines how these data were incorporated in a sound design of the whole torus exhaust pumping system. To do this, a dedicated

software package was developed which is able to describe gas flow in laminar, transitional and molecular flow regimes as needed for the gas coming through the divertor slots and along the pump ducts into the cryopumps. The entrance section between the divertor cassettes and each pumping duct was identified to be the bottleneck of the gas flow. The interrelation of achievable throughputs as a function of the divertor pressure and the cryopump pumping speed is discussed. The system design is completed by integrating performance curves for the roughing pump trains needed during the regeneration phases of the cryopumps.

IT/P3-18 · Carbon Erosion Mitigation by Beryllium Layer Formation in ITER

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Abstract: A beryllium-seeded deuterium plasma is used in PISCES-B to investigate mixed-material erosion and redeposition properties of ITER relevant divertor materials. The experiments are designed to reduce uncertainties in the prediction of tritium retention in redeposited mixed-materials. The beryllium containing plasma simulates the erosion of first wall material into the ITER sol plasma and its subsequent flow toward the carbon divertor plates. The beryllium impurity concentration in the ITER divertor plasma is expected to be in the 1 to 10% range. In PISCES, a small (0.15%) beryllium impurity concentration in the incident plasma onto a carbon target is seen to dramatically reduce the carbon chemical and physical erosion rates. The resultant plasma exposed surfaces contain a large fraction of beryllium coverage. Concomitant collection of redeposited material during the plasma exposure reveals the formation of beryllium-rich layers on surfaces outside the plasma column. These experiments show that the behavior of, and the accumulation of tritium in, the ITER divertor may be dominated by the beryllium impurities that will be present in the plasma rather than the carbon divertor plates.

IT/P3-19 · Progress with High Priority R&D Topics in Support of ITER/BPX Diagnostic Development

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Abstract: The development of diagnostic systems for next step Burning Plasma experiments (BPX) such as ITER requires R&D in some key areas. The International Tokamak Physics Activity Topical Group (TG) on Diagnostics has identified five topics as 'high priority' and these form the focus of the current work of the TG: (i) review the requirements for measurements of the neutron/ alpha source profile and assessment of possible methods of measurement; (ii) development of methods of measuring the energy and density distribution of confined and escaping alpha-particles; (iii) assessment of the thermal electromotive force on irradiated coils used for steady-state magnetic field measurements and developments of new methods to measure steady-state magnetic fields accurately in a nuclear environment; (iv) determination of the life-time of plasma facing mirrors used in optical systems; and (v) establishment of a Radiation Effects Database. The paper will present the recent progress in these areas.

IT/P3-20 · Objectives and Progress of the ITER Test Blanket Working Group Activities

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Abstract: The ITER Test Blanket Working Group (TBWG) has restarted its activity in October 2003 in order to reassess breeding blanket testing program and necessary R&D collaboration taking into account the need of the present six ITER Parties and the recent progress of breeding blanket technology. All parties agree on the extreme importance of breeding blanket testing in ITER. The main objectives of the work of the TBWG are to establish a meaningful and coordinated testing program and to fully define the interfaces between the three equatorial ports devoted to the testing and the ITER machine and buildings.

IT/P3-21 · Interaction of atomic hydrogen with charcoal at 77 K

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Abstract: When the inner surface of the ITER pumping duct is covered with a thin à-˜Ñ:H film, the hydrogen recombination coefficient can be reduced. In this case, atomic hydrogen can reach the cryopump region and interact with charcoal cryosorbent. The interaction of thermal hydrogen molecules and atoms with charcoal has been analyzed by sorption measurements and TDS at 77 K. A stream quartz reactor

with H2 RF discharge was used for the production of H atoms. The ratio of H and H_2 in mixture in the afterglow zone was 1/10000. After exposure in H/H_2 mixture the tube section with charcoal was warmed up to 300 K. In reference experiments the same sample of charcoal was exposed successively in H_2 and CH_4. After sample exposure in H/H_2 mixture, the TD peak shifted to higher temperatures from 125 K (peak temperature after exposure in H_2) to 150 K. The high temperature shoulder of this peak coincided with the temperature of methane release. The wide spectrum of heavy hydrocarbons formed at 77 K was registered by mass-spectrometry at charcoal heating up to 700 K. The specific adsorption volume of charcoal measured by N_2 adsorption at 77 K decreased by $10 - 15\%$.

IT/P3-22 · Study of ITER RWM control with semi-analytical models

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Abstract: The paper presents results of the study of ITER RWM control performed using semi-analytical models of the RWM. The models have been obtained by Y.Q.Liu and A.Bondeson with the code MARS for a set of ITER Scenario 4-type plasmas (9MA, weak negative magnetic shear). A multiple input, multiple output Linear Quadratic Gaussian (LQG) controller which determines the voltage in the correction coils was designed using these models for the control of RWMs with different degrees of instability. In spite of the screening effect of the vacuum vessel outer wall, the controller is able to suppress highly unstable RWM without using the second derivative of the measured poloidal magnetic field. The voltage required for RWM control was assessed using this controller. It was found that highly unstable RWMs can be stabilized with voltages less than 300V/turn. The effect of filtering of the diagnostics signal on the RWM control was studied with the goal to reduce AC losses in the superconducting coils. The cutoff frequency for a moderately unstable RWM can be as low as 60 Hz without significant deterioration of the control performance.

IT/P3-23 · Status of ITER Neutron Diagnostic Development

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Abstract: Due to the high neutron yield and the large plasma size many ITER plasma parameters such as, fusion power, power density, ion temperature profile, fast ion energy and spatial distributions in the plasma core can be well measured by various neutron diagnostics. Neutron diagnostic systems under study for ITER include: radial and vertical neutron cameras (RNC and VNC), internal and external neutron flux monitors, neutron activation systems and neutron spectrometers. The two-dimensional neutron source strength and spectral measurements can be provided by the combined RNC and VNC. The neutron flux monitors need to meet the ITER requirement of time-resolved measurements of the neutron source strength and can provide the signals necessary for the real time control of the ITER fusion power. Compact and high throughput neutron spectrometers are under development. A concept for the absolute calibration of neutron diagnostic systems is proposed. The development, testing in existing experiments and the engineering integration of all neutron diagnostic systems into the ITER is in progress and the main results will be presented in the paper.

IT/P3-24 · Modelling Studies of ITER Divertor Plasma

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Abstract: The latest results of divertor modelling for ITER are presented in the paper. We assess the modifications of the divertor operation engendered by variations of the production rate and residence time of the impurity ions in the edge, which are themselves the result of different assumptions on chemical sputtering coefficients and surface state under conditions of mixed plasma-facing materials. A set of effective boundary conditions for the core modelling for the steady state operation is presented. First results of implementing the neutral-neutral collisions and molecular dynamics are reported.

IT/P3-25 · Modelling of ITER Improved H-mode Operation with the Integrated Core Pedestal SOL Model

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Abstract: The Integrated Core-Pedestal-SOL (ICPS) model has been continuously modified and improved, so that it now models core energy transport with the MMM95 transport coefficients, stabilized by a combination of ExB velocity flow shear and magnetic shear in order to obtain a pedestal. ETG transport, which is not stabilized by flow shear, has now been added to the electron channel and the beam particle source has been corrected. These changes significantly improve the agreement between model and experiment for both AUG and JET. Previously, the boundary conditions for the core model had been self- consistently determined for ITER modelling only by scaling relationships, obtained from a database of B2-Eirene runs. This procedure is now applied also to determine the boundary conditions for the modelling of AUG and JET. To model Improved H-modes are modelled by reducing turbulent transport in regions for which the low order rational q surfaces are sparse. After calibration against AUG discharges, the model is used to investigate the implications of Improved H-mode operation for ITER.

IT/P3-26 · Numerical simulations for ITER divertor armour erosion and SOL contamination due to disruptions and ELMs

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Abstract: The divertor armour materials for ITER will be tungsten (as brushes or plates) and CFC. Disruptive loads with the heat deposition Q up to $100MJ/m^2$ on the time scale T of 3 ms or operation with ELMs at repetitive loads of $Q \sim 3MJ/m^2$ and T $\sim 0.3ms$cause enhanced armour erosion and produce contamination of SOL. Recent numerical investigations of erosion mechanisms with the anisotropic thermomechanics code PEGASUS-3D and the surface melt motion code MEMOS-1.5D as well as hot hydrogen plasma dynamics, heat loads at the armour surface, backward propagation of material plasma in SOL with the radiation-magnetohydrodynamics code FOREV-2D are survived. For CFC targets, the local overheating model is explained and numerically demonstrated. For the tungsten targets the numerical analysis of melt motion erosion on the base of MEMOS-1.5D calculations is developed and accompanied by numerical results. For validation of the codes at the regimes relevant to ITER disruptions and ELMs, the simulation results are compared with available experiments carried out at the plasma guns MK-200UG, QSPA-T, QSPA-Kh50, electron beam test facilities JUDITH and JEBIS, and the tokamak JET.

IT/P3-27 · Modeling of Noble Gas Injection into Tokamak Plasmas

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Abstract: The noble gas injection for the mitigation of the disruption in large tokamaks is simulated. Three stages are examined: neutral gas jet penetration through background plasmas, thermal quench, and radiative cooling with runaway electron generation. To model the gas jet expansion the numerical code LLP with improved radiation model is used. The model takes into account two most important opacity effects, ionization from the excited state and radiation capture in plasma volume. The beginning of thermal quench is determined using the kink-mode stability criterion. Conditions of jet penetration into central plasma region are found. Plasma cooling after thermal quench by noble gas radiation is simulated using the radiation model for optically thin plasmas. The radiation load on the first wall is estimated. The condition of runaway electron generation is obtained taking into account opacity effects in cooled plasma.

IT/P3-28 · Requirements for pellet injection in ITER scenarios with enhanced particle confinement.

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Abstract: Requirements for pellet injection parameters for plasma fuelling and ELM mitigation are assessed for ITER scenarios with enhanced particle confinement. The assessment is based on the integrated transport simulations including modeling of pedestal transport, ELM losses, reduction of helium transport and boundary conditions compatible with SOL/divertor simulations. The assessment of fuelling requirements is carried out for the steady state scenario with enhanced confinement with H > 1. A new type of SS scenario is considered with NBCD and ECCD instead of LHCD to extend the range of stable operation

and to avoid the reduction of the edge LHCD efficiency caused by pellet injection. The requirements for pellet injection for the inductive H-mode scenario (H = 1) are reconsidered taking account of a possible reduction of the particle loss obtained in some experiments at low collisionalities.

IT/P3-29 · Analysis of Disruption Scenarios and Their Mitigation in ITER

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Abstract: Representative disruption scenarios and associated electro-magnetic (EM) loads are investigated to check the robustness of ITER design. The quench rates and waveforms of plasma current are essential for estimating EM loads due to eddy and halo currents. The international disruption database is examined to derive an appropriate guideline both for linear and exponential waveforms of the plasma current. Numerical simulations by the DINA code are performed to evaluate detailed plasma behavior. Analyses of EM loads show that they are within a design target value for the expected disruptions in ITER. Massive noble gas injection developed in DIII-D is one of the possible techniques for disruption mitigation. Optimum species and amount of impurity to mitigate (i) the thermal load on plasma facing components during thermal quench, (ii) the EM load due to eddy and halo currents during the current quench, and (iii) runaway electrons, will be specified based on the experimental results in DIII-D as well as detailed calculations of impurity radiation and charge state coupled with the DINA code. Initial engineering studies in ITER are performed.

IT/P3-30 · Experimental assessment of the effects of ELMs and disruptions on ITER divertor armour materials

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Abstract: The response of plasma protection materials to thermal energy deposited during simulated Type I Edge Localised Modes (ELMs) and disruptions was studied. The paper describes the design and manufacture of special CFC and tungsten macrobrush targets, the experimental conditions achievable at simulating facilities and results of selected experiments. Experiments are conducted primarily under an EU/RF research collaboration in two plasma guns (QSPA and MK-200UG) located in TRINITI, Troitsk, Russia. The targets were exposed to a large number of repetitive pulses in QSPA plasma gun with heat loads varying in a range of $1-2MJ/m^2$ lasting $0.1-0.5ms$, with the purpose to determine the total expected erosion rate in ITER. MK-200UG experiments were focused on studying mainly vapor plasma production and impurity transport during ELMs. Moderate tungsten erosion less than 0.3 microns per shot was demonstrated for $1.5MJ/m^2$ energy densities. Energy density increasing up to $1.8MJ/m^2$ resulted in sharp growth of tungsten erosion, caused by intensive droplet ejection from irradiated tungsten surface. The program of further experiments is discussed.

IT/P3-31 · Effects of Alpha Particle Transport Driven by Alfvénic Instabilities on Proposed Burning Plasma Scenarios on ITER

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Abstract: The consistency of proposed burning plasma scenarios with Alfvénic instabilities driven by energetic ions is investigated. If the energetic particle pressure is above the threshold for resonant excitation of Energetic Particle driven Modes (EPMs), significant modification of the energetic particle pressure profile can take place. Model simulations are performed using the Hybrid MHD-Gyrokinetic Code HMGC retaining relevant thermal- plasma parameters, safety factor and alpha particle pressure profiles. ITER monotonic-q (scenario 2) and reversed-shear (scenario 4) equilibria have been considered. Also an ITER hybrid scenario will be studied and quantitatively compared with the previous ones. The transition from the low-amplitude Alfvénic instability saturation to the secondary excitation of a stronger mode will be addressed, and its effect on the energetic particle transport will be analyzed.

IT/P3-32 · The scaling of confinement in ITER with β and collisionality

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Abstract: The IPB98(y,2) scaling expression, presently used to predict the confinement time in ITER, is in conflict with results from single scan experiments in the dimensionless parameters beta and ν^*. The scaling expression has a strong degradation with beta and almost no ν^* dependence. Examination of the condition of the database using a principal component analysis shows that there is a problem with the 3 weakest principal components in that the errors in these directions are so large that standard ordinary least squares regression techniques do not give an unbiased fit to the data. Two main techniques are used to resolve this problem. The first is to improve the condition of the database by selecting an ITER relevant dataset and regressing with 3 less variables. The second technique is to use an errors in variables technique on the full database. Both techniques give scaling forms that have a significantly weaker beta dependence and stronger ν^* dependence. The confinement projections for ITER standard operation are very similar to that of the IPB98(y,2) expression, however at higher beta improved performance is predicted.

IT/P3-33 · Integrated modeling of the current profile in steady-state and hybrid ITER scenarios

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Abstract: This paper presents integrated modeling of steady-state and hybrid scenarios for ITER parameters, using various theory-based and semi-empirical transport models in conjunction with validated models for the various non-inductive current drive sources. The aim of the chosen current drive schemes is to optimize the q-profile for maximum fusion performance and non-inductive current fraction. The simulations are done using various transport modeling codes, self-consistently calculating heating and current drive with ITER design parameters. Various constraints like beta limits and power loss to the divertor are taken into account. The simulations address both the final stationary state and dynamic access to it. Resulting current profiles in ITER will be presented. A significant benchmarking activity has been undertaken as an integral part of this effort in order to validate the current drive models and reference scenarios. Both cross-code and code-experiment comparisons are reported for NBCD, ECCD and LHCD models. Comparison of simulations with a reference set of experimental discharges selected for their relevance to the ITER steady-state and hybrid scenarios are presented.

IT/P3-34 · Expected energy fluxes onto ITER Plasma Facing Components during disruption thermal quenches from multi-machine data comparisons

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Abstract: The expected energy fluxes during the thermal quench of the most probable disruption types in ITER has been evaluated on the basis of recent experimental measurements from tokamaks. For most disruptions (with the exception of VDEs and high beta ITB collapses) the plasma energy at the thermal quench is only 10–50% of that of the full performance plasma, leading to smaller energy fluxes on the divertor target for ITER disruptions than previously estimated. The disruption energy flux footprint on the divertor is significantly broadened, with respect to the full performance plasma and, thus, the ITER divertor wetted area will be in the region of $\sim$ 15–30m^2. These factors lead to an expected divertor thermal quench energy flux for ITER disruptions in the range $1-12MJ/m^2$, instead of $10-40MJ/m^2$ previously estimated. On the other hand, the large broadening of the power flux footprint at the thermal quench leads to significant power fluxes arriving to areas of the Be main chamber wall (particularly to the upper X-point region), which were not considered so far. The implication of these disruption thermal quench energy fluxes on PFC lifetime in ITER and operation will be discussed.

IT/P3-35 · H-mode threshold power dependences in ITPA threshold database

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Abstract: New contributions from a series of devices have recently been added to the international H-mode threshold database (IGDBTH). The modification of the basic plasma parameter ranges is primarily described. The new coefficients of the 'traditional' threshold power scaling law are expressed. The contribution from non-conventional tokamaks (MAST and NSTX) is used to study the influence of the aspect ratio on the threshold power. The possibility of extracting a new fit based on the low density turning point is discussed, as well as the influence of the effective charge (Zeff) for the low density discharges. The effect of plasma and divertor geometry is also addressed. Finally, a new prediction for the H-mode threshold power in future burning plasma experiments is given.

IT/P3-36 · Study of advanced tokamak performance using the International Tokamak Physics Activity database

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Abstract: Construction of an international database for advanced tokamak discharges is an activity coordinated by the ITPA. Analysis shows that study of improved confinement and performance from ASDEX Upgrade, DIII-D, FT-U, JET, JT-60U, RTP, T-10, TCV, TFTR and Tore Supra experiments is aided by using two categories: (i) discharges with central q$\sim$1 plus low magnetic shear, and (ii) discharges with central q above 1.5, with weak or strong reversed magnetic shear. An assessment of the operational domain and the potential use of both types of advanced scenarios in ITER is presented. The discharges with low magnetic shear have typically no internal transport barriers and operate at higher beta compared to standard H-modes, enabling long pulse operation at high fusion gain (hybrid regime). The reversed shear discharges have internal transport barriers, providing the possibility for non-inductive operation. From analysis of the database, areas for collaboration experiments for the two scenarios under the ITPA are indicated. This collaboration address more detailed transport studies for advanced scenarios and aim to document the gap between present day experiments and the requirements for ITER.

IT/P3-37 · Transport Modeling and Gyrokinetic Analysis of Advanced High Performance Tokamak Discharges

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Abstract: Predictive transport modeling and gyrokinetic stability analysis of demonstration hybrid and Advanced Tokamak (AT) discharges from the ITPA profile database will be presented. The level of commonality of the turbulent transport properties and the relative roles of the transport suppression mechanisms (i.e. ExB shear, magnetic shear, alpha stabilization, Ti/Te) are assessed. Various transport models are used, including the GLF23, Weiland, mixed Bohm/gyro-Bohm, and Current Diffusive Ballooning Mode models. Comparisons are made between the two regimes using DIII-D, JET, and JT-60U data from high performance H-mode discharges. In the hybrid discharges, high performance at reduced plasma current has been demonstrated under stationary conditions, enabling long pulse operation at high fusion gain. The hybrid scenario is being explored as a possible alternative to the conventional q95=3 ITER scenario. The AT scenario has demonstrated high performance with high bootstrap fractions and offers the potential of non-inductive operation. Obtaining a predictive understanding of the transport and the origin of the enhanced core confinement in both these regimes is sorely needed.

IT/P3-42 · Advanced neutron diagnostics for ITER fusion experiments

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Abstract: The magnetic proton recoil (MPR) spectrometer was developed for use on the main DT experiment at JET in 1997. It has demonstrated new capabilities of neutron emission spectroscopy (NES) diagnosis of plasmas, especially, fusion aspects of ITER relevance. This success gave thrust to including two further NES projects in the JET enhancement program. One concerns an upgrade of the MPR (MPRu) to facilitate high accuracy measurements at low background over the entire fusion neutron energy range down to about 1.5 MeV needed for diagnosing D plasmas. The other project is a time-of-flight spectrometer designed for optimized rate (TOFOR) to diagnose D plasmas at count rates a factor 100 higher than previously. This contribution will report on the principles of the most advanced NES diagnostics built for JET and their use to test diagnostic capabilities in fusion experiments that mimics ITER as close as today's generation of tokamaks permits. The demonstrations refer to the DTE1 campaign of 1997 and the TTE experiment of 2003, which was the first ever DT fusion experiment with an (active) advanced NES diagnostic where the latest achievement was absolute measurement of the fusion power.

IF

Inertial Fusion Experiments and Theory

IF/1-1Ra · Hydrodynamic Stability Experiments on the GEKKO-XII Laser Facility at the Institute of Laser Engineering, Osaka University

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Abstract: The Rayleigh-Taylor (RT) instability with material ablation through the unstable interface is the key physics that determines the success or failure of inertial fusion energy (IFE) generation, as the RT instability potentially quenches ignition and burn by disintegrating the IFE target. It is generally believed that some intended reduction of the RT instability is necessary for IFE. We will present two mitigation schemes of the RT growth rate by controlling the energy transport mechanism that dominates the ablation structure and hence the instability growth rate. One is to enhance the nonlocal nature of the electron heat transport by illuminating the target with long wavelength laser light, whereas the high ablation pressure is generated by irradiating short wavelength laser light. The second scheme is to generate double ablation structure in high-Z doped plastic targets. In addition to the electron ablation surface, a new ablation surface is created by x-ray radiation from the high-Z ions. Contrary to the previous thought, the electron ablation surface is almost completely stabilized by extremely high flow velocity. On the other hand, the RT instability on the radiative ablation surface is significantly moderated.

IF/1-1Rb · Effects of Magnetic Field, Shear Flow and Ablative Flow on the Rayleigh- Taylor Instability

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Abstract: The effect of magnetic field, shear flow and ablative flow on the Rayleigh- Taylor (RT) instability is investigated in the presence of sharp interface. RT instability can be suppressed by transverse magnetic field and even extinguished if magnetic field is strong enough. It is shown that the shear flow acts as a drive component of the RT instability and dominates over the gravity part for short wave, strong flow shear and small Atwood numbers. It is found that the effect of magnetic field is equivalent to the ablation velocity , which means the ablative RT instability could be limited even if the ICF target shell has not been accelerated to such high speed as deduced from previous model. The results agree with the numerical simulations and experiments reported by Glendinning et al. [PRL 69(1992)1201].

IF/1-2 · Overview of U.S. Heavy-Ion Fusion Progress

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Abstract: IAEA 2004 Abstract for: "Overview of US heavy- ion fusion research*" B. Grant Logan (on behalf the US HIF-VNL and VLT staff) Heavy Ion Fusion Virtual National Laboratory (HIF-VNL: LBNL, LLNL, PPPL) Significant experimental and theoretical progress has been made in the U.S. heavy ion fusion program on high-current sources, transport, final focusing, chambers and targets for inertial fusion energy (IFE) driven by induction linac accelerators seek to provide the scientific and technical basis for the Integrated Beam Experiment (IBX), an integrated source-to-target physics experiment recently included in the list of future facilities planned by the U.S. Department of Energy, and for IFE in the future. To optimize the design of IBX and future IFE drivers, current HIF-VNL research is addressing several key issues (representative, not inclusive): gas and electron cloud effects which can exacerbate beam loss at high beam perveance and magnet aperture fill factors, ballistic neutralized and assisted-pinch focusing of neutralized heavy ion beams, limits on longitudinal compression of both neutralized and un-neutralized heavy ion bunches, and tailoring heavy ion beams for uniform target energy deposition for high energy density physics (HEDP) studies. A multi-beam induction linac driven power plant study (9-03) shows detailed requirements for distributed radiator targets (spot size, power, symmetry and pulse shape) can be met with ballistic neutralized focusing of a 120 beam array over 6 meter focal lengths. Simulations and theory are investigating neutralized drift compression and focusing to see if modular induction linac driver systems with $\sim$ 20-40 linacs can meet the spot size, pulse shape and symmetry required for hybrid-distributed radiator targets, and for experimental targets for high energy density physics studies. Methods to accommodate or correct chromatic aberrations with neutralized drift compression are being investigated in this new study.
*This work was performed under the auspices of the U.S. Department of Energy by the University of California, Lawrence Berkeley and Lawrence Livermore National Laboratories under Contract Numbers DE-AC03-76SF00098 and W-7405-Eng-48, and by the Princeton Plasma Physics Laboratory under Contract Number DE-AC02-76CH03073.

IF/1-3 · Update On LMJ Target Physics

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Abstract: The objective of the ICF Program at CEA is to burn cryogenic DT capsules with the Laser Megajoule (LMJ), which will be composed of 240 beams producing up to 1.8MJ in 20ns at 0.35μm. LMJ prototype, LIL is composed of 4 beams and will be used for the first plasma experiments in 2005: we will present the previsions of these experiments designed to be representative of LMJ conditions. The LMJ point design has been chosen to provide enough margin with respect to the uncertainties relative to laser plasma interaction (LPI), to radiation symmetry, to hydrodynamic stability and to ignition. We will summarize recent studies made at CEA in order to quantify these safety margins, particularly the effect of random 3D laser imbalance and mispointing on the DT final deformation. The implosion of the capsule is very sensitive to the perturbations due to the target roughness. A chain of models and 2D simulations gave boundaries for the roughness compatible with the gain. Target fabrication is another challenge of the LMJ project: we will show the recent advances in capsule fabrication.

IF/1-4Ra · Direct Heating and Basic Experiments for Fast Ignition

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Abstract: Important issues related to fast ignition (FI) scheme have been extensively studied using our unique laser facility at the Institute of Laser Engineering, Osaka University, where the 1kJ-PW and GEKKO XII(12 green laser beams) laser systems could be used with a perfect synchronization at large laser energies ($<$ 1kJ at PW and $<$ 5kJ at GEKKO). Studied are (1) coherent transition radiation (CTR) in order to reveal the hot electron characteristics inside a target, (2) super-penetration which penetrated in an over-dense plasma, and (3) modeling fast ignition experiment with the direct irradiation of PW laser onto a highly compressed core. CTR indicated that there are two temperatures in the hot electron spectrum, important indication for efficient heating of the core. The super- penetration was demonstrated clearly for the first time at a PW laser power. The modeling experiment showed a factor 4 increase in neutrons production when CD targets were imploded first and then were irradiated with a PW laser for fast heating.

IF/1-4Rb · Comparative study of electron and proton heating for fast ignition

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Abstract: In ongoing research into fast ignition we have conducted collaborative experiments at the Vulcan laser facility in the UK and the Gekko laser in Japan using petawatt class beams to study isochoric heating by both laser generated electrons and by focused proton beams accelerated by the electrons. Experimental measurements are presented of isochoric heating diagnosed by excitation of K-alpha fluorescence and by XUV Planckian emission. Modeling with a hybrid PIC code is used to interpret the data. The relative prospects of electrons and protons for fast ignition are discussed.

IF/1-5 · Two-Dimensional Fokker-Planck Analysis of Core Plasma Heating by Relativistic Electrons

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Abstract: To clarify the core plasma heating by laser-produced fast electrons, we have developed a 2-D relativistic Fokker-Planck code for fast electrons. Being written for a cylindrical coodinate system with axial symmetry, the code can be used to calculate the core heating in spherically compressed targets. After describing the physics model, we examine the feature of energy deposition of beam electrons injected into dense core region. The energy deposition via long- range collective mode is shown to be comparable to that through binary collisions. The effects of self-generated electromagnetic field on the core heating are also discussed.

IF/P7-5 · A New Concept of Laser Fusion Experimental Reactor with Fast Ignition Target

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Abstract: We have analyzed the design windows of laser fusion power plants based on fast ignition targets, and examined feasibility of a small-sized laser fusion experimental reactor suitable for developing their power plants. Target gain curves are evaluated for power plants, which have 100∼200MJ fusion yields with 600kJ∼1MJ lasers, and for an experimental reactor (LFER), which has a 10MJ fusion yield with a 200kJ laser, 100kJ for implosion and 100kJ for heating. The pulse heat loads on the chamber wall of LFER are estimated at $2.5 J/cm^2$ for a 2.5-m-radius solid wall chamber, and $16 J/cm^2$ for a 1-m-radius liquid wall chamber. The fast ignition LFER can make its fusion output one order smaller than that of the central ignition, thus we can use a rather small solid wall chamber for the first stage of the LFER. We can also expect to decrease laser cost drastically, although for a heating laser we must develop the long life final optics. Through a fast ignition LFER, we showed a possibility to demonstrate net electric generation in a reasonably short time.

IF/P7-14 · Fabrication of Cryogenic Targets for Fast Ignition Realization Experiment

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Abstract: Development of the fabrication technology for the cryogenically cooled fuel targets has been initiated as a part of the Fast Ignition Realization Experiment (FIREX) Project at the ILE, Osaka University in the way of bilateral collaboration between Osaka University and National Institute for Fusion Science (NIFS). The present status of the study will be reported. Low density foam shells with a conical light guide will be fueled through a narrow pipe and will be cooled down to the cryogenic temperature. New ultra-low-density ($2 \sim 3mg/cm^3$) foam materials have been developed with $\sim 100nm$ lamella structure, and is suitable for the cryogenic foam target to ignite in the FIREX project.

IF/P7-26 · Study on the Pulsed Power Fusion at the Kurchatov Institute

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Abstract: Fast implosion of high-current Z-pinches is considered as possible way to the generation of X-ray pulse on the level of some dozens MJ aimed at IFE. In this talk, experiments are presented on sharpening the pulse by the plasma flow switches operating in nanosecond range with typical space scales ∼ 1 mm. In our experiments on the S-300 pulsed power machine, the extreme switching rate onto a tiny load about ten to the fifteenth power A/s has been obtained. In further experiments, the reproducible regime of switching on the level of 750 kA is achieved, with the consequent damping rate ∼100 ns. The radiative temperature of the inner wall of Hohlraum turns out to be as high as 40-50 eV. Experiments were carried out on the current-driven implosion of wire arrays composed of different fractions of Al and W. In the case of nested arrays, the effect of downfall the outer liner, while imploding, through the inner one was first discovered on the base of X-ray spectral analysis. The prospects of application plasma opening switches as output cascades of pulsed power generators of megajoule range is studied. By using the programmed fill the diode gap by a plasma, the suppression of pre-pulse has been achieved and shortening the pulse from 40 microseconds to 100 nanoseconds has been obtained.

IF/P7-27 · Fast Ignition Studies and Magnetic Field Generation

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Abstract: Experiments, theory and simulation have been carried out to describe intense, relativistic short pulse laser interaction with dense plasmas. Magnetic fields of up to 0.7 GGauss have been measured through polarization measurements of high order laser harmonies. Simulation and analytic theory show that these fields are associated with photon momentum deposition. The same mechanism can cause weaker fields in the speckle of long pulse laser in underdense plasmas, leading to termination of SBS. Other magnetic fields in gas-filled hohlraums lead to steep temperature gradients in regions where linear transport fails. A Vlasov-Fokker-Planck code has been developed to study these effects more accurately.

IF/P7-28 · Generation of Relativistic Electron Beam and its Anomalous Stopping in the Fast Ignition Scheme

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Abstract: We present experimental/theoretical results concerning two main physics issues related to the fast ignition scheme viz. the nonlinear mechanism of conversion of incident laser energy into a relativistic electron beam at the critical layer and its subsequent transport through an overdense plasma. Theoretical/numerical modelling of the experimental data, firstly shows that the conversion of the laser energy into an inward propagating electron beam occurs through the nonlinear mechanism of wave breaking of plasma waves excited at the critical layer and, secondly the transport of the electron beam through the overdense plasma is influenced by electrostatically induced and/or turbulence induced anomalous resistivity. In the case of fast ignition, the relative importance of these two mechanisms depend on the resistivity of the cold plasma at the core.

IF/P7-29 · Development of Fast Ignition Integrated Interconnecting Code (FI3) for Fast Ignition Scheme

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Abstract: The numerical simulation plays an important role in estimating the feasibility and performance of the fast ignition. There are two key issues in numerical analysis for the fast ignition. One is the controlling the implosion dynamics to form a high density core plasma in non-spherical implosion, and the other is heating core plasma efficiency by the short pulse high intense laser. From initial laser irradiation to final fusion burning, all the physics are coupling strongly in any phase, and they must be solved consistently in computational simulation. However, in general, it is impossible to simulate laser plasma interaction and radiation hydrodynamics in a single computational code, without any numerical dissipation, special assumption or conditional treatment. Recently, we have developed "Fast Ignition Integrated Interconnecting code" (FI^3) which consists of collective Particle-in-Cell code, Relativistic Fokker-Planck hydro code, and 2-dimensional radiation hydrodynamics code. And those codes are connecting with each other in data-flow bases. In this paper, we will present detail feature of the FI^3 code, and numerical results of whole process of fast ignition.

IF/P7-30 · Theory of ps-Laser Nonlinear Force Driven Ion Beams for Fusion

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Abstract: Theoretical results are presented for the experiments with clean TW-ps neodymium glass laser pulses where an anomalous ion emission was measured. The anomaly was seen in the fact that the maximum ion energy was 50 times lower than the usual value measured with longer pulses where relativistic self-focusing occurred. The number of ions with the clear pulses was independent of the laser power. This and further experiments all confirm the conditions that there was a direct plane wave interaction where the blocks are produced by nonlinear (ponderomotive) force acceleration as an extended skin layer process. In agreement with the measurements we present various numerical calculations showing how two plane density subrelativistic collinear blocks from or into the laser irradiated targets with deuteron current densities above $10^{10} A/cm^2$ are produced which are basically different to the relativistic MeV ion emission after self-focusing. The subrelativistic plane block skin layer acceleration is discussed in view of properties for the fast ignitor and possible light ion beam fusion now generated by the clean ps laser pulses.

IF/P7-31 · Innovative Ignition Scheme for IFE - Impact Ignition -

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Abstract: A new scheme for fast ignition is proposed, in which the compressed DT main fuel is ignited by impact collision of another fraction of separately imploded DT fuel. The ignitor DT shell is ablatively driven in the hollow conical target to hyper-velocities of about 10^8 cm/sec corresponding to temperatures > 5 keV on the collision with the main fuel, and this self-heatedportion plays the role of igniter. The igniter shell is irradiated typically by nsec-pulses at intensities $> 10^15 W/cm^2$ with blue laser to exert ablation pressures $> 100 Mbar$. A preliminary two-dimensional hydrodynamic simulation shows the generation of a dense hot core and thus the feasibility of the new scheme.

IF/P7-32 · Recent findings in NIF ignition target physics and potential implications for IFE

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Abstract: We report on two recent developments from the NIF indirect drive ignition program that potentially have significant implications for IFE using conventional hotspot ignition. One is a theoretical and experimental study suggesting the possibility of ignition with wavelengths longer than NIF's baseline $1/3\mu m$. The second is the discovery of a class of ignition capsules which have far less growth of hydrodynamic instabilities, seeded by surface roughness, than any previous class of ignition capsules. Taken together, these findings offer a possibility for high-gain, conventional IFE at longer wavelengths and lower intensities than is generally accepted.
*This work was performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract No. W-7405-Eng-48.

IF/P7-34 · Progress in Inertial Fusion Energy Modelling at DENIM

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Abstract: New results of the jet driven ignition target are presented both with direct and indirect drive. This target is based on the conical guided target used in fast ignition, but use only one laser pulse. The ignition of the target is started by the impact of a jet produced in the guiding cone, instead of using charged particles generated by a other high power laser. We have shown that a laser or X-ray pulse could be used to produce a high velocity jet of several hundred of km/s by an accumulative effect, and we use these ideas to design this new kind of targets. In order to increase the efficiency of the process, we scan in the simulations different materials, cone profiles and laser intensities. ANALOP is a code developed to calculate opacities for hot plasmas, using analytical potentials including density and temperature effects. It has been recently updated to include the radiative transport into the rate equations by mean of the escape factors, and in parallel a line transport code which solve self- consistently the rate equation and radiative transfer equation in 1D planar geometry has been also developed. We develop a sensitivity-uncertainty analysis method, providing the uncertainties of the different inventory responses functions due to the uncertainty of each of the reaction cross sections separately. Lately, we have developed and proved the excellent behavior of a Monte Carlo-based methodology in assessing the synergetic/global effect of the complete set of cross-sections uncertainties on calculated radiological quantities. The methods have been applied to the activation analysis of the National Ignition Facility and IFE concepts (HYLIFE and Sombrero). Research on multiscale modeling of radiation damage in metals will be presented in comparison with ?ad hoc? experiments. Theory and simulation to explain that physics of SiC radiation damage is being slowly progressing. The systematic identification of type of stable defects is the first goal that will presented after verification of a new tight binding MD technique. Research on simulation of Silica Irradiation Damage up to cascades of 10 keV will be presented. We also will present the role of ingestion by tritiated foods, when the most important chemical forms of tritium, elemental tritium and tritiated water derive in special form of tritium: Organically Bound Tritium.

IF/P7-36 · Laser ablation induced shock pressure amplification in multi layered thin foil targets

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Abstract: High power laser induced shock generation in solids is increasingly becoming important for high pressure research ranging from a few Mbar to less than a Gbar having generated with current short pulsed lasers. Both direct drive and indirect drive approaches are employed for shock generation. The ultra high pressures generated depend on the absorbed laser intensity at the surface of the target material. In the recent years, the thrust is on generation of smooth and planar shock front in the pressure range of few tens of Mbars (30 – 40) to study pressure ionisation effects. By using a high intensity ($> 10^{-14} Watt/cm^2$) laser beam, heating effects inhibit the shock pressure amplification beyond a certain limit. Thus to mitigate these effects, the absorbed laser intensity is desired to be $< 10^{-14} Watt/cm^2$ where the plasma is collisional. When a moving shock front passes from the one material with shock impedance Z_1 to the other material of shock impedance Z_2, then at the interface the pressure enhancement is given by the ratio of the shock impedance of the second to the first material. In this paper, we shall present the results of shock pressure

multiplication in three layered target of increasing shock impedances such as CH_2 – Al – Au using ‘’ impedance matching technique‘’. The technique has been used to generate pressure of 30 – 40 Mbar, using solid densities of the various composite materials. The results are based on coupled radiation hydrodynamic simulations using the code MULTI which uses multigroup opacities and realistic equation of state of various materials.

IF/P7-52 · Laser Driver for IFE: Novel Approach

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Abstract: New concept of creation of laser driver for IFE based on generation and amplification of radiation with controllable coherence is reported. The module of the laser facility has been triggered to check the validity of the proposed concept for constructing a laser driver for power stations, and the experimental results are reported. The experiments are shown a possibility of suppression of small-scaled self-focusing, formation of laser radiation pulses with required characteristics, simplification of optical scheme of laser, good matching of laser-target system and achievement of homogeneous irradiation and high output laser energy density without using traditional correcting systems (phase plates, adaptive optics, space filters etc.). The achieved value of the output energy density is equal to $9,3 J/cm^2$. Also the results of last experiments to achieve ultimate energy characteristics of developed laser system are reported.

IC

Innovative Confinement Concepts

IC/P6-6 · Sustainment and Additional Heating of High-Beta Field-Reversed Configuration Plasmas

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Abstract: Sustainment of high-beta field-reversed-configuration (FRC) plasmas have been successfully achieved for 4ms (longer than 20 times apparent energy confinement time) by the application of rotating magnetic field (RMF) at frequencies between ion and electron gyro frequency for the first time in a metal confinement chamber. The configuration was maintained as far as the RMF was applied, whereas, conventional FRCs did not have methods of driving or sustaining toroidal current and decayed in about 4 times energy confinement time typically, or in our case, about 0.5ms. The degree to which the internal magnetic field is reversed is intensively investigated mainly as a function of the RMF intensity and frequency, bias field and target density. NB heating experiment at the NB power of 250kW was accomplished in the high-beta FRC plasma. Improvement of confinement properties was observed. Careful study revealed that the improvement was ascribed to the rise in Te. Although confinement properties of FRC plasmas have so far been described only by Ti, Te dependence of them has been found for the first time.

IC/P6-16 · Potential control and flow generation in a toroidal internal-coil system – a new approach to high-beta equilibrium –

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Abstract: Potential control and flow generation have been studied on Proto-RT and Mini- RT. Each of these devices is equipped with an internal ring coil that produces stationary magnetic- surface configurations. On the Proto-RT device, we have generated a supersonic flow by biasing the surface of the internal coil. With negative biasing, we can produce an almost uniform electric field across the plasma. The potential contours coincide with the magnetic surfaces. The internal electric field (of order 1kV/m) yields the drift speed of order 100 km/sec that is much higher than the ion sound speed. Positive biasing, however, yields a gap (low density layer) between the electrode and the plasma. This is because the radial current (primarily ion current) sweeps the ions away from the coil surface. The relative permittivity of the plasma is of order 100, and hence, the electric field concentrates into the gap. The electromagnetic torque due to the radial ion current balances the collisional friction force between the flow and the neutrals. In the present experiment (Proto-RT), the plasma parameters still remain in the electrostatic regime because of the low density. If the density is raised and the flow velocity is comparable to the Alfven speed, the hydrodynamic pressure can produce a new type of high-beta (diamagnetic) equilibrium, so-called "double Beltrami (DB) state". Recent theory predicts "self-organization" of a DB state that may have Lyapunov stability.

IC/P6-33 · Identification of the sequence of steps intrinsic to spheromak formation

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Abstract: A planar coaxial electrostatic helicity source is used for studying the relaxation process intrinsic to spheromak formation. Experimental observations reveal that spheromak formation involves: (1) breakdown and creation of a number of distinct, arched, filamentary, plasma-filled flux loops that span from cathode to anode gas nozzles, (2) merging of these loops to form a central column, (3) jet-like expansion of the central column, (4) kink instability of the central column, (5) conversion of toroidal flux to poloidal flux by the kink instability. Steps 1 and 3 indicate that spheromak formation involves an MHD pumping of plasma from the gas nozzles into the magnetic flux tube linking the nozzles. In order to measure this pumping, the gas puffing system has been modified to permit simultaneous injection of different gas species into the two ends of the flux tube linking the wall. Gated CCD cameras with narrow band optical filters are used to track the pumped flows from the two ends of the flux tube, their convergence, and the association of this convergence with collimation of the flux tube. Preliminary results have been obtained showing the flows.

IC/P6-34 · Numerical Study of the Formation, Ion Spin-up and Nonlinear Stability Properties of Field-Reversed Configurations

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Abstract: Results of three-dimensional numerical simulations of the field-reversed configuration (FRC) using the HYM code are presented. Emphasis of this work is on the nonlinear evolution of magnetohydrodynamic (MHD) instabilities in kinetic FRCs. Kinetic simulations show nonlinear saturation of the n=1 tilt mode. The n=2 and n=3 rotational modes are observed to grow during the nonlinear phase of the tilt instability due to ion spin-up in the toroidal direction. The ion toroidal spin-up is shown to be related to the resistive decay of the internal flux, and the resulting loss of particle confinement. Three-dimensional simulations of counter-helicity spheromak merging and the FRC formation show good agreement with results from the SSX-FRC experiment. Simulations show formation of an FRC in about 30 Alfven times for typical experimental parameters. Growth rate of the n=1 tilt mode is shown to be significantly reduced compared to the MHD growth rate due to large plasma viscosity, field-line-tying effects, and strong toroidal flows generated during the reconnection phase.

IC/P6-35 · Solenoid-Free Toroidal Plasma Start-Up Concept Utilizing only the Outer Poloidal Field Coils

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Abstract: Eventual elimination of in-board ohmic heating solenoid is required for the spherical torus (ST) reactors and it is considered to be highly desirable for advanced tokamak reactors. A fundamental challenge for using only the outer poloidal field coils for the start-up purpose is the difficulty of creating a sufficiently high quality field null region while retaining significant poloidal flux needed for subsequent current ramp up. Here, we show through both static and dynamic calculations that a carefully chosen proper set of outer poloidal field coils can indeed offer a promising prospect of creating a good quality "multi-pole" field null while retaining sufficient poloidal flux, in particular, satisfying the "Lloyd" criteria for the inductive plasma start-up. For a single turn TF system envisioned for ST-based CTF and power plant, the polidal magnetic flux stored in the TF inner leg can provide additional significant flux. This concept can be readily extended to future devices for a multi-MA level start-up current due to the relatively simple physics principles and a favorable scaling with device size and toroidal magnetic field.
*This work is supported by KAIST and DoE Contract No. DE-AC02-76-CH0-3073.

IC/P6-36 · Present status of operation of the ETE spherical tokamak

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Abstract: The ETE is a spherical tokamak with aspect ratio A=1.5 (major radius of 0.3m and minor radius of 0.2m) under development at LAP/INPE. The ETE was designed to reach plasma current up to 440kA with magnetic field up to 0.8T at the plasma axis. These ultimate parameter values are limited by mechanical stresses in the demountable joints of the toroidal field coils and by stresses and heating of the ohmic solenoid ($\sim$ 0.25Wb). The ETE incorporates some innovative features that resulted in a compact and light weighted device with good plasma accessibility. Since the first plasma obtained at the very end of 2000 ($I_p = 12kA$, duration of 2ms, $B_o = 0.1T$), the machine is operational and improvements are being done in order to achieve the planned final parameter values for the first phase of operation ($I_p = 220kA$, duration 15ms, $B_o = 0.4T$), which are limited by the available capacitors. The efforts are being focused on incrementing the energy of the capacitor banks, lessening the stray magnetic fields, conditioning of the vacuum vessel wall, implementing diagnostics and optimizing the discharge parameters. Presently, plasma currents in the range of 40-60kA (duration of 6-10 ms) are routinely obtained.

IC/P6-37 · Steady Supersonic Rotation in the Maryland Centrifugal Experiment

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Abstract: The Maryland Centrifugal Experiment(MCX) studies enhanced confinement and stability produced by sheared supersonic rotation about a linear confining magnetic field. MCX has a mirror geometry of 2.6m length, mirror ratio 2-20,maximum mirror field 1.9T, maximum midplane field 0.5T. Biasing of an inner electrode relative to the outer wall produces a radial electric field which drives azimuthal rotation.

MCX has achieved high density ($n > 1020m^{-3}$) fully ionized plasmas rotating supersonically with velocities of ∼100 km/sec for times exceeding 8 ms under a wide range of conditions. Ion temperatures are 30 eV and confinement times ∼100 microseconds. Sonic mach numbers are 1-2 and Alfven mach numbers somewhat less than 0.5 for standard discharges. Plasmas remain grossly stable, or steady, for many milliseconds, much longer than MHD instability timescales for MCX, though significant magnetic fluctuations are clearly seen on magnetic probes. Ion density, confinement time, and radial plasma voltage all increase with magnetic field strength; rotation velocity saturates at higher B. Best performance requires mirror ratios of 5 or greater. Recently MCX has demonstrated an enhanced mode of operation with sonic mach numbers greater than 3, confinement times of several hundred microseconds and Alfven mach numbers near one.

IC/P6-40 · High frequency way of helium ash removal from stellarator-reactor

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Abstract: The paper deals with the problem of helium ash removal from stellarator- reactor. To solve this problem the lower hybrid heating of ash ions is proposed. The theory of ion stochastic heating, developed earlier by Karney, is generalized on the case of heating in stellarators. The features of the lower hybrid waves propagation and the ions motion in the stellarator confining field are taken into account. With proper choice of wave parameters (such as frequency, antenna position and initial spectrum of longitudinal refractive index) the slow mode of LH waves penetrates from the launching system to plasma core (and back) without conversion to kinetic plasma mode or to fast mode. With all these going on, the LH wave is absorbed by alpha particles only. The electron Landau damping is negligibly small, and there is no bulk ions stochastic heating. The motion of high energy (100 keV) ions in the LHD heliotron with inwardly shifted magnetic axis, as an example of stellarator type device, is calculated numerically using the single particle simulation code which couples modified Karney's ion stochastic heating theory. The effect of collisions was taken into account through the Monte Carlo equivalent of the Lorentz collision operator. It is shown, that due to interaction with lower hybrid wave, initially well-confined alpha particles are expelled from the plasma during the time period less then collision time. This effect is slightly dependent on alpha particle initial pitch value. At the same time, the low hybrid heating does not remove the ions with energy higher than 500 keV. Therefore, it is possible to use this method of RF heating for helium ash removal in stellarator-reactor. The required LH power is estimated to be of the order of 10 MW.

IC/P6-41 · Long Pulse FRC Sustainment with Enhanced Edge Driven Rotating Magnetic Field Current Drive

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Abstract: The pulse length on the quasi steady-state FRC experiment, TCS, has been extended to 10 msec, approximately 100 times the natural flux decay time. A new phenomenon has been noted, where the rotating magnetic field (RMF) sustainment switches to a higher performance mode, with lower overall effective plasma resistivity and stronger current drive. This result occurs for an edge driven mode where the RMF only penetrates the plasma near the separatrix. Detailed comparison with numerical modeling shows that this mode occurs due to a very non-uniform resistivity profile, with relatively low resistivity in the FRC center, and high resistivity in the driven, narrow, high electron velocity edge region. The switching between low and high performance is accompanied by subtle changes in the RMF penetration profile and some spontaneous generation of toroidal field. The results have important implications for the use of RMF as an FRC sustainment tool.

IC/P6-42 · Recent Results from the HIT-II and HIT-SI Helicity Injection Current Drive Experiments

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Abstract: Three important results are reported. 1) CHI startup has produce 100kA of closed current without using poloidal field (PF) coils or any transformer action. The initial equilibrium is then driven to 240kA with a 3V transformer loop voltage, indicating high quality plasma. 2) For the first time CHI alone has produced toroidal currents (330kA) that far exceed qaIinj, and with I_p/I_{tf} as high as 1.2. 3) The steady inductive helicity injection experiment has operated at 5kHz for 6ms with current amplitudes

up to 11kA in each injector. The helicity injection rate is nearly constant with the EXB flow always into the plasma and not into the walls.

IC/P6-43 · Scrape Off Layer Physics for Burning Plasmas and Innovative Divertor Solutions

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Abstract: Three distinct but related topics concerning scrape-off layer physics are examined. Recent experimental discoveries of large SOL transport are extrapolated empirically to a reactor. ARIES RS geometries have been simulated using UEDGE, including large convection in the far SOL similar to what is seen in experiments. The high edge plasma temperature plus large recycling from SOL transport give highly enhanced sputtering for a tungsten wall from CX neutrals. Empirical estimates of impurity screening imply a potential for core radiation collapse. A set of 2.5 D nonlinear fluid equations (without empirical parameters) is developed for SOL turbulence. This set readily enables computation to statistical steady state. They describe the birth of blobs by resistive ballooning turbulence near the separatrix, and their nonlinear, turbulent evolution across the far SOL to the wall. We also include impurity transport, and calculate important atomic processes via a kinetic neutral code. Preliminary simulations for C-MOD and DIII-D have shown that blobs transport a significant fraction of the particle flux to the main chamber wall rather than to the divertor, in qualitative agreement with data. The return flow in the region between blobs carries impurities from the wall to the main plasma. Simulations for ITER and reactors will be performed. Related work on novel magnetic divertor geometries is presented: 1) inducing a second axi-symmetric flux expansion region along the separatrix, and 2) the extraction of the separatrix flux to outside the TF coils with low plasma TF ripple. These allow the engineering advantages of a highly detached state to be realized without the poor confinement and disruptive tendencies found in conventional divertor magnetic geometries. Unoptimized coils for 1) and 2) are complex, so simpler designs are being developed with codes used for the National Compact Stellarator Experiment coils. 2-D neutral/plasma simulation of the divertor region will be performed to verify that stable detachment is possible.

IC/P6-44 · Direct Access to Burning Spherical Tokamak Experiment by Pulsed High- Power Heating of Magnetic Reconnection

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Abstract: The merging/ reconnection startup of high-beta ST has been developed in the TS-3/4 experiments, leading us to its new extension to the pulsed high-power heating for burning plasma formation. Two STs were produced inductively by swing-down of two or four PF coil currents without using any center solenoid (CS) and they were merged together for high-power reconnection heating. The reconnection outflow speed equals to the Alfven speed under no guiding field condition. The outflow energy is converted mostly into ion thermal energy through ion viscosity and/or fast shock, indicating that the ion temperature increment (and thermal energy) scales with squares of reconnecting magnetic field (Alfven speed). This unique method has the highest heating power MW-GW among all CS-less startups and the heating time much shorter than the energy confinement time and the electron-ion collision time. These facts indicate that the merging of two STs possibly provides a direct path to the burning plasma formation. The TS-3/4 scaling data suggest that two merging STs with $B = 1 - 3T$, $n = 10^{20}m^{-3}$ will be transformed into an ITER-like ST with T=20keV within reconnection time.

IC/P6-45 · Plasma Control for NCSX and Development of Equilibrium Reconstruction for Stellarators

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Abstract: The simulation of an entire NCSX discharge, the implications for plasma control, and work towards designing a set of magnetic diagnostics required to effect the necessary control are discussed. TRANSP is used to evolve the pressure and poloidal flux profiles in a 2D equivalent plasma model. Calculated profiles are input to STELLOPT to calculate the coil currents, 3D equilibrium, and constrained plasma physics properties. A set of magnetic diagnostics is being designed using a "control surface" and eigen-analysis database approach. Development of a stellarator equilibrium reconstruction tool V3FIT based on VMEC and EFIT are presented. Magnetic reciprocity relation is used to generalize the EFIT response function approach to 3D. Two tools V3RFUN and V3POST have been developed to efficiently compute 3D magnetic diagnostic responses and are being applied to support design of magnetic diagnostics.

A prototype 3D reconstruction code for examination of numerical features of the reconstruction process has been built and is being tested.
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IC/P6-46 · Interaction of Ambipolar Plasma Flow with Magnetic Islands in a Quasi- Axisymmetric Stellarator

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Abstract: We find that ambipolar plasma flow in quasi-axisymmetric (QA) stellarators can strongly suppress magnetic island formation by shielding rational surfaces against resonant perturbations. QA stellarators displaying this effect lie in a regime where the configuration is sufficiently close to quasi-symmetry to allow the plasma to flow in the toroidal direction with little damping, but where the deviations from quasi-symmetry are sufficiently large to produce a substantial ambipolar plasma flow. A reference NCSX equilibrium is calculated to lie in this regime. The layer physics at the rational surface is similar to that in tokamaks, but the momentum diffusion equation, which determines the viscous torque exerted on the rational surface, contains an additional term. As the electromagnetic torque exerted by a resonant perturbation causes the flow velocity to deviate from its ambipolar value, the resulting nonambipolar transport produces a radial current that exerts an opposing torque. Our modeling employs a 1D transport code, as well as the DEGAS code for estimating the momentum transfer rate to neutrals, and the PIES code for calculating the magnitude of the resonant perturbation.

IC/P6-47 · Evolution of Plasma Flow Shear and Stability in the ZaP Flow Z-Pinch

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Abstract: The ZaP Flow Z-pinch experiment at the University of Washington investigates the concept of using sheared flows to stabilize an otherwise unstable plasma configuration. The stabilizing effect of a sheared axial flow on the m = 1 kink instability in Z-pinches has been studied using linearized, ideal MHD theory to reveal that a sheared axial flow stabilizes the kink mode when the shear exceeds a threshold. The ZaP experiment generates a Z-pinch plasma that is 1 m long with a 1 cm radius with an embedded axial flow. Time-resolved Doppler shifts of plasma impurity lines are measured along 20 chords to determine the plasma axial velocity profiles. An azimuthal array of magnetic probes measures the fluctuation levels of the azimuthal modes m = 1, 2, and 3. After pinch assembly a quiescent period is found where the mode activity is significantly reduced. The plasma axial velocity evolves from a uniform to a sheared and back to a uniform flow profile. The sheared flow profile is coincident with a plasma quiescent period where magnetic mode fluctuations are low. The value of the velocity shear satisfies the theoretical threshold for stability during the quiescent period and does not satisfy the threshold during high mode activity.

IC/P6-48 · Liner compression of a self-organized MAGO / inverse-pinch configuration

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Abstract: In the "metal liner" approach to Magnetized Target Fusion (MTF), a 200-400 eV magnetized plasma target is compressed to thermonuclear temperature and high density (eg., $10^{26}/m^3$) by externally driving the collapse of a flux conserving metal enclosure, or liner, which contains the preheated plasma target. One possible plasma target is the hard-core diffuse z pinch. Such a configuration is studied in MAGO experiments at VNIIEF, using a unique dual-chamber formation method. The configuration can also be formed using an inverse pinch, and an experiment of that type is being constructed at the University of Nevada, Reno. Recently, two-dimensional resistive MHD simulations, which include non-ideal effects such as thermal conduction and radiation, have shown an intriguing feature of the diffuse z pinch. The non-linear evolution of m=0 interchange modes, arising from unstable pressure profiles, results in self organization into stable pressure profiles. A saturated level of turbulence is observed that results in convective thermal transport, but simulations still show substantial heating during liner compression. In addition, a liner experiment is being designed to study magnetic flux compression using the pulsed-power Atlas facility (23 MJ, 30 MA). The geometry for Atlas will be compatible with a diffuse z pinch but initial experiments will not involve plasma.

IC/P6-49 · Experimental Study on a New Spherical Tokamak Configuration Scheme Employing by Means of Spherical Snow-plough

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Abstract: The aim of this study is to identify the physical bases of an alternative self organization mechanism that exist on STPC-EX machine and to determine complementary features with respect to present compact toroid concepts. Here, either simplified operational properties of STPC-EX or the demonstration of spherical tokamak plasma (STP) creation using spherical snow-plough (SSP) with dual-axial z-pinch (DAZP) are presented. The STP in the envelope of SSP is shaped relating to m is 0 mode of DAZP. In this procedure, the basic objects to be characterised at the conventional STP are controlled by principal structural geometry of STPC-EX set-up and previously selected reference data of current-launcher. The main points achieved on this study have been: Aspect ratio is 1.2-1.6; averaged beta is 0.46- 0.62; elongation is 4-6; triangularity is 0.42-0.58; sustainment time is 4.3-6.5 ms; energy confinement time is 45-136 ms; Te plus Ti is 188-177eV; electron density is $10^{20} - 10^{22} m^3$. From the conceptual point of view this study has given a possibility for approach to the fission-fusion hybrids.

IC/P6-50 · Research on the Enhancement of the Thermonuclear Component of the Neutron Yield in Pinch Plasma Focus Devices

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Abstract: The possibility to enhance the thermonuclear component against the beam target component of the neutron yield in plasma focus devices is being studied. It is usually accepted for plasma focus devices operating in deuterium, that the total neutron yield Y, is $Y = Y_{th} + Y_{bt}$, where Y_{th} is the thermonuclear component and Y_{bt} is the beam target component. In reference [1], it is suggested that Y_{th} and Y_{bt} scale as $Y_{th} \alpha I^4 v^4$ and $Y_{bt} \alpha I^{4.5} v^{1.5}$ (I: peak current, v: velocity of the current sheath). In addition it is possible to consider $v \alpha I/a$. An analysis of data for Mather-type plasma focus devices for a wide range of sizes (energies from 1kJ to 1MJ), shows that v has practically a maximum fixed value, $10 cm/\mu s$ in the axial phase and $25 cm/\mu s$ in the radial collapse [2]. Increasing v (or I/a), it could be possible to increase the thermonuclear component of the neutron yield and to decrease the beam target component. In fact, in the limit keeping the anode radius constant whilst increasing the current, $Y_{th} \alpha I^8$ and $Y_{bt} \alpha I^3$ [1]. With this improved or enhanced yield dependence, the thermonuclear component of neutron yield will rapidly outstrip the beam target component. At present, the Chilean Nuclear Energy Commission, CCHEN has the experimental facilities and diagnostics in order to study plasma focus discharges in a wide range of energies (10J to 100kJ) and currents (10kA to MA). The devices at CCHEN are PF-50J, PF-400J, SPEED4 and SPEED2 [3-5]. As a part of our research program [4, 5] the possibility to study how to enhance the parameter I/a and its role in the thermonuclear component of the neutron yield has been recently included. We expect to obtain experimentally how scale the neutron yield with I and v using the devices at CCHEN. Preliminary results of this research program are presented. SPEED2 is a donation from Dusseldorf University to CCHEN.
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IC/P6-51 · Recent Advances in Quasi-Poloidal Stellarator Physics Issues

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Abstract: Quasi-poloidal (QP) stellarators achieve low effective ripple and approximate poloidal symmetry (in magnetic coordinates) through the use of a racetrack shaped magnetic axis and vertically elongated cross-sections in the regions of high toroidal curvature. This form of optimization has resulted in the design of attractive compact stellarator hybrid systems such as the two-field-period, very low-aspect-ratio (A = 2.7) QPS device. Through variable modular coil currents and plasma current drive (e.g., bootstrap, Ohmic) such devices can also be designed with a high degree of physics flexibility. The important physics issues for

QPS include flux surface fragility at low aspect ratio, plasma flow generation, plasma/energetic particle confinement, and plasma stability. Recently developed computational tools for evaluating these topics in 3D systems will be described and results for QP systems discussed. QP stellarators allow neoclassical transport to be reduced substantially below anomalous levels; flexibility studies have also shown that low collisionality transport levels can be varied by a factor of 25 about the design level. Novel anisotropies in the neoclassical viscosity tensor are characteristic of QP systems; calculations have shown that these differ significantly from those of tokamaks and other stellarators. For example, the poloidal viscosity is reduced by a factor of 10 below that of the equivalent tokamak in the experimentally relevant ion plateau regime. This feature is expected to be of importance for the generation of sheared flows and access to enhanced confinement regimes. Flexibility studies have shown that this viscosity can be varied by about a factor of 10. QP stellarators also offer bootstrap current levels compatible with steady-state operation, and ballooning second stable regimes. Second stability occurs for volume-averaged betas of about 6% while first stability ballooning limits are 2 to 2.5%; these can be lowered into ranges that can easily be tested. Ion diamagnetic effects (FLR) and shape control are expected to provide access between first and second stable regimes. Magnetic islands have been suppressed in QPS using the variable modular coil and plasma currents; island avoidance can be achieved either by tailoring the iota profile to remain between adjacent resonances or by targeting the residues of the dominant island chains.

IC/P6-53 · FRC plasma studies on the FRX-L plasma injector for MTF

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Abstract: To demonstrate the physics basis for Magnetized Target Fusion, we have designed a Field Reversed Configuration (FRC) target plasma to ultimately be compressed within an imploding metal liner. This new, high density FRC device, named FRX-L, is operating at Los Alamos as a compact "theta-pinch" formation FRC, and construction was largely completed in 2002. The system includes a 0.5 T bias field, 70 kV 250 kHz ringing pre- ionization, and a 1.5 MA, 200 kJ main-theta-coil bank. We show FRC data with plasma parameters approaching the desired MTF requirements, examples of large ohmic heating from magnetic flux annihilation, and measurements of plasma anomalous resistivity. Improvements which will increase the trapped flux in the FRC, and a reduction in main bank crowbar ringing, along with new diagnostics in 2004 will document performance at the design level, before implosion experiments are attempted. A prototype deformable flux-conserving liner with large entrance holes to accept the FRC has also been designed with MACH2 (2-D MHD modeling) and successfully imploded at Kirtland AFB on the Shiva Star pulsed power facility.

IC/P6-54 · Research and Development of a Compact Fusion Neutron Source for Humanitarian Landmine Detection

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Abstract: An Inertial-Electrostatic Confinement fusion (IECF) neutron source consists of a transparent hollow cathode at the center of a spherical vacuum chamber (serves as an anode) filled with a D_2 fuel gas, and glow discharge takes place between them, thereby, produced ions accelerated toward the center through the transparent cathode undergo fusion reactions through beam-beam or beam-background gas collisions. Taking advantages of its compactness, robustness, and long lifetime compared with conventional neutron emitters such as radioactive isotopes and accelerator-driven fusion devices, we have started R & D of advanced anti-personnel landmine detection system by using the IECF neutron source. Since explosives include C, N, O, H and Cl atoms in specific fraction, interaction with neutrons, such as neutron backscatter, and neutron-induced gamma rays, could be made use of as an innovative method effective to all-plastic mines, in particular, in order to speed up the humanitarian demining processes currently being done using conventional methods, such as metal detectors, and sniffer dogs. We have developed an extremely compact IECF device of 200 mm dia. with a titanium getter pump as a main exhaust pump to endure the vibration when it is installed at automobile. Also, a compact high voltage power supply system was designed. A prototype testing with an IECF device showed an extremely high neutron yield of 1.2×10^8 /sec in a pulsed operation for - 51 kV and 7.3 A peak. We expect a higher yield by the design goal of the pulsed power supply of 90 kV and 10 A. A magnetron-type built-in ion source of very simple configuration is also being developed for further enhanced yield by producing substantial ions in the vicinity of the spherical chamber to provide full energy to the ions with reduced charge-exchange processes under a reduced gas pressure. It is found that with the help of the ion source the fusion discharge is achieved even less than 2 mTorr D_2,

under which normal glow discharge never takes place, resulting in a high neutron yield of 2.1×10^7 /sec for a low input power of - 65 kV and 45 mA. Further design studies of the magnetron discharge system are being carried out by use of particle simulation codes to achieve a higher ion current supply. Also, the optimal setup position of the ion source is studied by use of a 3-D particle code.

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Fusion Technology and Power Plant Design

FT/1-1Ra · New Results in Development of MW Output Power Gyrotrons for Fusion Systems

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Abstract: The paper presents the latest achievements in development of MW power level gyrotrons for fusion installations. During two last years four new gyrotrons were designed and tested: a new version of 170 GHz gyrotron for ITER; multi-frequency (105-140 GHz) gyrotron for Asdex-Up, 84GHz gyrotron for LHD and 82.7 GHz gyrotron for SST-1. All these gyrotrons are equipped with diamond CVD windows and depressed collectors.The most efforts were spent for development of ITER gyrotron. The following gyrotron output parameters were demonstrated: 0.9MW/20 sec and 0.7MW/40 sec.

FT/1-1Rb · Performance of 170 GHz high-power gyrotron for CW operation

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Abstract: The development of a high-power millimeter wave source, gyrotron, is under way for fusion application. A performance of ∼10 s oscillation has been attained at 1 MW level output for 170 GHz frequency. Although the pulse extension was interrupted by a sudden outgassing in the gyrotron, the cause was confirmed to be the local heating of the internal component due to stray RF deposition. To suppress the heating, the inner surface of the component (bellows for RF beam steering mirror) was coated with copper, which reduced the Ohmic loss to 1/10 of the original one. Moreover, forced water cooling for the bellows was incorporated. As a result, no sudden pressure increase was observed, and a quasi-steady-state oscillation of 100 s with 0.5 MW power level was demonstrated at 170 GHz. The temperature of the major components of the gyrotron stabilized, which indicates a prospect for a 1 MW-CW, 170 GHz gyrotron.

FT/1-1Rc · Development of Steady-State 2-MW 170-GHz Gyrotrons for ITER

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Abstract: The development of high power gyrotrons in continuous wave (CW) operation is in progress in cooperation between European research institutions and European tube industry. A 1 MW, CW gyrotron for the stellarator W7-X in Greifswald/Germany has been developed successfully. This gyrotron operates in the TE28,8 mode. It is equipped with a highly efficient internal q.o. mode converter, a single-stage depressed collector and an edge- cooled, single-disk CVD-diamond window. After the test of a first prototype tube, an improved second prototype has been built and tested. For 180 s an output power of 890 KW has been obtained. The pulse length was limited by the capability of the HV power supply. At reduced current of about 25A an output power of 540 kW at a pulse length of 937 s with an efficiency of 41% has been achieved. This yielded an energy of 505 MJ per pulse. The limitation in pulse length is due to increase of the pressure inside the gyrotron. The pressure rise has been recognised to come from warming up of internal getter pumps due to microwave stray radiation. To avoid this problem in the series gyrotrons the pumps will be placed externally. Within a development program performed as ITER task at FZK the feasibility of manufacturing a 2 MW coaxial gyrotron operated in CW has been demonstrated and information necessary for a technical design has been obtained. Based on these results the development of a coaxial cavity gyrotron with an RF output power of 2 MW, CW at 170 GHz is in progress within the above mentioned European cooperation. The conceptual design of such a tube compatible with CW operation has been completed and the manufacturing process of a first prototype has been launched. The tube is designed to operate at a frequency of 170 GHz, an electron beam current of 75 A and an accelerating voltage of 90 kV. An RF output power of 2 MW is expected with an efficiency exceeding 45% with single-stage depressed collector. In parallel to the work on the industrial prototype, the design of the most critical gyrotron components will be verified under realistic conditions using a short pulse (5 - 10 ms) gyrotron at FZK. This experimental tube is equipped with the same cavity and RF-output system as designed for the industrial prototype. The result of the short-pulse experiments and a description of the prototype gyrotron will be given.

FT/1-2Ra · R&D on a High Energy Accelerator and a Large Negative Ion Source for ITER

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Abstract: The R&D of a 1 MeV accelerator and a large negative ion source have been carried out at Japan Atomic Energy Research Institute (JAERI) for the ITER NB system. At the previous conference, achievements of 1 MV voltage holding in the accelerator and H- ion production at low gas pressure ($300A/m^2 H^-$ at 0.1 Pa) were reported. The R&D is in progress at present toward: 1) demonstration of 1 A class H- ion acceleration at the current density of $200A/m^2$ up to 1 MeV as a "Proof-of-Principle (PoP)" of the ITER accelerator, and 2) improvement of uniform negative ion production over a wide extraction area. The latter has been one of major causes that limited the NB injection performance of the existing negative ion based NB systems such as JT-60U N-NBI. Recently, H^- ion beams of 1 MeV, 100 mA class have been generated with a substantial beam current density ($80A/m^2$). A key physics process of uniform negative ion production has been well understood in a large negative ion source. These encouraging progress supports R&D program planned for the ITER construction phase, in which next step is to test the first ITER NB injector at a testbed, before installation at the ITER site.

FT/1-2Rb · Improvement of Negative Ion Source with Multi-Slot Grids for LHD-NBI

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Abstract: In this article, we report the experimental results on a cesium seeded hydrogen negative ion (H^-) source newly designed for the neutral beam injector of the large helical device. The ion source consists of the accelerator with a multi-slot grounded grid, whose beam transparency is twice as large as conventional multi-aperture grounded grid. That is because voltage breakdowns between beam acceleration gap are considered due to striped electron, neutral gasses and sputtered ions from beam grounded grid caused by the beam exposure onto the grid. The heat loads onto the multi-aperture and multi-slot grids was water-calorimetrically measured for comparison. The heat load onto the multi-slot grid was reduced half as much as the load onto the multi-aperture grid. By exchanging the grid, the maximum injection power increased drastically up to 4.4 MW at the energy of 180 keV in fiscal 2002. Although the maximum beam energy increased using the multi-slot grid, saturation tendency of injection power was observed in a range of beam energy higher than 170 keV. The cause of the saturation was considered insufficiency of the optimization on production of H^- ions. In order to enhance the production rate, two-step improvement has been done. The first is to increase the temperature of plasma grid. The yield of negative ions depends on the surface workfunction of plasma grid. The temperature of the plasma grid was adjusted to higher temperature range where the H^- production rate is more enhanced. The next is an adjustment of distribution balance of arc plasma by increasing total filament number. After the improvements, the saturation tendency was removed and the maximum injection power reached 5.7 MW at the energy of 186 keV after improving the H^- production rate.

FT/1-2Rc · Status And Plans For The Development Of An RF Negative Ion Source For ITER NBI

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Abstract: The ITER reference design for Neutral Beam Injection heating is based at an arc source rated for 40 A of D^- ions extracted from an net area of $0.89m^2$. The main problem of an arc source is the limited lifetime of the filaments. Furthermore it is suspected that the arc current is responsible for the source non uniformity observed in large arc sources for negative ion production. Therefore RF sources, developed successfully at IPP for neutral beam heating based on H+ and D+ ions, offer substantial advantages for ITER neutral beam heating. The development of an RF ion source for negative ions has been started on a larger scale at IPP in December 2002 through an EFDA contract. So far current densities of $260A/m^2$ for hydrogen and $170A/m^2$ for Deuterium have been achieved for an extraction area of $0.07m^2$. The electron/ion ratio can be kept below 1 for hydrogen and deuterium if the filter field is sufficiently strong. Deuterium requires a stronger filter field than hydrogen. With the present set-up operation with strong filter field limits the useful RF power. Modifications to overcome this limitation are being prepared. An extension of the extraction area from 0.07 to $0.15m^2$ has already been demonstrated without loss of current density. This larger extraction area corresponds to 2/3rd of the area supplied by one RF driver in the ITER size source. Parallel to the source development the design and manufacturing of a test facility for pulses of up to 1 hour duration is proceeding, scheduled for commissioning towards the end of 2004. A scaled up

ion source with the same width and half the length of the ITER reference source will become available for commissioning early in 2005. The paper will present the latest results of the source development, design details of the half size ITER source and the status of the long pulse development.

FT/1-3 · Results and Implications of the JET ITER-Like ICRF Antenna High Power Prototype Tests

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Abstract: A high power prototype (HPP) of the JET ITER-Like ICRF Antenna has been built and tested in a joint effort by ORNL, PPPL, ERM/KMS, and EFDA-JET. For the first time, high power vacuum tests have been performed on an antenna with features similar to those found in the current ITER antenna design. Advances include a new matching network using internal capacitors that is insensitive to target plasma characteristics, and increased power density to minimize the amount of port space required. Three-dimensional electromagnetic modeling has been used extensively to improve the antenna design and assist in interpretation of test results. During the HPP tests, capacitor voltages greater than 45 kV were achieved for short (0.05s) pulses. Voltages greater than 35 kV, approximately the voltage needed to couple the full 7.2 MW design power into most JET plasmas, were sustained for moderate length ($\sim$0.5 s) pulses. Long (10s) pulse operation was limited by excessive heating in localized regions due to rf dissipation, a problem that will be corrected in the final antenna. In this paper test results, associated numerical modeling, and their impact on the design of the final device will be reviewed.

FT/1-4 · Progress of Reduced Activation Ferritic/Martensitic Steel Development in Japan

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Abstract: Extensive efforts for the development of the reduced activation ferritic/martensitic steels (RAF/Ms) have been accomplished in recent several years. They are, examinations of the effects of neutron irradiation on (1) Ductile to brittle transition temperature (DBTT) to explore lower temperature limit, (2) Enhanced He effect on DBTT shift for Ni/B doped heats, (3) Fatigue behavior, (4) Susceptibility to environmentally assisted cracking and (5) Flow stress-plastic strain, including (6) the improvement of ductility of RAF/M ODS alloys with high temperature strength and other supporting researches. Preparation of the design criteria for ITER test blanket module, closely relating to the results for (3), (4) and (5) above, is also one of the key tasks in the development of RAF/Ms (eg. F82H; 8Cr 2WTaV). Irradiation effect on fatigue properties was revealed to be not significant; except for the results with a very small cyclic strain range, irradiation did not introduce appreciable change in fatigue life. However, irradiation introduced fatigue mechanism change at a smallest plastic cyclic strain range of about 0.1% and smaller. Irradiation may also cause to increase susceptibility for cracking in high temperature pressurized water. Slow strain rate tensile tests in high temperature pressurized water environments were conducted. No obvious change in fracture mechanism was detected. Irradiation effect on constitutive equation of plasticity (flow stress-plastic strain relation) was analyzed. Results indicate that the effect was successfully expressed quite simply by introducing an equivalent strain for hardening. This finding drastically simplifies to estimate ductile fracture condition of the component under irradiation. One of the major life limiting issues is the degradation of DBTT shift by irradiation, and its enhanced effect by He. Heats doped with Ni and B of different isotope ratio to simulate the He effect in fusion neutron environment were irradiated and tested. The results including previous ones indicate that DBTT shifts almost linearly with hardening. In addition, effect of He is about one third or smaller than has been indicated previously. Simultaneously, suppression of hardening is clearly effective in reducing the DBTT shift. Application of ODS technique to expand upper bound temperature will also be introduced.

FT/1-5 · Assessment of plastic flow and fracture properties with small specimen test techniques for IFMIF-designed specimens

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Abstract: One of the primary missions of the International Fusion Material Irradiation Facility (IFMIF) will be to generate a material database for the design components. The IFMIF irradiation volume is quite limited so that the use of small specimens is required. In this paper we present recent progress made to extract the physical processes that govern the plastic flow and fracture properties on the EUROFER97 tempered martensitic steel. The fracture toughness data were obtained on sub-sized specimens in the brittle regime. As expected, the fracture toughness data obtained on these sub-sized specimens are size-dependent. We present a correction technique to account for this effect. The plastic flow was measured with non-standard punch tests that were supplemented with a finite element model (FEM). In order to simulate the punch curves of the specimens, an inverse methodology is presented, where the constitutive behavior assigned to the material in the FEM is adjusted until reconstruction of the experimental curve. We show that estimates of the yield stress and strain-hardening capacity deduced from the analysis of small ball punch tests are comparable to those obtained with standard methods.

FT/2-1Ra · From One-of-a-kind to 500,000 High Quality Ignition Targets Per Day

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Abstract: Over the last decade the US has made major investments in laser and Z pinch drivers for ICF with ignition as a major goal of the inertial confinement fusion (ICF) program. Advances in target fabrication are being achieved to reduce the risk associated with achieving ignition and to compensate for cost and schedule issues or lack of flexibility of the drivers. There is little doubt that ignition will be achieved on the NIF but it will be with one of a kind targets, with high target cost and shot rates measured in shots per day at best. To go from one of a kind inertial ignition to a power plant is an extrapolation of about 5 orders of magnitude in target production rate and target cost. We are working on demonstrating a credible pathway to a reliable, consistent, and economical target supply, a major part of establishing that Inertial Fusion Energy (IFE) is a viable energy source regardless of the driver technology.
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FT/2-1Rb · Development of Key Technologies in DPSSL System fpr Fast-ignition, Laser Fusion Reactor-FIREX, HALNA, and Protection of FInal Optics

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Abstract: Critical for the development of laser-fusion power plants is the production of a reliable, efficient, high-frequency energy driver designed to be appropriate for the reactor environment. At the Institute of Laser Engineering (ILE), Osaka University, elemental studies are being carried out on energy drivers for this use. A diode-laser-pumped, solid-state laser (DPSSL), HALNA-10, has been operated successfully at 5 J output power and 10 Hz repetition rate. Contamination of the final optics by metal vapor was studied using a 1:10 model of the beam duct. The results indicated that contamination can be controlled with high- speed shutters and a low-pressure buffer gas.

FT/3-1Ra · The Spherical Tokamak as a Components Test Facility

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Abstract: There would be significant benefits for the development of fusion if a volume neutron source were available (in addition to IFMIF) for testing large scale structures and components in a fusion environment. This paper explores the possibility that a spherical tokamak with a small major radius, R=0.75m, and aspect ratio, A=1.6, could provide a components testing facility (CTF). Such a compact device could provide the necessary neutron fluence with a sufficiently low tritium consumption (1kg per year) that it would not be reliant on its ability to breed tritium. We present detailed calculations of the neutral beam current drive efficiency, showing that the device could be sustained in steady state with ∼60MW of power (ECRH is also an option) with a modest enhancement ∼30% above the prediction of the IPB98(y,2) confinement scaling law. High current (8MA) and high field (2.8T) provide the conditions for 50MW fusion power (in DT). The normalised beta is held at a modest value of 3.5, which is routinely achieved on MAST, and stable solutions have been identified by ideal MHD stability analyses. Exhaust power loadings on target plates are maintained at $\sim 10 - 15MW/m^2$ by employing a system of cascading SiC pebbles which form a curtain in front of the outer target plates of the double null divertor. These remove 3/4 of the power in the divertor. Detailed neutronics analyses predict a peak neutron wall loading for unscattered neutrons of $1.6MW/m^2$(mid-plane modules, with $6m^2$ for testing) and $1.4MW/m^2$ (above and below mid-plane modules, with a further $6m^2$ for testing). A neutron fluence of $6MWyr/m^2$ could be reached within 10 yrs provided an availability of 40% can be achieved. Such a CTF would provide DEMO with flexible support for materials and components testing for optimising the in-vessel power plant designs, especially in accelerating the build-up of blanket reliability. However, CTF could also be operated earlier in pure D plasmas, providing valuable experience for operating high performance tokamak plasmas in steady state with high exhaust powers. This would be important for optimising operating regimes for DEMO, as well as for the subsequent operation of CTF as a test facility.

FT/3-1Rb · Physics and Engineering Assessments of Spherical Torus Component Test Facility

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Abstract: The results of a broadly based study of the physics and engineering characteristics of the Component Test Facility (CTF) using the Spherical Torus or Spherical Tokamak (ST) configuration are presented. The required testing capabilities of the CTF of high fusion neutron fluxes WL of $> 1MW/m^2$, large total testing area of $> 10m^2$, and intense testing fluence of $> 0.3MW - yr/m^2$ per year are found to set lower bounds on the CTF size. Testing of tritium self-sufficiency further pushes the aspect ratio toward 1.4. A typical CTF design is characterized by R = 1.2 m, A = 1.5, elongation = 3, Ip = 10 MA, BT = 2.5 T, producing a fusion power of 77 MW and WL of 1 MW/m2, assuming moderate normalized ST plasma parameters achievable without active feedback control of MHD modes. Assumption of the advanced physics regime with MHD mode stabilization would enable $WL = 4MW/m^2$ for testing at the level of demonstration power plants. The ST CTF device requires the use of a single-turn normal conducting center leg for the toroidal field coil without the induction solenoid and substantial neutron shielding. A current start-up RF power of 5 – 10 MW, and a ramp-up and sustainment RF and NBI power of 40 MW are estimated, based on latest data. A new systems code that combines the key physics and engineering requirements, limits, and performance of CTF are prepared and utilized as part of this study. The results show a high potential for a family of CTF devices to suit a variety of fusion nuclear testing and R&D missions.
* The work is supported in part by DOE Contract Nos. DE-AC02-76CH03073 and DE-AC05-96OR22464.

FT/3-2 · Result of the KSTAR Superconducting Coil Tests

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Abstract: The KSTAR (Korea Superconducting Tokamak Advanced Research) device is under construction using superconducting (SC) coils for long pulse operation. The KSTAR superconducting magnet system consists of 16 D-shaped toroidal field (TF) coils, 3 pairs of poloidal field (PF) coils, and 4 pairs of central solenoid (CS) coils. A prototype TF coil, TF00 coil, has been fabricated in the same size as the real coils using Nb3Sn SC cable-in-conduit conductor (CICC). The coil was installed and tested in a coil test facility in the Korea Basic Science Institute (KBSI). Major objectives of the test were (i) to prepare the SC coil test facility and to obtain operational experience of the test facility, (ii) to cool the coil down to operating temperature and to find any defects in the coil such as helium leaks, and (iii) to measure the thermo-hydraulic parameters during the current charge and discharge scenarios. The coil was cooled down to the operating temperature of 4.5K in 10 days and the SC phase transition was found at about 18 K. No noticeable defects of the coil were found such as helium leaks at cryogenic temperature, and helium circulation through the coil was sufficient to keep the coil at the operating temperature. The coil was charged in steps and this was followed by slow or fast discharges. The transient hydraulic parameters were measured according to the scenarios. The current test of the coil was stopped at 33.3 kA due to an insulation failure in the current feeder system. To verify the operational feasibility of the KSTAR CS coils, a pair of CS model coils has been fabricated and will be tested by the middle of this year. In this paper, the experimental results of the TF00 coil test and the preparation of the CS model coil test are presented.

FT/3-3 · Progress of the EAST project in China

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Abstract: The Experimental Advanced Superconducting Tokamak (EAST) project is one of the National Mega-Projects of Science Research (MPSR) in China, which was approved by Chinese government in 1997. The name of the project was changed from HT-7U (Hefei Tokamak 7 Upgrade) to EAST last October. EAST is a full superconducting tokamak with the elongated cross-section.The main parameters of EAST tokamak (for the first phase) are $B_t = 3.5T$, $R_0 = 1.75m$,$I_p = 1MA$, $a = 0.4m$, $\kappa(b/a) = 1 \sim 2$ with

double and single null divertor, $P_{LHCD} = 3.5 - 4MW$, $P_{ICRH} = 3 - 4MW$, $P_{ECRH} = 0.5MW$ and the maximum pulse long will be 1000 seconds[1]. If the working temperature can achieve at 3.8 K the second phase will be with $B_t = 4.0T$ and $I_p = 1.5MA$. The mission of EAST project is to investigate both the physics and technology for the steady-state and advanced tokamak as well as the power and particle handling under the steady-state operation condition. Since the EAST project was approved in 1998 the significant progress has been achieved including: the physical and engineering design; the most of R&D activities; cabling and jacking of CICC; vacuum vessel, thermal shield and cryostat; most of TF, CS and PF magnet and their testing; most of other subsystem. The final assembly of EAST is under way. It is plane to get the first plasma around end of 2005. The detail information of design and the testing results of TF, CS and large PF magnets as well as the experimental plane [4] will be given in this paper.

FT/3-4Ra · First experiments with SST-1 Tokamak

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Abstract: SST-1, a steady state superconducting tokamak, is at advanced stage of erection at the Institute for Plasma Research. The objectives of SST-1 include studying the physics of the plasma processes in tokamak under steady state conditions and learning technologies related to the steady state operation of the tokamak. These studies are expected to contribute to the tokamak physics database for very long pulse operations. The SST-1 tokamak is a large aspect ratio tokamak, configured to run double null diverted plasmas with significant elongation and triangularity. The machine has a major radius of 1.1 m, minor radius of 0.20 m, a toroidal field of 3.0 T at plasma center and a plasma current of 220 kA. Hydrogen gas will be used and plasma discharge duration will be 1000 s. Superconducting (SC) magnets are deployed for both the toroidal and poloidal field coils in SST-1. An Ohmic transformer is provided for plasma breakdown and initial current ramp up. SST-1 deploys a fully welded ultra high vacuum vessel, made up of 16 vessel sectors having ports and 16 rings with D- shaped cross-section, which are welded in-situ during the SST-1 assembly. Liquid nitrogen cooled radiation shield are deployed between the vacuum vessel and SC magnets as well as Sc magnets and cryostat, to minimize the radiation losses at the Sc magnets. In SST-1 tokamak, the auxiliary current drive will be based on 1.0 MW of Lower Hybrid current drive (LHCD) at 3.7 GHz. Auxiliary heating systems include 1 MW of Ion Cyclotron Resonance Frequency system (ICRF) at 22 MHz to 91 MHz, 0.2 MW of Electron Cyclotron Resonance heating at 84 GHz and a Neutral Beam Injection (NBI) system with peak power of 0.8 MW (at 80 keV) with variable beam energy in range of 10-80 keV. The ICRF system would also be used for initial breakdown and wall conditioning experiments. The assembly of the SST-1 tokamak is nearing completion. The cool down of the Superconducting magnets is scheduled to start by middle of year 2004. The SC magnets will be tested for current ramp up and stability. First experiments related to the breakdown and the current ramp up will subsequently be carried out. Results of the magnet cool down, testing of the Magnets and first experiments in SST-1 will be presented.

FT/3-4Rb · Superconducting Magnets of SST-1 Tokamak

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Abstract: The tokamak SST-1, at the Institute for Plasma Research, is the first tokamak that deploys superconducting magnets for both the toroidal field (TF) and poloidal field (PF) coils. Sixteen superconducting D-shaped Toroidal Field (TF) coils generate a field of 3.0 T at the major radius of 1.1 m, whereas nine superconducting Poloidal Field (PF) coils together with a pair of resistive PF coils inside the vacuum vessel provide various reference plasma equilibria envisaged in SST-1 operational scenarios. Each of the TF winding pack, consisting of 6 double pancakes, is shrink-fitted into a stainless steel case. Supercritical helium is fed from the high field region in the middle of each of the double pancake over a hydraulic path length of 47 m with an inlet pressure of 4 bar and temperature of 4.5 K. A 1 kW class helium refrigerator equipped with a cold circulation pump provides a nominal flow of 300 g/s at the rated parameter for magnet operation. The refrigerator has suitable cool-down, warm-up and quench bypass modes in its operation to accommodate both the normal and off-normal scenarios of the SST-1 magnet operation. Voltage taps across joints and termination locations are used for quench detection. The quench detection system ensures fail proof quench detection based on subtraction logic. Flow meters at the inlet of the magnets, temperature sensors at the critical locations and hall probes for field direction and magnitude measurement are the other sensors. A 20 V, 10 kA power supply will excite the TF magnets whereas the PF power supplies have voltages from few volts to more than 100 volts, to cater to the fast current ramp-up of the PF magnets during start-up scenarios. The quench protection of the magnets consists of dump resistances

in parallel with mechanical DC circuit breaker, connected in series with respective magnets. The design and fabrication of the magnets along with the description of power supplies & protection system for the magnets will be presented. The cooling scenario for the magnets will be discussed.

FT/3-5 · Physics, technologies and status of the Wendelstein 7-X device

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Abstract: Stellarators are developed because of their intrinsic characteristics to allow steady state operation. The last stellarator of the Wendelstein line, W7-AS, showed excellent performance at operational limits. Its successor, the sc stellarator Wendelstein 7-X is presently under construction. It is a fully optimised stellarator based on the concept of quasi- isodynamicity. The magnetic system of W7-X comprises 50 non-planar and 20 planar coils, which provide experimental flexibility. All plasma facing components are designed for continuous heating with 10 MW of ECRH. W7-X is at the transition to assembly now. By spring of 2004 more than 20 non-planar and 16 planar winding packages have been produced. The results of the low-temperature test will be reported. Two 1 MW, 140 GHz gyrotrons are assembled; the test results will be presented. An overview on the status of the project will be given. The mechanical structure of W7-X must withstand the high electromagnetic loads of the coil system. The implications of the more generic technical solutions will be presented. Specific attention is given to the accuracy of the components. The requirements on accuracy and reproducibility are met. Concepts to compensate undue errors will be discussed. The ITER relevance of the technologies employed in W7-X will be analysed and presented.

FT/3-6 · Improved Structure and Long-life Blanket Concepts for Heliotron Reactors

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Abstract: New design approaches are proposed for the LHD-type heliotron D-T demo- reactor FFHR2 to solve the key engineering issues of blanket space limitation and replacement difficulty. A major radius over 14m is selected to permit a blanket-shield thickness of about 1m and to reduce the neutron wall loading and toroidal field, while achieving an acceptable cost of electricity COE. Two sets of optimization are successfully carried out. One is to reduce the magnetic hoop force on the helical coil support structures by adjustment of the helical winding coil pitch parameter and the poloidal coil design, which facilitates expansion of the maintenance ports. The other is a long-life blanket concept using carbon armor tiles that soften the neutron energy spectrum incident on the self-cooled Flibe- RAF blanket. In this adaptation of the Spectral-shifter and Tritium breeder Blanket (STB) concept a local tritium breeding ratio TBR over 1.3 is feasible by optimized arrangement of the neutron multiplier Be in the carbon tiles, and the radiation shielding of the super-conducting magnet coils is also significantly improved. The key R&D issues to develop the STB concept, such as radiation effects on carbon and enhanced heat transfer of Flibe, are elucidated.

FT/P1-6 · Development of advanced superconducting coil technologies for the National Centralized Tokamak

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Abstract: Advanced technologies of fabricating superconducting coils for the National Centralized Tokamak (NCT) which is based on a modification of JT-60 have been developed. One of the developed technologies is the application of the react-and-wind (R&W) method to the fabrication of a Nb_3Al D-shaped coil. In order to form the D-shape that simulates the actual toroidal field (TF) coil curvatures, the winding technique of the conductor including "over bending" was developed. The energizing tests of the D-shaped coil were successfully carried out in 2003. We found that the bending strain due to the R&W method did not affect the critical current (Ic) characteristics of the conductor. This finding indicates the possibilities that manufacturing cost of large size coils can be reduced further by downsizing the heat treatment furnace and complex shape coils such as a helical type can be manufactured by using the Nb_3Al conductor. Another technology is an advanced winding technique, in which bending strains are loaded on the conductor, for the reduction of the AC losses of a Nb_3Sn coil, which is generally made by the wind-and-react (W&R)

method. Effectiveness of this technique was experimentally demonstrated. Other technology, such as a high copper/non-copper (Cu/non-Cu) ratio NbTi strand with fine filaments, is also developed.

FT/P1-7 · Development of advanced Nb3Al superconductors for a fusion demo plant

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Abstract: Nb_3Al superconductor has inherently outstanding features of larger critical current, higher critical magnetic field and excellent strain tolerance compared to Nb_3Sn. These features can provide a possibility to realize a fusion magnet that can be operated at around 16 T, higher than 13 T obtained by Nb_3Sn. Japan Atomic Energy Research Institute (JAERI) has therefore started the development of Nb_3Al conductor from the middle of 1980s, aiming at its application to the TF coil of a fusion demo plant, which will preferably be operated at more than 16 T. As an intermediate achievement of this development, the Nb_3Al conductor using a conventional jelly-roll method was developed, and a large-scale Nb3Al coil was successfully manufactured and operated at the rated field and current of 13 T and 46 kA, respectively. Recently, the National Institute of Material Science (NIMS) has successfully developed Nb3Al strand with a Rapid-Heating, Quenching and Transformation (RHQT) method, which has much higher performance than those obtained by the conventional jelly- roll method. The conductor of fusion magnets generally requires sufficient amount of copper for stabilization against perturbation. However, RHQT Nb3Al requires heat treatment at a very high temperature, which is more than the melting temperature of copper, 1083 °C, and followed by transformation annealing at about 750 °C. Therefore, the amount of copper to be included in the strand is very limited in this process. As an idea to overcome this issue, a method to supplement enough copper before the transformation annealing is proposed. A numerical analysis shows that the supplement of copper is significantly effective for stabilization. Thus, technologies at JAERI and NIMS are now being combined to develop advanced Nb_3Al conductor for a fusion demo plant.

FT/P1-8 · Evaluation of Tritium Breeding and Irradiation Damage for the EU Water Cooled Lithium-Lead Blanket Module in ITER-FEAT

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Abstract: Comprehensive neutronic analyses have been carried out for the EU WCLL TBM integrated in ITER-FEAT to assess tritium generation and parameters relevant to lifetime performance of TBM such as helium and atomic displacement production. The analyses have been performed utilizing models representing the complex ITER and TBM structure close to reality. The Monte Carlo transport code MCNP-4C and cross-sections from FENDL-2.0 data library have been used in the analyses. Theoretical estimates of radial distribution of tritium production density, total tritium production rate, radial distribution of helium and displacement formation along the TBM, poloidal distribution of helium and displacement generation in de-mountable hydraulic connections of TBM have been obtained.

FT/P1-9 · Key achievements in elementary R&Ds on water-cooled solid breeder blanket for ITER Test Blanket Module in JAERI

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Abstract: Development of key elementary technologies on the water-cooled solid breeder blanket for the ITER test blanket modules (TBMs) has widely been progressed in JAERI. On the basis of the established key elementary technologies, blanket development can be shifted into a new stage, where integrated performances of TBM structures will be demonstrated by scalable mockups of ITER TBMs. Demonstration of blanket functions under fusion environments, e.g., by installing TBMs in ITER, is considered as one of the most important milestones. This paper describes the significant R&D progress on the most essential issues for ITER TBMs, development of module fabrication technology, bonding technology of armors, measurement of thermo-mechanical properties of pebble beds, neutronics studies on a blanket module mockup, and tritium release behavior from Li_2TiO_3 pebble bed under neutron pulsed operation condition. By the improvement of heat treatment process for blanket module fabrication, a fine-grained microstructure of F82H, can be obtained by homogenizing at 1150 °C followed by normalizing at 930 °C after the Hot Isostatic Pressing (HIP) process. Moreover, a promising joining process for a tungsten armor and an F82H structural material was found by using a solid state bonding method based on uniaxial hot compression without any artificial compliant layer. As a result of high heat flux tests of F82H first wall mockups, it

was found that the thermal fatigue lifetime of F82H can be predicted by using Manson-Coffin's law. As a result of R&Ds on a breeder material, Li_2TiO_3, following results were obtained. Effective thermal conductivity of Li_2TiO_3 pebble was measured under compressive force simulating the ITER TBM environment. The increase of the effective thermal conductivity of the pebble bed was about 2.5 % at the compressive strain of 1 % at 400 °C. Neutronic performance of the blanket module mockup has been carried out by the 14 MeV neutron irradiation. It was confirmed that the measured tritium production rate agreed with the calculated values within about 10 % difference. Also, tritium release from the Li_2TiO_3 pebble was measured under neutron pulsed operation condition simulating ITER. It was observed that the ratio of tritium release rate to generation rate at the high power approached the saturated value of unity with a time constant of about 3 h.

FT/P1-10 · Operation problems of the blanket zone materials in an intense magnetic field

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Abstract: Tritium breeding materials of the blanket zone (lithium-containing ceramic and beryllium pebbles) in the fusion reactors (ITER, DEMO) will operate during a long term (up to 20,000 hours) not only at high temperature (up to 1123 K), intense radiation fluxes (up to $10^{18} nm^2 s^{-1}$) but also under intense magnetic fields (6-9 T). The experimental testing of the simultaneous action of these factors under the real operating conditions is not possible at present. Magnetic field can affect the radiolysis of ceramic materials, the tritium release, the mutual corrosion between the ceramic and structural materials, to cause the electrical polarisation of ceramic materials at the moments of the plasma ignition and decay etc. The influence of the magnetic field (2.4 T) on the radiolysis and the tritium release from the lithium orthosilicate and metatitanate ceramics is investigated. The possible electrolysis of the ceramic material as the result of the rapidly changing magnetic field is estimated as well. The mutual corrosion of EUROFER-97 with lithium orthosilicate or titanate increases under the simultaneous action of high temperature, radiation and magnetic field.

FT/P1-11 · Development of fabrication technology for low activation vanadium alloys as fusion blanket structural materials

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Abstract: Vanadium alloys are promising candidates for fusion blanket structural materials, because of their low activation property, high temperature strength, high resistance to neutron irradiation and good compatibility with liquid lithium. It has been reported that large scale V_4Cr_4Ti alloy ingots designated as NIFS-HEAT were melted in a collaboration program between the National Institute for Fusion Science, Japanese universities, and industries. In the melting, the level of interstitial impurities, such as C, N and O, was reduced to improve mechanical and irradiation properties. In the present study, high purity vanadium alloy products, such as plates, wires, tubes and weld joints, were fabricated by using the technologies applicable to industrial scale fabrication. During the breakdown and the welding processes, the impurity behavior and its effect on mechanical properties were investigated. In the plate fabrication process, which consists of hot working and subsequent cold rolling into 0.25-26 mm-thick plate, no contamination with C, N, or O was detected, as a result of canning at hot working, management of surface cleanness and use of Ta or Nb getter at annealing in a vacuum. Impact properties of the plate were improved by an increase in the degree of cold rolling. It was found that clusters of Ti-C, N, O precipitates were produced during the hot working process. The clusters of the precipitates were redistributed into precipitate bands aligned to the rolling direction with an increase in the degree of cold working. The clusters of the precipitates are thought to induce the brittle fracture, whereas the precipitate bands blocked the primary crack propagation by producing secondary cracking along the bands. In the case of vanadium alloys, it was revealed that the transition from precipitate clusters to bands by sufficient working is a critical technology for high ductility. In tubing and welding processes, behavior of the precipitate bands, their dissolution and re-precipitation was also investigated. It was concluded that the mechanical properties of the products were improved significantly not only by reducing the impurity levels of C, N and O, but also by controlling density, size and distribution of the Ti-C, N, O precipitates. Thus the large scale fabrication of vanadium alloy products is considered feasible with appropriate management of the process.

FT/P1-12 · Basic Principles of Lead and Lead-Bismuth Eutectic Application in Blanket of Fusion Reactors

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Abstract: One of the main requirements of advanced nuclear-power engineering is inherent safety of power installations. It initiates R&D of heavy liquid metals (lead, lead- bismuth eutectic) application in fission reactors as substitute of sodium. The same requirement makes advisable R&D of the lead and lead-bismuth eutectic application in blanket of fusion reactors as substitute of lithium. High magnetohydrodynamic pressure drop is an important issue for liquid metal blanket concepts. To decrease MHD-resistance authors propose to form electroinsulating coatings on internal surface of blanket ducts at any moment of fusion reactor exploitation. It may be achieved easily if lead or lead-bismuth eutectic is used and technology of oxidative potential handling is applied. A number of experiments carried out in NNSTU show the availability of the proposed technology. It bases on formation of the insulating coatings that consist of the oxides of components of the structural materials and of the coolant components. In-situ value of the electroinsulating coatings characteristics r_d (r – specific resistance of coatings, d –thickness) is $\sim 10^{-5}\Omega\cdot m^2$ for steels and $5,0 \times 10^{-6} - 5,0 \times 10^{-5}\Omega \cdot m^2$ for vanadium alloys. Thermal cycling is possible during exploitation of a blanket. The experimental research of the insulating coatings properties during thermal cycling have shown that the coatings formed into the lead and lead-bismuth coolants save there electroinsulating properties. Experience of many years is an undoubted advantage of the lead-bismuth coolant and less of the lead coolant in comparison with lithium. Russian Federation possesses of experience of exploitation of the research and industrial facilities, of experience of creation of the pumps, steamgenerators and another equipment with heavy liquid metal coolants. The unique experience of designing, assembling and exploitation of the fission reactors with lead-bismuth coolant is also available. The problem of technology of lead and lead-bismuth coolants for power hightemperature radioactive facilities has been solved. Accidents, emergency situations such as leakage of steamgenerators or depressurization of gas system in facilities with lead and lead-bismuth coolants have been explored and suppressed.

FT/P1-13 · Experimental Studies on Tungsten-armor Impact on Nuclear Responses of Solid Breeding Blanket

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Abstract: In fusion DEMO reactors, the blanket is required to provide a tritium breeding ratio (TBR) of more than unity by neutron induced reactions in lithium in the blanket. The solid breeder blanket being developed by JAERI for tokamak-type DEMO reactors consists of Li_2TiO_3 or other lithium ceramics, beryllium, reduced activation ferritic steel F82H and water. In the blanket design proposed by JAERI, the local TBR is around 1.4 - 1.5 for the case without an armor on the first wall. From the view point of sputtering damage of the first wall, tungsten (W) armors are the effective protection. However, from the view point of nuclear responses, W armors may reduce the TBR, and TBR might not satisfy the requirement in case. In order to evaluate this issue experimentally, neutronics experiments have been performed by using DT neutrons at Fusion Neutron Source (FNS) facility of JAERI. A breeding blanket mockup, composed of a set of slabs of 16 mm thick F82H, 12 mm thick Li_2TiO_3 and 200 mm thick Be with about 660 mm height and about 660 mm width each, was installed at about 450 mm distant from the DT neutron source. In the experiments three types of mockups were tested: without the armor; with 12.6 mm thick W armor; and with 25.2 mm thick W armor. Metal foils of Nb, Al, Au, were installed at the boundary surfaces inside the mockup. After the irradiation, induced radioactivities were measured by decay gamma ray intensity with a high purity Ge detector, and neutron fluxes corresponding to each reaction rate were evaluated. Fifteen slices of Li_2CO_3 pellets were embedded inside the Li_2TiO_3. After the irradiation, induced radioactivities were measured by beta ray intensity of these pellets with a liquid scintillation counter, and tritium production rate (TPR) was evaluated. By installing the 25.2 and 12.6 mm thick W armors, the TPRs were reduced by 19 and 18 %, respectively, in maximum, and the integrated TPRs in the Li2TiO3 were reduced by 8 and 3 % relative to the case without the armor, respectively. Numerical analyses were conducted by using the Monte Carlo neutral particle transport code. The calculation results almost showed the same tendency as the experiment ones with the C/E ratios below 1.15 for the integrated TPRs. In the blanket design proposed by JAERI, it is expected that the reduction of the TBR is less than 2 % as the thickness of the W armor is less than 5 mm.

FT/P1-15 · Testing on ISTTOK of the liquid metal limiter concept

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Abstract: The application of liquid metals to fusion research has suffered a renewed interest due to the possibility offered by these materials in solving the problem of power exhaustion from large size devices. Among several candidates Lithium is the one that has received the main attention due to its high compatibility with plasmas (low Z). Another liquid metal that has very good thermal properties (low fusion and high ebullition temperatures) is Gallium. In tokamak ISTTOK this element has been chosen in order to investigate the behavior of a thin (2 mm diameter) liquid metal jet interacting with plasmas. This jet will be placed in the vicinity of an already existing moveable stainless steel limiter to allow for a controlled exposure of Gallium to the plasma. In this paper we present the technical description of this project and some of the results achieved both in an experimental rig for a transient magnetic field influence on the jet and in the tokamak.

FT/P1-16 · Steady state hydrogen plasma interactions with solid and liquid lithium

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Abstract: It is widely recognized in the magnetic fusion community that high performance confinement plasmas often favor low edge recycling conditions. Therefore, wall conditioning such as boronization has routinely been conducted in many confinement devices. Unfortunately, however, due to the surface saturation, the efficacy of boronization has finite lifetime, necessitating re-conditioning. This points to a need for enabling wall concept development to maintain reduced recycling even at steady state. As a possible resolution, the concept of moving-surface plasma-facing component (MS-PFC) was proposed while ago. Recently, a laboratory-scale proof-of-principle (PoP) experiment has successfully been conducted, employing a continuously Ti- or Li-gettered rotating drum exposed to steady state hydrogen plasmas, and the results indicate that wall recycling reduced down to 90∼95% at steady state [2, 3]. Following these successful experiments, our interest has now been directed to moving liquid surfaces, namely, the "Liquid waterfall wall concept". In the present work, steady state hydrogen plasma interactions with solid and lithium have been observed in a dedicated facility: Vehicle-1 (for the Vertical and Horizontal configurations Interchangeable test-stand for Components and Liquids for fusion Experiments). The Vehicle-1 facility employs a 1kW ECR plasma source and produces hydrogen plasmas with densities of the order of $10^{10} cm-3$ and electron temperatures around 3eV. Also, Vehicle-1 is equipped with a scanning Langmuir probe, CCD video camera, multi-channel optical spectrometer, IR-pyrometer, and thermocouples for well diagnosed experiments. Sticking coefficients measured for as-received solid lithium are 1×10^{-5} and 3×10^{-1}, respectively, for hydrogen molecules and hydrogen plasma particles. It is also found that helium plasma bombardment increases the molecular hydrogen sticking coefficient up to 0.03, indicating the effect of removal of surface oxides. When it is liquidized at elevated temperatures, lithium has been found to trap almost all incoming hydrogen plasma particles until bulk saturation occurs. The break found in the Arrhenius plots of reciprocal hydrogen recycling time constant data suggests that the effective diffusivity of hydrogen penetration into lithium changes at temperatures around 300°C.

FT/P1-17 · Blister and Bubble Formation Mechanism on Tungsten Irradiated by High Flux Hydrogen/Deuterium and Helium Plasmas

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Abstract: We report comprehensive study on blister and bubble formation on tungsten (W) exposed by hydrogen isotope (H, D, T) and/or helium (He) ion with an ion energy below 100 eV in the linear divertor plasma simulators, NAGDIS-I and II. Many blisters with a diameter above 100 micron are formed on a powder metallurgy tungsten (PM-W) sample irradiated by H ions at the surface temperature below 950 K and at the ion fluence above $10^{25}/m^2$. A stratification structure with a layer thickness of 1micron is seen in the PM-W. The blistering occurs by cleaving PM-W along the stratified layers, which could be made while manufacturing process. On the other hand, there are no large blisters observed in a single crystal W and re-crystallized PM-W in the same experimental condition, which have much fewer intrinsic defects than the PM-W. We also used PM-W rolled with a much higher processing temperature, which also leads to no blister formation. These experimental results indicate that the appropriate manufacturing method of tungsten is one of the key factors to avoid the blister formation due to H plasma irradiation. In a case

of He ion irradiation, many bubbles with a micro-meter diameter are observed in the W at the surface temperature above 1600K . It is also found that a threshold value of incident He ion energy for the bubble formation is around 5eV. The micron-sized He bubbles are formed by following mechanism: (1) He ions, whose incident energy exceeds the surface barrier potential of W to He (theoretical calculation predicts 6 eV), can penetrate into W bulk, (2) He atoms diffused in W can be trapped in the thermal vacancies, which could be dominating trap sites of He atoms for the bubble nucleation at high W surface temperature above 1600 K, (3) thermal vacancies with He atoms aggregate to form bubbles. Thermal desorption spectra of deuterium released from W with and without He bubbles indicates that deuterium retention is strongly influenced by the He bubble formation because the many bubbles near W surface can contain hydrogen gases.

FT/P1-18 · Studies On High Energy Proton Degradation In Optical Transmission Materials

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Abstract: We studied at the Bucharest TANDEM accelerator the 3 and 12.6 MeV proton irradiation-induced modifications on ultraviolet transmission properties on KU1 quartz glass samples. The optical transmission properties (absorption, transmission and reflectivity) in the UV region have been measured with a Cary 4 VARIAN spectrophotometer. For 3 MeV, absorption peaks at 215 nm and 240 nm, similar in shape, but smaller in intensity to gamma irradiation case, can be observed and a supplementary 202 peak appeared, splitting the 215 nm peak. For 12 MeV, we observed the presence of 215 nm peak (due to both electron and nuclear collisions stopping) and a big reduction of 240 nm peak (due only to nuclear collisions). We evaluated the dose rate of our 12.6 MeV proton irradiations at 200 Gy/s and the total irradiation doses at 10 and 20 MGy. Comparing our spectra (mainly the intensity of 215 nm peak) with the results for gamma and high energy electron irradiations, we can conclude for the 12.6 MeV proton irradiation that the saturation effect in absorption is obtained after a 10 MGy dose, as compared with 4-5 MGy for gamma and with 11-12 MGy for electrons, suggesting the ionization process is essential for defect absorption centers in all the cases.

FT/P1-19 · In-vessel Tritium Inventory in ITER Evaluated by Deuterium Retention of Carbon Dust

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Abstract: The evaluation of in-vessel tritium inventory is one of important issues in ITER. Systematic measurement on fuel hydrogen retention of carbon dust, however, has not been conducted so far, although it is suggested that the carbon dust may have a largest tritium inventory. In this study, the measurement on deuterium retention of carbon dust prepared by electron beam evaporation of graphite was carried out after the exposure to deuterium gases and deuterium ions, using a technique of thermal desorption. The co-deposited carbon dust was also prepared in the deuterium arc discharge device with carbon electrodes. The deuterium retention due to the gas absorption was observed to be very small. The deuterium retention due to the ion irradiation was roughly the same as the case of graphite. The deuterium concentration of co-deposited carbon dust increased with the discharge pressure and decreased with increase of substrate temperature. Compared with the case of the ion irradiation, the deuterium concentration of co-deposited carbon dust was higher when the gas pressure was higher than 1Pa and the substrate temperature higher than 573K. In the ITER condition, 1Pa and 573K, the deuterium concentration of co-deposited carbon dust in atomic ratio was D/C=0.2, which is comparable with that of the ion irradiation. In the DT discharge, the tritium concentration becomes T/C=0.1.This value is several times lower than the value estimated so far. In ITER, the in-vessel tritium inventory is limited below several hundreds grams. When the inventory exceeds this value, the periodic cleaning of dust is required. If such the cleaning has to be conducted often, the plasma discharge number is limited. The present result suggest that the period may be several times lengthened compared with that estimated so far. References [1]T. Hino et al, Fusion Eng. and Design, 61-62(2003)605. [2]H. Yoshida, T. Hino et al, Fusion Sci. Technol., 41 (2002)943.

FT/P1-20 · Dynamic erosion of plasma facing materials under ITER relevant thermal shock loads in electron beam facility JUDITH

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Abstract: Thermal shock loads in the order of several $10MJ/m^2$ with duration of a few ms (disruptions) are predicted in ITER. Recent studies show a strong erosion of carbon based materials due to particle release caused by brittle destruction (BD) during thermal shock loads. Macroscopic erosion such as BD is associated with a substantial material loss because the released particles are not re-deposited on surfaces but create directly dusts. Moreover, brittle materials such as tungsten are also concerned in terms of material loss due to BD. In the present study, dynamic erosion under intense thermal loads was studied by optical diagnostics. The thermal shock loads were performed in the electron beam facility, JUDITH. The incident power density can reach values up to $15GW/m^2$. In addition to a pyrometer, a photodiode array (PDA) and photodiodes were newly installed to observe the dynamic erosion. As the first step of experiments, fine grain graphite and CFC were loaded by the electron beam. The PDA detected thermally radiating particles due to brittle destruction. The detected signals from graphite consist of continuum and sharp peaks, which are corresponding to "small particles" and "large particles". These particles originated from the graphitized binder phase and gain clusters in graphite, respectively. On the other hand, such different particle species were not distinguished clearly in CFC. The observation volumes of the PDA were positioned at 52 and 69 mm above the surface. By using the distance between the two volumes and the time lag, the speeds of the particle were estimated. The mean speed of small particles from graphite was determined to be about 300m/s at $2.4GW/m^2$, increasing with decreasing power density. The released speed may be related to the particle size and the maximum speed of large particles was estimated to be 330 m/s at these conditions. Time-evolution of line emission such as CII (589 nm) was detected by photodiodes with interference filters. It was shown that the large particles released from graphite were fast rotating and emitting carbon atoms from its surface. The features of the particles must be related to the release processes. Beside detailed analyses of material erosion of carbon based materials, erosion of tungsten and the alloy under ITER relevant thermal shock loads will be also discussed.

FT/P1-21 · Towards a Reduced Activation Structural Materials Database for Fusion Reactors

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Abstract: The development of First Wall, Blanket and Divertor materials which are capable of withstanding many years the high neutron and heat fluxes, is a critical path to fusion power. Therefore, the timely availability of a sound materials database has become an indispensable element in international fusion road maps. In order to provide a related materials database for design, construction, licensing and safe operation of the ITER Test Blanket Modules and of a DEMO reactor, a wealth of R&D results on the European reduced activation ferritic-martensitic steel EUROFER, and on oxide dispersion strengthened (ODS) variants have become available, mainly in the temperature window 250-700 °C. Industrial EUROFER-batches of 3.5 and 8.0 tons have been produced with a variety of semi-finished, quality-assured product forms. Extensive chipless shaping and joining experience taking into account different welding procedures and powder technology product forms have demonstrated that EUROFER type steel complies with a wide range of established manufacturing processes. EUROFER is also resistant to high temperature aging, and the existing creep-rupture properties (∼30000 h) indicated long term stability and predictability. To increase the thermal efficiency of blankets beyond 45%, high temperature resistant SiCf/SiC channel inserts for liquid metal coolant tubes are developed. Mechanical and thermal properties of various SiCf/SiC composits have been measured after neutron radiation. Regarding radiation damage resistance of blanket structural materials, a broad based reactor irradiation programme counts several steps from <5dpa (ITER TBMs) up to 75 dpa (DEMO). In addition to the blanket activities, a modular He gas cooled divertor is being developed within the European power plant conceptual study. As $\sim 10MW/m^2$ needs to be removed, the design is presently based on tiles made of W (∼2000 °C), as well as on structural materials like W-alloy (∼700-1300 °C) and RAF(M)-ODS steel (∼650°C). Severe plastic deformation of pure W and W alloys improves ductility, but does not prevent from re-crystallisation between 850 and 1200°C. For the European divertor designers, a materials data base including tensile, fatigue and long-term creep rupture properties is presently being prepared for pure W and W alloys, and related reactor irradiations start with temperatures from 650 -1000 °C.

FT/P1-22 · Integral Benchmark Experiments of the Japanese Evaluated Nuclear Data Library (JENDL)-3.3 for the Fusion Reactor Design

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Abstract: JENDL-3.3 is a neutron cross section data of 337 nuclei evaluated from the latest experimental data. JENDL-3.3 introduces double differential cross sections, which are energy and angle dependent ones of the scattered secondary neutrons, and are important for anisotropic neutron transport calculations for the fusion reactors design. This paper overviews benchmark experiments carried out for key fusion related nuclei such as Iron and Vanadium, and the results of analyses with JENDL-3.3, together with JENDL-3.2 and FENDL-2 for a comparison purpose. The experiments have been carried out at the Fusion Neutron Source (FNS) of JAERI. During the neutron injection into the assemblies, neutron and secondary gamma-ray spectra have been measured inside and outside the assemblies. For the test assemblies, we have used Iron, Copper, Vanadium and Tungsten as a single element material, and SS316L, $LiAlO_2$ and SiC as a compound material. From the integral benchmark experiments it was confirmed that the accuracy of JENDL-3.3 has been improved well compared with JENDL-3.2 and FENDL-2 by the re-evaluation using latest experimental data, and JENDL-3.3 is suitable for the nuclear analysis of the fusion reactor.

FT/P1-23 · Technologies for the ITER divertor vertical target Plasma Facing Components

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Abstract: The ITER divertor vertical target (VT) has to sustain heat fluxes up to 20 MW/m2. The concept developed for this Plasma facing Component (PFC) working at steady state is based on Carbon Fibre Composite (CFC) armor for the lower straight part and tungsten (W) for the curved upper part. The main challenges of such component are to be able to remove the high deposited heat fluxes and to join mechanically and thermally armor to the metallic heat sink, despite of the mismatch of the thermal expansions. Two solutions based on the use of CuCrZr hardened copper alloy and AMC process were investigated during the ITER EDA phase: the first one called "flat tile technology" was mainly developed for Tore Supra pumped limiter, the second one called "monoblock technology" was developed by Europe. This paper presents a review of these two technologies and analyses their assets and drawbacks: pressure drop, critical heat flux, surface temperature and expected behavior during operation, risks during the manufacture, controls of the armor defects during the manufacture and at the reception and possibility to repair defected tiles.

FT/P1-24 · Development of optical diagnostic system for burning plasma machine

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Abstract: A possibility of applying a compact optical system to diagnostics of the 14MeV neutrons has been studied, utilizing radiation resistant optical fibers and radio-luminescence (radiation-induced luminescence) materials sensitive to the 14MeV neutrons. Experimental results clearly showed that the proposed optical systems would satisfy severe demands in the ITER burning plasma diagnostics. Several candidate materials were attached at an end of a synthesized fused silica (SiO_2) made radiation resistant optical fiber whose core-diameter was 0.2mm and were exposed to the 14MeV neutrons and to gamma-rays from cobalt-60 sources. Three materials, silver activated zinc sulfide (ZnS;Ag), copper activated zinc sulfide (ZnS;Cu), and a strontium aluminate doped with europium and dysprosium ($Sr_xAl_yO_z;EuD_y$), were found to be radioluminescent, being sensitive to the 14MeV neutrons with a radioluminneescence peak position at 450nm, 570nm, and 500nm, respectively, having a half width of 75-150nm. The ZnS;Ag had the strongest luminescence among three but its intensity decreased with the increase of neutron fluence. In contrast, the other two, the ZnS;Cu and the $Sr_xAl_yO_z$;EuDy had relatively weak luminescence intensity but their intensity did not change substantially with the fast neutron fluence up to $10^{20}n/m^2s$. At present, the ZnS;Cu is evaluated to be the best as a sensing element for fusion neutron diagnostics. The ZnS;Cu in the present system can detect the fast neutron flux down to $10^{11}n/m^2s$ with an integration time of 10 s. However, it is estimated that the system can detect a fast neutron flux down to $10^9n/m^2s$ with an integration time less than 1 s, with a photomultiplier system working at a fixed optical wavelength region. A dimension of a sensing element will be in the order of 0.1mm. Thus, the present results showed that the optical fast neutron detecting system with a high space resolution can be composed with a radiation resistant optical fiber and a neutron sensitive ZnS;Cu element. Development of the optical fusion-neutron

diagnostic system, which is compact, robust and easy to install and dismantle in a limited space, will give a strong and positive impact in the development of a burning plasma machine such as the ITER.

FT/P1-25 · Investigation of radiation damage in copper using MCNP4C2 and TRIM98.01 simulation codes

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Abstract: The work is focused towards the radiation damage of the first wall material of International Thermonuclear Experimental Reactor (ITER). There were performed Monte Carlo simulations by MCNP4C2 and TRIM 98.01 codes to design the experimental part for Positron Annihilation Spectroscopy (PAS) measurements of neutron treated material of ITER first wall based on Cu-alloys. MC simulations by MCNP4C2 code have been performed to determine the distribution of neutron fluency and primary-knock on-atoms (PKA) creation and to predict the probable activation of Cu sample. The TRIM 98.01 code was used for the simulation of defects creation (vacancies, voids) in pure Cu by hydrogen H of 95 keV, only. The aim was to find out the distribution of protons in the defined depth. The main defects production was identified in the region of about 400-800 nm. Data obtained from computer simulations were compared with experimental data from PAS technique and are useful also for irradiation experiment preparing currently in FZ Rossendorf in the TUD neutron laboratory.

FT/P1-26 · Plasma jet source parameters optimisation and experiments on injection into Globus-M spherical tokamak

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Abstract: Results of theoretical and experimental research of plasma source and injection of plasma jet produced by modified source into tokamak Globus-M are presented. Experimental test stand was developed for investigation of intense plasma jet generation. Optimisation of pulsed coaxial accelerator parameters by means of analytical calculations is performed aiming to achieve the highest flow velocity at fixed coaxial electrode length. Optimal parameters of power supply to generate plasma jet with minimal impurity contamination and maximum flow velocity were determined. Comparison of experimental and calculation results is made. Plasma jet parameters are measured, such as: impurity species content, pressure distribution across the jet, flow velocity, plasma density and degree of ionisation. Experiments on interaction of higher density and kinetic energy plasma jet with magnetic field and plasma of the Globus-M tokamak were performed. Experimental results on plasma jet injection into different Globus-M discharge phases are presented and discussed. Results are presented on investigation of plasma jet injection as the source for discharge breakdown, plasma current start up and initial density rise.

FT/P1-27 · Recent Results in Plasma Facing Materials Studied at SWIP

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Abstract: Recently, it is focuses on that tungsten coating, innovation liquid metal free surface and multi-elements doped carbon base materials for plasma facing materials (PFMs) researches at Southwestern Institute of Physics (SWIP), China. Using B_4C, Ti and Si doped graphite and C/C fiber composite (CFC), tungsten coatings on carbon and copper substrates by IPS and VPS are developed, and their thermal physics properties are investigated by experiments with special devices and HL-M Tokamak. It is investigated that MHD stabilities of liquid metal free surface NaK jet in a gradient magnetic field on experiment and theory.

FT/P1-28 · Use of High Temperature Superconductors for Future Fusion Magnet Systems

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Abstract: With the construction of ITER the feasibility of a fusion machine will be demonstrated. For a commercial success of fusion it will be essential to reduce losses in future fusion power plants. One major component where losses can be reduced strongly is the cooling system. With a magnet system working at 20 K a fusion machine would be much more efficient on electric power consumption for cryogenics due

to a reduction by a factor of 5-10. And even better than that would be a machine with a magnet system operating at 65 K to 77 K. In this case liquid nitrogen could be used as coolant which saves money for investment and operation costs. Such an increase of the operating temperature of the magnet system can be achieved by the use of High-Temperature Superconductors (HTS). In parallel the use of HTS would allow a much lower effort for thermal shielding and alternative thermal insulation concepts may be possible, e.g. for an HTS bus bar system. This contribution will give an overview about status, promises and challenges of HTS conductors on the way to an HTS fusion magnet system beyond ITER.

FT/P1-29 · Oxidation of Beryllium - a Scanning Auger Investigation

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Abstract: Beryllium is considered a candidate material for the first wall of nuclear fusion plasma devices, e.g. ITER. For this application, the interaction of beryllium with oxygen is important for several reasons. One aspect is the reaction of hot beryllium with air in case of a catastrophic leak, the other is the action of beryllium as a getter, binding oxygen impurities and thus helping to keep the level of contamination in the plasma low. We therefore investigated the interaction of beryllium with air at elevated temperatures up to 600°C on a microscopic level, using a high resolution Auger electron microscope. At 390°C, a thin protective oxide film is formed, while at 500°C oxidation starts to enter into the grain boundaries, leading to the loosening of small particles of beryllium already at 600°C. This temperatures are considerably lower than the previously reported onset of catastrophic oxidation at 750°C. This could be due to our more sensitive methods of analysis or due to the higher impurity content of the plasma-sprayed and sintered samples we have used. The expected diffusion of oxygen from the surface into the bulk has not been observed up to 390°C, the highest temperature to be safely applied in UHV inside the Auger microscope. Thus, an operation of beryllium liners as a non-evaporable getter (NEG) is not to be expected in this temperature range. Getter activity linked to the transport of beryllium from the liner to some deposition areas is however possible. Carbon impurities included in the beryllium samples diffused to an atomically clean surface produced by sputtering already at room temperature. This points to a possible problem of carbon contamination of the plasma, when carbon containing beryllium material is used as a first wall.

FT/P1-30 · Advances in the Ignitor Program

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Abstract: Significant advances that have been made within the Ignitor program are described. On the physics side these involve the study of oscillatory states for the plasma pressure near ignition that can be obtained by both external and internal forms of control, a comparison of the confinement properties of plasmas with peaked density profiles produced in experiments relevant to Ignitor, and the identification of regimes with double X-point configurations in Ignitor. On the engineering side, we include the construction of second generation prototypes of the toroidal magnet plates, the completion of the design of a plasma chamber withstanding the estimated disruption forces, of the Mo first wall tiles and supporting plates, of the remote handling system, and other relevant R&D activities (i.e. construction of a fast pellet injector). All elements of the poloidal field system have been optimized. The analysis of the connection of Ignitor to a node of the European grid has been completed with positive results. The maximum current pulse has been optimized, to minimize the requirements of the machine electrical power supply system.

FT/P5-36 · Grain boundary sliding and migration in copper: Effect of vacancy

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Abstract: The atomic structure and mechanism for the interface sliding of the *f*Ã=5 symmetric tilt grain boundary (GB) in copper and their interaction with a vacancy at elevated temperature has been studied using a computationally efficient potential based on Embedding Atom Method in connection with finite - temperature Monte Carlo technique. Copper was chosen here because special copper alloys were selected as some good candidates for heat sink in the design of ITER Plasma Facing Components. We found that the grain boundary sliding energy profile in the presence of a vacancy placed at the interface increases the GB energy, but reduces the sliding energy. The sliding process invokes the interface migration in such way that vacancy effectively migrates at the more convenient position and reduces the GB energy. Finally, the sliding and migration properties of the GB partially depend on the position of the vacancy in the core.

FT/P6-38 · Vertical CT Injection into the STOR-M Tokamak

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Abstract: The University of Saskatchewan Compact Torus Injector (USCTI) has been modified to allow vertical CT injection into the STOR-M tokamak. The modification includes the addition of a 90 degree bending tube and a cone reducer. The characteristics of the traveling CT through the bend and reducer have been measured and the results confirmed that the CT can preserve its configuration while travelling through the bend and the reducer. Preliminary vertical injection experiments seem to indicate that the vertical CT injection is feasible. Further experiments are in progress to study the effects of CT injection on tokamak discharges.

FT/P7-1 · Fusion/Transmutation Reactor Studies Based on the Spherical Tours Concept

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Abstract: Fusion/transmutation concept will enable improved pathways to fusion power as well as possible competitive applications earlier than the power application of fusion energy. In this paper, a conceptual design of a Compact Fusion Experiment Reactor, CFER- ST, for transmutation Minor Actinide (MA) nuclear waste based on the Spherical Torus (ST) tokamak configuration was performed. A design study of plasma parameters suitable for the transmutation nuclear waste are presented. The blanket structure is characterized by a 20-cm thick transmutation zone with suitable MA composition rate and tritium breeding zone of 20- cm thickness. A solid $LiAlO_2$ was chosen as the tritium breeder in the tritium breeding zone. A integration design, including the transmutation neutronics, the structure, the thermohydraulic and the liquid curtain wall design for two types of the transmutation blankets which the LBE (Lead-Bismuth Eutectic) cooled and FLiBe eutectic self-cooled blanket were performed and relevant calculation results were also given. Magnet shield optimization was performed to design a shield which provides adequate magnet protection. The main conclusion of this study is that a small-scaled compact (R/r = 0.91/0.7) and lower fusion power (67MW) tokamak transmutation reactor, CFER-ST, have attractive advantages if it is used as a tool of transmutation the MA wastes. Proposed fusion/transmutation system can be designed based on existing ST tokamak physics and fusion technology databases.

FT/P7-2 · Stationary Compact VNS Tokamak for Transmutation

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Abstract: The concept of stationary VNS tokamak with aspect ratio A=2 based on moderate physical and technical assumptions is presented. Non-inductive stationary mode can be achieved by using of tangential NBI. Using of the confinement multiplier $H_{IPB98}(y, 2) = 1.6$ was shown to be enough to getting a stationary mode with $PNB \sim 45MW$. The dependence of NB absorption profile on neutral energy requires to apply neutral beams with energies 140 and 500 keV. The carried out calculations have shown possibility of tangential NBI to provide fully non-inductive current drive. The portion of bootstrap current fbs is ~ 0.4. Efficiency of CD is $\sim 0.06A/W$. The beam-target interactions during the fast ion slowing down are taken into account. An averaged neutron loading is about $0.3MW/m^2$. As specific Ohmic heat in toroidal coils does not exceed $12W/cm^3$, the heat can be removed from coils by usual ways. The divertor problem is suggested to solve by using of a design similar ITER with 24 reception cartridges. The blanket with neutron multiplication factor keff ~ 0.95 and lithium heat-transfer agent was chosen. The volumetric blanket power is $\sim 15MW/m^3$.

FT/P7-3 · Status of a Mirror Type 14 MeV Neutron Source Project in Novosibirsk

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Abstract: On the basis of the gas dynamic trap concept, a high power 14 MeV neutron source for fusion material testing can be developed. The device is essentially a modification of the Budker type mirror machine with axially symmetric magnetic field and a high mirror ratio, operated with relatively cold and dense plasma. Oblique injection of high energy deuterium and tritium neutral beams into the warm target plasma results in build up of population of fast anisotropic ions. Their density is strongly peaked near the turning points, in which neutron flux as high as $2MW/m^2$ (or even more) can be generated within

$\sim 1 - 2m^2$ testing zone. The paper discusses status of experiments at a scalable model of the gas dynamic trap and current upgrade of the device. Major goal to be achieved in the nearest time is an increase of the electron temperature up to Te ~300 eV that will immediately prove a practicability of moderate power neutron source with a flux of $0.45 - 0.5MW/m^2$.

FT/P7-4 · Conceptual design of a demonstration reactor for electric power generation

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Abstract: Conceptual study on a demonstration plant for electric power generation, named Demo-CREST, was conducted based on the consideration that a demo-plant should have capacities both (1) to demonstrate electric power generation in a plant scale with moderate plasma performance, which will be achieved in the early stage of the ITER operation, and foreseeable technologies and materials and (2) to have a possibility to show an economical competitiveness with advanced plasma performance and high performance blanket systems. The plasma core was optimized to be a minimum size for both net electric power generation with the ITER basic plasma parameters and commercial-scale generation with advance plasma parameters, which would be attained by the end of ITER operation. The engineering concept, especially the breeding blanket structure and its maintenance scheme, is also optimized to demonstrate the tritium self-sustainability and maintainability of in-vessel components. Capacities of stabilization of reversed shear plasma and the high thermal efficiency are additional factors for optimization of the advanced blanket.

FT/P7-6 · Energy Conversion in Reactor Chamber for Fast Ignition Heavy Ion Fusion

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Abstract: One of the major goals of conceptual studies of inertial fusion energy (IFE) is a description of energy fluxes and transformations in working media lines and trains and at interface between components of the system. Among the key problems to be solved is a transfer of the fusion energy to the cooler. In this paper the concept of fast ignition heavy ion fusion (FIHIF) is discussed. New data on the reactor chamber design and its response to fusion energy release are presented.

FT/P7-7 · Critical beta analyses with ferromagnetic and plasma rotation effects and wall geometry for a high beta steady state tokamak

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Abstract: The critical beta is shown to be decreased by ferromagnetic effect by about 8% for m/m0~2, m and m0 denote the permeability of ferromagnetic wall and vacuum, respectively, for standard aspect ratio tokamak. The existence of the stability window opened by both effects of the toroidal plasma rotation and the plasma dissipation, which was not observed for high aspect ratio tokamak, is found for tokamak of aspect ratio 3. The effect of ferromagnetism on them is also investigated. The critical beta analyses of NCT (National Centralized Tokamak) plasma using VALEN code are started with stabilizing plate and vacuum vessel geometry with finite resistivity, and the results for passive effect of stabilizing plate are obtained. The calculation will continue to include the effect of active feedback control.

FT/P7-8 · Design study of National Centralized Tokamak facility for the demonstration of steady state high beta plasma operation

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Abstract: Design studies are shown on the National Centralized Tokamak facility (formerly called JT-60SC). The machine design is performed to satisfy the following requirements as a supporting tokamak in parallel with ITER towards DEMO : (1) a super- conducting device with break-even-class plasma performance, (2) capability to sustain non- inductively driven high beta ($\beta_N = 3.5 - 5.5$) plasma for more than 100 second, and (3) flexibility in terms of plasma aspect ratio (A=2.6-3.6), plasma shape control (S=4-7), and feedback controls (internal RWM coils).

FT/P7-9 · RFX: new tools for real-time MHD control

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Abstract: The RFX device (Padova, Italy) has evolved in time from the passive control of the magnetic field configuration to the active control of both the field errors and the m=0 field harmonics. Encouraging results were obtained in terms of reduction of localized plasma-wall interaction and of induced rotation of MHD modes: on this basis the RFX load assembly has been recently modified, reducing by a factor of ten the time constant of the stabilizing shell and covering the whole plasma surface by a system of 192 saddle coils, each of them independently fed by a fast amplifier. Moreover, the new toroidal field power supply allows to produce robust m=0 rotating field harmonics, to drag the locked modes by non-linear coupling among different modes. These features realize the most powerful control system of plasma dynamics in any fusion device, with the best space resolution and the fastes response time. Specific real-time control technologies and software tools have been developed with the aim of offering a variety of control scenarios: driven rotation of MHD modes, controlled formation of Quasi-Single Helicity states, feedback stabilization of resistive wall modes, combinations of the above schemes. The work is supported by theoretical simulations of plasma response to the various control actions performed by the digital regulators and the fast amplifiers. The scientific programme to be pursued by means of these real-time control tools is considered to be highly relevant not only for Reverse Field Pinches, but also for Tokamaks, in particular when operating under advanced confinement scenarios.

FT/P7-10 · Integrated Plasma Control for High-Performance Tokamaks

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Abstract: Sustaining high performance in a tokamak requires controlling many equilibrium shape and profile characteristics simultaneously with high accuracy and reliability, while suppressing a variety of MHD instabilities. Integrated plasma control, the process of designing high-performance tokamak controllers based on validated system response models and confirming their performance in detailed simulations, provides a systematic method for ensuring high control performance. We describe the method, results of design and simulation tool development, and recent research which has produced novel approaches to plasma shape and MHD control. Use of the method is illustrated in applications to NTM suppression in DIII-D and gap control in ITER. This approach can reduce the need for machine time dedicated to control optimization, and can allow design of high-reliability controllers prior to producing the target equilibrium experimentally. A full set of tools needed for the approach has recently been completed and applied to DIII-D and NSTX. This approach has proven essential in the design of new devices including KSTAR, JT-60SC, and ITER.
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FT/P7-11 · Development and reactor integration of Helium cooled in-vessel components for DEMO

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Abstract: As the physics of a burning plasma are coming closer to being demonstrated in ITER, the focus of fusion research will shift towards the technology needed to build a demonstration reactor. The EU supports the development of its reference breeding blankets mainly within its long-term breeding blanket and materials R&D programmes. The recent European Power Plant Conceptual Study (PPCS) points out that a plant based on Helium- cooled in-vessel components, i.e. breeding blanket and divertor, while assuming limited technological extrapolation, has the potential for producing electricity in a competitive future market. While this outcome is based on careful analysis of key functions of the in-vessel components, the complexity of the task has prevented a comprehensive design of a blanket/divertor system fully integrated within the reactor vessel of a tokamak fusion machine. The paper presents a study of design concepts available for reactor integration, based on the recent development in Europe of blankets and divertors, and focussing on the EU modular Helium cooled pebble bed (HCPB) blanket concept. A geometry relevant for a demonstration reactor was created by scaling a PPCS reactor model down to 750 MWe. Conceptual design solutions are proposed for (i) blanket segmentation; (ii) curved blanket modules covering the tokamak surface; (iii) shield modules with integrated in-vessel manifolding; (iv) divertor cassettes and manifolding; and, (v) attachment schemes for the different components. The requirements imposed by in-vessel integration have implications on blanket performance, since they affect blanket coverage, the

amount of structural steel required, and radial space requirements. The work provides important feedback especially on current breeding blanket design, and input for the definition of a future European DEMO study.

FT/P7-12 · Helium Loop Karlsruhe (HELOKA) - large experimental facility for the in- vessel ITER and DEMO components

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Abstract: The design of a new planned FZK experimental facility, Helium Loop Karlsruhe (HELOKA) is presented in this paper. This facility will be dedicated to the testing of various components for nuclear fusion facilities: ITER Test Blanket Module (TBM), ITER Test Divertor Module (TDM), IFMIF High Flux Test Module (HFTM). The design of HELOKA has been elaborated in close co-operation with the future potential users. Based on their requirements, HELOKA is planned to be composed of three subloops: HELOKA-Low Pressure for IFMIF/HFTM, HELOKA-High Pressure for TBM and HELOKA-High Pressure for TDM and DEMO with different pressure- and temperature profiles. The facility would be worldwide unique and will offer the opportunity to investigate problems of helium cooling covering a variety of applications in different areas thus offering the possibility of a wide range of co-operations.

FT/P7-13 · The European Development of Helium-cooled Divertors for DEMO

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Abstract: Helium-cooled divertors are considered a suitable solution for fusion power plants, as they are compatible with likewise He-cooled blanket systems. They are also recommended for those blankets, where water-cooling of in-vessel components would lead to considerable concerns in terms of safety (e.g. steam-beryllium reaction with H production). Furthermore, it allows for a relatively high gas outlet temperature, i.e. a high thermal efficiency of the power conversion systems. Within the framework of the European power plant conceptual study (PPCS), two conceptual design concepts HETS and HEMP are being investigated at ENEA Frascati, Italy, and Forschungszentrum Karlsruhe, Germany, respectively. Both He-cooled divertor concepts are based on a modular design which helps to reduce thermal stresses. The design goal is to achieve a high heat flux of at least 10 MW/sq m. The development and optimisation of the divertor concepts require a close link of and iterative approach comprising the main issues of design, analyses, materials, fabrication technology, and experiments. Discussion of these issues and of the state of the art of divertor development shall be the subjects of this report.

FT/P7-14 · Fabrication of the KSTAR Toroidal Field Coil Structure

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Abstract: The KSTAR toroidal field (TF) coil structure is under fabrication with completion of engineering design and prototype construction. One prototype TF coil structure has been fabricated within allowable tolerance. Encasing of the prototype TF coil (TF00) in the prototype structure has been carried out through major processes involving coil alignment, fiducialization, final welding, vacuum impregnation, and final machining. During the final welding of enclosing the cover structure, we have measured temperature on the coil surface and stress on the coil surface and structure. Measurement has also been made of the magnitude of welding deformation due to the final welding. Based on the prototype fabrication results, we have finally chosen a SUS316LN as material of real TF coil structure. We used the narrow-gap Tig welding method. Thickness of the case and connection plate has margin of 5 mm for acceptance of welding deformation. To reduce residual stress due to welding and machining, a vibration method was adopted. To meet assembly tolerance, we have designed an adjustable spacer for the shear key and tapered ring for the conical bolt, which allows assembly tolerance of $\pm$ 2 mm. The fabrication of the 16 TF coil structure, which started in January 2004, is to be completed in early 2006 by Doosan Heavy Industries & Construction Co.

FT/P7-15 · Recent Advances in the Long Pulse Heating and Current Drive System for KSTAR

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Abstract: The heating and current drive systems for the KSTAR tokamak are being developed to support long pulse, high β, advanced tokamak physics experiments. The use of multiple heating technologies such as tangential NBI (beam energy < 120 keV, 8 MW), ion cyclotron waves (frequency range of 25-60 MHz, 6 MW), and lower hybrid waves (frequency of 5.0 GHz, 1.5 MW), and ECH (frequency of 84 GHz, 0.5 MW) will provide functions of current drive and profile control. Key technologies relevant for high power, long pulse operation are under development. Substantial progress has been made on areas such as ion source, RF launchers, and high power supplies. The results of the development will make the advanced tokamak operation of the KSTAR be obtainable and maintained for long pulse operating condition.

FT/P7-16 · Status of the KSTAR Superconducting Magnet System Development

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Abstract: The Mission of Korea Superconducting Tokamak Advanced Research (KSTAR) Project is to develop a steady-state-capable advanced superconducting tokamak for establishing a scientific and technological basis for an attractive fusion reactor. Because the KSTAR mission includes the achievement of a steady-state-capable operation, the use of superconducting coils is an obvious choice for the magnet system. The KSTAR superconducting magnet system consists of 16 Toroidal Field (TF) and 14 Poloidal Field (PF) coils. Both of the TF and PF coil systems use internally-cooled Cable-In-Conduit Conductors (CICC). The TF coil system provides a field of 3.5 T at the plasma center and the PF coil system is able to provide a flux swing of 17 V-sec. The major achievement in the KSTAR magnet system development includes the development of CICC, the development of a full size TF model coil, the development of a background magnetic field generation coil system, the construction of a large scale superconducting magnet and CICC test facility. TF and PF coils are in the stage of the fabrication for the KSTAR completion in the year 2006.

FT/P7-17 · Progress in the Assembly of the KSTAT Tokamak

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Abstract: As there is active progress in fabrication and delivery of the major components of the KSTAR, the site assembly task was launched at the start of 2004. The assembly work refers to all of the details of the KSTAR assembly plan that was finally established in 2003. The assembly work scope mainly consists of the assembly procedure, specifications, jigs & tools, measurement & alignment plan, welding procedure, cleaning plan, and other details that are related to the assembly. Among the major components of the KSTAR, the cryostat support beam, cryostat base, and magnet gravity support have been successfully assembled within specifications by February 2004. The 337.5° sector of the vacuum vessel with thermal shield will be assembled by November 2004. Moreover, special jigs & tools for assembly of the TF magnet have been successfully fabricated and constructed on the tokamak pit in March 2004. In this paper, design features, progress, and future plan of the KSTAR assembly will be reported.

FT/P7-18 · Recent developments in ICRF antenna modelling

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Abstract: The antenna systems presently developed for ion cyclotron resonance frequency (ICRF) heating of the ITER plasma consist of an array of a large number of radiating straps, 16 to 24 depending on design options. Such a large number is needed to deliver the high power density ($\sim 8MW/m^2$) required for the launcher without exceeding the radio-frequency (RF) voltage standoff of its straps. The straps will unavoidably be mutually coupled as they are radiating in the same medium. In this paper we discuss important recent developments in the modelling of such a demanding system. We specifically address: (i) The advances brought about by recently developed commercial software in the coupling analysis and optimisation of ICRF antenna arrays. (ii) Appropriate ways to convincingly include a realistic plasma description in such codes. (iii) The validation of the resulting numerical tools by comparison with the existing JET A2 ICRF array. (iv) Advances in the design of appropriate antenna test bed conditions: the

use of salted water as a means of creating realistic loading conditions, and the appropriate scaling of a mockup in order to preserve the same impedance matrix with scaled-down dimensions.

FT/P7-19 · Research and Development of Steady-State EC/ICRF Heating in LHD and an Optimal Remote Steering Antenna

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Abstract: Steady-State plasma sustaining by RF such ECH and ICRF heating is one of main objectives in the Large Helical Device. An ICRF plasma with a line-averaged density of $6 \times 10^{18}/m^3$ and temperature of 2 keV was sustained during 150 seconds with injection power of 0.5MW. The duration is determined due to increases in density and radiation power which result from hitting carbon tiles of divertor by high energy ions. On the other hand, with injecting ECH power of 72kW, an ECH plasma with averaged density of less than $1 \times 10^{18}/m^3$ and averaged ECE radiation temperature of 240 eV was sustained without radiation collapse during 756 seconds. Waveguide temperature and outgas in the evacuated waveguide raised with time and then an interlock in degree of vacuum in the evacuated waveguide terminated the injecting ECH power. To extend operation range of a remote steering ECH antenna, the characteristics and optimization of a square corrugated waveguide antenna with all four sides were examined theoretically and experimentally. In addition to the well-known result for small-angle injection, the existence of many branches with high recursive efficiency was found and put to practical use for large-angle injection up to 25 degrees.

FT/P7-21 · Magnum-psi, a plasma generator for plasma-surface interaction research in ITER-like conditions

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Abstract: The FOM-Institute for Plasma Physics – together with its TEC partners – is preparing the construction of Magnum-psi, a magnetized (3 T), steady-state, large area (100 cm^2) high-flux (up to 10^{24} ions $\mathrm{m}^{-2}\mathrm{s}^{-1}$) plasma generator. Magnum-psi is being developed to study plasma-surface interaction in conditions similar to those in the divertor of ITER and fusion reactors beyond ITER. Magnum-psi is to be embedded in an integrated plasma-surface laboratory including in situ and ex situ, in vacuo surface analysis. The scientific program includes a strong modeling effort. A pilot experiment (Pilot-psi) has been constructed to explore the techniques to be applied in Magnum-psi. In Pilot-psi, the required hydrogen ion flux can be achieved with a high-pressure plasma source (cascaded arc), by applying a longitudinal B-field of 1.6 T. Results of extensive diagnostic measurements on Pilot-psi (a.o., Thomson Scattering and high-resolution spectroscopy), combined with numerical studies of the source and the expansion of the plasma, are presented and discussed in the frame of the feasibility of the large Magnum-psi plasma generator.

FT/P7-22 · Component Manufacturing Development for the National Compact Stellarator Experiment (NCSX)

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Abstract: NCSX is the first of a new class of stellarators called compact stellarators which hold the promise of retaining the steady state feature of the stellarator but at a much lower aspect ratio and using a quasi-axisymmetric magnetic field to obtain tokamak-like performance. Although much of NCSX is conventional in design and construction, the vacuum vessel and modular coils provide significant engineering challenges due to their complex shapes, need for high dimensional accuracy, and the high current density required in the modular coils due space constraints. Consequently, a three-phase development program has been undertaken. In the first phase, laboratory / industrial studies were performed during the development of the conceptual design to permit advances in manufacturing technology to be incorporated into NCSX's plans. In the second phase, full-scale prototype modular coil winding forms, compacted cable conductors, and 20 degree sectors of the vacuum vessel were fabricated in industry. In parallel, the NCSX project team undertook R&D studies which focused on the windings. The third, or production phase will begin in late calendar year 2004.

FT/P7-23 · High-Beta Steady-State Advanced Tokamak Regimes for ITER and FIRE

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Abstract: An attractive tokamak based fusion power plant will require the development of high-β steady-state advanced tokamak regimes to produce a high gain burning plasma with a large fraction of self-driven current and high fusion power density. Both ITER and FIRE are being designed with the objective to address these issues by exploring and understanding burning plasma physics in the conventional H-mode regime, and in the advanced tokamak ($\beta_N \sim 3$ - 4, $f_{bs} \sim$50 - 80%) regime envisioned for an attractive steady-state high power density fusion power plant. ITER has employed conservative scenarios, as appropriate for their nuclear technology mission, while FIRE has employed more aggressive assumptions aimed at exploring the scenarios envisioned in the ARIES power plant studies. The main characteristics of the advanced scenarios presently under study for ITER and FIRE are compared with advanced tokamak regimes envisioned for the European Power Plant Conceptual Study (PPCS-C) and the US ARIES-RS Power Plant Study. The physics and plasma technology issues of ITER and FIRE are very similar, and technical solutions for one will likely be applicable to the other. The goal of the present work is to develop AT modes that would fully exploit the capability of ITER and FIRE. This paper will summarize the status of the work and indicate critical areas where further R&D is needed.

FT/P7-24 · The IFMIF Test Cell – Design and Neutronics Overview

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Abstract: The International Fusion Materials Irradiation Facility, IFMIF, is an intense neutron source driven by two 40 MeV deuteron beams striking a joint lithium target producing neutrons with a peak around 14 MeV. The neutrons produced within a footprint of 20 cm width by 5 cm height are penetrating the high-flux (0.5 litres, 20-55 dpa/full power year), medium-flux (6 litres, 1-20 dpa/fpy), and low-flux (>100 litres, <1 dpa/fpy) test modules. Irradiation simulation calculations on the basis of recently developed nuclear data libraries and advanced neutron transport calculations have shown that IFMIF offers not only favorable conditions for structural materials in the high flux volume but can also be used as very suit-able test bed to achieve for the neutron multiplier Beryllium DEMO-specific dpa, Helium and 3H production rates. The recently improved design of the high flux test module (i) increases the available high flux test volume by $\sim$ 20%, (ii) reduces the flux gradients by $\sim$ 10%, (iii) maximises the available space for specimens by $\sim$ 30%, and (iv) allows individual rig temperatures with practically homogeneous temperature profiles during irradiation and during beam-off periods.

FT/P7-25 · Advanced Computational Tools and Methods for Nuclear Analyses of Fusion Technology Systems

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Abstract: The worldwide efforts in fusion technology aim at developing, on the long- term, power reactors which can contribute substantially to the supply of electricity. The construction of the experimental fusion device ITER ("International Thermonuclear Experimental Reactor") and the intense neutron source IFMIF ("International Fusion Material Irradiation Facility") are considered as important next steps towards this long-term goal. The availability of qualified computational tools and nuclear data for the neutron transport simulation and the calculation of relevant nuclear responses is a pre-requisite to enable reliable design calculations for these facilities. Significant effort has been spent over the past few years to provide suitable neutronics tools and nuclear data. In this paper we focus on the computational tools and methods developed recently for nuclear analyses of fusion devices such as ITER and the IFMIF neutron source. These include (1) Monte Carlo based computational schemes for the calculation of 3D shut-down dose rates, (2) a CAD interface programme for the Monte Carlo code MCNP, (3) a method for Monte Carlo based sensitivity/uncertainty calculations, and (4) computational techniques and data for IFMIF neutronics and activation calculations. An overview of the recent achievements in these areas is presented.

FT/P7-35 · Technological and Environmental Prospects of Low Aspect Ratio Tokamak Reactor "VECTOR"

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Abstract: The optimization of the VECTOR design parameters led to a fusion output of 2.5 GW with a small reactor weight of 8,000 tons. Because of its compactness, CO_2 emission in the life cycle of the VECTOR power plant was estimated to be as low as 2.9 g-CO_2/kWh, being lower than that of an ITER-sized DEMO reactor (4.9 g-CO_2/kWh). As to the waste management of VECTOR, on the basis of reactor design and radiological considerations, we suggested reusing a liquid metal breeding material (PbLi) and a neutron shield material (TiH2) in successive reactors. According to this waste management, the disposal waste would be reduced to as low as 3,000 tons, which is comparable with the radioactive waste of a light water reactor (4,000 tons in metal).. Furthermore, it was numerically confirmed that such a low-A reactor would have an advantage over a-particle confinement.

SE

Safety, Environmental and Economic Aspects of Fusion

SE/P3-15 · Benchmark calculation for spills of cryogenic He into the ITER VV used as basis for an experimental

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Abstract: Code validation activities have been promoted inside the EFDA (European Fusion Development Agreement) to test the capability of codes in simulating accident phenomena in fusion facilities and, specifically, in the ITER (International Thermonuclear Experimental Reactor) plant. This work concerns a benchmark between three different computer codes (CONSEN, MAGS and MELCOR) and one analytical model (ITER model) in simulating cryogenic helium leaking into a vacuum vessel (VV) which contains hot structures. The task is the evaluation of the transient pressure inside the VV. The results will be used to design a vent duct (equivalent diameter, length and roughness) which allows helium pressure relief towards controlled volumes. A maximum pressure of $2.0 \times 10^5 Pa$ inside the vacuum vessel is allowed during the accident transient. The reference geometry is a simplified scheme preserving the main features of the full scale ITER design. Based on the results of the simulations, a matrix of experiments will be proposed to validate the calculated results and to design the vent duct for the ITER VV. The experiments are planned to be performed in the EVITA facility, located in CEA Cadarache (France).

SE/P3-38 · Fusion power in a sustainable future

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Abstract: This paper gives an integration of the key economics, safety, and environmental results from extensive recent European studies in the interlinked fields of: conceptual designs of commercial fusion power plants; analyses of their safety, environmental impacts and economic characteristics; analyses of the incorporation of fusion power into economic scenario modeling. The results demonstrate the following points. Fusion has very well attested and attractive inherent safety and environmental advantages, to address global climate change and gain public acceptance. The cost of fusion electricity is likely to be comparable with that from other environmentally responsible sources of electricity generation. Economically acceptable first generation fusion power plants, with major safety and environmental advantages, are those that could be accessed by a "fast track" route of fusion development, through ITER without major materials advances: there is potential for more advanced fusion power plants. Fusion, if constrained by earlier plans for the rate at which it would be deployed, would capture twenty percent of the electricity market by the end of this century, or earlier if a "fast track" development were adopted. The discounted risk-adjusted value of fusion development, derived from a probabilistic economic analysis, is substantially positive, in spite of the fact that commercial realization is not certain: thus fusion would massively over-recover the costs of its development. Overall, these results, together with fusion's large resource base, show that fusion is an attractive option to contribute in the medium and long term to sustainable energy generation; it would be particularly suited to baseload electricity supply and would ideally complement renewable energy sources in a future energy mix.

SE/P3-39 · Nuclear Fusion as New Energy Option in a Global Single-Regional Energy System Model

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Abstract: Nuclear fusion is a very promising option among new energy supply technologies due to its safe operation, nearly inexhaustible resources and its potential for a CO2 emission free production of base load electricity. Fusion reactors are expected to be commercially available for power generation about the middle of the century. Within a subtask of the Socio-Economic Research on Fusion (SERF) programme of the European Commission the central goal is to identify, whether there is a window of opportunity for fusion on the electricity market under "business as usual" conditions, and if not, how the boundary conditions have to look like to open such a window. Using the TIMES model generator, a global single-regional energy system model has been developed at the IPP Garching in co-operation with the ITP Graz. Within an EFDA-SERF project task, the ITP Graz studies the potential role of fusion power in a future energy system. The work is mainly focused on long-term scenario analysis until 2100 with special attention to nuclear fusion and its most likely long-term competitors. In a next step, the global TIMES model is extended with respect to the level of detail and accuracy of the description within the different sectors of the energy system. In addition, the advanced modelling features Endogenous Technological Learning and elastic demand are included in the model. In a first preliminary assessment, the impact of all these features on the model results, especially on the potential role of fusion power and its long-term competitors, is investigated.

SE/P3-40 · Introduction condition of a tokamak fusion power plant as an advanced technology in world energy scenario

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Abstract: The plasma performance required for a tokamak fusion power plant to satisfy the economic introduction condition of fusion energy in a world energy scenario is analyzed by using the fusion power plant system analysis code (FUSAC). The typical value of the economic introduction condition is the break-even price of electricity for introduction of fusion energy into a world energy scenario. In the present study, the world energy scenario including fusion energy is investigated by a linearized version of the DNE21 model, which are quite sophisticated energy supply technology assessment models. The break-even price of fusion energy in the year 2050 is estimated at the range from 65 mill/kWh to 135 mill/kWh under the condition of 550 ppm CO_2 constraint. The width of the break-even price depends on energy demand scenarios, capacity utilization ratio of options in energy/environment technologies, and so on. If the cost of electricity for fusion energy can achieve the lowest break-even price of this range, i.e. 65mill/kWh in the year 2050, about 20% share of fusion energy in all produced electricity in the year 2100 is expected. Consequently, fusion energy can substantially contribute to reduce the CO_2 emission. Hence, we consider the introduction year 2050 of fusion energy as the year we should aim at. We carried out the parametric analysis to clarify the plasma performance required to achieve the break-even price in the year 2050. The major engineering conditions are maximum magnetic field 16T, net electric power 1000MW, thermal efficiency 40% and plant availability 60%. To reach that break-even price with a tokamak reactor, the range of $3.0 < \beta_N < 5.5$ is found to be required. This result is based on the database of about 100,000 operation points, where the conceivable range of plasma parameters after ITER is applied. As for β_N, high performance steady state operation of ITER has a good potential to reach this range. On the other hand, $\beta_N > 5.0$ with the major radius $R < 6.0$ m is required to achieve the most severe case of the break-even price 65mill/kWh. In conclusion, although β_N as high as possible is desirable, the range of $3.0 < \beta_N < 5.5$ (with B_{tmax}=16 T) is a possible target range to achieve the economic introduction condition.

SE/P3-41 · Socio-economic Aspects of Fusion

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Abstract: Fusion deployment scenarios have been developed for the last half of this century. These scenarios are analyzed to determine the material resources required, the waste produced, and the reduction in carbon dioxide relative to fossil power production. The material requirements for these scenarios are significant but should be within world production capability. The fusion goal is to produce only low level activated waste. The amounts produced should be manageable. The positive impact on carbon dioxide in the atmosphere does not reach significant levels until the beginning of the next century as a result of the slow response of the environment to carbon dioxide emissions due to the large reservoirs involved.

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