21st International Toki Conference (ITC-21)

Integration of Fusion Science and Technology for Steady State Operation

Book of Abstracts

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Ceratopia Toki, Toki-City, Gifu, Japan

The International Toki Conference (ITC) is an international conference for the discussion and presentation of research activities related to nuclear fusion and plasma. It has been held annually by the National Institute for Fusion Science in Toki city since 1989. Despite recent progress, steady state operation of fusion energy systems remains a major issue of research. The ITC-21 will focus on the integration of key science and technology towards high performance fusion energy systems with the capability for steady state operation. The conference topics cover a wide range of research in fusion science and near-term and long-term technology oriented to steady state operation.

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(Monday 28th November)

PL-1

Physics and Technology of Steady Sate Operation in LHD

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Steady state plasma sustainment for more than one hour is one of the main missions of the LHD project. A fusion reactor should be inevitably a steady state operation device. LHD is a helical type device and well-suitable to form the steady confinement configuration and to control steady state plasma. It is contrasted with high autonomous tokamak plasma which has to hold large internal plasma current and also to control its radial profile. The steady state plasma operation has been conducted from the beginning of the LHD project. LHD has the highest record of 1.6GJ (500kW/54 minute) in the incidence energy. The targeted value of the LHD mission is the incident power of 3MW and one hour plasma sustainment. However, there are many issues to be solved to the steady state operation. Controlling the particle balance that is a result of particle feeding, wall recycling, wall and vacuum pumping, and also control of temperature rise of divertor, wall and heating unit are key issues.

An avoidance of impurity accumulation is also an important physics issue. Accumulation of impurities in the plasma is observed in some limited conditions of pulsed operation in LHD. We know the conditions to avoid the accumulation now. Usually, the exhaust of plasma particles include impurities tends to occur in ECH heating, and more interesting character is exhaustion of high Z ions observed in the high power NBI heating in pulsed operation. This confinement character has been paid attention as a good feature for the fusion reactor.

Heat balance at divertor plate and vacuum wall, and steady driving of heating tools (ICH, ECH and NBI) are also important technical issues. Thermal loading at the diverter plate is mitigated using the magnetic axis swing technique changing the poloidal coil currents which is a noble and unique technique only usable in heliotron magnetic configuration. Moreover, the booster ECH pulse (around 0.4MW) was synchronously injected to avoid a radiation collapse against occasional impurity flake drops and it demonstrated successfully to continue the steady state operation.

The steady state operation was mainly performed using ICRF and ECR heating, and it was terminated by radiation collapse due to the sudden increase of plasma density caused by dropping of flakes from the divertor or vacuum wall. Recently it is clarified that the flakes are mixed-material of stainless steel and carbon generated by helium glow discharge for wall conditioning and the usual plasma discharges.

The physics and technology issues relate to the steady state plasma operation connect to those of the future fusion reactor and should give us valuable information for reactor design.

PL-2

Materials Science under Extreme Conditions: Overview of Materials Challenges for Realizing Practical Fusion Energy

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Recent advances in plasma physics are bringing the prospect of fusion energy close to achievement of steady state operation at fusion reactor-relevant conditions. Durable, high-performance structural and plasma-facing materials (along with numerous special-purpose materials systems, ranging from tritium breeding and plasma heating and diagnostic systems to superconducting magnets) will be key for the successful development of fusion energy. The anticipated temperatures, heat fluxes and radiation damage levels for plasma-facing and structural materials represent an extraordinary jump compared to the conditions experienced by fuel cladding and core internal structures in existing fission power plants. Similar daunting operational challenges face the tritium breeding and plasma heating systems, due to issues such as high heat flux gradients, intense radiation fields, high temperature coolant compatibility requirements, and the necessity for low absorption of transmitted plasma heating power. A science-based approach will be crucial for the successful resolution of these extreme materials challenges. This talk will focus on the challenges associated with neutron radiation damage, high heat flux and high operating temperatures on materials for potential steady-state demonstration fusion reactors, and how multiscale materials modeling and experimental validation can expediently result in the development of high-performance radiation-resistant materials.

Progress Towards Steady-State Operation in NSTX Through Advances in Plasma Control and Plasma-Wall Interactions

M.G. Bell, S.P. Gerhardt and the NSTX Team

Princeton Plasma Physics Laboratory

Spherical tokamaks for fusion energy development will operate at high-beta, and require excellent confinement and non-inductive current drive to maintain a steady state. In pursuit of these goals, the National Spherical Torus Experiment (NSTX) has developed improved control of its strongly shaped plasmas, implemented routine feedback control of n = 1 error fields and resistive wall modes and utilized extensive lithium coating of its plasma facing components. The combination of these techniques has produced non-inductive current fractions up to 70% which were sustained for several current-relaxation times. These discharges have also achieved the highest stored energy in NSTX and, in different combinations of the toroidal and poloidal magnetic fields, the highest sustained values of toroidal and poloidal beta. In these H-mode discharges, which became essentially free of ELMs as a result of applying lithium beforehand, the scaling of energy confinement with current is somewhat stronger than was evident in previous experiments, whereas the toroidal field dependence previously observed is nearly absent. Passive stability to modes with toroidal mode number n = 1 has been facilitated by shaping the plasma cross-section and maintaining a broad pressure profile. Reducing intrinsic n = 1 error fields and active n = 1 mode control using a set of non-axisymmetric coils to generate controlled n = 1 radial field perturbations further improved reliability at high beta, although n = 1 kink and tearing instabilities in the plasma core often terminated the high-performance phase. Simulations with several MHD stability codes show that these particular modes can be avoided by maintaining an elevated central safety factor q. The relaxed current profile in MHD-quiescent phase of these discharges is well modeled with essentially neoclassical physics, although it is necessary to take into account the effect of large Toroidal Alfven Eigenmode (TAE) "avalanches" which occur repetitively in redistributing the beam-driven toroidal current and thereby elevating the central q. This effect has been simulated heuristically with the transport analysis and modeling code TRANSP by applying a transient radial diffusivity to the unthermalized beam-ion component coincident with each observed burst of TAE activity. The results of the simulations then match well the measured toroidal current profile and reproduce the observed drops in the DD neutron emission which is dominated by beam-target reactions in NSTX. Using this physics model, simulations with TRANSP and the DCON stability code show that the higher toroidal field and off-axis beam current drive in the planned NSTX-Upgrade can produce a range of ideally stable, fully non-inductive plasma scenarios, and partially inductively sustained scenarios with toroidal beta exceeding 25%.

Work supported by US Department of Energy Contract DE-AC02-09CH11466

Recent TCV results – innovative plasma shaping to improve plasma properties and insight

A.Pochelon for the TCV-Team

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The TCV tokamak experiment is used to study the influence of unconventional plasma shapes on core and edge confinement. In low collisionality L-mode plasmas with electron cyclotron resonance heating (ECRH) confinement increases with increasingly negative triangularity δ, The confinement improvement correlates with a decrease of the very inner core electron heat transport, even though the triangularity of the plasma boundary quickly decreases towards the core, which points to the effect of global – non-local – transport properties. This confinement improvement remains for a large range of radial ECRH power deposition profiles. TCV is also used to study negative triangularity effects in H-mode plasmas. In contrast to the L-mode results H-mode confinement is known to improve towards positive triangularity, due to increasing pedestal height with triangularity, though plagued by large edge localised modes (ELMs). An optimum triangularity can thus be sought between steep edge barriers (δ>0) with large ELMs, and improved core confinement (δ<0). This is significant for a reactor, since it opens the possibility of having H-mode-like confinement time within an L-mode edge, or at least with reduced ELMs. ELMy H-modes with δ_{up}<0 have already been obtained in TCV. Another approach to reduce the effect of ELMs is the exploration of different divertor shapes, like for instance the snowflake divertor. The characteristics of such a hexapolar null configuration are compared to those of the conventional quadrupolar X-point divertor. TCV experiments confirm some of the advantageous properties of the snowflake configuration such as the distribution of the exhaust power on more strike points than the two that characterize conventional X-point divertor. Plasma shaping may further help understanding the coupling process of sawteeth to NTMs, leading to confinement degradation. TCV experiments show that sawtooth crashes can almost instantaneously (<30μs) generate harmonics resonant at q>1 flux surfaces as resolved instantly by spatial Fourier analysis of continuous toroidal magnetic probe arrays, potentially giving rise to 3/2 and 2/1 modes. These examples demonstrate the effectiveness of plasma shaping, both to improve confinement and MHD properties in the experiment, and to give further insight in stability or gyrokinetic modelling.

This work was supported in part by the Swiss National Science Foundation.

Overview of Experimental Results in the EAST Tokamak

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The Experimental Advanced Superconducting Tokamak (EAST) is a fully superconducting tokamak with a flexible poloidal field system to accommodate both single null (SN) and double null (DN) divertor configurations, and its main mission is to establish steady-state high performance plasma and study related physics and technologies. Since the first plasma was successfully achieved on Sept. 2006, the new capabilities including actively water-cooled PFCs, Heating & CD system, Plasma Control System, and Diagnostics have been developed and validated in the past five years in the EAST Tokamak. A series of wall-conditioning technologies in superconducting machines (RF-DC, HF_GDC and Li-Coating) are explored, which will make unique contributions and produce results of significance for ITER in the future. Integrated operation scenarios (plasma startup, and ramp up/down) were focused on superconducting tokomak and reliable plasma operation with Ip=1.0MA was obtained. In the last experimental campaign on 2010, long pulse diverted plasma operations up to 100s were realized by LHCD of 0.8MW. Stationary H-mode plasma discharges, with either LHCD only or combined auxiliary heating of ICRF and LHCD, have been achieved and sustained up to 6.5s. The power threshold for H-mode access in EAST follows the international tokamak scaling and the energy confinement for H-mode in LHCD has been identified. In addition, experiments of SOL & Divertor physics, first evidence of the role of Zonal Flows for the L-H transition, intrinsic rotation and toroidal flow driven in LHCD plasma, full non-inductive CD, effective ICRF heating, etc. are encouraging. The future plan of EAST in next 3-5 years also will be presented in this paper.

Suppression of Edge Localized Modes in the DIII-D Tokamak with Small 3D Magnetic Perturbations

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Edge Localized Modes (ELMs), associated with large edge pressure and current profile gradients in high confinement (H-mode) plasmas, are a significant concern for the next generation of large tokamaks such as ITER where the stored plasma energy is expected to exceed several hundred megajoules. The largest transient energy bursts driven by ELMs in ITER are predicted to release as much as 6%-7% of the plasma energy in less than a millisecond. This is expected to cause a substantial increase in the divertor target plate erosion rate resulting in a reduction of the operational lifetime of these essential components. In addition, the eroded divertor target plate material may cause an increase in the influx of impurities into the core plasma that could limit the fusion power output of the device and trigger instabilities that are known to lead to a rapid termination of the discharge. Over the last few years, a series of experiments done in the DIII-D tokamak have demonstrated that large ELMs can be reproducibly suppressed over a wide range of H-mode conditions and plasma shapes including ITER similar shapes with ITER pedestal collisionalites. In these experiments, it is found that small 3D magnetic perturbations ($b_{3D}/b_{tor} \sim 2x10^{-3}$), primarily localized to the edge of the plasma, can be used to suppress the first ELM following the L-H transition and to maintain an ELM suppressed discharge for up to 24 energy confinement times, limited only by the heating duration of DIII-D neutral beams. During this time the line average density, pedestal density, Z_{eff} and core radiation rate all remain constant, in contrast to a conventional ELM-free H-mode, suggesting that robust steady-state H-mode conditions, with an H-mode factor H89Y2 of approximately unity, have been achieved. ELM suppression has been obtained using both $n=3$ and $n=2$ resonant magnetic perturbation fields from the DIII-D internal non-axisymmetric coils. The current requirements for ELM suppression and plasma response to the applied perturbation fields are most sensitive to the safety factor at the 95% flux surface and the shape of the outer flux surfaces. In this talk, examples of the plasma characteristics observed in DIII-D ELM suppressed H-mode plasmas will be discussed along with a summary of the operational space required for suppression and an overview of the physics mechanisms that are believed to be important for obtaining ELM suppression in DIII-D. Implications for scaling these results to ITER will also be discussed.

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

Excitation of Beta-induced Alfven Eigenmode during Strong Interchange Mode in Large Helical Device

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The beta-induced Alfven eigenmodes (BAEs) were observed and investigated under different conditions in tokamak plasma, including that driven by fast ions [1], energetic electrons [2], and large magnetic islands [3-5]. The BAEs are very important because they can be used as energy channels to transfer the fusion-born-alpha-particle energy to the thermonuclear plasma.
In a helical plasma, the BAEs during strong interchange mode, whose mode-numbers are m/n=2/1, have been recently observed for the first time. The first harmonic frequencies of these oscillations range from 30 to 70 kHz in Large Helical Device (LHD) , much lower than the toroidal-Alfven-eigenmode (TAE) frequency, and are provided with the same order of the low-frequency gap induced by finite beta effects. The magnetic fluctuation spectrogram indicates that the BAEs often occur in pairs, and their mode-numbers are m/n=2/1 and -2/-1. The analysis reveals that the modes propagate poloidally and toroidally in opposite directions, and form standing-wave structures in interchange-mode rest frame. The frequencies of the pair mode are associated with the Te/Ti ratio, and the frequency difference of the pair modes is determined by the frequency of interchange mode. The novel experimental results indicate that this phenomenon is very similar with that of BAE during strong tearing mode in tokamak plasma, but the BAE frequencies do not completely depend on the intensity of interchange mode. The new finding shed light on the underlying physics mechanism for the excitation of the low frequency Alfvenic fluctuation. Comparing with the tearing mode, although the free energy, which drives the interchange mode, is different, the steepen pressure gradients at the vicinity of resonant surface both potentially excite the type of BAE instability.

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Studies of MHD stability in Heliotron J Plasmas

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To clarify the MHD instabilities is important because the MHD instabilities could lead to the deleterious displacement of the equilibrium as well as the increment of the anomalous transport, thus resulting in the degradation of plasma performance. Therefore, the MHD stability is extensively being studied in many tokamaks and helical systems.
The main purpose of Heliotron J, which is the middle size helical device with the pole number of helical coil l=1 and toroidal field period N_p=4, is to optimize the magnetic configuration aiming at the simultaneous improvements in particle transport and MHD stability. With regard to MHD stability, the magnetic configuration of Heliotron J has a low magnetic shear in combination with a magnetic well by which MHD instabilities particularly at rational surfaces with low order can be avoided or would be stabilized for pressure driven MHD instabilities. We experimentally investigate the MHD stability of Heliotron J plasmas by utilizing the scan of rotational transform and magnetic well/hill in order to demonstrate the advantage of our concept with low magnetic shear and magnetic well.
Two kinds of MHD instabilities are observed in Heliotron J plasmas. One is the low-n pressure driven resistive interchange mode with m/n=2/1 and m/n=5/3 in the plasmas with low order rational surface 0.5 and 0.6, respectively. When these MHD instabilities are observed, the bulk plasma parameters such as plasma stored energy is saturated or decreased indicating that they affect global energy confinement. It should be noted that we have observed no coherent mode in the plasma without rational surface. The other is the energetic ion driven MHD instabilities including GAEs and/or EPMs in NBI heated plasmas. They might affect the energetic ion transport from the results that some plasma parameters are simultaneously modulated with the bursting GAEs. We will also report the characteristics including spatial profile of these instabilities and parameter dependencies in Heliotron J plasmas.

Progress in LHCD experiment and modelling to approach AT regime on Alcator C-Mod

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Assessing the performance of heating and current drive techniques at ITER relevant parameters is critical for designing non-inductive current drive systems on that device and for other future steady-state experiments. On Alcator C-Mod, excellent lower hybrid (LH) current drive (CD) efficiency (η = 2.0-2.5 × 10^{19} $AW^{-1}m^{-2}$) has been observed during fully non-inductive, reversed shear plasma operation at line averaged density of 0.5-0.6×10^{20} m^{-3} and magnetic field of 5.4T, close to what is expected in ITER steady state scenarios [1]. Some of these discharges exhibit an ITB characterized by an abrupt increase of core electron temperature (r/a < 0.2). The ITB formation coincides with reversal of magnetic shear about 200 - 300 ms after the LH turn-on. A key issue to approach high f_{BS} (> 50%) advanced tokamak regimes from these plasmas is to maintain the good CD efficiency at high density (1.0-1.5 × 10^{20} m^{-3}), where a significant fraction of LHCD power was observed to be absorbed either directly or indirectly in the SOL region, especially in diverted configurations [2]. LHCD simulations using both ray-tracing and full wave codes have been performed to better understand this unfavorable phenomenon and parasitic edge losses were shown to be a candidate to explain the experimental results. Although multiple mechanisms, such as fast electron diffusion and collisional absorptions of LH waves, can contribute to such losses, these losses tend to become more important in multipass absorption regimes, suggesting that stronger single pass absorption could be generic methods to improve CD efficiency at high density. Indeed, recent discharges using ICRF mode-conversion heating to increase the electron temperature (T_{e0} > 4 keV) indicate that such a recovery is possible. The extensive modelling effort and direct comparisons with experiment improve our understanding of LHCD, giving a solid guideline for planned experiments at twice the LHCD power (>2MW) on C-Mod and enhancing our capability to predict the performance of LHCD for future fusion devices.

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Plasma Potential in Toroidal Devices: T-10, TJ-II, CHS and LHD

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Direct measurements of the potential φ have been undertaken in the T-10 tokamak [1] (B=1.5-2.5 T, R=1.5 m, a=0.3 m, P_{ECRH}<2 MW), and stellarators TJ-II [1] (B=1 T, <R>=1.5 m, <a>=0.22 m, P_{ECRH}< 0.6 MW, P_{NBI}< 0.9 MW), CHS [2](B=1 T, <R>=1.0 m, <a>=0.2 m, P_{ECRH}<0.6 MW, P_{NBI}<1.5 MW), and LHD [3] (B=1.5-3 T, R=3.6-3.9 m, <a>=0.63 m, P_{ECRH} <0.6 MW, P_{NBI}<19 MW) using the Heavy Ion Beam Probing. L- and H-modes with and without plasma current were considered. Despite the differences in machine sizes, heating methods and magnetic configurations, and different expected mechanisms for the electric field E_r generation in tokamak and stellarators, the observed φ shows the striking similarities: (i) Similar magnitudes of E_r; (ii) For low densities, n_e <0.5×10^{19} m^{-3} (unattainable in T-10), φ is positive, and an increase in n_e is associated with the decrease of positive φ and formation of a negative E_r; (iii) For higher densities, n_e>(0.5-1)×10^{19} m^{-3}, both φ and E_r tends to be negative despite different heating methods: Ohmic and ECR heating in T-10, ECRH and/or NBI in TJ-II, CHS and LHD; (iv) Application of ECRH, causing a rise in T_e, results in more positive values for φ and E_r. Analysis shows that for stellarators the main features of φ dependences on n_e and T_e agree with neoclassical predictions within the experimental and simulation precisions. Spontaneous and biased transitions to the H-mode, observed in TJ-II and T-10, are associated with more negative E_r and broadband turbulence suppression. Similar effects were observed in JFT-2M. Potentials on the four devices, similarities and differences in the mechanisms of E_r generation are discussed.

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Optimizing the current ramp-up phase for the hybrid ITER scenario

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The current ramp-up phase of ITER is a critical stage: MHD instabilities have to be avoided, flux consumption has to be minimized, and this has to be achieved within the narrow operational window of ITER. Ramp-up for the hybrid scenario is more critical than for the standard (H-mode) scenario, since the q profile must be shaped carefully: q_{min} should stay near or slightly above 1 and, for an optimized fusion performance, the q profile should have the typical hybrid shape with a wide flat region [1]. This paper reports on a systematic effort within the ITER Scenarios Modelling working group (ISM), part of the European Integrated Tokamak Modelling (ITM) Task Force, to optimize the current ramp-up phase for the ITER hybrid scenario, and to assess the sensitivity of the results to the assumptions made.
Validation on the ramp-up phase of JET, AUG and Tore Supra [2] has shown that both empirical scaling based models and the semi-empirical Bohm/gyro-Bohm model (L-mode version, ITB shear function off) yield a good reproduction of this phase for considered discharges, in terms of T_e and q profile and l_i. Therefore these models have been used in the reported work, which was carried out with the CRONOS integrated suite of codes.
Following assumptions from the ITER team were adopted: *(i)* An expanding ITER shape is used with early X-point formation at 3.5 MA; *(ii)* A flat Z_{eff} profile is assumed, decreasing in time, as in [3]; *(iii)* A rather low density of $n_e = 0.25\, n_e^{\mathrm{Gw}}$ is taken; *(iv)* The n_e profile is assumed parabolic with a moderate peaking factor $n_e(0)/\langle n_e \rangle = 1.3$.
The simulations start 1.5 s after breakdown, when $I_p = 0.5$ MA. The heating systems are used within their designed limitations. The total input power is taken to be below the L-H threshold during the whole ramp-up phase. Other assumptions ($T_{e,i}$(edge), initial $T_{e,i}$ and l_i) are based on experimental evidence. Regarding sensitivity of the results to the assumptions, following parameters were varied: $T_{e,i}$(edge) (by 40%); n_e (by 60%), profile shape (parabolic vs. flat) and Z_{eff}.
The simulations show that the available heating systems allow the attainment of a hybrid q profile at the end of the current ramp-up. This is reached by using off-axis NBI, ECCD (from UPL) and LHCD. A heating scheme with only NBI and ECCD is slightly less effective.
The optimum heating scheme depends on the chosen transport model. Moreover, the simulations show that modified assumptions on n_e peaking, edge $T_{e,i}$ and Z_{eff} can be easily accounted for by a shift in time of the heating scheme. By post processing the simulation results with the free boundary equilibrium code FREEBIE, run in Poynting mode, it has been checked that the reference case, both with and without additional heating, is safely within the boundaries put by ITER coils.

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Poster Session 1

(Monday 28th November)

Assessment of Plasma Performance in a Magnetic Configuration with Reduced Poloidal Coils for a Helical DEMO Reactor FFHR-d1

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The performance of high-density plasmas heated by neutral beam (NB) injection in a vertically elongated configuration using 4 poloidal-coils has been investigated in LHD and shown to be better than that in the normal configuration using 6 poloidal-coils. Reduction of the poloidal-coil is favorable in designing an LHD-type helical reactor FFHR-d1 [1], from both points of view of cost reduction and realization of large maintenance ports. In the normal condition in LHD experiment, the toroidally averaged plasma cross-section is kept circular by cancelling the quadrupole components of the magnetic field for 100% with 6 poloidal-coils. The toroidally averaged plasma cross-section becomes vertically elongated when the 2 of 6 poloidal-coils are not used, while the acceptable size of maintenance port is maximized. A better energy confinement was observed in the circular configuration for low-density plasmas heated by electron cyclotron heating [2]. On the other hand, vertical elongation is effective to mitigate the Shafranov shift in high-beta plasmas [3]. In this study, the plasma performances in the vertically elongated and circular configurations have been compared for high-density NB heated plasmas, using the direct profile extrapolation (DPE) method, which has been developed to predict the radial profiles in fusion reactors from the profile data obtained in the experiment [4]. Although no clear difference between the two configurations is recognized in the relation between the central density and the central pressure, a factor C_{exp} used in the DPE method, which is proportional to the reactor size, $R_{reactor}$, at a given reactor magnetic field $B_{reactor}$, as $R_{reactor} \propto C_{exp} B_{reactor}^{-4/3}$, is smaller in the vertically elongated configuration. The difference in C_{exp} between the two configurations is enhanced to > 10 % as the plasma beta increases. This might be due to the mitigated Shafranov shift in the vertically elongated configuration [3], which results in a larger plasma volume at a given plasma beta.

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Parameter dependence of MHD mode rotation in the Large Helical Device

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The characteristics of MHD mode rotation have been investigated on Large Helical Device (LHD). It is widely known in tokamaks that MHD mode rotation is a key for maintaining the "stable" plasma state. In the case of tearing mode, reduction of the plasma flow induces the mode locking, leading the disruption finally. Also, since the resistive wall mode, which limits the achieved beta, is destabilized by insufficient plasma flow. On the other hand, in helical plasmas, the rotation mechanism of the interchange mode and the effect of the plasma flow on the stability is unclear so far. In the heliotron configuration without net current, an interchange mode is one of key instabilities because of magnetic hill formation. Especially, the modes excited in peripheral region of plasma are dominantly observed when the beta increases and/or L/H transition occurs. Several modes are excited in the plasma edge with the stochastic layer and they are key instabilities for the high beta plasma production and/or the formation of the edge transport barrier. In the previous experiments, the onset and parameter dependence of the modes have been investigated in various configurations [1]. Here, these rotations were compared with the plasma flow in order to understand the characteristics of mode rotation. The density scan experiment was performed so as to change the poloidal plasma flow because the toroidal flow is almost negligible in the peripheral region. The electron density was changed to 2-5 × 10^{19} m^{-3} and then the volume averaged beta value, $\langle\beta\rangle$, was maintained at about 1.5 %. The m/n=1/1, 3/4 and 2/3 modes where the resonant surfaces are located at $\rho > 0.9$ were dominantly observed and rotate in the electron diamagnetic direction in the laboratory frame. The mode rotation is compared with the electron and ion flows taking into account the diamagnetic drift frequency. The experimental results show that each mode rotation is consistent with the electron flow within the measurement error, which suggests that the observed mode freeze the electron fluid as well as the tearing mode [2]. The plasma was heated by co-NB in the experiments, whereas it is confirmed that this tendency is almost the same as the counter-NB case. In high beta discharges with $\langle\beta\rangle > 2$ %, MHD modes such as the m/n = 1/2 mode in the stochastic region (outside the last closed flux surface) are excited typically. On discharges where the 1/2 mode dominantly observed, this rotation frequency is close to the ion flow rather than electron one. As a result, this mode seems to rotate in the ion diamagnetic direction in the plasma frame.

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Study of Confinement Properties on Unfavorable LHD Configurations for Ideal MHD Instabilities using TASK3D-code

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The Large Helical Device (LHD) has various magnetic field configurations, by changing the vertical field coils and the pitch parameters of the helical coils, which include the unfavorable ones for the interchange type of ideal MHD instabilities. In deed, in the LHD experiments with the MHD unfavorable configurations like the magnetic axis torus outwardly shifted configurations and the high aspect configuration with either high magnetic hill and the low magnetic shear, the beta collapses and the flat electron temperature profiles are observed. On the above discharges, the thermal transport analyses are done by the subset of the 3D integrated-transport simulation code, TASK3D [1] in addition to a liner MHD instability analysis. The effects of the global MHD instability on the confinement will be discussed from the comparative study between the transport and the stability analyses.
Moreover, the above results are compared with the prediction of the pressure profiles calculated by the MHD instability effects evaluation module on the confinement of the TASK3D, TASK3D/EQ+TR+MSSH [2], which leads to the validity check of the 'TASK3D/EQ+TR+MSSH ' module.

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Impact of T_e/T_i on energy confinement properties in a tokamak plasma

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Scaling expressions for thermal energy confinement time of tokamak H-mode plasmas have a very important role in predicting the performance of future fusion devices. The IPB98(y,2) scaling law [1] has been widely used. In the compiled database for IPB98(y,2), the ratio T_e/T_i is mainly smaller than unity. In future burning plasmas with strong alpha heating, on the contrary, it is expected that T_e/T_i is larger than unity.

The objective of the present study is to understand the impact of T_e/T_i on confinement properties. We have studied the updated DB3v10 database[1, 2], paying attention to the T_e/T_i effect. The 67 data from AUG and 68 from C-Mod have been extracted from the database. For these data, the density is lower than 60% of the Greenwald density limit n_{GW}. As a consequence, we have obtained following scaling expressions in two density profile regimes.

$\tau_{sc1} = 0.0777 I_p^{0.91} B_t^{0.11} n_e^{0.28} P_L^{-0.59} R^{1.73} M^{0.27} \varepsilon^{0.54} \kappa^{0.41}$: $1.0<n_{e0}/n_e<1.1$ (flat density case)

$\tau_{sc2} = 0.0594 I_p^{0.90} B_t^{0.08} n_e^{0.34} P_L^{-0.63} R^{1.78} M^{0.16} \varepsilon^{0.41} \kappa^{0.72}$: $1.1<n_{e0}/n_e<1.6$ (peaked density case)

(τ_{sc}: s, I_p: MA, B_t: T, n_e: 10^{19}m^{-3}, P_L: MW, R: m, M: ion mass number, ε: inverse aspect ratio, κ: ellipticity, n_{e0}/n_e: ratio of central density to line averaged density) Examining the impact of T_e/T_i on these two regimes separately, we have found that it is weak for flat density case while it becomes apparent for peaked density case. For the latter, we have further developed a scaling expression including T_e/T_i by adding 795 JET data. We incorporated 3 nondimensional variables [3], namely $B_t R^{1.25}$, n_e/n_{GW}*, and safety factor q in addition to T_e/T_i. The dependences on n_e/n_{GW}* and q were weak, and the following new expression was obtained:

$\tau_{sc3} = \tau_{sc2}\{1-0.157(T_e/T_i-1)(B_t R^{1.25})^{0.59}\}$.

The standard deviation was reduced remarkably. Here, we find that the confinement time decreases with increasing of T_e/T_i[4].

The impact of T_e/T_i on the plasma transport is investigated using the code TOPICS[5] with the GLF23 transport module. The GLF model suggests that the transport is enhanced for large T_e/T_i domain[6]. Transport simulations for JT-60SA plasmas are carried out in a wide range of plasma conditions, especially focusing on the ratio between heating power for electrons and that for ions. Simulation results will be checked against above scaling analyses.

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Effect of Nuclear plus Interference Scattering on Fast Alpha-particle Orbit and Confinement in DT Plasmas

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To realize the fusion reactor system it is important to understand the fast alpha-particle confinement and slowing-down process in magnetic field, and hence many numerical simulations have been made in various reactor systems [1-3]. In the previous alpha-particle orbit analyses, however, the effect of nuclear plus interference (NI) scattering [4] on the alpha-particle behaviors has not been included. The NI scattering is a non-Coulombic, large-angle scattering process, and a large fraction of the fast-ion energy is transferred in a single event. In actual DT plasmas, some of the alpha-particle orbits would be changed significantly due to the NI scattering. In such a case, confinement of the alpha-particle and plasma heating process may be influenced.

So far we have investigated the effect of NI scattering on plasma burning characteristics and plasma diagnostics scenarios on the basis of Boltzmann-Fokker-Planck model[5]. In this paper we newly incorporate the NI scattering effect into the alpha-particle orbit simulation code system [6]. The NI scattering effect on the alpha-particle orbit in the magnetic confinement device and the influences on the alpha-particle confinement are examined. It is shown that due to the NI scattering effect, fraction of lost energetic alpha-particle to total generation decreases to some extent. The detailed mechanism and effect of plasma parameters are discussed.

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Non-modal renormalized gyrokinetic approach for time-dependent plasma shear flow

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The investigation of the long-time mode of behavior of the regimes of the improved confinement of plasma, where ExB shear stabilization mechanisms play a key role, is the theme of major importance in the physics of the continuous operation of the fusion systems. In spite of the great progress in gyrokinetic treatment of tokamak drift turbulence, there still remain key issues in analytical investigations of the long-time evolution of the turbulence in plasma shear flows.
New approach is developed and applied to analyze a temporal evolution of drift ITG turbulence of inhomogeneous plasma flow across the magnetic field with time-dependent velocity shear in plane geometry. In contrast to typical gyrokinetic treatment, we use a method of shearing modes (also named as 'nonmodal' approach) which consists in transforming of Vlasov-Poison (V-P) system to sheared (in space and velocity) coordinates convected with shear flow and accounted for the effect of the waves stretching by shear flows. This transformation is followed by transformation to renormalized guiding center and Larmor orbit coordinates, which accounted for the effect of turbulent scattering of particles by ensemble of shearing modes with time dependent wave numbers. This renormalization procedure precedes the standard gyroaveraging and retains effect the gyration angle turbulent scattering in gyroaveraged V-P system. That approach allows strong time-dependent velocity shear rate to be treated, while retaining the effect of turbulence.
Developed theory reveals characteristic time scales and corresponding evolutionary processes, which display different statistic properties. It was obtained, that main nonlinear effect, which is absent in conventional gyrokinetic theory, but is responsible for extremely fast suppression of the instability, is effect of the gyration angle scattering by sheared perturbations.
The renormalized non-modal quasilinear equation, which accounted for the effect of ions scattering by ensemble of the sheared drift waves on the ion equilibrium distribution function, is obtained. It is applied to the investigations of the multistage processes of the temporal evolution of the anomalous transport of ions density and energy in plasmas with time dependent velocity shear. The time dependent diffusion and thermal conductivity coefficients, which display fast decay with time due to velocity shear, are obtained.

P1-8

Reduced Size LHD-type Fusion Reactor with D-shaped Magnetic Surface

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LHD-type magnetic configuration (Heliotron configuration) is produced by continuous helical and vertical coil systems. LHD experiments have achieved an average beta value of 5 % without beta collapse. Hence, LHD-type fusion reactor is indicated the possibility of compact, high-output and high-performance, in steady state.

LHD-type reactors require a large major radius to attain the self-ignition condition with a sufficient space for blankets. To reduce the major radius size, small helical pitch parameter (γ) configuration is considered[1]. Small γ configuration requires relatively high current density for helical coils. The volume of the last closed magnetic flux surface(V_{lcfs}) of vacuum field becomes small, because the magnetic surface of the present LHD is located close to the inboard-side of vacuum vessel wall, with elliptic shape.

In the present paper, we propose a new winding law for the helical coils: helical coils winding along the geodesic curve of a torus. It become possible that the magnetic axis of the magnetic surface with excellent characteristics coincide with the center of the helical coils. The magnetic surface in the vertically elongated cross section becomes D-shaped, so that wide blanket space become possible. This indicates that the LHD-type fusion reactor can be achieved in reduced size.

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P1-9

Another approach to problem of controlled nuclear fusion

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Currently, in research on controlled nuclear fusion tokamaks play the dominating role. Here, the essence of the approach to solving problem is to “detach” the plasma from the walls ($\beta\beta \ll 1$), to create a “vacuum” gap between the plasma and the wall, thereby preventing the destruction of the walls. However, we know from experience that at $\beta\beta \ll 1$ the first wall is rapidly destroyed. The wall damage and disruption instability are those vulnerable aspects which do not permit, with complete confidence, considering tokamaks as a foundation for future fusion power setups. Hence, there naturally, arises a question: if the chosen way to develop tokamaks, i.e. permanent increase of their sizes (and cost) the only one correct? Or are there other, more simple solutions? We offer an alternative approach to the problem, which is appropriate neo-magnetic confinement, when unlike the conventional magnetic confinement the magnetic field is mainly used for thermal insulation of the plasma, while the pressure of plasma itself is confined by a high-pressure gas. This will make it possible to reduce significantly the magnitude of the magnetic field and simplify the complex. Besides, the formation of a powerful plasma-gas layer near walls [1] should contribute largely to better confinement and more reliable protection of the wall. Such confinements regimes are possible due to the fact that during fusion burning in a magnetic field, the “naturally boundary” of the plasma, where the plasma temperature formally vanishes, is at a finite distance from the region with a maximum temperature. As will be show in the paper, under conditions of neo-magnetic confinement we deal with interesting and practically important phenomenon of a self-sustaining magnetic field predicted by author [2]. This phenomenon can provide effective thermal insulation of the plasma. In the framework of the proposed concept, we have formulated a simplified physical model that allows an approximate analytical study of the problem of a self-sustaining fusion reaction. We consider the possibility of solving a particular important task – the creation of a deuterium reactor. By the example of the toroidal geometry of the plasma, we consider the physical conditions of the plasma, and present the estimates of the possible parameters of a fusion power reactor, using DT-, DD- and D3He- reactions.

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Laser nuclear fusion on the basis of mini-torus shaped target

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Successful solutions of the problem of harnessing nuclear fusion energy based on inertial confinement depend, first and foremost, on the choice of the target design. Early studies in this approach [1] proposed several simple target variants. There were considered target where a heavy shell was used to confine hot high-dense plasma spread [2] and proposed to amplify the reaction energy yield by initiating a thermonuclear combustion wave [3]. Also, there was considered the effect of magnetic field [2, 4] on the thermal insulation of plasma in the case of a cylindrical geometry of the target. According to estimates [2] a self-sustaining thermonuclear reaction in the target with a shell required a laser pulse energy of ~ 1 MJ if the magnetic field B ~ 1 MG was used. At the same time, for the cylindrical target, the technical difficulties due to the need to "close" the butt ends of the target arise. After Teller's group [5] published the concept of programmed laser compression of the spherical target, practically all researches switched to this line. In view of their preliminary estimates, only ~ 10 kJ was required to ignite some ideal target, and this made such models very attractive. More careful subsequent investigations showed that, in fact, a much higher energy of ~ 4-5 MJ was required to ignite the target, if to take into account the real physical conditions. Herewith, no less than 100 laser beams are required to provide the symmetrical compression of the target. Thus, the assumed reactor facility becomes too complicated.

In this paper a simple target in the form of a mini-torus is considered, in which the heavy shell is used to confine the plasma spread. A significant decrease of heat losses is achieved using an external magnetic field and/ or as result of a self-sustaining magnetic field that is generated in the plasma [5]. A mini-torus has exceptional advantages in comparison with other geometrical shapes (sphere, cylinder etc.) because the magnetic field can be most efficiently used here. In fact, the mini-torus is an ideal theoretical model, since this target has not butt ends and, thus, the heat losses of the plasma itself in the axial direction are excluded as compared with, e.g., a cylinder target. On the basis of an approximated physical model, the conditions for "ignition" are discussed. It is shown that the ignition energy can be significantly reduced up to 50-100 kJ if the mini-torus shaped target with magnetized plasma is used. It is especially important that, in contrast to targets with programmed compression, the considered mini-torus target does not require a complicated multi-beam system of laser pulses.

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Design of a vacuum pumping system for a closed helical divertor for steady state operation in LHD

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Requirements for a vacuum pumping system in future fusion reactors are helium ash removal, impurity reduction in the core plasma, neutral particle pumping for active control of the peripheral plasma density profile. Efficient particle pumping is also required for steady state operation in fusion reactors.

Two test modules of a closed helical divertor (CHD) were installed in the inboard side of the torus of the Large Helical Device (LHD) in the last experiment campaign (2010y). The CHD consists of three components: slanted divertor plates, a roof shaped dome and target plates. It successfully enhanced the neutral particle density behind the dome by more than one-order of magnitude compared to that in the original open divertor [1]. In the next experimental campaign, a vacuum pumping system will be introduced behind the dome along the space between two helical coils in the inboard side. Because the minimum distance between the divertor plates to the inlet of the vacuum pumping system is very short (~0.1m), heat loads by radiation from the divertor plates and by thermal conduction due to neutral particles have to be carefully considered for designing the pumping system.

The particle pumping efficiency and the heat loads on the pumping system were investigated using finite element method based software for multi-physics analysis (ANSYS) and a fully three-dimensional neutral particle transport simulation code (EIRENE) [2]. It proposed a possible candidate which is composed of water cooled (WC) blinds, liquid nitrogen (LN2) cooled chevrons and gas/liquid helium (LHe) cooled panel. The investigation indicates that the heat load onto the LHe cooled panel due to radiation is not dominant. It also shows that buffer plates on the water/LN2 cooled components and on the inner vacuum vessel significantly reduce the heat load due to the conduction of neutral particles without serious degradation of the pumping efficiency [3]. The detailed analysis using the neutral particle transport simulation code predicts that a modified structure of the WC components enhances the pumping efficiency. An enlarged WC blind inlet and a slit at the bottom of the WC component are effective in improving the pumping efficiency. This paper reports calculations of the pumping efficiency for modified WC components and propose an improved design of the pumping system for steady state operation in LHD.

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1D model study on the effect of impurity radiation cooling on LHD SOL plasma

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Recent progress in fusion plasma performance gives us perspectives on future fusion devices like DEMO and helical-type reactor, FFHR[1]. The increasing heat load to the plasma-facing wall in these devices, however, can exceed the engineering limit if the same operation is carried out. The material of divertor plates in reactor-class devices, i.e. tungsten, reduces carbon impurity in the plasma. Although it is beneficial for the energy confinement, the heat load has to be reduced by other means. One of possible candidates is gas puffing of impurity such as neon and nitrogen. The cooling effect of the gas puffing with impurity radiation has been demonstrated in the 14th experimental campaign of LHD in 2010 to 2011. It is found that the temperature and particle flux onto divertor plates is reduced significantly. The physical understanding of the plasma response is desirable to apply the gas puffing to the future operation. The cooling effect of neon on the SOL plasma is studied here by means of 1D two-fluid model based on our previous model[2]. Since sustaining radiation is observed after an instantaneous gas puffing in the experiment, the model treats a steady state plasma. The radiation cooling is modeled as a source term in the balance equation of energy flux. Another role of the impurity as a plasma source is ignored under the assumption of small amount of impurity, e.g. a few percent. Numerical solution for various connection length from meters to kilometers in the LHD ergodic layer was obtained. A significant reducing effect on particle flux and temperature was recovered. The physical mechanism of the reduction is discussed in the paper.

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Effect of divertor target geometry on detached plasma in divertor simulator TPD-SheetIV

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In order to achieve the high performance plasma for high power and long pulse operation, the divertor design for stable detached plasma formation should be optimized to handle such high heat and particle fluxes. In JT-60SA or ITER, the bottom part of the divertor chamber forms a distinct corner (V-shaped) with the target was proposed for high gas conductance between the divertor legs. Recently, the closed Helical Divertor (HD) in LHD is planned to accomplish an active neutral particles control to improve plasma confinement and to sustain high performance long pulse discharges. Therefore, the divertor target geometry to be compatible with the high performance plasma is one of key significant issues on detached plasma.
In this study, we present the experimental simulation of the deivertor target geometry via detached plasma formation of hydrogen plasma in a linear divertor plasma simulator TPD-SheetIV[1]. Three types of target geometry (V-shaped(long), V-shaped(short) and oblique targets) have been investigated with steady state or pulse flow plasma. Measurements of the electron density, n_e, the electron temperature, T_e, and Balmer series emission intensities were carried out in hydrogen detached plasma with hydrogen gas puff. The ionization and recombination events are discussed by Collisional-Radiative model. In the V-shaped target, detached condition with high radiation loss is produced easily in steady state and pulse flow plasma. Also, T_e and Q rapidly decrease with increasing gas pressure. The V-shaped target enhances the recycling and detachment plasma is attained there effectively.

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P1-15

Calculation of Ion Orbits in the Divertor/Dipole Regions of the GAMMA10 A-divertor

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The design of the divertor mirror cell, which is to be installed in the present GAMMA10 tandem mirror, (called tentatively the GAMMA10 A-divertor) is in progress[1,2].
The divertor mirror cell plays a role of exhausting the ions diffused radially in the central cell to the dipole region, where the divertor plate is installed.
That is, the GAMMA10 A-divertor is planning to perform the simulation experiments such as the divertor of a big torus.
The calculation of ion orbits, therefore, is important to know how ions move into divertor mirror cell and then go into dipole region through x-point.

The numerical code for calculation of ion orbits was developed for the above purpose.
In order to determine the magnetic field $\boldsymbol{B}$ at the local ion position, numerical meshes were introduced.
The numerical meshes correspond with the flux coordinates (ψ, θ, χ), where one axis χ is taken along a magnetic field line and $\boldsymbol{B} = \nabla\psi \times \nabla\theta = \nabla\chi$.
The great merit of these meshes is to be able to introduce the electrostatic potential easily as a function of ψ, θ, χ.

The energy and angular momentum of ions in the code are conserved with good enough accuracy.
The 10,000 ions at the mirror throat initially were followed in the GAMMA10 A-divertor with a divertor plate of the dipole region.
Almost all ions enter the dipole region due to the non-conservation of magnetic moment around x-point.
The ions stay near x-point for a long time and penetrate into the dipole region irregularly and then hit the divertor plate finally.
The mean loss time of ions to divertor plate is a few msec depending on the size of the divertor plate.

As a summary the code was developed in order to investigate the behavior ions in the divertor/dipole regions of the GAMMA10 A-divertor and the behavior of ions is being made more and more clear.

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P1-16

Effects of Radial Losses of Particle and Energy on the Stability of detachment Front in a Divertor Plasma

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Operation with partially detached divertor plasmas [1] is considered to be a promising way to reduce divertor heat loads. Although many theories and modelings of PDD in tokamak devices have been developed so far, physics of formation and stability of the detachment front has not been fully understood yet. Recently the cross-field transport of plasma was found to be an important factor for the formation of PDD [2] and for the stability of the detachment front [3]. In the present paper we further study the impact of cross-field transport on PDD plasmas. We extend the previous works in [3] where a “two-layer” one-dimensional model for PDD plasmas was proposed. Particularly, we analyze effects of the radial loss of particle and energy toward the first wall from a SOL-divertor region, which were not taken into account in the previous work. Radial diffusive fluxes of plasma particle and heat in a SOL-divertor region towards a first wall, Γ_{loss} and q_{loss}, can be described in terms of the cross-field particle and heat diffusion coefficients, D and χ, and the radial decay lengths of the plasma density and temperature, λ_n and λ_T. In the one-dimensional model framework in [3], such plasma particle and energy losses can be described as sink terms in the plasma-fluid equations. As for the neutral particles, their spatial profile in the direction parallel to the magnetic field was given by a simple exponential decay model [3]. In the present work we take into account the radial transport of neutral particles. We describe this neutral particle transport with a characteristic radial transport length λ_{N0} which is an input parameter in the present work. Comparison of the numerical results with the experimental results in ASDEX [4] is also presented.

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P1-17

Hydrogen Isotopes Retention in the Tile Gaps of JT-60U

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Evaluation of fuel retentions in the tile gaps, or side surfaces of plasma facing tiles, is urgently needed for modeling a fuel retention build-up in fusion devises. In this study, retention of hydrogen isotopes (Hydrogen (H) and Deuterium (D)) on the side surfaces of plasma facing carbon tiles used for first wall in JT-60U were measured by TDS.
The tiles were used in experimental campaigns from 1992 to 2004, and their plasma facing surfaces were predominantly eroded. Discharges in JT-60U were mostly with deuterium (D) plasma (DD discharge). At the end of the each series of the DD discharges, discharges with hydrogen (HH discharge) were performed to remove tritium (T) retained on the plasma facing surface. For the measurement of the H and D retentions by TDS analysis, two samples were taken from each side surface of the tiles. One was from the entrance side of the gap (front side) and the other from the bottom side of the tile (bottom side).
TDS spectra obtained for the samples both the front and bottom sides showed broad desorbed peaks of HH, HD and DD with each maximum at around 900 K. Integrating the total desorbed amount for the present TDS spectra, hydrogen isotopes (H+D) retentions in the samples of the front and bottom sides were $1\sim3 \times 10^{22}$ H+D atoms/m^2 and $0.8\sim1.9 \times 10^{22}$ H+D atoms/m^2, respectively. Comparing with previously obtained TDS spectra for plasma facing surface of the tile, the H+D retentions on the side surfaces were appreciably smaller. Interestingly, D/H ratios in the retention on both the side surfaces and the plasma facing surfaces were nearly the same with a value of ~1. In previous works [1], it has been shown that retained D on the plasma facing surface is partly replaced by H during the HH discharges and the value of D/H ratio is simply determined by the tile temperature. Furthermore, the H+D retentions ($1\sim3 \times 10^{22}$ H+D atoms/m^2) on the side surfaces were similar and/or smaller than those on the plasma facing surface ($\sim3 \times 10^{22}$ H+D atoms/m^2). All these results suggest that the temperature rise of the first wall was volumetric and the temperatures of the plasma facing surface and side surfaces were nearly the same and accordingly the retention mechanism for both would be similar, though the former were eroded while the latter re-deposited. Although hydrogen isotopes (H and D) retention on the all side surfaces of each tile was smaller than those on the plasma facing surface, the integrated retention on the side surfaces over whole first wall tiles would be ~1/10 of the total fuel retention of JT-60U.

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P1-18

Tritium retention to the first wall of JT-60U

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For economic and safety reasons, tritium accumulation on plasma-facing wall of nuclear fusion reactor is strictly limited. Since the carbon tile is one of the main candidates for the use in vacuum vessel components of fusion reactor, it is important to get a clear understanding of how tritium is trapped by carbon.
In this study, the amount of tritium retained in the graphite tiles used as the first wall of JT-60U was measured by the full combustion method. All the samples used here were the surface tiles located at the equatorial position of the first wall. Each examined tile was exposed to a different DD discharge period respectively in operation years of 1991 to 2003. At that time, T atoms produced by DD reactions were ranging from $\sim 10^{18}$ to $\sim 10^{20}$ T atoms. For experimental preparations, samples were cut into $10 \times 10 \times 1$ mm^3 size pieces including plasma facing surface or taken from bulk. The samples were heated up to 1000°C for the complete combustion and purged by humid argon gas. Released tritium(HT) from each sample was transferred to a copper oxide bed at 350°C to convert HT to the tritiated water (HTO) and collected to the dual pure water bubblers. It was found that T was only retained near plasma facing surfaces sides of the carbon tiles and the amount of retained T increased from $\sim 10^{11}$ to $\sim 10^{13}$ T atoms/cm^2 with increasing the exposed discharge period of the tiles. Integrating the T retention with the total surface area of first wall tiles, T retention rate in the first wall was determined to be 13 % of the total tritium production rate calculated by neutron production rate. This value agrees with the previous estimation by a tritium imaging plate technique [1] All these confirms that the part of energetic tritium produced by DD reaction were implanted within a few μm from the eroded plasma facing surface and retained there without being replaced by H during HH discharges performed after each DD discharge campaign.

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Toroidal distributions of gas retention and impurity deposition for 13th experimental campaign in LHD

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Plasma wall interactions (PWIs) such as impurity deposition and fuel gas retention are key issues for the improvements of confinement and safety operation in fusion devices. Material probe analysis is very useful for evaluation of the PWIs in the fusion devices [1]. In the present study, PWIs during 13th experimental campaign in the Large Helical Device (LHD) were investigated by the material probe analysis. The toroidal distributions of gas retention and impurity deposition were discussed based upon the plasma discharges and the positions of plasma heating.
The material probes made of silicon and 316L SS were placed on the wall at every toroidal sector. These were exposed to the plasmas in the 13th experimental campaign. The numbers of hydrogen and helium main discharges were 3200 and 1200, respectively. Hydrogen and helium glow discharge cleaning (GDC) was conducted for 133h and 36h, respectively. After the campaign, the material probes were taken out and then the change of surface properties was analyzed. The toroidal distributions of gas retention, impurity deposition and change in surface morphologies were investigated by using thermal desorption spectroscopy, Auger electron spectroscopy and scanning electron microscope, respectively.
The boron deposition was large at near the anodes, and the deposition of carbon was large in the vicinity of tangential NBI.
The hydrogen desorption behaviors largely depended on the impurity deposition on the material probe. Namely, most of material probes with a low impurity deposition, showed a sharp desorption at around 950K. The material probes with thick carbon and boron, showed a broad peak at the lower temperature regime.
The amount of retained hydrogen was large near the anodes for GDC. This result indicates the hydrogen is implanted mainly during the GDCs. On the other hand, the amount of retained helium was small near the anodes, perhaps due to the small discharge times.

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P1-20

Impurity Effects of Hydrogen Isotope Retention on Boronized Wall in LHD

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Boronization is thought to be one of useful wall conditioning techniques in the Large Helical Device (LHD) of the National Institute for Fusion Science (NIFS) for steady state plasma operations. It is considered that boron is deposited on the first wall with the impurities and energetic hydrogen isotopes escaped from plasma will implant into the boron film. Therefore, the retention behavior of hydrogen isotopes in impurities containing boron film should be elucidated for the evaluation of hydrogen isotope recycling in fusion devices. In our previous studies, it was found that the retention of hydrogen isotope trapped by impurities was enhanced by hydrogen plasma exposure due to existence of impurities from plasma into the boron film, which would strongly affect on hydrogen isotope retention. In this study, the chemical states of the boron films boronized in LHD were investigated and the correlations between the chemical states of the boron films and the hydrogen isotopes retention behaviors were discussed.
The LHD boron films were prepared by boronization in LHD and preheated to remove residual impurities and hydrogen. Thereafter, the samples were exposed to hydrogen plasma in LHD or implanted to D_2^+ at Shizuoka University. The chemical states of boron films before and after hydrogen plasma exposure or D_2^+ implantation were measured by X-ray photoelectron spectroscopy (XPS) and the hydrogen retention in the boron films were investigated by Thermal Desorption Spectroscopy (TDS).
The XPS measurement showed that the impurity concentrations of oxygen and carbon were 7 and 15%, respectively. The deuterium was bound to boron as B-D-B and B-D bonds for pure boron film, whose desorption temperatures were around 500 K and 700 K, respectively. In the case of LHD boron film, the additional deuterium desorption stages were observed around 800-1100 K, assigning to desorption of deuterium trapped by carbon as B-C-H and oxygen as B-O-H bonds. 70% of total hydrogen retention for the hydrogen plasma exposed boron film was retained by impurities. After exposure to hydrogen plasma, the atomic concentrations of impurities were increased, resulting in the increase of hydrogen retention trapped by carbon and/or oxygen. It was considered that hydrogen isotope retention would increase with increasing the operation period of D-D plasma, which leads the increase of tritium inventory in LHD.

P1-21

Tritium Retention on Stainless Steel Surface Exposed to Plasmas in LHD

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From viewpoints of controlling particle balance in the reactor core and the amount of tritium retention in the PFM's, it is of great important issue to clarify the retention behavior of tritium in PFM's exposed to plasmas. In this study, we have studied retention and release behavior of tritium on the surface of stainless steel samples, which were specially fixed on the plasma-facing surface in the Large Helical Device (LHD) and were exposed to plasmas for one cycle.
Small sample plates used in this study were fixed at four different locations on inner walls of LHD before start in each experimental cycle (12th, 13th and 14th cycle): namely, the locations are 1.5U, 5.5U, 6.5L, and 9.5L. The samples were analyzed by X-ray photoelectron spectroscopy and laser Raman spectroscopy. Behavior of tritium retention and release was examined using &beta-ray induced X-ray spectrometry.
Results of surface analyses showed that surfaces of all samples were covered with the deposition layers, and those of 1.5U and 5.5U samples were thinner than that of 6.5L and 9.5L. Constituent elements (Fe, Cr and Ni) of the stainless steel were observed in the all samples. Large difference in the order of tritium retention was observed in the location of sample and experimental cycle. The order was as follows: 1.5U<6.5L~5.5U<<9.5L for the 12th cycle, 1.5U<5.5U<9.5L<<6.5L for the 13th cycle, and 1.5U<5.5U<9.5L<<6.5L for the 14th cycle. Maximum tritium retention in the 6.5L and 9.5L samples was ten times and more in comparison with other samples.

Dust Dynamics Released from Plasma-Facing Components in HL-2A Tokamak

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Dynamics of a dust particle with a spherical shape is studied in the SOL/divertor plasma of the HL-2A tokamak with a lower single-null closed divertor configuration [1]. The dust particle in relatively low temperature plasmas at SOL/divertor region is negatively charged due to the high mobility of plasma electrons. The charging time is as fast as the order of nanoseconds, which is much faster than the dynamics time of the order of milliseconds. This allows us to apply the equilibrium charge of the dust particle as the charge state [2]. In this study the forces by ion drag forces due to Coulomb scattering and due to ion absorption, gravity, Lorentz force are taken into account. In order to calculate the dynamics of the dust particle the background plasma and magnetic field configuration parameters are given by the B2-EIRENE code [3], where the power transferred from the core plasma is 200 kW and the corresponding density is 7.5×10^{18} m^{-3}. The plasma density and temperature around the mid-plane of the SOL/diverter are the order of 10^{18} m^{-3} and 10 eV, respectively, where parallel flow speed is around 10 km/sec.

So far, mechanisms and positions of dust generation are not clarified. Therefore, in this study it is investigated the dynamics of dust particles released from the strike points of the separatrix on the divertor plates made of copper. The dust particles are ejected from the strike point to the poloidally isotropic directions. Results indicate that the dust dynamics released from the inner divertor plate and the outer one is quite similar. For example, the 90 % of released dust particles of the radius 3 μm with higher speed than 10 m/sec are redeposited on the first wall. The rest are redeposited on the dome region, where the release direction is almost parallel to the divertor plate. On the other hand the 20 % of the slower dust particles with 1 m/sec are returned to the divertor plates due to the friction force of the poloidal plasma flow toward the divertor plate, which is the order of a few km/sec.

The effects of the dust particles released from the other plasma-facing components such as the entrance of the closed divertor to the core plasma are discussed. These results will be compared to the experimental observations in HL-2A, which might lead to the modelling of the dust dynamics to be modified.

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Deuterium retention and desorption behavior of co-deposited carbon film produced by deuterium arc discharge with carbon electrodes

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Evaluation of tritium retention in the vacuum vessel is an urgent issue for ITER and next step fusion devices, since it greatly effects on safety and limits the operational schedule. Graphite material, such as carbon fiber composite (CFC), is candidate for divertor target. The carbon material easily erodes by fuel hydrogen and the eroded carbon particle co-deposits on the walls with fuel hydrogen. It is known that the fuel hydrogen retention of the co-deposited carbon film is significantly high. Thus, it is important to investigate the fuel hydrogen retention of the co-deposited carbon film, in order to evaluate the in-vessel tritium inventory.
In fusion devices, the co-deposited carbon film is produced not only on plasma-facing surface but also at the gaps between armor tiles, and the removal of the fuel hydrogen in the gaps is not easy. In the present study, the co-deposited carbon film was prepared on the gaps between the tiles, in the deuterium arc discharge with carbon electrodes [1]. The deuterium retention and desorption behavior of the co-deposited carbon film was examined. The deuterium concentration was obtained as functions of discharge pressure and gap position. The relation between the deuterium desorption behavior and microstructure of the co-deposited film is also discussed.
The deuterium arc discharge was conducted with discharge current of 50A and voltage of 12-16V. The sample holder with gap width of 8-50 mm and gap depth of 100 mm was placed in the discharge chamber. The discharge pressure varied from 0.8 to 36 Pa. During the discharge, both the carbon atom and the deuterium deposited on Mo substrates placed on the wall of the gap. The substrate temperature measured by using thermocouples during the discharge was up to 330 K. After the discharge, deuterium concentration and desorption behavior of the co-deposited carbon films were investigated with thermal desorption spectroscopy (TDS). The crystal structure was examined by Raman spectroscopy.
The deuterium concentration in atomic ratio of D/C increased with the discharge pressure. In the case of 36 Pa, the D/C reached 0.9. The deuterium retained in the film desorbed in the forms of D_2, HD, CD_4 and C_2D_4. The desorption behavior of D2 depended on the degree of deuterium concentration. In the film with a low deuterium concentration (D/C = 0.1), a single desorption peak appeared at around 1000 K, while in the film with a high deuterium concentration (D/C = 0.9), three desorption peaks appeared at around 670, 850 and 1000 K. The crystal structure with a high deuterium concentration was amorphous. The film with D/C of 0.9 partly contained polymer-like structure. The gas species desorbed also depended on the deuterium concentration. In the film with a low deuterium concentration, major gas species were HD and D_2 while in the film with a high deuterium concentration, the desorption amount of CD_4 and C_2D_4 increased.
The film thickness decreased exponentially with increase of the gap depth. In the case of narrow gap width, the film thickness rapidly decreased with increase of the gap depth. The clear relation between the deuterium concentration and the gap depth was not observed in the present co-deposited carbon films.
The results obtained in this study offers important data for the evaluation of in-vessel tritium inventory.

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P1-24

Deuterium retention in graphite and its removal by inert gas glow discharge

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Glow discharge cleaning using inert gas has been widely used in order to reduce the fuel hydrogen retention so far. Carbon fiber composite is used for divertor tiles in ITER because of its low atomic number and its high fracture toughness to high heat load. However, the removal amount of fuel hydrogen due to the glow discharge cleanings in graphite has not been investigated in detail. In the present study, the amounts of retained or desorbed deuterium for graphite during the glow discharge [1] were measured and the effect of the inert gas glow discharge on the reduction of the deuterium retention was investigated.
The graphite sheets of IG-430U were used for a liner in the glow discharge apparatus. In order to implant the deuterium into the liner, the deuterium glow discharge was conducted between the liner cathode and copper anode. After that, the helium, neon or argon glow discharge was conducted in order to reduce the deuterium retention. The gas pressure before the discharge was kept 8 Pa using a mass flow controller. The liner temperature was room temperature. The amounts of retained or desorbed deuterium were measured by residual gas analyzer. The changes in partial pressure of gases containing deuterium were measured by a quadruple mass spectrometer. The obtained results were compared with the cases of other material liners, 316L stainless steel and tungsten.
The amount of retained deuterium in the graphite by deuterium glow discharge was very large, compared with that of 316L stainless steel or tungsten. This is owing to high capability of carbon to trap the deuterium. It was found that the reduction of the deuterium retention by helium glow discharge was most effective among the inert gas species. The implantation depth of helium ion is highest among these inert gas ions, so that the impact desorption of helium ion might have well desorbed the deuterium.
In addition to the reduction of deuterium retention in graphite by inert gas glow discharge, the inert gas retention is also presented.

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P1-25

Hydrogen Atom Behavior with Structural Change of Carbon Materials

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BCA-based Monte Carlo simulation codes, for example, EDDY, TRIM, and ACAT have been applied to studies for plasma-surface-interaction on physical sputtering, erosion, and ion implantation. These codes employ the Monte Carlo method, which implies that the target material is assumed to have an amorphous structure of randomly placed atoms.

In our previous research [1], we extended BCA-based simulation to any structure of target materials, including monocrystals, polycrystals, crystals with defects, and amorphous crystals. In that code, however, atoms are injected into the same initial material for each injection. The structural change of target materials is cleared prior to the next injection.

Under bombardment of plasmas whose thermal energies are higher than several eV, it is expected that the structures of target materials dynamically change. For example, when single crystalline graphite material is bombarded by hydrogen plasmas, the surface of the graphite becomes hydrogenated amorphous carbon by the chemical and physical reaction of hydrogen atoms. This structural change affects the processes of retention, reflection and sputtering.

In this study, BCA-based simulation is further extended to handle cumulative structural changes of materials. This extension reveals the following issues:
・Time evolution of the structural change of target materials;
・The effect of structural change to retention, reflection and sputtering yields;
・The effect of retention of atoms in target materials to retention, reflection and sputtering yields.
We perform the extended BCA-based simulation and investigate hydrogen behavior with structural change of carbon materials.

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Time Development of Agglomeration Process on Carbon Dusts in Hydrocarbon Plasma

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Carbon dusts in a fusion device are expected to absorb a considerable amount of tritium atoms and increase tritium inventory in the device. The source of these dusts is carbon made first walls of the fusion device. Carbon walls emit carbon clusters and hydrocarbons by hydrogen plasma-carbon wall interaction, which diffuse into plasma and agglomerate to form dusts in low temperature edge plasma. The time development of carbon dust formation in edge plasma should govern the quantity and characteristics of formed carbon clusters, and affect the amount of overall tritium retention. Thus, the fundamental processes governing the dust growth is important for future fusion operation. Information on the size and density of carbon dusts in the edge plasma, as well as the density distributions carbon hydride molecules are indispensable to clarify the dust growth process in edge plasma. As one promising way to quantify densities of carbon hydrides the infrared absorption spectroscopy system can be installed to observe edge plasma. A small experimental setup has been designed and built to investigate that this scheme can be effective to study dust formation by hydrogen plasma-carbon wall interactions.
The designed experimental system confines hydrogen plasma inside of a graphite container. The discharge region is 80 mm diameter and 300 mm long. A carbon hollow cathode is installed inside of the all graphite wall container that serves as the anode of the discharge. The graphite made plasma container, together with graphite made hollow cathode can be taken out of vacuum system which is a 120 mm diameter and 350 mm long cylindrical glass chamber. Carbon dusts have been confirmed formed on the graphite wall by running Ar discharge in the present experimental set up. An optical path is opened on both sides of the discharge container, so that infrared absorption spectra can be measured. The observed infrared spectra will be compared with vibration energy levels of large hydrocarbon molecules calculated with an ab initio molecular orbital code Gaussian 03. The particle size distribution of the dust formed in our experimental setup will also be compared with molecular dynamic simulation.

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P1-27

Retention of Hydrogen Isotopes in Tungsten Exposed to ELM Plasma

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Tungsten has been selected as a candidate material of the ITER divertor plate because of its high melting temperature, good thermal conductivity and highly resistive nature against erosion due to high threshold energy for sputtering [1]. Divertor plates in a fusion device are exposed to high intensity heat fluxes of energetic particles. Many experiments [2,3] have indicated that tungsten retains tritium and deuterium under their bombardment of hydrogen isotope plasmas. The resultant retentions of the isotopes lead to the safety and economic problems, and should be minimized for future fusion reactor operations. Thus, efforts are being made to find out which factors is the most influential upon hydrogen isotopes retention in tungsten.

We have been studying hydrogen isotope retentions in tungsten utilizing a dynamic Monte Carlo simulation code ACAT-DIFFUSE [4]. So far, a uniform temperature model is assumed to evaluate the retentions. However, large temperature gradient near the surface of plasma facing component will largely alter the transport of hydrogen isotopes in tungsten. To realize more accurate simulation calculation under high heat flux exposures as observed in edge localized mode (ELM) conditions, the code has been modified to take into account of the diffusion of hydrogen isotopes by including subroutines of heat transfer calculation based upon the finite element method. Some initial results of the code will be introduced.

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P1-28

Deuterium Trapping by Radiation Defects in Tungsten

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Tungsten is recognized as a primary candidate of plasma-facing material. Radiation defects such as vacancies and voids can act as trapping sites against hydrogen isotopes, and hence tritium accumulation in neutron-irradiated tungsten has to be carefully evaluated. A common way to investigate the trapping effects of radiation defects is to damage tungsten by heavy ions and measure hydrogen isotope retention after exposure to high flux plasma in divertor simulators (linear plasma machines). Heavy ions, however, can create defects only in near-surface regions, and hence defects induced by irradiation of heavy ions may be modified by exposure to high flux plasma. If such modification by plasma takes place, the effects of neutron irradiation in the bulk of tungsten cannot be simulated well by implantation of heavy ions. In the present study, deuterium was introduced in tungsten damaged with heavy ions by exposure to gas or low energy neutral atoms at moderate flux, and trapping of deuterium was examined by thermal desorption technique. The obtained results were carefully compared with those provided by plasma exposure to understand the effects of damage modification by plasma.

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P1-29

Simulation of Deuterium Retention in Tungsten Exposed to Divertor Plasmas

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An important issue in the choice of plasma-facing materials for ITER is the tritium retention, which must be reduced for safety reasons. Dominant retention mechanism for tungsten (W) is implantation. Implanted T atoms are highly mobile at elevated temperatures and they are retained in radiation damage sites or defects of the bulk. In the previous study, models for diffusion, including trapping and detrapping, of thermalized atoms in the bulk and the surface recombination were incorporated into a binary collision approximation code, EDDY.

In the present study, deuterium (D) retention characteristics for the inner and outer divertor plates, made of W, are discussed in a detached plasma condition of ITER. The edge plasma parameters used for simulation are calculated with B2-EIRENE. The magnetic field lines intersect the plate at very shallow angles between 1 and 3 degs, where the gyro-motion and electric field influence the angle of incidence of the impinging ions. Therefore, the mean angle of the ions at each position of the plate is calculated using particle-in-cell plasma simulations for different magnetic angles. The mean energies of D ions exposed to the plate are tens of eV or less. At such low energies, the reflection coefficient at the relevant energy and angle ranges to the divertor plates is calculated by a molecular dynamics code, which reveals that most ions impinging the plate are reflected and thus much less implantation of the ions. Therefore, the incident fluxes to the plates are reduced according to the reflection coefficient.

At the inner divertor W plate, at the early stage of the discharge, the number of mobile D atoms reached a steady-state value due to a balance between implanted and re-emitted atoms. During the discharge, most of implanted D atoms were retained in traps and after saturating available traps, inward diffusion and subsequent trapping increase the inventory. Most of trapped atoms remained in the bulk even after the discharge, whereas the number of mobile atoms is steeply reduced. Further detailed calculation and discussion on surface processes at such low energy will be an important issue for the use of W as a divertor plate in ITER and devices beyond.

Simulation Study of Tungsten Exposed to Helium / Neon / Argon Based on Binary Collision Approximation

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It has been experimentally observed that nanostructured tungsten is formed due to exposure of helium plasma [1], and that the fomation of nanostructure differs from irradiation nuclear species, say, helium, neon, or argon.

We applied a binary collision approximation (BCA) based simulation [2] to the noble gas plasma exposure to tungsten material and found that the BCA model could qualitatively explain differences of nanostructure formation between irradiation nuclear species.

In this presentation, we report simulation results based on the BCA model of tungsten exposed to helium, neon, and argon plasma, and refer to comparisons between simulation and experimental results.

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P1-31

Cross-Field Motion of Plasma Blob-filaments and Related Particle Flux in an Open Magnetic Field Line Configuration on QUEST

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Transport of particles and heat by collective motions at the boundary of magnetically confined plasmas is an important problem for steady state operation in controlled thermonuclear fusion research [1]. Recently experimental and theoretical work suggested the importance of ‘blob’ type fluctuation in plasma edge, scrape-off layer (SOL). Blob causes huge amount of energy and particle losses [2]. Blob-filaments have been observed by combined measurement with a fast camera and a movable Langmuir probe in an open magnetic field line configuration of electron cyclotron resonance (ECR) heating plasma in QUEST [3]. The experimental observation show that the intermittent visible blob-filaments extended along field lines do correspond to over-dense plasma structures and propagated across the field lines to the outer wall. The typical radial velocity of the blob-like structure, V_b, was measured as ~ 1 km/s at the intermediate area and increased along major radius, R (outward). The cross field movement of blob was dominantly driven by the E × B force. The radial velocities of blob-like structures in QUEST corresponds to roughly 0.02 ~ 0.07 of the local sound speed (C_s). V_b/C_s decreases with local plasma density and plasma temperature.
The large blob structures, occurring only 10% of the time, can carry more than 60% loss of the entire radial particle flux. This indicates blob structures will cause the dominant losses channel in the SOL regions. It strongly interacts with the first wall, and thus the first wall might be damaged. The particles of main plasma loss by blob radial motion and thus the particle confinement time will be affected by blob losses. The radial velocity, size of the blob and radial flux driven by blob showed good agreements with the results obtained by sheath-connected interchange theoretical model. In most of big tokamak devices, ITER, JET, DIII-D etc., the blobs closure schemes are also satisfied with sheath-connected interchange model. A roughly prediction for the blob regime of ITER can be given according to the QUEST and other tokamak device's results.

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P1-32

Analysis of end-loss-ion-flux for application studies of the plasma flow from the end-mirror exit of GAMMA 10.

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To perform a good simulation experiment for the divertor system of the actual fusion reactor, the test plasma should achieve very high energy and density. As one of the device which is capable of generating such high-performance plasma, the world's largest tandem mirror device, GAMMA 10 has been noted. Since the mirror-magnetic field is open-ended, there exists an axial loss flux of the plasma which is called the end-loss flux witch is quite useful for a diverter simulation experiment. More detailed study of the end-loss flux is needed before we proceed to the divertor simulation experiment. The end-loss ion energy analyzer (ELIEA) is used and the energy distribution and the current density of the ion flux are examined in the west end-cell of GAMMA 10. The objective of this study is to obtain the knowledge of characteristics and controllability of parameters in the end-loss ion flux.

GAMMA 10 is an axisymmetric tandem mirror which consists of a central cell, two anchors and plug/barrier cells. Parameters of the loss plasma will be changed due to the change of parameters of the main plasma. In the GAMMA 10 experiments, the power of ICRF heating input and the amount of gas-puffing particle supplied to the main plasma are varied and the change of end-loss ion flux is investigated. In a typical GAMMA 10 plasma (without confining potential), the ion current density is measured to be about 0.4mA/cm^2. It is found that the least square method fitting to the measured energy distribution of the end-loss ion flux works better if we assume that the energy distribution to be double-component Maxwellian rather than single-component Maxwellian. Here, we defined the effective temperature of ions from the ratio of two components with respect to the total ion flux. The evaluated effective ion temperature is about 400eV in typical plasma of GAMMA 10. The particle flux of the end-loss ion can be increased by increasing the amount of gas-puffing. However, injecting too much gas causes the degradation of the energy or collapsing of the plasma. Therefore, in order to increase the particle flux to the west end-cell without increasing the amount of gas, the additional heating in the east anchor-cell using a newly installed ICRF antenna is proposed and tested. With an input of the new-ICRF in the east anchor-cell, significant increase of the ion-particle flux was ofserved. In the paper, detailed results will be described and discussed.

P1-33

Material dependence on plasma shielding induced by laser ablation

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In nuclear fusion research, it is one of the important themes to study the damage on plasma facing components (PFC). For example, expected heat load are 10 to 100 MW/m^2 at magnetic fusion energy (MFE) divertor and 10^9 W/cm^2 or higher at inertial fusion energy (IFE) first walls at plasma densities 10^{19} /m^3 or higher and temperature 1 eV to keV or even to MeV. With the plasma thermal loading, any solid material may be ablated, therefore, some way to protect the reactor wall is necessary. However there are some active functions to protect themselves like vapor shielding [1] and plasma shielding effects. Shedding light on this effect can be of use.
Plasma shielding is studied using the experimental set up “LEAF-CAP” (Laboratory Experiments on Aerosol Formation by Colliding Ablation Plumes) [2]. In this set up, the ablated Tungsten or Molybdenum plumes by a third harmonic beam of YAG laser (6ns, 10Hz) can be crossed orthogonally. This laser ablation scheme can cover pretty wide range of intensity regime as a heat source at its laser focus spot from 10^3 to 10^{14} W/cm^2 and the ablated plasma plume temperature and densities can range from 0.1 to a few eV from 10^{11} to 10^{13} /cm^3.
Plasma plume will give a material deposition when the quartz thickness monitor is placed in the forward direction. When another plasma plume crosses this plasma plume, the original plume may collide at the crossing point and may be absorbed through the collision. If any energy of the plume is absorbed, the plasma is considered to be shielded. Shielding rate is reduced from the deposition rates with or without the shielding plasma plume.
As a result when shielding Tungsten plume intersects the original Tungsten plume, the deposition rate decreases from 5.2×10^{13} to 3.7×10^{13} particles/pulse/cm^2 at the laser energy density 10 J/cm^2. About 30% Tungsten are intercepted by the intersecting plume. In the case of Molybdenum, the reduction of deposition rates is similar, and shielding rate is 20%. In the ICCD camera fast movie observation shows that there is clearly a collision of two plasma plumes. In case of Carbon, the collided part stagnates for more than microsecond.
The dependence of deposition of Molybdenum on laser energy density is higher than that of Tungsten. However, the dependence of shielding rate is regardless of laser energy density, 20 to 30%. The dependence of the material considering for the PFCs is measured by comparing values of quartz thickness monitor.
These experimental data indicate the possibility of plasma shielding effects to heat load flux at the divertor or first wall.

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Fundamental study on hydrogen retention in carbon aerosol relevant to inertial fusion reactor

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【Introduction】Chamber wall of an IFE reactor is subject to the bombardment of high-energy, short -pulse X-ray, unburned DT and He particles from implosions. Therefore solid wall could be ablated leading to the formation of aerosol in the target chamber. If aerosol is formed in the hydrogen atmosphere including tritium and attached to chamber wall, this affects its radiation safety. Therefore, it is extremely important to understand the carbon aerosol behavior of retaining hydrogen using a laboratory-scale experimental setup.
【Experimental】In our work, a YAG laser beam (1064nm, 6nm, 10Hz) is first converted into the 3rd harmonic (355nm), and it's half-split into two beam and line focused to two arc-shaped carbon targets positioned in vacuum <10^{-3} Pa. Therefore ablation plasma plumes collide with each other and create aerosol in the center-of arc region. The laser energy density is 10J/cm^2/pulse relevant to the IFE reactor situation.[1] The chamber is filled with hydrogen gas as the substitution of tritium. Aerosol is collected in different hydrogen pressure condition 10, 20, 50Pa, and its hydrogen retention is examined by TDS.
【Results and discussion】Examining carbon aerosol created in non-hydrogen atmosphere by Q-mass, C_2^+, C_3^+, C_4^+, C_5^+, C_6^+ cluster were detected. As the laser energy density was increased, low number were clusters decreased, and high number ones were increased. The result indicates low number clusters were used for composing high number molecules through the collision processes. In fact, collecting carbon aerosol created in laser energy density 2.2J/cm^2/pulse and observing its surface and structure by digital microscopy, SEM and TEM, CNTs were clearly found in the collected samples. It indicates that the combination reaction of cluster occured because CNT can be made by C_5, C_6. Colliding carbon plume experiments have also been conducted in the LEAF-CAP facility with a backfilled hydrogen gas. Under the conditions which already explained "Experimental", carbon aerosol have been collected and examined its hydrogen retention by TDS. Expressing result as atomic ratio of H/C, it is proportional to hydrogen gas pressure, and indicates maximum H/C≈0.28 at the sample collected in gas pressure is 50Pa. This is still low compared to the saturated co-deposition ratio of H/C~0.4.[2] However our result indicates a possibility to retain fair amount of tritium in IFE chamber.

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Optical emission characteristics of an atmospheric Ar/N2 microwave plasma jet in relation to superhydrophilic surface treatment of stainless steel

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An atmospheric-pressure microwave plasma jet of Ar/N2 was used to enhance the wettability of stainless steel samples. The contact angles of the samples were investigated in terms of varying gas ratios of Ar and N2 in the plasma. Optical emission spectroscopy was used to determine and compare the relative populations of the active species in the Ar/N2 plasmas. The observed wettability changes were related to the contact angle measurements.The morphology, elemental composition, and roughness of the treated sample surfaces were examined using SEM, EDX, and AFM, respectively.

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Proposal of New Type Diplexer for ECCD System

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Neoclassical tearing modes (NTM) can be controlled by the local current drive in a magnetic island with electron cyclotron current drive (ECCD) [1]. For improving a stabilizing efficiency of NTM, the fast directional switch (FADIS) had been developed [2, 3].
In this paper, the new type diplexer as a fast switching device of high power millimetre wave is proposed for ECCD system in fusion devices. The principle is a ring resonator of oversized corrugated circular waveguide. The two half mirrors (dielectric disks or slotted antennae) are integrated in a ring resonator. The switching operation of a diplexer has been simulated with the code using finite differential time domain (FDTD) method. The switching operation of the diplexer by frequency shifting has been confirmed in the cases of dielectric disks or slotted antennae, where the loss caused by mode conversion and misalignment in a resonant ring are considered. A diplexer for 1 MW short pulse at a frequency of 170 GHz is designed and fabricated.

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ECCD Experiment Using an Upgraded ECH System on LHD

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Electron cyclotron current drive (ECCD) is an attractive tool for controlling plasmas. ECCD can control the plasma current and rotational transform profiles, which affect magnetohydrodynamic (MHD) activity. In tokamak-type plasma confinement devices, the effectiveness of ECCD for stabilizing the neoclassical tearing mode has been demonstrated. Also, for stellarators/heliotrons that do not need plasma current for plasma confinement, the current profile control capability enables fine plasma control. In the large helical device (LHD), ECCD experiments have been performed by using EC-waves with the frequency of 84 GHz. The capability of long-pulse operation of the 84 GHz gyrotron and the wide setting range of EC-wave beam direction enabled by a 2-D widely tiltable mirror antenna at a bottom port were suitable feature for ECCD. However, the available injection power for the long-pulse operation was low, and the maximum driven current was ~10 kA with 100 kW and 8 s power injection. These years, an upgrade of ECH system on LHD is ongoing. A high-power, long-pulse gyrotron was newly installed. The oscillation frequency, the output power for several seconds and the output power for continuous operation are 77 GHz, 1.5 MW and 0.3 MW, respectively. The mirror antenna system at a horizontal port used for the 77 GHz gyrotron also furnishes widely tiltable plane mirror so that the 77 GHz ECH system can be used for ECCD experiments with high-power and long-pulse. An ECCD experiment with 775 kW injection power was performed in the second harmonic condition, that is, the magnetic field on the magnetic axis was 1.375 T. The discharges were started with 300 ms pulses of 84 GHz and 82.7 GHz EC-waves, and the 77 GHz waves of 8 s pulse duration sustained the plasmas. The EC-wave beam direction was scanned toroidally, keeping the beam direction aiming at the magnetic axis in X-mode polarization. In spite of the change in the EC-wave beam direction, the electron density and the plasma stored energy were kept nearly the same values for the discharges, ~0.3×1019 m–3 and ~30 kJ, except for the plasma current. The plasma current showed a systematic change with the change in the beam direction, and at an optimum direction the plasma current reached its maximum, ~40 kA.

Improvement of output power and electric efficiency for 77 GHz gyrotrons by externally controlled anode voltage in LHD

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In the Large Helical Device, three 77 GHz gyrotrons with each output power of over 1 MW are operational for plasma generation and heating [1]. The MIG type of the 77 GHz gyrotrons is triode and the anode voltage (V_A) can be controlled using the preset waveform. High power and high efficiency operation was successfully achieved such as the output power of 1.41 MW with the electric efficiency of 49.1 % for 1 sec by applying the two-step-rise V_A. In the first step period, enough small V_A was applied for adequate duration (> 50 ms), in which the electron beam flows inside the tube but RF was not generated. After that, V_A was increased to optimum value for oscillation startup.
The charge neutralization of the gyrotron electron beam in the cavity [2] is considered to be a key for the improvement of the electric efficiency by the stepwise V_A applying. In order to examine the effect of the beam charge neutralization on the gyrotron oscillation, the gyrotron output characteristics of the stepwise V_A operation were investigated and compared with those of the normal operation. In the stepwise V_A operation, the output power increased with the extension of the first step duration and was saturated for the longer duration than 50 ms. This indicates that the drop of the accelerating voltage recovered through the charge neutralization process in the first step period and the space charge was enough neutralized for ~50 ms. In the normal operation with the rectangular V_A waveform, it was found that 5~6 kV higher acceleration voltage was required to generate the similar output power with the stepwise V_A operation with the first step duration of 100 ms. The increment of the beam acceleration voltage in the normal operation case corresponds approximately to the drop of the acceleration voltage due to the space charge effect. These results suggest that the gyrotron operational parameters are able to be optimized for the fully accelerated electron beam by applying the stepwise V_A with the adequate duration of the first step, leading to the improvement of the output power and the electric efficiency.

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Experimental results of electron Bernstein wave heating on the LHD and consideration for a helical type fusion device

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As a heating method in fusion devices, electron cyclotron heating (ECH) has the great advantage that the launcher of the electron cyclotron (EC) wave can be placed apart from the plasma. In an envisioned high density operation in a helical type fusion reactor, the EC wave cannot access the electron cyclotron resonance (ECR) layer because it is shielded by the cutoffs. ECH by the electron Bernstein wave (EBW) is expected in such a high density operation, because the EBW has no density limit in propagation and is absorbed at the ECR layer by cyclotron damping.
In the large helical device (LHD), two methods of EBW excitation and heating have been studied by launchers placed in ports apart from the plasma. One is to inject the extraordinary (X-) mode extremely obliquely from the high field side [1]. This method is applicable even the electron density is less than the cutoff density. In the plasma where n_{e_bar}= 5 x $10^{19}m^{-3}$, when the 84GHz, 0.43MW EC wave was launched, increase of the stored energy (W_p) and the ECE signals were observed. (n_{e_bar} : line averaged electron density.) The result of the ray tracing calculation suggests 10~30% of the launched power was absorbed as the EBW and the rest is absorbed as the X-mode. Almost all of the launched power can be absorbed by this method. In the helical type fusion device, this method may be applicable in its high density operation if the electron density is less than two times of the cutoff density. Another is to inject the O-mode from the low field side obliquely to the external magnetic field [2]. Precise adjustment of the launching direction is required whereas this method has no density limit. Increase of W_p and the second X-mode ECE signals were observed when the 77GHz, 1MW EC wave was launched into a plasma where n_{e_bar} = 9 x $^{19}m^{-3}$. However the estimated heating efficiencies were about 15%. Improvement of the heating efficiency including changing the EC wave frequency and position of the launcher is now under consideration. Application of these methods to a helical type fusion device is also under consideration with reference to these experimental results.

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ICRF heating experiment using phasing antenna in LHD

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Ion cyclotron range of frequencies (ICRF) heating is one of the heating methods for fusion reactor and has been carried out successfully in large helical device (LHD). In LHD, ICRF heating has been used for long pulse plasma discharge and the pulse length of more than 54 minutes was achieved so far [1]. Ion heating was observed in minority ion heating scheme and high-energy ion tail more than 2 MeV was detected. During the long pulse discharge by the ICRF heating, the divertor temperature near the antenna location increased especially and the plasma was terminated by influx of metallic impurity into the plasma. It is important for the steady state operation by the ICRF heating to reduce the local heat load near the antenna and edge interaction caused by RF waves.

We designed [2] and installed new type of ICRF antenna in LHD in order to control the wave number in the direction of the magnetic field line. The antenna has two straps side-by-side and can change the parallel wave number by phasing of the strap current. We call the new antenna as HAS antenna named after the antenna shape and Hasu-Seigyo (wave number control) in Japanese. HAS antenna has an improved design by experience of the experimental results of existing poloidal array antenna. There is a possibility of control of sheath potential at plasma edge region generated by antenna electric field by phasing of antenna. It is expected to suppress the unnecessary acceleration of edge particles and reduce the impurity generation caused by RF injection. Carbon side protectors were replaced by CFC material. The shape of current strap and Faraday shield and water-cooling channel were re-designed in consideration of the steady state operation.

In the experiment, the phasing effect was studied in the minority heating experiment. The heating efficiency of 0-pai phasing was higher than that of 0-0 phasing. Achieved electron density was higher in 0-pai phasing in the same heating power. These experimental results are better than that of the poloidal array antenna. We will conduct the high power and long pulse discharge and evaluate the antenna performance in steady state operation.

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Study of ICH Heating Power Profile Evaluation Method in LHD Type Plasma

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In the LHD, long plasma discharges with a temperature of several keV are maintained for more than one hour mainly by ICH. In these plasmas, a total plasma heating power by the ICH is evaluated from a loading resistance of ICH antennas. With the heating power, global confinement properties are investigated, and it is found that a global energy confinement performance in the ICH plasma is almost same with that in the ECH and NBI plasmas. However, the ICH heating power profile and the local thermal transport property in the LHD has not been clarified yet. To establish the method for evaluating the ICH heating power profile and the local thermal transport property is an important issue for development of the long time plasma discharges with the higher confinement capability, especially in the comparatively high density plasma.
In order to evaluate the ICH heating power profile in the LHD experiments, a full wave field solver, TASK3D/WM, on the basis of the Maxwell's equation is being developed [1]. On the other hand, in a LHD type fusion reactor, which is about three times larger than the LHD, the ICH heating power profile was evaluated by a simple ICH heating power evaluation code based on the Ray-tracing method [2].
As a pre-stage for establishing the ICH heating power profile evaluation code in the LHD, the ICH heating power profiles in the LHD type plasmas, of which sizes are larger than that of the LHD, taking the finite beta effects on MHD equilibrium into account are calculated by using the TASK3D/WM and the Ray-tracing code. In this presentation, the validity of the Ray-tracing code by changing the device size is shown. The effect of the plasma kinetic effects on the propagation and absorption of the ICRF wave will be discussed.

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Improvement of ICRF antenna loading in the minimum-B configuration on GAMMA 10

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On the GAMMA10 tandem mirror, waves in the Ion-Cyclotron Range of Frequency (ICRF) are used for the plasma production, the ion heating and keeping the magneto-hydrodynamic (MHD) stability. High-β plasmas in the minimum-B configuration at the anchor cells supply the MHD stabilizing effect to the whole of GAMMA 10. It is necessary for high performance plasmas that the ions are heated efficiently in the anchor cell. In the standard discharge, ICRF waves (around 10MHz), which have resonance layers in the anchor cell, are excited by NAGOYA Type-III antennas installed in the central cell and propagate to the anchor cell. It is difficult to control only the heating effects in the anchor cell because these waves are used also for the plasma production in the central cell.

In this paper, the direct anchor heating by using antennas in the minimum-B configuration is studied. Firstly, a simple bar-type antenna has been used in the experiment and confirmed the stabilization of GAMMA 10 plasmas [1]. However, the loading of the bar-type antenna has been found to be very small values (< 20%). To calculate the loading resistance of antennas in the minimum-B configuration, a three-dimensional full wave numerical code is used, which is developed by one of the authors (A. Fukuyama). To improve the loading, a new antenna has been designed and replaced in the anchor cell. The loading of the new antenna is compared with that of the old bar-type antenna in the experiment and calculation. According to calculation, changes of the shape from the single strait line to double elliptic arcs and the antenna location are effective for the improvement. In the initial experiment, the loading of the new antenna is improved up to 50%. Secondly, the phase control between the anchor antenna and the NAGOYA Type-III antenna in the central cell has been used and confirmed clear anchor heating effects. For the effective anchor heating, it is planned to calculate wave excitation in the minimum-B configuration and optimize the anchor antenna configurations.

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Quasi-Optical high purity HE_{11}-mode Exciter for Oversized Corrugated Waveguide Transmission

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Oversized circular corrugated waveguides will be mainly used for electron cyclotron heating 20 MW transmission lines in the ITER burning fusion-plasma experiments. The higher order modes as well as a main HE_{11} mode are excited due to the misalignment of the coupled beam and due to the manufacturing error of the waveguide components. These cause undesirable events such as overheating and/or arcing in the transmission lines. The modes excitation is concerned with the transmission of a high power millimeter wave without remarkable transmission losses and arcing events. Therefore the aim of the study is to experimentally evaluate of the unwanted mode excitation. A scalar feed horn antenna for Gaussian beam excitation was designed with a numerical analysis based on the method of moment. A field radiated from the designed antenna was measured and confirmed to be Gaussian filed distribution. The HE_{11} mode in the waveguide is matched to the TEM_{00} mode (Gaussian beam) in a free space with the conversion efficiency of 98%. The Gaussian beam (TEM_{00} mode) was often used for the HE_{11} mode excitation in the waveguides. The HE_{11} mode purity over 99% should be prepared to study the loss and the mode excitation with 1% in the transmission line. The conversion loss from Gaussian beam into the HE_{11} mode comes from the differences of kurtoses in field distributions.
Two Gaussian beams may be prepared to excite a high purity HE_{11} mode. The beams should have opposite phase curvatures but the same size. These were combined at the waveguide aperture with phase matching technique. The excited HE_{11} mode purity results 99.6% in theoretical analysis. We designed a quasi-optical system for the HE_{11} mode exciter system. It is composed of the scalar feed horn antenna and quasi-optical mirrors. The beam from the antenna was separated with a half mirror. Each beam was led into ellipsoid reflective mirrors for focusing and expanding. Finally the beams were integrated with the half mirror again and were shaped with the desired beam profiles at the aperture. The theoretical analysis and experimental configuration was considered for the high purity HE_{11} mode exciter. In the paper, the system design and experimental results will be reported in detail.

This work was realized in the collaboration between FURUKAWA C&B and Kyushu University.

Finite-Difference Time-Domain Simulation on Millimeter-Wave Propagation in Corrugate Waveguide

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We have developed the Finite-Difference Time-Domain (FDTD) simulation with the Drude-Lorentz model [1] to investigate optical phenomena related to the electromagnetic response of metals using near-field scanning optical microscope (NSOM) [2-5]. As one of our FDTD simulation results, the oscillating pattern for the transmission image of a gold nanorod using NSOM was obtained by the FDTD simulation [3].
In the present paper, we propose the FDTD simulation, which has been developed in the nano optics, to estimate the propagation loss of the millimeter-wave through the corrugate waveguide. In this proposal, we pay an attention to the following fact: the ratio of the wave length to the typical structure spatial scale between the nano-optics phenomena and the millimeter waveguide phenomena becomes the same order.
First we model the corrugate waveguide, which is actually made of Al, as the perfect electric conductor (PEC). Using the PEC, it is expected to investigate the behavior of the electromagnetic wave propagation qualitatively. Moreover, as the next step, we also include the Drude-Lorentz model in the FDTD simulation, to obtain the propagation loss inside material in the waveguide.

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Study of magnetic reconnection heating in UTST measured by Doppler spectroscopy

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The merging start-up of spherical tokamak (ST) plasma has been accomplished in the UTST device by using external poloidal field coils [1]. The plasma current up to 180 kA has been attained with assistance of the center solenoid (CS) coil. In this merging start-up method, significant ion heating caused by magnetic reconnection is expected to help the formation of high-beta equilibrium. The reconnection outflow velocity measured by carbon impurity spectroscopy showed good agreement with the bulk plasma’s Alfven velocity, suggesting the conversion from magnetic energy to plasma kinetic energy [2].
We have developed a Doppler spectroscopy system using an 8 × 8 channel PhotoMultiplier Tube (PMT) assembly to observe the temporal evolutions of ion temperature and flow. Line-integrated emissions from the merged ST plasma were analyzed by using the developed spectroscopic system. Increase of the bulk ion temperature was observed by helium impurity spectroscopy at the same time as the plasma merging was completed. The heating efficiency of the plasma merging will be discussed quantitatively.

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Effect of Gas Fueling Control on Plasma Performance in Heliotron J

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The optimization of gas-fueling scenario has been studied to improve the plasma performance in Heliotron J, which is a low magnetic-shear device with an L/M = 1/4 helical coil ($<R_0>$ = 1.2m, $<B_0>$ < 1.5T) based on the helical-axis heliotron concept [1]. This paper discusses the effects of the gas-fueling scenario on the plasma performance based on the experimental observations.
A gas fuelling by supersonic molecular beam injection (SMBI) was applied to ECH/NBI plasmas [2]. When the plasma density is increased with conventional gas-puff fueling (GPF), the stored energy W_p becomes saturated or decreased at a density level, probably due to the edge cooling caused by excess neutrals. Local fueling with a short pulse of SMBI can increase the core plasma density avoiding the confinement degradation. In a combination heating condition of ECH and Co-NBI, W_p reached about 50% higher value than the maximum one achieved under the similar heating condition with GPF. On the other hand, it was observed that a large increment of W_p after turned-off of an intense GPF in a high power NBI plasma [3-4]. By applying an intense GPF, W_p initially increases with increase of the plasma density and then decreases, as usually observed in a case of excess GPF. If the intense GPF is stopped at a proper timing during the NBI pulse, however, W_p started to recover and finally reached to a much higher value than the first peak of W_p. This suggests that the plasma density/temperature profile during the decreasing phase of W_p is changed to a preferable one for NBI.
Detailed analyses of these observations will give us an insight into the optimization of fueling scenario.

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Improvement in the fueling efficiency of supersonic gas puffing in LHD

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Two kinds of improvement in the fueling efficiency of supersonic gas puffing (SSGP) have been observed in the Large Helical Device (LHD). The fueling efficiency improves when the edge-density is kept high and a strongly hollow density profile is maintained. The fueling efficiency also improves suddenly when the target plasma is close to the density limit. In SSGP, fuel gasses are ejected through a fast solenoid valve equipped with a Laval nozzle. Because of the solenoid valve and the Laval nozzle, higher working pressure of < 8 MPa and more convergent gas flow than those of ordinary gas puffing are available. Three solenoid valves with different Laval nozzles are installed in the SSGP device. The efficiency of SSGP, which is defined as a ratio of increase in the total number of electrons in the plasma due to SSGP to the number of electrons in the injected gas, is about 10 - 30 % in the case of hydrogen gas injection. As the electron density of the target plasma before SSGP injection increases, the fueling efficiency decreases. After SSGP injection, a hollow electron density profile is formed. The particles are transported from the plasma edge region to the core region through the diffusion process. The fueling efficiency strongly depends on the edge electron density before SSGP. In order to obtain high efficiency, a steep inward gradient should be maintained. This can be realized when the edge electron density before SSGP is low enough, or the amount of density increase due to SSGP is higher than the edge electron density before SSGP. In the case of helium SSGP, where high recycling from wall is expected, inward density gradient was maintained for a longer time than in the case of hydrogen SSGP. The fueling efficiency of helium SSGP was higher than that of hydrogen SSGP. In the case of hydrogen SSGP, abrupt increase in the density increase rate was observed when the pulse-length of SSGP was extended. As a result of this, the fueling efficiency was improved two times even though the difference in the number of supplied particles was less than 20 %. Similar phenomena are frequently observed when the target plasma is close to the density limit.

Production and Measurement of Negative Hydrogen Ions in a FET based RF Ion Source

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High performance ion sources are required for neutral beam injection heating in the next generation fusion devices. The use of metal-oxide-semiconductor field effect circuit transistors (MOSFET) for the RF power supply enables us to operate RF ion sources with lower frequency (<1 MHz) that is feasible because of large skin depth for high-density plasma production. We have developed a compact FET based negative hydrogen (H^-) ion source [1]. The characteristics of the ion source have been studied, and the source performance has been improved.
The ion source consists of a driver, an expansion region, and a set of three beam extraction grids [1]. Helmholtz coils are attached around the driver for the improvement of the plasma confinement. A pair of permanent magnets is attached near the plasma grid to generate a magnetic filter field. A Cesium oven is attached to the expansion chamber, and the temperatures of the oven and the plasma grid can be controlled. The typical RF operation frequency is 0.3-0.5 MHz. Plasma parameters were investigated in the driver, and upstream and downstream of the expansion region. The H^- ions near the plasma grid were measured by laser photodetachment.
The effects of the axial magnetic field and the magnetic filter field were investigated. The electron density increased as the axial magnetic field increased implying improvement of the plasma confinement in the driver. The electron densities attained more than 10^{19} m^{-3} in the driver region and 10^{18} m^{-3} in the expansion region. The electron temperature downstream of the filter field drastically decreased to ~1 eV, and the density ratio of H^- ions to the electron doubled to 10%. The H^- beam was extracted at ~15 keV. The RF power loss at the isolated transformer should be minimized for effective plasma production. The developed transformer recorded high RF power transmission efficiency of almost 90%. Cs effects on the H^- production and H^- beam extraction will also be presented.

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Study of Edge Fluctuation Characteristics using Multiple Langmuir Probes in Heliotron J

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Turbulence behavior around last closed flux surface (LCFS) and the contribution to transport are key issues in plasma fusion researches since performance of plasma confinement is substantially influenced by turbulence characteristics around edge region, as observed for transitions to improved confinement modes. A variety of phenomena regarding to edge fluctuations, for example, transport barriers, meso-scale structures and blob transport, have been focused in many years. Recently, various advanced analysis techniques and multi-channel diagnostic tools have markedly progressed and contributed to the characterization of the phenomena in these studies.
In an advanced helical axis heliotron device, Heliotron J, physics of edge fluctuation is being studied using multiple Langmuir probes installed at different toroidal/poloidal positions[1-3].
In the plasma heated by neutral beam injection, it was found that low frequency fluctuation component less than 100 kHz plays a dominant roll in turbulent transport and characteristics of the fluctuation changes depending on the radial location. From the perspective of statistical approach, the distortion of probability density function, characterized by the parameters of kurtosis and skewness, was observed in signals of ion saturation current and floating potential, which may be relative to the intermittent transport such as blob transport. Moreover, phase relationship between ion saturation current and poloidal electric field fluctuations, which is directly linked with particle flux caused by fluctuation, varies with probe location although the power spectra of the fluctuations at different radial positions are similar. These results may imply that mechanism of particle flux has different characteristics at different location around LCFS. In the presentation, details of characteristics and the spatial dependence of the fluctuation behavior around LCFS will be shown and discussed.

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Configuration of flows in a cylindrical plasma device

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The investigation of flow velocity fields in turbulent magnetized plasmas is a crucial issue for the understanding of a wide range of phenomena. Recently, the observation of toroidal flows has increased the interest in studies of parallel momentum flux and its link to plasma turbulence. Detailed investigation of flows is a challenging task, especially in bigger devices, due to the complexity of the plasma configuration and the limitations for diagnostics. Laboratory plasma devices with their high reproducibility and easier experimental access can help to understand fundamental relationships between flows and turbulence.
In the cylindrical magnetized plasma device PANTA, the ion flow in the helicon argon plasma is studied with three different methods. The time delay estimation technique can be used to measure the apparent phase velocity of fluctuations between two electrostatic probes. This technique will take into account not only the mean plasma flow, like the $E\times B$ rotation in azimuthal direction, but also the propagation of waves [1]. The cross-field convective particle flux due to plasma instabilities is measured using three-tip Langmuir probes. Finally, the mean ion flux is studied in azimuthal and axial direction using Mach probes [2]. Measurements of mean axial ion flow measurements show a reversed flow direction in the outer part of the plasma column in high neutral pressure conditions. In the same conditions an inwards directed radial cross-field particle transport is observed close to the radial position of axial flow inversion. This contribution presents results of flow studies in different discharge conditions and discusses possible links between plasma turbulence and flows in the radial, axial and azimuthal directions.

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Time Evolution of Power Spectrum Density during Spontaneous Transition in Cylindrical Magnetized Plasma

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Several stable turbulent states exist in magnetized plasmas and fast transitions between them are observed both in linear and toroidal plasmas. Understanding of such turbulent states and transitions is an important issue both in physics of non-equilibrium system and in nuclear fusion research. In the Large Mirror Device-Upgrade (LMD-U), which is a low temperature cylindrical plasma device, mainly two kinds of turbulent states were successfully generated depending on the neutral gas pressure condition P_n [1,2], i.e., a state of solitary waves in the high P_n case and that of broadband in the low P_n case. A transition phenomenon between them was studied in detail. Recently, a new transition among states of broadband, which have different turbulent spectra, is found in the Plasma Assembly for Nonlinear Turbulence Analysis (PANTA, the successor device of the LMD-U) in low neutral gas pressure condition at the case of P_n = 1.00 mTorr. Under this condition, the fluctuation spectra and radial profiles transit between two different states. At lower and higher neutral pressure conditions, spectra and profiles are stable. First, we analyzed transition properties for radial profiles. The ion saturation current profile (which can be index of the electron density fluctuation in the PANTA case [3]) changes ∼ 40% within the order of millisecond. The transition propagates axially with a speed of ∼ 1 km/s that is 30–50% of the ion sound speed. Second, we obtained a time evolution of two-dimensional frequency and azimuthal mode number power spectrum at the transition, using short time FFT technique. We discuss the time scale of both growth and decay of fluctuation amplitudes.

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Evaluation of Electron Temperature Fluctuations Using a Conditional Technique on PANTA

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To understand the turbulence structures and their formation mechanism in magnetized plasma, fluctuations are measured with multi-Langmuir probe arrays in the PANTA linear plasma device. Observation of temperature fluctuations is an important issue to evaluate fluctuation-driven heat transport. Langmuir probes are useful tools to measure plasma parameters and electron temperature can be evaluated from their current-voltage (I-V) characteristic curve. However, it is difficult to obtain I-V curves with high temporal resolution.An additional technique to evaluate electron temperature fluctuation is required. One candidate is the conditional averaging method[1][2].

In the conditional method, the I-V curves are re-sampled based on the phase of the fluctuation of ion saturation current or floating potential measured with another probe. From the reconstructed I-V curves, we could obtain simultaneous measurements of electron temperature fluctuation and other fluctuating plasma quantities. In PANTA, correlation between electron temperature fluctuation and floating potential fluctuation was observed using this method in conditions where the fluctuation spectrum has multiple peaks[3] . In this paper we establish the electron temperature fluctuation measurement based on the conditional averaging method. For simplification, we excited a coherent fluctuation which has single peak in spectrum (~3KHz). We extracted correlation between electron temperature fluctuation and floating potential fluctuation and the one between electron temperature fluctuation and ion saturation current fluctuation, and compared the results with the triple probe method. Effectiveness of the conditional method is thus verified in plasmas with two different fluctuation patterns. In addition, we will discuss the application of this method to plasmas where broad fluctuation spectra are observed.

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Magnetic Helicity Injection Experiments for Double-null Startup of the UTST spherical tokamak

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A spherical tokamak (ST) plasma is produced by double-null merging (DNM) method by using two pairs of external poloidal field coils in the University of Tokyo Spherical Tokamak (UTST) device [1]. In the late phase of the DNM formation, a single ST is connected to the external poloidal coil flux, which enables magnetic helicity injection from the helicity source (the coil flux) into the helicity sink(the ST plasma).
We observed the relationship between magnetic helicity and magnetic reconnection by two dimensional pickup coil arrays, which are located in the r-z plane [2].
We performed a parameter scan to check how crucial and effective the plasmoid ejection is for helicity injection. Thus, we investigated on what parameter the plasmoid formation depends and how helicity injected becomes when the formation is changed.

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Modeling of Vacuum Field in Start-up in EAST

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Enough loop voltage and large null field region is necessary for EAST to break down and initiate the plasma current. Eddy currents in conducting structural elements perturb the loop voltage and field not only spatially, but also temporally. Due to the individually powered poloidal field magnets with specially designed compensation coils for the field, currents in each coils have to be adjusted to achieve enough large null field region and desired loop voltage for the breakdown and initial ramp-up. This paper summarized the key factors which influence the plasma start-up[1]. Three algorithms were used to calculate the vacuum field. The first algorithm uses both magnetic flux loop and magnetic probe to calculate the vacuum field by dividing the vacuum vessel into 80 elements and estimating the eddy current distribution. The second algorithm using only the measured values of the magnetic flux loop is used to calculate the vacuum field based on the boundary integral equations[2]. The third algorithm is used to calculate the vacuum field directly from the magnetic sensor signals similarly with Cauchy condition surface method[3]. Calculation results of these three algorithms agree well by comparing the results of the three algorithms, which proves the reliability of the models.

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FDTD simulation analysis for improving the two dimensional electron density distribution measurement by using phase imaging method in GAMMA10

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GAMMA10 is the tandem mirror device utilizing plasma confinement by magnetic mirrors and electrostatic potentials. It consists of the central, anchor, and plug/barrier mirror cells. In GAMMA10, plasma is produced and contained by the ion cyclotron heating. The ion and electron confining potentials are formed by applying the electron cyclotron heating at plug/barrier cells, respectively. Microwave interferometers are placed at each cells for measuring the plasma electron densities. A phase imaging interferometer system is set horizontally to measure upper half of the plasma in the plug cell. It consists of two microwave oscillators (69.85 and 70.00 GHz), lenses, mirrors, and two dimensional detector. The detector position can be moved in the radial or axial directions. The phase imaging method is the method to measure the phase difference between the microwave beam passing through the plasma and the reference microwave beam which depends on the line integrated electron density. We can obtain two-dimensional line integrated electron density distributions by using the two dimensional detector. With application of the Abel transform technique to the line integrated electron density distributions, we can obtain the electron density radial distributions. Normally, the Abel transform is assumed axisymmetric distribution of plasma density. However the plasma profile is not always axisymmetry. Then, using Abel transform method for nonaxisymmetric plasma may have large error in the calculated electron density. The purpose of this study is to optimize the assumption of density distribution of the Abel transform with the use of Finite Difference Time Domain (FDTD) method, and to improve the measurement accuracy of two dimensional density distribution by use of the phase imaging method. We compare the phase differences obtained by experimental measurement and those obtained by FDTD simulation. Next, we use calculated density distribution which was good agreement to the assumption of the Abel transform and FDTD simulation. In this paper, we report the improvement of the electron density distribution measurement by using this method.

Observation of spectral broadening of LH waves in Alcator C-Mod

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Lower hybrid current drive (LHCD) experiment in Alcator C-Mod indicates that lower hybrid (LH) waves are not strongly Landau-damped inside separatix in high density plasmas with the line averaged density over 1020 m-3[1]. While the collisional loss of LH power in the scrape off layer (SOL) is a working model to explain this phenomenon in Alcator C-Mod, little attention has been given to the effect of spectral broadening of LH waves.
In order to characterize the spectral width of LH waves near separatix in Alcator C-Mod, we have up-graded the existing O-mode reflectometer system into a scattering diagnostic to detect LH waves away from the LH launcher. The measured full width at half maximum (FWHM) is order of 1 MHz, which is wider than FWHM of incident waves (500 kHz). This broadened width of scattered signals can be explained by the model proposed by Andrews and Perkins in which multiple scattering of LH waves by low frequency density fluctuations determines the frequency width of LH waves as LH waves cross turbulent SOL [2].
In addition to scattering diagnostics, the spectral width observed by the Langmuir probe indicates that spectral width of LH waves at -35dBc becomes narrower from 17 MHz to 7 MHz as the distance between the inner wall and the last closed flux surface decreases from 1.5 cm to 0 cm in the upper null configuration. Interestingly, strongest non-thermal emission is observed when the spectral broadening is minimum. This broadening is more likely due to parametric decay instabilities (PDIs) because it is typically accompanied with enhancement of sidebands with 30 MHz spacing. A numerical study of PDIs will be performed to estimate the role of spectral broadening on the spectrum of parallel refractive index.

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Identification of MHD Mode Structure using ECE measurement

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It is recognized that the controlling of MHD instability is important issue to improve the plasma performance. However the mode structure is less well understood even though the study using the measurement of SX(soft-Xray) has been done in LHD [1]. In this study, the result of the measurement of mode structure used by ECE (Electron Cyclotron Emission) [2] is reported.

Usually, MHD or high-beta experiment are conducted in the condition of low magnetic filed Bt > 1.0 T in order to clear the effect on the magnetic field. In that condition, ECE measurement cannot be used because of the cut-off in LHD. Instead, SX emission measurement is a helpful tool for fluctuation studies. However it is not adequate to identification of mode structure, in principle, since the obtained SXR emission intensity is included the integral effect along the line-of-sight. If the result of identification of mode structure used by SX is supported by the ECE measurement. This study helps the confinement study of the low filed discharge.

The experiment is conducted using ECE measurement in the condition of Rax=3.6m,Bt=1.5T, which is suitable for ECE to measure in $0.1<\rho<1$. The MHD fluctuation of low-order mode which is seem to be m/n = 1/1 are observed by SX measurement, magnetic probe (MP) and ECE. There is the significant coherence among SX fluctuation, magnetic fluctuation and electron temperature fluctuation around $\rho = 0.85$, which corresponds to the area $\iota = 1$. To clarify the mode structure, we calculated the displacement amplitude ξ. When SX signal can not show the structure clearly, we can see the mode structure around $\rho \sim 0.9$ using ECE data, which is local measurement tool. The coherence radial profile calculated by the ECE shows also the mode width clearer than that of SX measurement.

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Fast, nonlocal, radially coherent changes of T_e during ELM crashes observed by 2D ECE imaging in KSTAR

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Local electron temperature (T_e) changes throughout the entire cross-section that is correlated with large ELM crashes of H-mode plasmas have been observed by a 2D ECE imaging system in KSTAR. During the pedestal collapse, the core region showed a fast T_e drop and recovery in a time scale less than 100 μs whereas the intermediate region showed the opposite T_e trend. Interestingly, the core density and stored energy during the ELM crash were observed to be recovered in a diffusive time scale (~ 100 ms). The preliminary analysis in the perspective of non-locality will be discussed.

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Electron Density Fluctuation Measurement with 3-D Microwave Imaging Diagnostics in LHD

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By measuring the local electron density, the micro and macro instabilities can be investigated experimentally. The Microwave Imaging Reflectometry (MIR) is one of the most promising diagnostics of the local electron density. In MIR the plasma is uniformly illuminated with the microwave, and the plasma wave reflects it at the cutoff density. Imaging optics makes the image of fluctuating plasma on the imaging detector. The electron density fluctuation causes the fluctuation of phase of the reflected wave primarily. Also the amplitude fluctuation is caused by the density fluctuation. Therefore both the phase (&phi) and the amplitude (A) of the reflected wave are detected in this system. The phase is detected with the I-Q demodulator that makes the I (~cos &phi) and Q (~sin &phi) signals.
In the large helical device (LHD), we have developed 3-D MIR system [1]. In this system, the illumination wave consists of 4 frequencies (RF1: 60.410 GHz, RF2: 61.808 GHz, RF3: 63.008 GHz, RF4: 64.610 GHz) in order to observe 4 different plasma layers. The imaging detector is a newly developed 2-D Horn-antenna Mixer Arrays (HMA). By using 4 frequencies illumination and the 2-D HMA, we can measure the electron density fluctuation 3 dimensionally.
In the case of low density plasma, where the cutoff surface is absent, the I and Q signals are sinusoidal, of which amplitude is constant. The phase of I signal is 90 degrees ahead of that of Q signal, as I and Q correspond to cos &phi and sin &phi, respectively. The microwave is reflected at the vacuum vessel wall, and the phase change indicates the electron density change.
In LHD, the edge harmonic oscillation (EHO) is often observed. The feature of EHO is the frequency spectrum that consists of equally separated higher harmonics. EHO is a fluctuation in the edge plasma, and it was observed in the VH mode plasma in DIII-D when the ELM is absent. Usually, LHD plasma has a steep electron density gradient at the edge. The MIR image shows that EHO has a narrow structure along a field line. The EHO is localized near the iota=1.5 surface, where R=4.525 m. Since the channel separation is 2 cm, the width of EHO may be 5 cm along the flux surface. Thickness of EHO in the radial direction is 2 cm. As the phase signals (I, Q) change rapidly, the reflection surface moves rapidly in the radial direction. This is the first 3-D MIR measurement in the world.

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The development of the Infrared TV system for KSTAR

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In the KSTAR tokamak, the thermal imaging camera was developed for to measure heat flow to the poloidal limiter, to the ICRH antenna, to the inboard limiter and to the passive stabilizer and have been installed. The long view ports make a periscope inevitable to get a field of view for diagnosing the internal space of the vacuum vessel and better observation of plasma discharge on KSTAR. The specifications of the Infrared camera(FLIR/ThermoVision SC6000) are 640×512 pixels resolution, full frame rate of 125Hz, and Noise equivalent temperature difference(NETD) of <25mK. We can determine both the location and the intensity of the heating on the poloidal limiter, on the passive stabilizer and inboard limiter during heating power into plasma. The ExminIR which is provide from FLIR systems was used for the camera control and Data storage. Data is stored in digitized forms.

Cylindrical Surface Wave on Periodically Corrugated Metal Cylinder

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Electromagnetic waves from microwave to terahertz wave are demanded for widespread applications such as plasma heating, plasma diagnostics, telecommunication systems, radar systems. In such a wide range, conceivable electromagnetic wave sources are Smith-Purcell free electron laser (SP-FEL) [1,2] and backward wave oscillator (BWO) [3]. In SP-FEL, an electron beam passes near the grating surface and a spontaneous radiation may be excited. SP-FEL in Ref. [1] realizes a stimulated radiation and attract attention to develop a compact source of high intensity terahertz wave. The stimulated SP emission requires a beam bunching by BWO operation at the beam interaction with the surface wave of the grating. The SP devices mentioned are commonly based on plane geometry. Another realistic geometry is cylindrical one, which is very effective for high-power operations of BWOs. An idea of SP-FEL using cylindrical SWS seems to be very attractive to improve the radiation intensity as shown in Ref. [4]. To realize intense SP emissions, properties of cylindrical surface waves should be studied more definitely.

In this work, we study cylindrical surface waves on metal cylinders with rectangular corrugations. The properties of the surface waves due to the corrugation are examined. In the practical experiments, the metal cylinder has a finite length. Reflections from both ends quantize the electromagnetic modes into axial resonant modes. The axial modes formed by the surface wave are measured using an experimental setup based on a network analyzer. There is another kind of cylindrical surface wave caused by surface plasmons, i.e., Sommerfeld wave. We confirm the axial modes by Sommerfeld wave and discuss two kinds of cylindrical surface wave. For Sommerfeld wave, the dispersion characteristics are fixed by the physical property of metal surface. On the other hand, the cylindrical surface wave due to periodic structure can have an arbitrary dispersion [5], and might be very useful to cylindrical SP-FEL and electromagnetic waveguiding up to terahertz wave.

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Vertical profiles of EUV spectral emissions from edge impurity ions observed at different toroidal angles in LHD

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Magnetic field configuration of Large Helical Device (LHD) has three-dimensional structure. The elliptical poloidal cross section mainly formed by two helical coils rotates five times during a toroidal turn. The presence of thick stochastic magnetic field layer surrounding the last closed flux surface (LCFS), so called 'ergodic layer', is a typical character indicating the three-dimensional structure of magnetic filed lines in LHD. The magnetic field lines in the ergodic layer are radially deviated when they toroidally move, while the connection length of field lines is long enough to confine the edge plasma. Therefore, the three-dimensional measurement of edge plasmas is very important to study the transport mechanism of the stochastic magnetic field layer. A space-resolved EUV spectrometer working in range of 50-500A has been recently improved for measuring plasma poloidal cross section at different toroidal angles by adding a toroidal scanning mechanism, although the vertical profile of EUV emissions has been observed at horizontally elongated plasma cross section, in which the observation chord passes through both the inboard and outboard X-points. Then, direct observation of the inboard and outboard X-points is possible at different toroidal angles by scanning horizontal angle of the observation chord in the EUV spectrometer. Vertical profiles of several spectral lines emitted from ergodic layer have been observed at different toroidal angles with vertical spatial resolution of 15 mm. The edge boundary of LHD plasmas is studied near inboard and outboard X-points using CIV (312.4A) with ionization energy of 64eV located in the farthest edge of ergodic layer. Analysis on the edge boundary near both the X-points is presented with vertical profiles of EUV emission lines in the ergodic layer at different toroidal angles.

Extension of wavelength range in absolute intensity calibration of space-resolved EUV spectrometer for LHD diagnostics

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Absolute intensity calibration of a flat-field space-resolved extreme ultraviolet (EUV) spectrometer has been carried out in wavelength range of 80-400A by comparing the radial profile of EUV bremsstrahlung continuum with that of visible bremsstrahlung continuum at 5300A emitted from high-density plasmas in Large Helical Device (LHD). Recently, the EUV spectrometer has been improved to explore the impurity line emissions existing in shorter and longer wavelength ranges beyond the original wavelength range, and the observable wavelength range could be successfully extended to 30-500A. The extension of wavelength range makes possible to measure the first resonance lines of BIV (61.1A), BV (48.6A), CV (40.3A) and CVI (33.7A) in addition to NeVII (465.2A: $2s2p\text{-}2s^2$), which are essential to edge impurity transport study in LHD. The absolute intensity calibration has been also done in such the extended wavelength range of 30-500A with the same manner as before. The result on the calibration is presented and compared with the previous result. The present calibration extended to shorter wavelength range can also improve an alternative approach to Zeff diagnostics based on EUV bremsstrahlung continuum. In particular, the stronger bremsstrahlung intensity in shorter wavelength range can lower the density limit ($n_e \sim 3\times10^{13}cm^{-3}$) in the Zeff measurement. Detailed examination on the calibration factor in shorter wavelength range can also bring an interesting result on the reflectivity of EUV emissions for presently using holographic grating.

EUV Spectroscopy of Highly Charged Tungsten Ions with Electron Beam Ion Traps

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Tungsten is a strong candidate for the material of the plasma-facing components, such as the first wall and divertor plates, in ITER because it has excellent physical properties, such as a low sputtering yield, a high melting point, a low tritium inventory, and so on. However, since high particle and heat fluxes would cause serious damages to such components, tungsten is considered to be one of the most abundant impurities in the ITER plasma. Emission lines of highly charged tungsten ions thus play an important role in the spectroscopic diagnostics of the ITER plasma, and consequently the spectroscopic data of tungsten ions are strongly needed. To date, the spectroscopic studies of highly charged tungsten ions have been carried out by using high temperature (several keV) plasmas [1-2] and electron beam ion traps (EBITs) [1, 3-4] with relatively high electron energies (several to several tens keV). However, there is not yet enough data. In particular, the spectral data of moderate charge state tungsten ions ($W^{10\text{-}30+}$) are quite limited.
For the systematic spectroscopic studies of tungsten ions, we have been using the low-energy compact EBIT (CoBIT) [5,6] developed for spectroscopic studies of moderate charge state ions. We have constructed two CoBIT, and installed them at the UEC and National Institute for Fusion Science. The electron energy range of CoBIT is 0.1 - 2.5 keV, and the accessible charge states are about 10 – 40. In this paper, we present extreme ultraviolet (EUV) obtained with the electron energy range 100 - 2000 eV. Grazing-incidence flat-field grating spectrometers especially designed for CoBIT were used for EUV emission in the 1 - 20 nm range. The electron energy dependence of line intensity and its application to the plasma diagnostics are also discussed.

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Collisional-Radiative Modeling of moderate ionized Tungsten Plasma

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Tungsten (W) attracts the research interests recently because it will be a good candidate for the divertor or wall material in the next generation magnetic confinement fusion reactors due to it's favorable properties. It also has motivated a number of theoretical and experimental investigations of its radiative properties over a rather large temperature range. Plasma electrons collide with tungsten ions that produce excited states and radiate high energy photons in the main core plasma, while it can promote significant cooling in the divertor region.

A detailed fine structure collisional radiative model with thousands of levels is developed to calculate the radiative properties of W^{27+} tungsten plasma based on the atomic data from relativistic configuration interaction calculation by using FAC (Flexible Atomic Code) code. The radiative transition, electron collision excitation and their inverse process are taken into account to construct the rate equation. The intensity ratio between some neighboring transitions in EUV and X-ray region has been found to be sensitive to the electron density which could be good candidate to diagnose the electron density of plasma. The present work should also be of interest to the continuing efforts to model the radiative losses in large magnetic fusion devices.

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Steady state recombining plasma in a radio-frequency plasma device for divertor-detachment study

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Transient increase of high energy particle flux flowing into the divertor region, which is induced by the edge localized modes (ELMs), is a topic of fusion reactor development. Those high energy particles interact with ions and electrons in the plasma. The plasma density and temperature are then concerned to be changed transiently, which might results in reduction of recombination rate in the detached plasma[1].

Use of a table top linear device with an energy controlled ion/electron beam source is expected to support the divertor development in a viewpoint of quantitative understanding of atomic processes. However, compatibility among (1) beams penetrating plasma source, (2) high density plasma production, and (3) low temperature plasma in a test region is an troublesome matter.

In this presentation, development of an steady state recombining plasma using a radio-frequency (RF) plasma device, DT-ALPHA is described. The DT-ALPHA device has the beam penetrating compatibility since the high density plasma is produced using a cylindrically winded RF antenna. The electron temperature required for detached/recombining plasma study is on the order of 1 eV, while typical electron temperature in RF plasmas are about 10 eV. In order to obtain such a low temperature plasma from the RF plasma source, a gas cooling method was applied in the test region. Orifices for gas conductance control and an differential pumping system were installed in the DT-ALPHA device to prevent the cooling gas from affecting plasma production and beam penetration. When the target gas pressure increased up to 50 Pa, change of the source pressure was within a factor of two showing good isolation of plasma production condition from the cooling gas. Then a plasma with about 2 eV of the electron temperature was obtained. Recombining spectrum is also discussed in the presentation.

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P1-67

Initial Studies of Microwave-Induced Atmospheric Plasma Jets Using Optical Emission Spectroscopy

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Preliminary studies on a microwave-induced atmospheric plasma jet were done using optical emission spectroscopy (OES). The microwave plasma jets were generated and sustained by a continuous-wave magnetron. The active species from various atmospheric microwave-induced plasma discharges of Ar, Ar-N_2, Ar-O_2, and He were determined. The effects of the absorbed power and gas flow rate on the relative population of the active species were also investigated. The results were found to be useful in various future plasma applications.

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Optical emission spectroscopy studies on the effect of negatively biased copper shield to low pressure glow discharge plasmas

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This paper investigates the effect of a negatively biased copper shield on the low-pressure DC glow discharge oxygen, water and hydrogen peroxide plasmas generated using a Plasma Enhanced Chemical Vapor Deposition (PECVD) facility. The negatively biased copper shield served as a focusing mechanism to influence the electric field intensities, affect plasma formation on the system and confine the active plasma species responsible for bacterial inactivation. This mechanism can be utilized to improve plasma sterilization techniques on materials with deep cavities and complex geometries. Optical emission spectroscopy was used to identify plasma species present and examine the effect of the copper shield on spectral line intensities of chemically active plasma species involved in sterilization processes.

P1-70

Improved Charge Exchange Measurements in Large Helical Device

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The pellet charge exchange technique (PCX), which is a combination of the compact neutral particle analyzer (CNPA) and an impurity pellet, is a unique method to observe the radial energetic particle distribution. The polystyren pellet of 1 mm-diameter is ablated after the injection to the plasma and it occurs a charge exchange reaction between the plasma ion (mainly proton). There are not only the charge exchange reaction between the hydrogen in pellet and the proton, but also between the ionize carbon in pellet and the proton. We measure the energetic particle enhancement at the resonance layer in the ion cyclotron resonance heating. However the current system has a problem about the poor spatial resolution. According to the CCD imaging with filter, the pellet cloud is expanded to 5 cm or more along the magnetic line. However the main core, which consists of mainly neutral atoms, is still within 2 cm. This means that highly time resolved measurement is meaningful. We prepare 10 times high repetition clock system to obtain the 10-micro seconds resolution. If the pellet velocity is 500 m/s, the spatial resolution of 0.5 cm can be obtained.

Three-dimensional camera system is also introduced to observe the accurate pellet trajectory. At asymmetric tangential injection of the neutral beam injection, the pellet trajectory is not straightforward. More accurate spatial distribution of the energetic particle can be expected by using the system.

Concept for Numerical Calculation of 3D MHD Equilibria with Flow and FLR Effects

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Equilibrium flows and 3D effects can significantly impact plasma stability and energy confinement. Further, in equilibria with flow, FLR effects can play an important role. Presently, there exist a number of codes which can calculate MHD equilibria with a subset of the above effects, such as: the FLOW code,[1] the PIES code,[2] and a recently developed code for calculating axisymmetric equilibria with flow and hot ions in the large aspect-ratio limit.[3] Using insights gained from these codes, the concept for a new code for calculation of 3D MHD equilibria with flow and FLR effects is being being developed; the code is called the Kyoto ITerative Equilibrium Solver (KITES).

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P1-72

MHD Equilibrium Analysis of SDC Plasmas in LHD

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The MHD equilibrium is the basis of both most theoretical considerations and physics interpretation of the experimental results. As a standard technique to calculate the 3D MHD equilibrium, an inverse equilibrium solver VMEC, assuming the existence of perfect nested flux surfaces, is widely used. In such a technique, a magnetic coordinate system is directly constructed so as to satisfy the force balance, or, MHD equilibrium equation. For low-β equilibrium, since the magnetic field sustains clear flux surfaces, the standard technique is acceptable. However, by nature, 3D MHD equilibrium has magnetic islands and stochastic regions in the plasma because of the absence of toroidal symmetry. For high-β equilibrium, the degradation of flux surfaces by the finite β effect is not avoidable, so that the standard technique based on the nested flux surfaces could not be directly applicable to them. On the other hand, in recent experiments, various types of 3D MHD equilibrium are obtained, namely, low-shear 3D MHD equilibrium with magnetic islands, 3D MHD equilibrium with multiple magnetic axes, 3D MHD equilibrium with zero rotational transform, two-dimensional (2D) MHD equilibrium with current hole near the magnetic axis. The standard technique based on the nested flux surfaces is not suitable in such situations. In order to analyze such MHD equilibria with magnetic islands and stochastic magnetic field, other techniques are required such as the HINT2 and PIES codes.

As mentioned above, the HINT2 code is a powerful and user-friendly code. In this study, we apply the HINT2 code to study SDC (Super Dense Code) plasmas in the LHD. Since the SDC plasma has the peaked pressure profile, the Shafranv shift is large. In addition, the weak magnetic shear leads the stochastization in the peripheral region. We study the stocahstization due to finite β effects.

MHD equilibrium analysis with anisotropic pressure in LHD

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In the LHD (Large Helical Device) experiments, high beta plasmas are sustained only by tangentially-injected neutral beam (NB) lines under the low density and the low magnetic field strength. Because of the long slowing-down time of the energetic ions by tangential NBs and the large beam pressure comparing with the thermal one, the velocity distribution of the plasmas is expected to be anisotropic. From a Monte Carlo simulation, the beam pressure is estimated ～30% of the total pressure in a high beta plasma with ～5% volume averaged beta value[1]. The MHD equilibrium theory predicts that anisotropic plasma pressures parallel and perpendicular to field lines are not flux surface quantities. The method to identify the distortion of pressure from the surface average in experimental plasmas is not established. In this study, the effects of the distortion on measurable values are investigated in LHD plasmas. The MHD equilibria with the anisotropic pressure are analyzed using a 3D MHD equilibrium code, ANIMEC[2]. We study the position of magnetic axis[3] and the magnetic flux obtained by magnetic diagnostic as candidates of characteristic parameter in the MHD equilibrium. As the results, we find that the position of magnetic axis is not sensitive to the distortion from the flux surface average, but the magnetic flux is sensitive to the distortion. This suggests that the magnetic diagnostics has a possibility to estimate the distortion from flux surface average.

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Numerical analysis of axisymmetric toroidal equilibria with flow in single-fluid and two-fluid MHD models

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Numerical analysis of axisymmetric toroidal equilibria with flow is performed based on single-fluid and two-fluid magnetohydrodynamic (MHD) models. Effects of toroidal and poloidal flow comparable to the poloidal sound velocity, two-fluid, ion finite Larmor radius (FLR), pressure anisotropy and parallel heat fluxes on high-beta toroidal equilibrium is studied by solving reduced MHD equilibrium equations. The equations to solve are the Grad-Shafranov (GS) type equilibrium equations for high-beta tokamaks derived from the fluid moment equations for collisionless, magnetized plasmas [1]. The equations for the anisotropic pressures derived in the formalism include the heat fluxes. The gyrovisosity and other FLR effects cause the so-called gyroviscous cancellation of the convection due to the ion diamagnetic flow induced by the two-fluid effects in the equilibrium equations of momentum balance, pressure and heat fluxes. We have solved the reduced GS equations numerically by means of the finite element method. The two-fluid effects induce the diamagnetic flows, which result in asymmetry of the equilibria with respect to the sign of the E×B flow. Higher order terms of quantities like the pressures and the stream functions show the shift of their isosurfaces from the magnetic surfaces due to effects of flow, two-fluid and pressure anisotropy. We will also show the numerical solutions for the generalized GS equation for non-reduced single-fluid MHD equilibria with flow and compare with the analytical solution of the reduced GS equations in the single-fluid limit [2] to examine the validity of the reduced GS equations.

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MHD equilibrium including a static magnetic island for the reduced MHD equations in straight heliotron configuration

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We study the interaction between static magnetic islands and interchange modes in straight heliotron configurations by using the reduced MHD equations[1]. In our previous studies for the interaction[2,3], a pressure profile corresponding to nested magnetic surfaces was used as the equilibrium pressure profile. In the results, interchange modes grow as in the case without the islands. This is due to the fact that the finite pressure gradient exists in the region of the magnetic islands. In general, the equilibrium pressure corresponding to the magnetic surfaces including magnetic islands can be flat inside the magnetic islands. Therefore, the magnetic islands can affect the stability of interchange modes through the local change of the pressure profile. In order to study island effect on the stability, an MHD equilibrium having the pressure profile consistent with the magnetic islands is required.
In this work, we numerically calculate such an MHD equilibrium including a static magnetic island for the reduced MHD equations in a cylindrical geometry by using a two-step approach. In the first step, we obtain the plasma pressure which is constant along the field lines with the magnetic field fixed. In the second step, we solve the equilibrium equation corresponding to the vorticity equation with the pressure fixed. The two steps are iterated until the island width is converged. We utilize three kinds of method for the first step. At first, we employ a diffusion equation parallel to the field to obtain the pressure. As a result, the obtained pressure profile has local flat structure at both the O-point and the X-point of the magnetic island. Next, we include the pressure diffusion perpendicular to the field in the first step. In this case, a pressure profile with a flat region only at the O-point, not the X-point, is obtained. Furthermore, we employ an averaging method as a different approach in the first step. In this approach, the resultant pressure profile is also locally flat only at the O-point.

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P1-76

Simulation of R-T and K-H hybrid instabilities in a 2D slab

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Linear and nonlinear evolutions of Magnetohydrodynamic (MHD) unstable modes such as interchange/ballooning modes can play key roles to understand experimental results in magnetic confinement devices. While these unstable modes starting from an equilibrium without a flow have been well studied (see [1,2], for example), it can often happen that the unstable modes are coupled with other kinds of unstable modes, such as the Kelvin-Helmholtz instability, when a flow is driven.
In this paper we study a simultaneous growth of the Rayleigh-Taylor and the Kelvin-Helmholtz modes in a 2D slab. Numerical simulations of the compressible MHD equations in a 2D slab geometry are carried out for various parameters to clarify various aspects of interactions between the two kinds of modes. Influences of the wave numbers of the two modes, magnitudes of the initial perturbations, the ratio of the mass density, the velocity difference on the growth rates of the unstable modes, evolution of the mixing range, and transition to turbulence are concerned.
We are also concerned about some two-fluid effects on the hybrid evolution since both of the two types of the instability have the same nature such that the growth rate of the unstable linear eigen-mode is larger for a larger wave number. Because of the nature, the wave length of the unstable modes can be comparable to or smaller than the ion skin depth and/or the ion Larmor radius, influences of which are not included in the single-fluid MHD equations. We will try to clarify a few aspects of the two-fluid effects on the hybrid evolution of the two unstable modes.

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Development of a portable AMR module applicable to various fluid/particle simulations

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Enabling a high-resolution simulation is a prerequisite subject to study a complex dynamical behavior of hot plasma. It is required not only to understand detailed physics of small-scale events such as saturations of short-wave instabilities, micro-turbulence and magnetic reconnections, but also to construct a macro-scale model to predict long-time behaviors of the plasma because small-scale events often affect the long-time behaviors through changing local pressure gradients and/or magnetic configurations.

For the purpose of carrying out numerical simulations with a high resolution and a high efficiency, we develop an adaptive mesh refinement (AMR) module which is easily transplanted to various numerical codes, especiall in the Numerical Simulation Research Project (NSRP) of NIFS. While the module is based on the block-based domain decomposition approach and the self-similar refinement, it is designed to be applicable to a cell-based decomposition easily. Computational domains generated as a consequence of the refinement are distributed to the computational nodes according to the Morton method[1]. It is aimed that the ion skin depth, the ion Larmor radius and some other scales are well resolved by intrducing the AMR module while macroscopic behaviors are also simulated simultaneously.

In this paper, we demonstrate the effectiveness of the developed AMR module on fluid simulations by carrying out magnetohydrodynamic (MHD) simulations. The MHD equations are approximated either by the use of the 4th order central finite difference scheme or by the use of the 8th order compact finite difference scheme, which have been applied for the nonlinear MHD simulations of LHD[2,3]. The suitability of the compact finite difference scheme to the AMR approach will be discussed.

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Numerical MHD Analysis of LHD Plasma in Magnetic Axis Swing Operation

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When we consider the steady state operation in a fusion reactor, we have to avoid disruptive phenomena. Heliotron configurations including Large Helical Device have an advantage that they are free from the disruptions which inherently occur in tokamaks. However, minor pressure collapses have been observed even in the heliotrons even in the LHD configurations with the vacuum magnetic axis shifted inward. They are not as dangerous as the disruptions, however, they should be suppressed for the stable steady state operations. Recently, experiments with a real time control of the vertical field was carried out for the precise study of the collapse property in the inward shifted configurations[1]. In the experiments, the position of the corresponding vacuum magnetic axis is shifted toward the inward side during each discharge. This operation is called a magnetic axis swing operation. Collapse in the profile of the electron temperature, therefore in the beta profile, is observed when the corresponding vacuum magnetic axis becomes smaller than about Rax=3.55m.

In the present work, we simulate the experiment with the magnetic axis swing by means of the nonlinear MHD calculation. In the simulation, we take the change of the equilibrium quantities due to the vertical field change into account. However, the time scale of the equilibrium change is quite slow compared with the time scale of the MHD dynamics of the perturbation. We have to treat both time scale simultaneously in the simulation. On the other hand, we have developed a multi-scale simulation scheme for nonlinear MHD simulations for beta-increasing heliotron plasma[2]. In the scheme, we can follow the development of the equilibrium due to the increase of beta as well as the nonlinear dynamics of the interchange modes. Thus, we apply this multi-scale scheme to the simulation of the magnetic axis swing experiment. This can be achieved by exchanging the equilibrium development due to the increase of beta to that due to the vertical field variation. We discuss the mechanism of the collapse of the pressure and the stability limit in the view point of the nonlinear collapse based on the simulation results.

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P1-79

Dynamic Domain Decomposition for 3D PIC simulation with Adaptive Mesh Refinement

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To investigate multi-scale phenomena in space plasma including plasma kinetic effects, we started to develop a new electromagnetic Particle-In-Cell (PIC) code with Adaptive Mesh Refinement (AMR) technique. In AMR simulation, spatial grid size and time step intervals are defined according to the hierarchy levels, where high and low levels correspond to the fine and coarse grid systems, respectively. As the simulation system evolves, some complex micro-scale phenomena can locally and intermittently occur in a hierarchical domain (Level L). If the grid size in Level L is too coarse to simulate the local complex phenomena, A higher hierarchical domain (Level L+1) is adaptively created in which the grid spacing size and the time step interval become half of those used in the domain of (Level L). AMR-PIC simulations require smaller amount of memory and shorter computation time than conventional PIC simulations.
In parallelizing the code for a distributed many-core system, we adopt a scheme of dynamic domain decomposition which realizes the load balance between distributed sub-domains. In conventional domain decomposition scheme, one of the issues is load balance between processors because the number of spatial grids and particles belonging to each sub-domain is not always constant due to the AMR procedure. Since the cost of particle calculation is dominant, which is about 70-80% of the total cost, the number of particle calculation loops should be balanced between each processor. For this purpose we introduced dynamic domain decomposition called DDD. We are currently incorporating this scheme in the three dimensional version of AMR-PIC code and are performing a test simulation to examine the parallelization efficiency as well as the scalability with respect to the number of processors. In the presentation, we will show you the detail of DDD scheme as well as some preliminary results of AMR-PIC

Nonlinear stability of externally driven magnetic islands

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Physics of resonant magnetic perturbations (RMPs) in torus plasmas is of great interest to the magnetic confinement fusion, since it might be applicable to the control method of the magnetohydrodynamic (MHD) instability. The RMPs due mainly to external current coils produce current sheets at corresponding resonant magnetic surfaces, and under certain condition they drive the forced magnetic reconnection and generate magnetic islands, even in the absence of the intrinsic tearing instability. In the Large Helical Device, which is a stellarator, sudden transition between equilibria with and without the externally driven magnetic islands is observed[1]. It is also observed that such bifurcation affects the amplitude of the pressure-gradient-driven MHD instability[2].

The goal of the present study is theoretical understanding of the influence of the RMPs on stellarator plasmas. For this purpose, the following approaches are examined.

Firstly, a theoretical model of the forced magnetic reconnection in a stellarator is developed, which is an extended version of that in our previous work[3]. In the model, zero-dimensional equations of the width and the phase angle of magnetic islands are coupled with an one-dimensional equation of the neoclassical poloidal flow produced by the rippled toroidal magnetic field. It is found that nonlinear saturation state of the externally driven magnetic islands has two branches, i.e., non-rotating, small magnetic islands and non-rotating large magnetic islands, and the transition and the hysteresis between these states are observed. These phenomena are caused by the screening effect of the poloidal flow and the slip of the magnetic island rotation in the rest frame of rotating plasmas.

In the next step, the three-dimensional nonlinear dynamics of the externally driven magnetic islands is simulated, using a reduced set of two-fluid equations for a stellarator. The nonlinear bifurcation state of the externally driven magnetic islands is also observed in the simulation, and the results of the theoretical model are confirmed. Influence of the RMPs on the pressure-gradient-driven MHD modes is also discussed.

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Effect of resistivity on mode structure of interchange instability in heliotron plasma

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It is one of the important subjects for realization of the fusion reactor to investigate the Magnetohydrodynamics (MHD) instabilities. The interchange mode which is one of the pressure-driven instabilities is considered to play an important role in plasma confinement property in the heliotron plasma.
In the Large Helical Device (LHD), the magnetic fluctuations around the peripheral rational surface are observed. The level of such fluctuations increases with decreasing magnetic Reynolds number [1]. It is important for understanding the interchange instabilities to investigate relationships between the magnetic islands produced by the magnetic fluctuations and the mode structures of instabilities. However, such relationships have not been investigated especially for cases with the high magnetic Reynolds number.
In this study, we numerically analyze the interchange instability of the straight heliotron plasma with various magnetic Reynolds number based on the linearized reduced MHD equations. The relationships between the width of the magnetic island and the mode structure will be presented. We focus on interchange instability with (m, n) = (1, 1) mode. (m is the poloidal mode number and n is the toroidal mode number.) In addition, the dependencies on the magnetic Reynolds number and the β value will be discussed.

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P1-82

Development of an MHD Code Based on the CIP Method to Consider the Advection with Alfvén Velocity

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The analysis of Magnetohydrodynamics (MHD) instabilities is one of the important issues to realize
the nuclear fusion reactor. The advection of the Alfvén wave must be solved with small numerical errors in the simulations of the instabilities since the fluctuations propagate at the Alfvén velocity. The advection term described with the Alfvén velocity explicitly appears in the MHD equations when Elsässer variables are introduced [1]. It is well known that the Constrained Interpolation Profi le (CIP) method [2] is useful for stably and accurately solving the advection equation.
In this study, we have developed an MHD code in which the MHD equations using Elsässer variables are solved with the CIP method. Using the developed code, it is investigated whether the initial equilibrium of tokamak plasmas is maintained. We first apply the developed code to a hypothetical rectangular cross sectional tokamak plasma. The developed code is also applied to the JT-60U plasma.
The detail of the developed code will be described. The time evolutions of several quantities will be
presented. The comparison will be made between our results and those by use of the CIP-MOCCT method [3].

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P1-83

Modeling of Formation of Helical Structures in Reversed-Field Pinch

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From a point of view of the magnetohydrodynamic (MHD) stability, spontaneous formation, or self-organization, of a quasi-steady state is a promising nature of plasma for a long-term sustainment of the magnetic confinement. In this study, we aim at modeling the formation of three -dimensional structures in the reversed-field pinch (RFP) plasma, which is experimentally observed with several types of aspect, by means of a nonlinear MHD simulation. In RFPs, the tearing modes tend to be destabilized because of the internal current by nature. The confinement is therefore deteriorated by the overlaps of multiple rational surfaces. In order to avoid this situation, a control method is proposed both experimentally and theoretically, in which the perturbations are concentrated at a small number of the modes, and kept to quasi-steady states termed the quasi-single helicity (QSH) and the single helical axis (SHAx) states. We apply the general-purpose nonlinear MHD codes, MIPS, to the RFP configuration with the reconstructed equilibria of the RELAX device solved by the RELAXFit code[1] as the initial condition. The long-term solutions of the simulation reproduce the growth of the resistive modes dominated by the m/n=1/4 component and the helical deformation of the whole torus. There is also a qualitative agreement in the dependency on the equilibrium parameters to the concentration to the single n=4 mode. We also discuss the transition processes between the single- and multi-helicity states.

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P1-84

Study of resistivity effect on MHD stability beta limit in LHD using TASK3D-code

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The MHD Stability beta limit in the Large Helical Device (LHD) has been analyzed using a hierarchy-integrated simulation code TASK3D[1] in order to explore the capability of the LHD configuration. For the analysis of the MHD stability beta limit, the MHD equilibrium module VMEC, the transport module TR and the linear MHD stability module MSSH are used. The numerical model for the effect of the MHD instabilities is introduced such that the pressure profile is flattened around the rational surface due to the MHD instabilities. The width of the flattening of the pressure gradient is determined form the width of the eigenmode structure of the MHD instabilities. It is also assumed that there is an upper limit of the poloidal mode number of the MHD instabilities; mc which directly affect the pressure gradient. Recent experimental results suggest that eigenmodes of the MHD instabilities with poloidal mode number of $m<5$ dynamically affect the pressure profile. In our analysis, $m_c=4$ is used as the standard mode of the calculation of the achievable beta value. In the previous study[2,3], ideal interchange modes, which are most important instabilities in helical plasmas, were considered to give the MHD stability beta limit. It has been found that the achievable volume averaged beta value is expected to be beyond about 6% for $1.5 < \sigma < 1.6$, where σ is the peaking factor of the pressure profile. In this study, the linear MHD stability module MSSH has been extended to analyze resistive modes and the effect of the resistive interchange modes on MHD stability beta limit is investigated. Dependence of the achievable beta value on the plasma resistivity will be discussed.

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Development and experimental application of integrated transport code, TASK3D, for helical plasmas, ~equipment and extension of neoclassical transport module~

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The integrated code for helical plasmas, TASK3D, has been developed both by modifying modules in TASK [1] to e applicable to three-dimensional magnetic configurations, and by adding new modules for stellarator-heliotron specific physics. In this paper, these module developments so far are introduced, and those experimental applications are collectively described.
The Large Helical Device (LHD) experiments have steadily expand the parameter regime of helical plasmas such as electron and ion temperatures, beta value, electron density, pulse length and total input energy to plasma in a long-pulse operation as summarized in Ref.[2]. The wide-range applicability of the integrated code is mandatory, and, in the mean time, experimental verification of each module and its integration should be performed by utilizing the experiment database.
The radial transport aspect of TASK3D employs the diffusive transport equation [TR] with the local diffusivity (particle and heat) and the particle and heat source/sink term. Taking three-dimensional configuration's impact into account such quantities have been main focus of module modification/supplement in TASK3D development.
The presence of the ripple transport (so called 1/ν diffusion) is one of specific features in non-symmetric magnetic configurations. The ambipolarity of neoclassical (NC) particle fluxes has been demonstrated to well describe the bifurcation nature of the radial electric field (Er) in LHD and other helical plasmas [3,4]. Thus, it is of a great importance to accurately calculate the neoclassical fluxes, and then Er. For this purpose, the diffusion coefficient database, DGN/LHD [5], has been prepared for a wide-range of equilibria by utilizing advantages of NC transport codes, DCOM [6] and GSRAKE [7]. This highly-accurate NC module has made it possible to compare with the power-balance analysis (TR-snap package, composed of several modules of TASK3D) [8]. Recently, DGN/LHD database has been extended to higher-beta range, so that accurate NC transport calculations for LHD high-beta plasmas and predictive analyses for parameter regime in a reactor such as FFHR [9] can be performed. Such example calculations will be reported in the paper.

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P1-86

The radial electric field formation in high T_e plasmas with the electron internal transport barrier in LHD

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Very high electron temperature (T_e ≃ 15 keV) plasmas are successfully obtained with the ECH heating in recent LHD experiments [1]. These plasmas are called Core Electron-Root Confinement (CERC) plasmas [2] since they are accompanied by the strong positive (electron-root) radial electric field (E_r) in the core region. CERC plasmas also have the characteristic steep T_e gradient (electron internal transport barrier, eITB) at the core. It is also observed that the electron-root E_r has the steep shear there corresponding to the eITB formation. The finite orbit width (FOW) effect of electrons becomes important in such high T_e helical plasmas [3]. As the electron-root E_r and the steep T_e gradient are formed in the core of CERC plasmas, the width of the local flattening, namely, the region of approximately zero temperature gradient, becomes narrower. Since E_r in helical plasmas are predominantly determined by the ambipolar condition of the neoclassical (NC) transport, the numerical evaluation of the NC transport and the ambipolar E_r is the key issue to understand the relation between the improvement of the transport and E_r in CERC plasmas.

In this study, we focus on the NC ambipolar E_r formation in the core region of CERC plasmas. Since the poloidal flow velocity is deeply related to E_r through the $E \times B$ drift and it affects the NC transport, the radial profile of the poloidal flow is examined. This enables ones to investigate the reduction processes of the transport by E_r and its relation to the poloidal flow in CERC plasmas. For this purpose, NC transport analyses for CERC plasmas are carried out using δf Monte-Carlo particle code, FORTEC-3D [4], which has been extended to be applicable to electron species and can calculate the NC transport including the electron FOW effect. The numerically obtained ambipolar E_r is compared to the experimental observations. The formation of the E_r and its shear in the core region, where either the local flattening or the steep gradient of T_e is observed, are also investigated. It is also investigated how the particle orbit affects the NC transport improvement in the core.

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Dependence of thermal diffusivity on plasma parameters in perturbed magnetic field in toroidal plasma

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We consider the modeling of energy transport under effect of noise caused by resonant magnetic perturbations (RMPs) in a toroidal plasma. It is found in the previous study [1] that the radial thermal diffusivity χ_r in the perturbed region is represented as $\chi_r=\chi_r^{(0)}(1+c|\delta B_r/B_t|^2)$, where $|\delta B_r|$ is the strength of RMPs, $|B_t|$ is the strength of toroidal magnetic field at the magnetic axis, $\chi_r^{(0)}$ is thermal diffusivity when $|\delta B_r|=0$, and c is a coefficient. In the present paper, dependence of the coefficient c on plasma parameters is investigated by a Monte-Carlo simulation code calculating time evolution of an ion guiding center distribution function, where the guiding center distribution function evolves with time from the Maxwell distribution under effects of both the Coulomb collision and noise caused by RMPs. Recently, we develop a transport simulation code without the assumption of nested flux surfaces [2]; the code is named KEATS. The code is programmed by expanding the well-known Monte-Carlo particle simulation scheme based on the δf method. By using KEATS code, it is possible to execute the investigation. We find that the coefficient c depends on the collision frequency ν in wide range of collisionality and on the particle mass m.

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Research on burn control of core plasma with the transport code

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For the commercial fusion reactor, one will be required to control the reactor parameter, like fusion power P_{fus}, diverter heat flux q_{div}, and avoid the disruption. One can't satisfy these requests directly, so one need to control some mid parameters like plasma shape (κ,δ), position, density n, current density j, temperature T, HH-factor, troyon factor β_N and etc. To control these parameters, many diagnostics and actuators are needed. In the future reactor, one must take COE (Cost Of Electricity) or large fusion neutron flux into account and the diagnostics and actuators which is extrapolated into the reactor are limited. In addition, the actuators, like gas puff, impurity injection, NBI injection, RF, external coil flux, and etc, and plasma parameters are not necessarily one-to-one correspondence. For these reasons, construction of control theory is required for the future reactors. The control of plasma shape or position are discussed in [1], then, in this research, we will consider the control of the core plasma and the control theory is expressed as follows,
Au(t)=x(t)
y(t)=Bx(t)
where x(t),u(t),y(t) are reactor parameter vector, actuator vector, mid parameter vector respectively, and A, B are tensors. The choice of the components of x(t), y(t), u(t) are very important and difficult problem. In this research, as the first step of the construction of control theory, we use the simple model.1) We assume B tensor as a unite tensor, in other words, we can measure the plasma parameters accurately.2) We assume the dimension of u(t) and x(t) are 2, and A tensor is a 2×2 tensor.3) We assume the components of u(t) are P_{puff} and P_{LH}, and x(t) are P_{fus} and q_{min}. In most of previous studies, they control one plasma parameter, and the study of multiple parameters control with multiple actuators has not been much done. For this reason, we use two - two model. P_{fus} is the most important parameter to make electricity, and q_{min} has strong influence on high-β. The analysis of safty factor q control is done in [2-3]. In this research we will analyze the A tensor with 1-D transport code, and in this conference, we will present about the control simulation.

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The neoclassical transport and viscosity in helical plasma consisting of multi-species ions including high-Z impurities

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The neoclassical transport has important roles on the transport phenomena in helical plasma, such as the radial electric field, the poloidal and toroidal viscosities, particle flows, bootstrap current. In addition, this transport becomes significantly large at high temperature. Though high-Z ions is considered to have greater influence on the neoclassical transport and viscosity, the effects of multispecies ions on the neoclassical transport in helical plasmas have not been well known yet,. Therefore in this paper, by using the moment method, those effects on the neoclassical transport and the neoclassical viscosity in helical plasmas are investigated.

P1-90

Electron fluid perturbation and related electron transport in a field-reversed configuration

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The transport mechanism of a field-reversed configuration (FRC) plasma is still unclear. Resistivity is anomalously high, and a trapped poloidal flux soon decays in less than 100 micro-seconds. The rapid flux decay may cause toroidal spin-up of an FRC. We have proposed that direct conversion from the magnetic flux to the kinetic angular momentum takes place as long as axisymmetry holds. The resultant rotational instability with the toroidal mode number n=2 of an FRC plasma is known as the most often observed global instability. The FRC current just after formation is primarily carried by electron current, while ions are approximately at rest. The ions, however, gradually gain angular momentum in the ion diamagnetic direction before the onset of the instability. Conversion to the angular momentum, however, also occurs for electrons, and it plays the electron current drive and consequently maintains the magnetic flux.
Therefore, we suppose that the presence of an anomalous loss of the electron angular momentum due to an electron fluid perturbation with higher frequency than the ion cyclotron frequency. Spatial structure of the perturbation in an FRC is studied analytically. Calculation of electron orbits in the prescribed fields is carried out in order to find the loss of the angular momentum, i.e., the resistivity and the end-loss rate.

P1-91

Computation of Neutral Gas Flow Generation from a CT Neutralization Fuel-Injector

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"CT Neutralization Fuel Injection" is an expansion of the "CT Injection" method.[1,3] The "CT Injection" method was proposed as a fuel-injection method for large-scale power generation plasmas such as the ITER. However, feasibility is still unclear because of the difficulty concerning the translation process of a plasmoid across the vertical magnetic field of the fuel injection target. To resolve this problem, the "CT Neutralization Fuel Injection" method utilizes a neutralization cell which transforms the injection plasmoid into an ultra-fast neutral gas flow. It is expected to be faster than conventional methods such as the "Gas-Puff" or "Pellet Injection". Therefore, there is a high possibility of reaching the core of the fuel injection target.[2,4]
In this study, we will check if the neutral gas generation is enough to meet above quotas. We run a computer simulation on the neutralization cell of the "CT Neutralization Fuel Injection" device. In our calculations, we focus on the plasmoid itself moving through the cell. The plasmoid is treated as a stationary target upon which the neutral gas particles of the cell collide and charge exchange with. Therefore only gas pressure inside the cell and injection speed of the plasmoid is changed. The collision processes are reproduced by the Monte Carlo calculation. The calculation determines the plasmoid's degree of neutralization, and as a result optimum cell length is found.
When the central ion temperature is 100 [eV] and the injection speed of CT at 50-300 [km/s], it was determined that 0.4 [m] is the optimum cell length due to thermal diffusion of the generated gas flow. Any cell longer will be not efficient, owing to the increased chance of the neutral gas particles colliding with the cell and translation tube walls. This will likely cause a reduction of the gas injection speed. Furthermore, it was calculated that the gas pressure inside the neutralization cell and the plasmoid's injection speed does not largely affect the neutralization level and process. Also, on the subject of interactions with the cell wall, we have determined that the incident angle for most of the impacts is 30 to 60 degrees.
We also show preliminary results of a hybrid simulation on CT plasma. The hybrid simulation model under development will take into account the neutralization process to investigate an electromagnetic behavior inside the neutralization cell. In particular, it should be clarified that magnetic flux decay occurs by reducing the plasma current and associated radial expansion of the plasma caused by imbalance of the radial force. We will also study the influence of the axial electric field generated by the friction force between electrons in the moving plasmoid and cold ions.

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Toroidal spin-up caused by an anomalous loss of the electron angular momentum of a field-reversed configuration

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The rotational instability with the toroidal mode number n=2 of a field-reversed configuration (FRC) plasma is the most often observed global instability. The current of FRC plasmas just after formation is primarily carried by electrons, while ions are approximately at rest. The ions, however, gradually gain the angular momentum in the ion diamagnetic direction before the onset of the instability. The rotation mechanism of the FRC plasma has been often explained by a selective loss of ions [1], or end-shorting [2].
We have proposed another possible spin-up mechanism, which is direct conversion from the magnetic flux to the kinetic angular momentum as long as axisymmetry holds [3]. In this spin-up mechanism, plasma ions in the core of the FRC plasma can gain the angular momentum. Therefore, plasma rotation is possible to start near the field-null; the core rotation start-up is good agreement with experiments. Conversion to the angular momentum, however, also occurs for electrons, and a rapid loss of the electron angular momentum is needed to explain a rapid loss of the magnetic flux.
Therefore, we suppose that the presence of an anomalous loss of the electron angular momentum due to an electron fluid fluctuation with higher frequency than the ion cyclotron frequency. Hybrid simulation is carried out to find the time evolution and the spatial profile of the toroidal flow velocity. Comparison of the calculation results with experimental results is also

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P1-93

Nonthermal effects on the resonant instability of the dust-acoustic wave in semi-bounded Lorentzian plasmas

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The nonthermal effects on the resonant instability of the surface dust-acoustic wave are investigated in semi-bounded Lorentzian dusty plasmas containing elongated rotating dust grains. It is found that the nonthermal effects suppress the frequency domain and enhance the growth rate of the resonant instability for the case of positively charged dust grains. For negatively charged dust grains, however, the nonthermal effects on the frequency domain and the growth rate are found to be negligible. In addition, it is found that the growth rates of the resonant instabilities in nonthermal plasmas would be greater than those in thermal plasmas.

Numerical Analysis of Quantum-Mechanical Grad-*B* Drift II

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We have developed a code to solve the time-dependent Schrödinger equation [1], [2], with standard notations, $i\hbar\,(\partial\psi / \partial t) = (1 / 2m)\,(-i\hbar\,\nabla - q\mathbf{A})^2\psi$, in the presence of a non-uniform magnetic.

In the previous paper [2], we have shown that the quantum mechanical variance σ^2 in position may reach the square of the interparticle separation in a time interval of the order of 10^{-4} sec for typical magnetically confined fusion plasmas.

In this paper, as an extension of the paper [2], we investigated the dependence of the variance σ^2 in position on parameters such as m, q, v_0, B_0 and L_B, where m is the mass of the particle, q is the charge, v_0 is the initial speed of the corresponding classical particle, B_0 is the magnetic field at the origin and L_B is the gradient scale length of the magnetic field. Here we have assumed a vector potential $\boldsymbol{A} = (-B_0\,y\,(1 - y / 2L_B), 0, 0)$, which corresponds to the non-uniform magnetic field of $\boldsymbol{B} = (0, 0, B_0\,(1 - y / L_B))$.

We solved the two dimensional time-dependent Schrödinger equation for a single particle in the presence of the non-uniform magnetic field for various sets of m, q, v_0, B_0 and L_B. It was numerically found that the variance, or the uncertaintly, in position σ^2 is given by, $d\sigma_r^2 / dt = 4.04\hbar v_0 / qB_0L_B$.

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Quantum Mechanical Plasma Scattering II

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We have solved the unsteady Schrödinger equation in order to analyze a real particle's motion [1, 2]. We will numerically solve the two-dimensional Schrödinger equation for a single particle in the presence of a uniform magnetic field and field particles.
The unsteady Schrödinger equation for wavefunction $\psi(\boldsymbol{r},t)$, at a position $\boldsymbol{r}$ and a time t, is given by
$i\hbar\partial\psi/\partial t=[(-i\hbar\nabla-q\boldsymbol{A})^2/(2m)+q\varphi]\psi$,
where φ and $\boldsymbol{A}$ stands for the scalar and vector potentials, m and q the mass and electric charge of the particle under consideration, $i\equiv(-1)^{1/2}$ the imaginary unit, and $\hbar\equiv h/2\pi$ the reduced Planck constant. When the corresponding classical particle has an initial momentum $\boldsymbol{p}_0=m\boldsymbol{v}_0$ at a position $\boldsymbol{r}=\boldsymbol{r}_0$, the initial condition for the wavefunction is given by
$\psi(\boldsymbol{r},0)=\exp[-(\boldsymbol{r}-\boldsymbol{r}_0)^2/(2l_B^2)+i\boldsymbol{k}_0\cdot\boldsymbol{r}]/(\pi^{1/2}l_B)$,
where $\boldsymbol{r}_0$ is the initial center of ψ, $l_B\equiv(\hbar/qB)^{1/2}$ is the magnetic length, and $\boldsymbol{k}_0=m\boldsymbol{v}_0/\hbar$ is the initial wavenumber vector.
Let the vector potential be
$\boldsymbol{A}=(-B_z y,0,0)$,
and the scalar potential be, in the cylindrical coordinates (R,θ,Z),
$\varphi(\eta)=\Sigma_f\varphi_f(\eta)=\Sigma_f q_f/(\pi^2\varepsilon_0 l_B)\int_0^\infty K[4\eta\eta'/(\eta+\eta')^2]\exp(-\eta'^2)\eta'/(\eta+\eta')d\eta'$,
where f is the label for field particles, ε_0 is the vacuum permittivity, K is the complete elliptic integral of the first kind, and $\eta=R/l_B$.

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Theoretical study of ultra-relativistic laser electron interaction with radiation reaction by quantum description

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With the advent of ultra?relativistic laser, it will reach to the Intensity 10^{22}W/cm^2. If an electron is under this laser when laser intensity is larger than 10^{18}W/cm^2, an electron will carry out relativistic behavior. The most important phenomenon in this regime is an effect of ponderomotive force, and an electrons are pushed out in the propagation direction of laser. If we can get the laser intensity 10^{22}W/cm^2, the strong bremsstrahlung might be caused. Moreover, simultaneously with it, “radiation reaction force (or damping force)” works to charge particles. We could reach to describe this reaction force working on an electron in Minkowski spacetime, it is as follows[1]:

$$f_{reaction}{}^{\mu}=m_0\tau_0\, d^2w^{\mu}/d\tau^2 + m_0\tau_0/c^2\, g(dw/d\tau,dw/d\tau)w^{\mu} - m_0\tau_0/c^2\, (w^0w^{\mu}-c^2\delta^{\mu}{}_0)/((w^0)^2-c^2)\, d^2w^0/d\tau^2$$

Where $\tau_0=Q^2/6\pi\varepsilon_0 m_0 c^3$, w, τ and g is the relativistic 4-velocity, proper time and Lorentz metric. Before this equation appears, the Landau-Lifshitz (L-L) equation[2] as the approximation of Lorentz-Abraham-Dirac (LAD) equation[3] is used. But, the L-L equation breaks the relation of relativistic covariance. Our equation is satisfied with the covariance.
In the next laser intensity generation (over 10^{24}W/cm^2), we consider the electron-positron pair creations and something effects appear[4]. This intensity level requires us to treat phenomena as QED. However, we can consider an electron as quantum description under 10^{24}W/cm^2, without the pair creations. In this presentation, we suggest the quantum model of the radiation reaction without the electron-positron pair creation using the Dirac equation.

$$[i\hbar\gamma^{\mu}(\partial_{\mu}-ie/\hbar A_{\mu})+P_{radiation}-mcI^{4\times4}]\psi=0$$

Here, A_{μ} means the external laser driver field. The radiation reaction is one of the self interactions. Therefore, we put the new interaction term of $P_{radiation}$. In our model, the bremsstrahlung-photon field is treated as quantum field.

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Laser Guiding Through an Axially Nonuniform Magnetoplasma Channel in the Weakly Relativistic Limit

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In this paper, laser guiding through an axially nonuniform plasma channel is investigated in the weakly relativistic limit, under the effect of static magnetic field. Self defocusing of the prepulse leads to the axial nonuniformity of the plasma channel. The propagation of the guided laser beam through the preformed axially nonuniform plasma channel results into the alternate convergence and divergence of the beam. Unbalanced diffraction and refraction phenomenon through magneto-plasma channel results into periodic beam width variation with the distance of propagation. The wave equations governing the propagation characteristics of the ionizing prepulse and delayed pulse through an axially non-uniform plasma channel have been solved by moment theory approach. Effect of the Magnetic field, Beam width of the guided laser beam, intensity of the guided laser beam and the axial nonuniformity of the plasma channel on the laser guiding has been studied. Laser guiding up to several Rayleigh lengths has been observed. We have seen a very important effect of magnetic field on the guidance of the laser beam through the plasma channel and have noticed that laser guidance increases with the increase in magnetic field. Results of the present analysis are very important for the successful implementation of the fast Igniter concept, in which, it is required that the intense guided pulse must propagate through the plasma region without being absorbed and without beam break-up due to filamentation up to the edge of the core.

Charge-transfer cross sections of ground state He^+ ions in collisions with He atoms and simple molecules in the energy range below 4.0 keV

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Helium is the second most abundant element in the universe. Indeed, helium ions were observed in the outer space of the Earth by an extreme ultraviolet scanner on Mars Orbiter Planet-B [1]. Some observations of X-ray emissions from comets have attracted much attention and the origin of this emission has been interpreted as being due to charge transfer into solar-wind ions from the cometary gases [2]. Helium will be also one of typical impurities in D-T fusion plasmas as ashes, if the D-T fusion reactions really occur in controlled thermonuclear fusion devices. Thus, the accurate knowledge on charge-transfer collisions between He^+ ions as well as He^{2+} ions and common impurity molecules is critically important in proper handling of the helium ashes in the divertor region, better understanding of the behavior of low temperature edge plasmas, and their plasma-modeling in the fusion devices [3].
Total cross sections of these charge-transfer collisions have been extensively measured since the 1955s [4]. In the low collision energy region, however, serious discrepancy is observed among existing experimental data. Therefore, in the present work, we have systematically measured the charge-transfer cross sections for the ground state He^+ (^1S) ions colliding with He atoms and simple molecules (H_2, D_2, N_2, CO, CO_2) in the energy range of 0.2 to 4 keV by applying an initial growth rate method. Furthermore, in He^+ + He collisions, we will discuss about the detection efficiencies of a micro channel plate detector for He^+ ions and He atoms. The present observations have been found to provide reasonably reliable cross-section data for the charge transfer of He^+ ions in collisions with these molecules in the energy range below 4.0 keV. In He^+ + D_2 collisions, any isotope effect was not observed over the present energy range, compared to H_2 molecules.

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Application of Collocation Meshless Method to Eigenvalue Problem

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Many meshless approach [1], [2] have been proposed and have yielded excellent results in the fields of engineering and science. In spite of the convenience, meshless approaches are plagued by two difficulties. First, the method for implementing the essential boundary condition is different according to meshless approaches. Second, both the essential boundary condition and the natural one are not exactly fulfilled on the boundary. If a new implementation method of not only the essential boundary condition but also the natural one were proposed without dependence on meshless approaches, the above demerit could be completely resolved.

In the previous studies, Kamitani *et al.* [3] have proposed the new method for implementing the essential boundary condition to the meshless Galerkin/Petrov-Galerkin approach. The results of computations show that the accuracy of the proposed method is higher than that of the standard one. In addition, Saitoh *et al.* [4] have investigated the applicability of the proposed method to the nonlinear Poisson problem.

The purpose of the present study is to develop numerical code for solving the eigenvalue problem on the basis of the collocation meshless method and to investigate the performance of the collocation EFGM by use of the code.

The numerical results will be shown in the conference.

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Numerical Simulation of Electromagnetic Wave Propagation using Time Domain Meshless Method

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In the Large Helical Device (LHD), the electron cyclotron heating device is used for plasma heating. The electrical power which is made by the gyrotron system transmits to LHD by using long corrugated waveguide. However, it is not clear that the shape of curvature of the waveguide or transmission gain of electromagnetic wave propagation theoretically.
Generally, Finite Differential Time Domain (FDTD) method is applied for electromagnetic wave propagation simulation. FDTD method has provided the solution of Maxwell equation directly. Furthermore, FDTD method has great advantages in terms of parallelization and treatment of problems and so on. However, the numerical domain should be divided into rectangle meshes if FDTD method is applied for the simulation, and it is difficult to treat the problem constructed by arbitrary shapes.
The meshless approach has even developed, such as the element-free Galerkin (EFG) method and the meshes local Petrov-Galerkin (MLPG) method and the radial point interpolation method (RPIM). As is well known that the meshes approach does not require finite elements or meshes of a geometrical structure. And these methods are applied to a variety of engineering fields and the fields of computational magnetics. In particular, meshes approaches based on RPIM are applied to time dependent problems[1, 2].
The purpose of the present study is to develop numerical code for analyzing electromagnetic wave propagation in arbitrary shapes of waveguide using meshes approach based on RPIM.

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Implicit Function with a Natural Behavior over the Entire Domain

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Many kinds of meshless methods have been proposed, and have been applied to numerical simulations of various fields including plasma physics and fusion science. In the meshless methods, although the elements representing a geometrical structure are not necessary, an analysis domain has to be defined. To define the analysis domain, an implicit function [1, 2] is employed in the eXtended Boundary-Node Method (X-BNM) [3]. In the X-BNM, it is assumed that, by using the implicit function $f(\boldsymbol{x})$, the boundary of the analysis domain is represented as $f(\boldsymbol{x}) = 0$.

In general, an implicit function $f(\boldsymbol{x})$ has properties such as $f(\boldsymbol{x}) < 0$ and $f(\boldsymbol{x}) > 0$ for inside and outside of the surface, respectively. For generating an implicit function $f(\boldsymbol{x})$ from a large scattered point data, the Partition of Unity (PU) based methods such as the Multi-level Partition of Unity implicits (MPU) method [1] are often employed. However, the generated implicit function $f(\boldsymbol{x})$ has an effective range. Namely, $f(\boldsymbol{x}) = 0$ is distributed not only on the surface but also outside of the effective range. This is not a natural behavior of an implicit function, since the above properties are not satisfied. Especially in the X-BNM, since a procedure of finding the boundary $f(\boldsymbol{x}) = 0$ is indispensable for evaluating the influence coefficients [3], it is desirable that $f(\boldsymbol{x}) = 0$ is only distributed on the boundary.

The purpose of this study is to generate an implicit function satisfying the above properties over the entire domain. To this end, two kinds of implicit functions are generated by using the PU based method and the radial basis function based method [2], respectively, and these implicit functions are smoothly combined by using an appropriate weight function.

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A new De-noising Method of Laser-produced Plasma Penumbral Images by Principal Component Analysis

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Penumbral imaging technique[1] can be applied to highly penetrating radiations such as neutrons. The large aperture is used and the convolved image (penumbral image) is recorded on the detector. The source image can be recovered by the deconvolution. Therefore the technique can be used to inertial confinement fusion research. In the inertial confinement fusion research, the fast ignition [2] approach can increase the number of fusion. However, the γ rays produced by the fast-heating laser pollute the penumbral image as noise. Since the conventional deconvolution methods like the Wiener filter cannot obtain the clear reconstructed image from the noisy penumbral image. Other reconstruction methods are proposed to obtain the reconstructed image from the noisy penumbral image. But these methods cannot obtain the reconstructed image since the signal-to-noise ratio (S/N) on the detector is extremely low due to the γ rays by the fast-heating laser.
In this paper, we propose a new reconstruction method by the principal component analysis[3]. The methods can efficiently remove the noise by high S/N training images which are obtained from other experiments. We used the (2D)2PCA[4] method as a noise reduction method, which cannot obtain clearer images but calculation cost is lower compared with the conventional PCA methods. The efficacy of the proposed method is demonstrated by the computer simulations.

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P1-103

Studies on Plasma Production and Acceleration Using Rotating Electromagnetic Fields for Electrical Propulsion Engine

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In the space engineering, VASIMR (Variable Specific Impulse Magneto-plasma Rocket) researched by NASA[1] is attracting considerable attention. It has an important feature of independent control of thrust and specific impulse by using helicon wave plasma production and ion cyclotron heating. This two-stage radio frequency (RF) system makes the equipment bigger and complicated. The authors have proposed a simple system: an employment of the Rotating electromagnetic Field Antenna (RFA) for plasma production and heating, which consists of a pair of double half turn antennas staggered by 90 degrees in the azimuthal direction. By controlling the phase difference between currents of two straps (φ), the ratio of excitation fields of left and right hand polarizations can be controlled. Each polarization field will lead excitations of slow and fast waves, and thus, ion heating and plasma production, respectively.
In the electrical propulsion rocket experimental simulator, the effectiveness of RFA has been examined. To the base hydrogen plasma produced by another RF power, an RF application by RFA provides sinusoidal variations of ion saturation current (Iis) and ion temperature (Ti) to φ. Under the ion cyclotron resonance condition, the Iis and Ti show maximum values of 120% and 320% increments for $\varphi=\pi/2$ and $-\pi/2$, respectively. Moreover, 200% increment of Ti has also been observed under the second harmonic resonance condition.
As a next stage, a study of simultaneous performance of plasma production and ion heating is in progress. Using a helical antenna, forward left hand polarized wave and backward right hand polarized wave will be excited simultaneously, and ion heating and plasma production will be performed, respectively. The experimental results on this scheme will be presented in the conference.

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Exact trajectory of a charged particle in several non-uniform magnetic fields

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We have derived exact expressions for trajectory of a charged particle in the presence of non-uniform magnetic fields. The types of magnetic field B include:
(1) $B(x) = (1 + a_1 x + a_2 x^2 + a_3 x^3)B_0$, or the cyclotron motion in x-y plane.
In the simplest case, where $B(x)=(1-x/L_B)B_0$, the exact drift speed V_D is given by
$V_D/v_0 = 1 - (2/m)(1-\mathrm{E}(m)/\mathrm{K}(m))$,
where v_0 is the initial speed of the particle, K(m) and E(m) are the elliptic integrals of the first and second kinds whose argument m is defines as
$m = 4\rho_0 / (\mathrm{L_B} + 4\rho_0 \sin^2(\alpha/2))$
where ρ_0 is an initial cyclotron radius at the origin, α stands for the initial phase angle of the gyration.
(2) $B_\varphi \propto 1/R$ in cylindrical (R,φ,Z) coordinate, or a simple torus.
The exact drift speed V_D in Z-direction for the initial pitch angle of $\pi/2$ is given by
$V_D/v_0 = \mathrm{I}_1(-\rho_0/R_0)/\mathrm{I}_0(-\rho_0/R_0)$,
where I_0 and I_1 are the modified Bessel functions of the zero-th and the first kinds, R_0 stands for the initial radial position of the particle. Note that the drift speed is independent from the initial phase angle α.
Also presented will be the exact trajectories and the drift speed for the arbitrary pitch angle concerning (2).

Application of the Binary Interaction Approximation to Plasma Oscillation

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The authors have developed a code to solve multibody problems for plasma oscillation using the BIA [1].
In a binary system with an impact parameter $b = b_0\cot(\chi/2)$, a typical relative velocity change Δg is given by
$\Delta g = 2g\sin(\chi/2) \sim \varepsilon g,\ \varepsilon \equiv 2b_0/\Delta l = U/K,$ (1)
where b_0 corresponds to $\chi = \pi/2$ scattering, Δl is the average interparticle separation, U and K stands for the potential and kinetic energies. In N-body systems with $\varepsilon << 1$, such as the fusion plasma, Equation(1) means that three-or-more body interaction is of order of ε^2 and can be ignored. It should be noted that the Debye lengths λ_D in fusion plasma generally satisfy $\lambda_D >> \Delta l$, thus typical interaction is characterized by the nondimensional parameter ε.
Let us assume that there is a single positive charge fixed at the origin and the electrons are randomly distributed initially around the positive charge.We will solve such a case using the BIA schem.

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P1-106

Accuracy Assurance in Binary Interaction Approximation for *N*-body Problems

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The authors have developed an algebraic model for multibody problems, and have shown that the momentum transfer cross-section with our model is in excellent agreement with the exact one [1]. In a binary system with an impact parameter $b=b_0 \cot\chi/2$, a typical velocity change in the relative velocity Δg is given by
$\Delta g=2g\sin \chi/2\sim \varepsilon g, \varepsilon\equiv b_0/\Delta l$(1)
where b_0 corresponds to $\chi=\pi/2$ scattering, and Δl is the average interparticle separation. In *N*-body systems with $\varepsilon\ll 1$, such as the fusion plasma, Eq. (1) means that three-or-more body interaction is of order of ε^2 and can be ignored. It should be noted that the Debye lengths λ_D in fusion plasma generally satisfy $\lambda_D\gg\Delta i$, thus typical interaction is characterized by the nondimensional parameter ε. This parameter coincides approximately with U/K, where U and K stand for the potential and kinetic energies.
For a given time interval Δt and a set of initial conditions $\mathbf{r}_i(0)$ and $\mathbf{v}_i(0)$ for *N*-particles, first calculate $\mathbf{r}_l^*(\Delta t)$ and $\mathbf{v}_i^*(\Delta t)$ using the exiting BIA code.
Second, calculate $\mathbf{r}_i(\Delta t)$ and $\mathbf{v}_i(\Delta t)$ from the same initial conditions then calculate $\mathbf{r}_i(\Delta t)$ and $\mathbf{v}_i(\Delta t)$ from $\mathbf{r}_i(\Delta t/2)$ and $\mathbf{v}_i(\Delta t/2)$. If $|\mathbf{r}_i^*(\Delta t)-\mathbf{r}_i(\Delta t)|$ and $|\mathbf{v}_i^*(\Delta t)-\mathbf{v}_i(\Delta t)|$ for any i are less than the ERROR TOLERANCE, then calculation will be proceeded for $t>\Delta t$. If not, the time interval is reduced to $\Delta t/2$ this procedure will be repeated until errors become within the ERROR TOLERANCE.

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Localization method using Microsoft Kinect for indoor structures

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Most nuclear fusion reactor is generally large and its buildings are also large. In such a large space, a user cannot easily recognize where he/she is. Therefore it is useful to provide the user current location information. The popular way to get the location information is to use Global Positioning System (GPS); however, GPS cannot be used inside the building. For overcoming this problem, many researches for localization have been proposed from the stand point of machine vision with cameras. Many vision-based localization methods using markers have been proposed [1]; however, it is necessary to install and allocate many artificial markers in large space. It costs a lot of manpower and the markers are eyesore. One method without the artificial markers is the natural feature point landmark database [2]. This method consists of two stages. In the first stage, we reconstruct the environment from omni-directional image sequences. We find out natural feature points by processing the images and create the natural feature points database in which the camera position and posture and image data is stored. In the second stages, when the user moves with camera, the current position is calculated by using this database. As a result, the user can know the current position and this system can provide suitable information to the user. For example, if he wants to operate one device, this system can provide the information of operation of the device in front of the user. An omni-directional multi-camera system is generally used for grabbing environmental image, however this type of camera is very expensive. Moreover, it is difficult to measure self-position accurately since experimental error of depth tends to be large only using a RGB camera. On the other hand, the Microsoft Kinect sensor, which has a RGB image camera and a depth camera, can improve this problem. Finally, we provide one example, which reconstructs the room shape.

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Plenary / Invited / Oral Sessions

(Tuesday 29th November)

2-D Microwave Imaging Experiments on KSTAR

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The 2-D Electron Cyclotron Emission Imaging (ECEI) system has been a major diagnostic tool for the study of various MHD modes during the last two KSTAR campaigns (2010 and 2011). Following the first application to sawtooth physics, the detailed study of the growth of ELMs [1] and a newly observed core instability [dual flux tubes only with electron cyclotron heating (ECH)] in both campaigns will be reviewed in this paper. The preliminary images obtained during the suppression experiment of ELMs using in-vessel control coils (IVCC) and meso-scale T_e fluctuations during the H/L transition will be addressed. A toroidal separated second ECEI system for 3-D reconstruction of the MHD instabilities which will verify the high *m* and *n* mode numbers and rotation speeds (poloidal and toroidal) for 2012 campaign will be discussed. At the same time, a unique 2-D Microwave Imaging Reflectometry system [2] which is capable to measure density fluctuation will be deployed for the 2012 campaign to study the turbulence based transport physics on KSTAR.

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Super-X Reality: Implementing an Innovative Divertor

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One of the most prominent challenges for a fusion power plant based on the magnetic confinement concept is that of handling the exhaust plasma. For this a divertor is implemented in current devices, where a thin layer (DEMO: ~1cm) of scrape-off-plasma is guided magnetically onto special targets. In reactor size devices such as DEMO the power density reaching the targets needs to be reduced significantly to avoid excessive target erosion, e.g. by radiation. At present it is not clear if a viable divertor solution for a full-scale reactor exists. The Super-X divertor concept could play a significant part in the design of large-scale fusion devices, in particular for tight aspect ratio, by offering a way of mitigating the power flux to the target. In this concept, the poloidal magnetic field of the divertor is used to take the target strike zone through an intermediate region of low poloidal field out to a larger major radius. The lower total field at the larger radius leads to a reduced parallel heat flux, whilst reducing the poloidal field enlarges the plasma wetted area. In addition, the intermediate region of reduced poloidal field increases the connection length of the plasma in a volume some distance away from the core plasma, allowing for volumetric effects to further reduce the power density. The current project to upgrade MAST incorporates two closed, actively pumped divertors in double null configuration capable, of realising, for the first time, this Super-X concept. Moreover, 2 sets of 8 coils controlling the divertor field allow the comparison of various divertor geometries with different strike point radius, connection length and flux expansion; including conventional divertor geometries. This flexibility, together with an extensive set of diagnostics, will be used to test the divertor physics leading to a reduced power flux at the target such as increased connection length, role of collisional processes, reduction of parallel power density, and the approach to detachment. SOLPS and other tools were used to assess the balance between competing design choices like recycling at the entrance, closure, pumping and achievable connection length. For example the geometry of the divertor throat represents a compromise between sufficient closure and tolerable recycling and modelling shows neutral compression factors in excess of 100 with the chosen design, with the promise of improved performance of the MAST core plasma. Furthermore, engineering challenges needed to be overcome, such as the control requirements of coil currents to better than 1% out of ~5kA total for some divertor coils to ensure the stability of the intermediate low poloidal field region. In this paper we will present the current Super-X design in MAST upgrade, discuss the basic design principles outlined above and the challenges they pose for the design and specification of the plant, as well as the expectations of the physics programme that will be possible.

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Influence of the isotope effect on the charge-exchange process between hydrogen isotopes and ions and atoms of plasma facing component materials

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The plasma-wall interactions remain one of the main areas of research in the physics and technology of controlled thermonuclear fusion. The conditions in the peripheral region largely determine also the processes in the main plasma. Numerical simulations of the near-wall and divertor region plasma in fusion devices are based on the knowledge of the rates of the elementary processes involving plasma particles and ions and atoms sputtered from the plasma facing components (PFC). The energies of collisions in the peripheral region are very low (about 1 – 500 eV/u). Under these conditions the charge-exchange becomes the dominant process for the neutralization and population of the excited states of plasma impurities and plays an important role in radiative cooling, particle transport and ions charge distribution. The isotope effect, found recently in charge-exchange reactions between hydrogen isotopes and α-particles [1-3] and PFC's ions (Li, Be, C) [4], manifests in a significant difference (up to several orders of magnitude) in the charge-exchange cross sections: the heavier the isotope the larger the cross section. This effect occurs at very low collision energies due to rotational interaction in close collisions – the scattering angle depends on reduced mass of colliding particles. Since DT burning plasma experiments are planned in ITER, hydrogen isotopic effects on transport of the sputtered particles is an important issue. We present here calculations of the charge-exchange cross sections in slow collisions of Li, Be, C and W ions with H, D and T. The isotope effect is studied in adiabatic approximation [5] which, in the theory of atomic collisions, is used to describe electronic transitions when the collision velocity is small and the nuclear motion can be treated classically. In this theory, there are no assumptions on the specific form of the electronic Hamiltonian, and only the smallness of the relative nuclear velocity is used. It results in a deeper understanding of the nature of nonadiabatic transitions. Since the isotope effect occurs at collision energies where the adiabatic theory applies, the adiabatic approximation is a natural theoretical framework for studying the effect. The cross sections are calculated with ARSENY code [6] based on the hidden crossing method. The charge-exchange cross sections in collisions of $W^{+}(6s)$ with H, D, T, which are presented here for the first time, at the collision energy of 30 eV/amu are 1.56E-19 cm^2, 1.18E-18 cm^2, and 1.25E-16 cm^2, correspondingly. This significant difference in the cross sections for H and T targets (three orders of magnitude) indicates the necessity for accounting the isotope effect in edge plasma modeling.

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O-6

Divertor Simulation study using the GAMMA 10 end-mirror cell

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Heat load of plasma particles led to divertor plates approaches tens of MW/m^2 on steady-state operation. Therefore, it is important to develop the divertor plates which attain the level of high heat load from the divertor plasma. The divertor simulation experiments are widely performed by using linear devices. In the GAMMA 10 tandem mirror, divertor simulation experiments were planned and started [1-3]. GAMMA 10 consists of a central-cell, two anchor-cells, two plug/barrier-cells and two end-cells. Plasma heating systems which are ICRF, NBI and ECH are installed for plasma production/heating and confinement. By applying ICRF in the central-cell and anchor-cells, end-loss plasma can be generated. In this plasma state, contribution of ion dominates in heat flux. In the case of superimposing ECH in plug/barrier-cell on ICRF-produced plasmas the effects of electron become significant. Thus, characteristics of end-loss plasma toward the divertor plate can be changed by controlling heating systems in GAMMA 10.

The purpose of the study is analysis of heat and particle flux in end-loss plasmas for divertor simulation experiments. Heat and particle fluxes were measured by using directional probes and calorimeters. In the case of only ICRF heating, heat-flux has strong dependence on diamagnetism which is time integrated in central-cell. While, particle fluxes are proportional to electron line-density in central-cell. These results mean that the characteristics of end-loss plasma are significantly influenced from the central-cell plasma parameters. It is also found that spatial profiles of heat and particle fluxes have peaks on the axis and are governed by the magnetic field configuration in the end-mirror region.

In this paper, we report the detailed results of characteristics of heat and particle fluxes from the end-mirror exit.We also describe an initial result of the background plasma simulation for understanding the experimental data by using two-dimensional fluid code

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I-6

Advances in Lower Hybrid Current Drive for Tokamak Long Pulse Operation: Technology and Physics

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Progress and achievements of LHCD technology for long pulse operation in tokamaks are reported, including the development of CW high power klystrons, and its evolutions towards ITER.
So far, the 3.7GHz / 700kW CW klystrons have been produced in series by Thales Electron Devices (TED) for Tore Supra. First series of eight klystrons, are operating now (3 MW/ 43s achieved after several days of operation). A prototype of 500 kW CW klystron operating at 5 GHz has been developed for KSTAR by Toshiba Electron Tubes and Devices (TETD) for the steady-state RF source. This prototype klystron has produced RF output powers of 300 kW / 800s and 450kW / 20s using during the commissioning.
The review also reports the situation on wave coupling and antennas, with the latest Tore Supra results of the new CW Passive-Active Multi-junction launcher; the antennae concept foreseen for ITER LHCD system.
Finally, the activity in view of ITER of the worldwide LHCD community is updated. An LHCD system capable of fulfilling several important tasks on ITER is presented. Based on experimental results, simulations of ITER scenario using LHCD have been revisited, including advanced modes of operation in tokamaks as well as the LHCD-assisted start-up. LHCD is found to be a key tool for two tasks: i) extend the burn duration, up to 500s (save Volt-seconds, from early plasma phases); ii) help accessing and sustaining Steady-State (drive far off-axis current, complementarily to Bootstrap Current, NBCD and ECCD).

ECW / EBW Heating and Current Drive Experiment Results and Prospects to CW Operation in QUEST

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The Q-shu university experiment with steady state spherical tokamak (QUEST) has been conducted in Kyushu University. Electron Bernstein wave heating and current dive (EBWH/CD) is one of attractive candidates to sustain the steady-state ST plasma.The plasma current of ~15kA with an aspect ratio of 1.5 was started up and sustained by only 8.2GHz RF injection. The long pulse discharges of 10 kA and 15 kA were attained for 37 s and 20 s in the limiter configurations, respectively. The new density window to sustain the plasma current was observed in the overdense plasma. The single-null divertor configuration with the high plasma current (<~25kA) was attained in the 17 s plasma sustainment.
In order to attain the steady state ST configuration, several topics have been focused in the QUEST. The phased array antenna system has been developed to excite an elliptically polarized O -mode in the oblique injection for the EBW/CD experiments. In the long pulse discharge with the limiter configuration, the hot spots were appeared, and the metal impurity contamination was observed. The plasma position and shaping for the divertor configuration has been optimized to avoid the hot spot formation. The hot first wall is being designed to reduce hydrogen retention at the wall which causes the discharge termination due to the hydrogen desorption in the long pulse discharge. Some trials for the CW operation in the QUEST will be introduced and discussed.

Recent Results from the Development of the Electron Cyclotron Heating System for JT-60SA toward High-Power Long-Pulse Operations

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Electron cyclotron (EC) waves are widely used in tokamaks for many purposes, such as electron heating, current drive and instability control, due to their capability of highly localized deposition profile. In addition, EC waves can be used to assist plasma initiation and to clean the first wall. In JT-60SA, an electron cyclotron heating (ECH) system will be installed and used for the above purposes. The frequency of the EC wave is 110 GHz, and the total injection power is expected to be 7 MW using 9 gyrotrons. Although the ECH system for JT-60U is reused as much as possible, modifications and developments of almost all components, such as power supplies, gyrotrons, transmission lines, antennas and control systems, are required, because the specifications for JT-60SA are much beyond those for JT-60U. In particular, required pulse duration at an output power of 1 MW is significantly extended from 5 s to 100 s. In 2010, pulse duration for 1 MW output was extended to 31 s by installing a new mode convertor, which was designed to reduce stray radiofrequency power [1]. An increase in the temperature of coolant water for the DC break of the gyrotron, which hindered the extension of the pulse duration, was found to decrease to about half of its previous value. And the temperature reached a steady state in about 10 s with a temperature increase of ~35°C. From this result, it was concluded that there is no serious limit for the gyrotron in extending the pulse duration. On the other hand, the temperature on the surface of the transmission line without cooling continuously increased; the temperature reached about 50-70°C for a 1 MW, 15 s output. To suppress such a temperature increase, whose typical time constant is on the order of 100 s, the straight section of the transmission line was covered with copper plates having water pipes. The effectiveness was confirmed in a 99 s operation at 0.3 MW. In 2011, in order to achieve longer pulse duration, 60.3 mm-diameter components were installed for the transmission line in place of the previous 31.75 mm-diameter components. A low-power transmission test at typically 1 mW and a subsequent high-power transmission test at ~0.5 MW showed no problem. Furthermore, a cathode resistor unit was modified to enable long-duration operations of the gyrotron power supply, and the capability of heat resistance and heat removal in the transmission line was improved. After these modifications, gyrotron operations toward the target value of 1 MW for 100 s were restarted and are now ongoing. In addition to the upgrade of the present gyrotrons, a new gyrotron is being developed. The main objective is to deposit the EC waves in the central region of JT-60SA plasmas for the toroidal magnetic field of 2.25 T; EC wave frequency of 130-140 GHz is desired for that purpose. In order not to increase the total number of gyrotrons, a dual-frequency gyrotron was designed. The second EC wave frequency was chosen to be 138 GHz from the viewpoint of the oscillation mode and the wavelength. In this combination, the oscillation modes for 110 GHz and 138 GHz are $TE_{22,8}$ and $TE_{27,10}$, respectively, and the ratio of these wavelengths is 5:4. A superconducting magnet enabling higher magnetic field is also being fabricated. A gyrotron output window made of synthetic diamond with a thickness of 2.3 mm will also be fabricated to allow the transmission of both frequencies. The fabrications will be finished in March 2012, and the operations of the new gyrotron will be started after its installation.

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Steady State Operation of Ion Sources for Fusion Plasma Heating

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The neutral beam injection (NBI) system for ITER will extract negative deuterium ions from radio frequency (RF) driven ion sources instead of cathode arc discharge sources [1]. However, quasi-steady state operation of ion sources with RF plasma excitation may also exhibit some problems like high temperature cathode discharge. A comparative research to excite ion source plasma at various RF frequencies has been started to clarify potential problems associated with steady state RF ion source operation for NBI.

Microwave power at 14 GHz has produced plasma in a test ion source with the volume less than 50 cm^3. Change in the magnetic field configuration has largely altered the performance of the ion source. The magnetic mirror geometry with the axis set perpendicular to both beam extraction and the electric field of the incident electromagnetic wave, created a dense plasma in the ion source, but the field hindered extraction of negative hydrogen ions (H^-) from the source [2]. A permanent magnet induces a strong axial magnetic field in the ion source of the current design. It produces a field large enough to realize the electron cyclotron resonance condition, while the field near the extractor hole realizes the “tent filter” geometry suitable for H^- extraction [3]. Experimental results of the internal conductor design will be also reported.

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Microwave Reflectometry Diagnostics: Present day systems and challenges for future devices

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Microwave Reflectometry, based on the reflection of a microwave beam at the plasma cut-off layer, is applied to measure not only the electron density profile but also plasma turbulence and plasma rotation. During the 1990s different approaches were developed to overcome the deleterious effect of the turbulence on the density profile measurement, becoming nowadays an almost routine diagnostic in many devices. Similarly, turbulence measurements have been extended from the initial single frequency systems to multiple frequency or ultra-fast sweep frequency systems. Moreover, in the last decade a new technique, Doppler reflectometry, has been developed to measure simultaneously plasma turbulence and flows with excellent spatio-temporal resolution. All these advances make reflectometry a very attractive diagnostic presently used in almost all fusion devices. Although it is not straightforward, the extension to future devices like ITER is still possible partially due to the limited access needed to accommodate the antennas and the possibility to use stainless steel or carbon-based materials for the antennas keeping the sensitive elements as microwave sources and detectors outside the radiation area. In ITER several reflectometry systems are being developed for profile and turbulence measurements [1]. Besides, reflectometry will be used as an alternative approach to the magnetic systems in long pulse operation for plasma position control [2]. All these reflectometry systems face some difficulties linked to the intense radiation and the long pulse operation. This imposes a careful design of the front-end to keep radiation and thermal and mechanical stresses at acceptable levels and to allow maintenance by remote handling [3]. Relativistic effects as well as problems associated with non-absorbed microwave stray-radiation should be also considered as they could hinder the achievement of the required measurement accuracy in some plasma scenarios. Besides, an intelligent handling of the huge amount of data is mandatory to allow real time monitoring of long pulse discharges.

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ITER instrumentation and control system toward long pulse operation

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ITER is a long-pulse tokamak with elongated plasma. The nominal inductive operation produces a D-T fusion power of 500 MW for a burn length of 300-500 s, with the injection of 50 MW of auxiliary power. With non-inductive current drive from the H&CD systems, the burn duration is envisaged to be extended to 3000 s.

The term ITER Instrumentation & Control (I&C) includes everything required to operate the ITER facility. It comprises three vertical tiers; conventional control, interlock system and safety system, and two horizontal layers; central I&C systems and plant system I&C. CODAC (Control, Data Access and Communication) system forms the upper level of the hierarchy, and is the conventional central control system of ITER architecture. CODAC system is responsible for integrating all plant system I&C and enable operation of ITER as a single integrated plant. CIS (Central Interlock System) and CSS (Central Safety System) also form the upper level of the hierarchy to supervising and integrating all plant system interlock and safety functions. Plant system I&C forms the lower level of the hierarchy, and provide dedicated plant data acquisition, plant status monitoring, plant control and plant protection functions to perform individual plant system operation under the supervision of central I&C systems.

Due to the long pulse operation of ITER, pulse scheduling and plasma control system shall provide flexibility be able to change discharge scenario and plasma control algorithms according to the phenomenon during the pulse. The concepts of plasma discharge scenario associate with pre -defined control algorithms are considered and further investigating collaborate with ITER parties.

ITER produces two different data streams, engineering data and scientific data. Engineering data contains plant system status data and alarms during plant system operation. The majority of scientific data is dominated by plasma diagnostics and some of these data shall also be used for real-time plasma control.

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Diagnostic- and other in-vessel components in the high-power μ-wave background of ECR heated steady-state discharges

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Incompletely absorbed Electron Cyclotron Resonance Heating (ECRH) beams which are multiply reflected and scattered at the highly reflective walls result in a nearly isotropic μ-wave background in the plasma vessel. During long pulse operation this stray radiation may cause severe overheating for μ-wave absorbing in-vessel components such as gaskets, bellows, windows, ceramics and cable insulations. The level of unabsorbed μ-wave radiation is particularly high in scenarios with moderate single pass absorption only such as high-density scenarios with X2-heating close to cut-off conditions, O2-, X3-heating or OXB mode-conversion heating. Extrapolations to W7-X with 8MW ECRH (140 GHz) and O2-heating yield values up to 150 kW/m ^{2}. With well absorbing plasma (X2, medium density) the stray radiation level is reduced by a factor of ~100.
In-vessel components of W7-X are qualified for 30 min loads of isotropic 140 GHz radiation at power flux densities of up to 50 kW/m^2 in the MIcrowave STray RAdiation Launch facility, MISTRAL, operated at IPP Greifswald. Microwaves are provided from a power gyrotron with adjustable duty cycle via a corrugated transmission line. An issue is the development of calibrated sensors for the isotropic radiation also as an operational diagnostic of W7-X.
Measures to protect sensitive poorly cooled items can be tight metallic shields with the remaining holes for pumping being smaller than λ/2 (<1mm). Additionally the power load may be reduced by μ-wave absorbers consisting of ceramic layers on cooled metal structures. The spurious sensitivity of a bolometer diagnostic to the μ-wave radiation background was reduced by a factor of 300 via μ -wave absorber coating in the detector enclosure and - additionally - μ-wave reflecting meshes in front of the detector. Windows are closed for isotropic μ-waves with conducting Indium Tin Oxide (ITO) layers while remaining transparent in the visible and UV. Critical issues under investigation are the shielding of cable isolations, the rather high effective μ-wave absorption of poorly cooled meshes, e.g. protecting flexible cooling pipes, the shielding of cryo-pumps and port-bellows.

Fatal damages due to breakdown on a diagnostic mirror located outside the vaccum vessel in JT-60U

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In fusion devices, the initiation of arcing has been an important plasma surface interaction issue, because it could cause damages on divertor and first wall materials and deteriorate the plasma performance consequently [1]. Recently, enhancement of the initiation of arcing on divertor surfaces has been pointed out due to the surface morphology change and transient heat loads, such as ELMs and disruptions [2]. Inside the vacuum vessel, it seems that the arcing has been frequently observed, and the existences are more or less recognized. However, it is likely that arcing or breakdown phenomena take place not only inside the vessel, but also outside the vessel. In this study, we will report that a diagnostic mirror in JT-60U, which was located outside the vacuum vessel, for Thomson scattering measurement optics in JT-60U has been seriously damaged by some discharge phenomena. Because the damages were so fatal that the mirror had to be replaced after the breakdown. Details of the characteristics of the damage are revealed from the observation of the trails. Furthermore, the mechanism to cause the damages is discussed based on the observation. Many fine trails were found on the surface. The trails could be categorized into two different types with respect to the trail width. It is thought that surface discharge was initiated on the mirror. The mechanisms to lead the damages were discussed based on the observation. This study issues warning on the components to be installed in ITER and future fusion devices both inside and outside the vacuum vessel.

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Summary of Japanese Researches on Tritium Science and Technology for Fusion Reactor

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In order to establish a D-T fusion reactor as an energy source, economical conversion of fusion energy to electricity and/or heat, having enough margins in tritium breeding, and insuring tritium safety must be simultaneously achieved. Scientists and researchers working on Tritium in Japan are now tackling with T related problems. Their research subjects can be categorized into two, (1) Science and technology and (2) Tritium safety.
Researches on (1) Science and technology have been carried out mainly Japanese University and JAEA under the project of “Tritium science and technology for a fusion reactor” supported by Grant in Aid for Scientific Research, Ministry of Education, Culture, Sports, Science and Technology (MEXT), Priority area No.467. (2007-2011) [1]. In addition JAEA is doing its original research works mainly related to ITER. Some works have also been done under the frame work of NIFS-LHD collaborations through Toyama Univ.
Researches on (2) Tritium safety have been carried out mainly under the frame work of LHD Project Research Collaboration of NIFS. Detailed subjects are (2-1) Behavior of Tritium in Environment includes (i) Water analysis(rain, river, groundwater, etc) , (ii) OBT behavior (measurement, Incorporation), (iii) Modeling (water, air, OBT, dose) and (iv)) Atmospheric analysis (Chemical forms) and (2-2)Studies on biological Effects include (i) establishment of a hypersensitive assay system that can be applied for experiments in radiation biology (cultured cells, transgenic mice), (ii) Biological responses to low-dose (rate) tritium radiation, especially to tritiated water (HTO) exposure and (iii) Molecular mechanisms of DNA damages and repair.
Many researchers from various universities, and institutes such as NIFS, JAEA and IEA (Inst. Environmental Science) in Japan are involved in the research programs given above. Most of their recent results are presented in 9th International Conference on Tritium Science and Technology, Nara Japan Dec.2010) [2], co-organized by the project “Tritium science and technology for a fusion reactor”and NIFS. Present paper summarizes recent results given by the project “Tritium science and technology for a fusion reactor” focusing on tritium issues to be solved for establishment of a fusion reactor.

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O-11

Estimation of decay heat in fusion DEMO reactor

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Decay heat of activated materials is important in safety of fusion DEMO reactor against loss of coolant-flow accidents. Effects of decay heat for main components of the SlimCS [1] DEMO reactor are studied with a one-dimensional code THIDA-2 [2]. The major parameters of SlimCS are a plasma major radius of 5.5 m, aspect ratio of 2.6 and fusion power of 2.95 GW [1]. The main reactor components consist of the inboard (IB) and outboard (OB) blanket modules, and divertor. The decay heats of IB blanket, OB blanket and divertor are calculated under the neutron wall loading of 2.5 MW/m^2, 3.5 MW/m^2 and 1.5 MW/m^2, respectively. Here, the blanket is assumed to be composed of structural material with F82H, tritium breeder with Li_2TiO_3 pebbles and neutron multiplier with $Be_{12}Ti$ pebbles. The blanket is filled with the mixture Li_2TiO_3 pebbles and $Be_{12}Ti$ pebbles. The surface of the blanket is covered with 0.2 mm-thick tungsten (W). The divertor is assumed to be made of W mono-block and F82H. For the reactor with a fusion power of 2.95 GW, the decay heat of IB blanket, OB blanket and divertor are roughly estimated to be as high as 29 MW, 146 MW and 6 MW, respectively, immediately after the shutdown of operation. This means that the blanket and divertor need to be actively cooled during several year storage in the reactor or hot-cell. Three days after the shutdown, the total decay heat decreases to 15 % of the initial one. However, this value is higher than decay heat for fission reactor. The total decay heat for a fission reactor with an electrical generating power of 460 MWe is 5.2 MW at three days after shutdown. In this paper, main radionuclides contributing to decay heat are estimated for different fusion powers in the fusion DEMO reactor. Finally, a detailed comparison of between fusion and fission reactor is discussed from view point of decay heat.

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O-12

OVERALL BURNING EFFICIENCY OF TRITIUM AND TRITIUM BALANCE IN A D-T FUSION REACTOR

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Tritium consumed in the plasma vacuum vessel must be supplied from the breeding blanket to maintain the self-sustainable D-T nuclear burning because no effective tritium resources from outside are expected at present. Then, the attainable tritium breeding ratio which is estimated from the neutron usage and the recovery efficiency of the bred tritium in the blanket system must be larger than the required tritium breeding ratio which is estimated from the tritium balance in a fusion reactor. Accumulation of tritium is also required for preparation of the initial inventory of next reactors to be constructed.

Recent plasma-wall interaction studies have revealed that a considerable amount of tritium introduced into the plasma vacuum vessel is trapped at the re-deposition layer, though the trapping behavior under the power plant condition is not qualitatively understood yet.

It is considered that trap of tritium to the co-deposits or tritium loss due to plasma driven permeation increases with proportional to the tritium introduction rate to the plasma vacuum vessel. The tritium inventory in the fueling system also proportionally increases with increase of the tritium introduction rate. The tritium introduction rate is decided from the overall burning efficiency and the energy output. Some tritium is also considered to introduce to the pedestal plasma region for the ELM control or for the Divertor gas cooling. Then, it is important to know the efficiency in plasmadization of tritium introduced and the efficiency in transfer of plasmadized tritium to the burning core together with the ideal burn-up in the core.

Effect of the overall burning efficiency on the tritium balance is discussed in this paper.

Poster Session 2

(Tuesday 29th November)

Modification of DOHEAT for optimization of coolant conditions in DEMO blanket

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For demonstration of power generation fusion reactors (DEMO), the engineering feasibility of blanket is a key issue. The difficulty of blanket design is that the blanket has the competing requirements: (1) fuel self-sufficiency, (2) heat removal, and (3) structural strength against thermal stress and electromagnetic force. For conceptual blanket design of fusion reactors, a two-dimensional (2-D) nuclear-thermal-coupled analysis code, DOHEAT, has been developed [1]. Use of the code has showed outstanding usefulness in the blanket design where detailed evaluation of neutron flux, nuclear heating rate, tritium breeding ratio (TBR) and the temperature of materials is required for various blanket concepts and trial-and-error-basis iteration is sometimes necessary. However, in the previous version of DOHEAT, the coolant temperature is given as an input data. Actually, a temperature change of the coolant along the cooling tube needs to be determined implicitly when the inlet temperature, pressure and flow speed are given. The coolant conditions are the parameters that have effect not only on blanket design, but also on system design including power generation system. For more systematic design, DOHEAT has been modified. The coolant condition calculation module was added into the 2-D thermal analysis module, and the temperature profile in the blanket was provided based on the nuclear heating rate profile and coolant temperature. A validation calculation indicates that DOHEAT provides reasonable results on the temperature profile in the blanket.

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P2-2

Removal of deuterium in lithium titanate by gas exposure

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In the fusion blanket, tritium is produced by nuclear reaction with neutron irradiation. Lithium titanate (Li_2TiO_3) pebbles will be employed as tritium breeding material in a water cooled solid breeder blanket [1], and the tritium generated in and released from the pebbles will be swept and collected by using sweep gas. As the sweep gas, helium with a little hydrogen gas will be employed. However, the mechanism on the removal behavior of tritium by the sweep gas has not been investigated in detail and must be clarified in order to improve the tritium recovery.
In the present study, Li_2TiO_3 pebbles were irradiated by deuterium ions in order to simulate the tritium generated in the pebble. The effects on the sweep gas exposure on the removal of implanted deuterium in the pebbles were evaluated.
The Li_2TiO_3 pebbles with a diameter of 2 mm were used as a sample. The pebbles were heated at 997K for 1h in a vacuum for degassing. After the degassing, the pebbles were irradiated to deuterium ions by using an ECR ion source. The ion energy was 1.7keV and the ion fluence was 3.0×10^{18} D/cm^2. In order to remove the implanted deuterium, the pebbles were exposed to helium gas for 1h. The exposure pressure was 0.01 MPa, and the exposure temperature was taken at RT, 373K, 473K, 573K or 773K. After the gas exposure, the amount of residual deuterium in the pebbles was measured by thermal desorption spectroscopy, TDS. The samples were heated from RT to 973K with a heating rate of 10 K/min and then kept at 973K for 1h. The desorbed gases which contain deuterium atoms were quantitatively measured by a quadrupole mass spectrometer. Deuterium retained in the pebbles desorbed in forms of HD, D_2, HDO and D_2O, and the desorption of HDO was the largest. The desorbed peak of HDO appeared for 580K with the pebbles after exposure at RT. This peak disappeared for the pebbles after the exposure above 473K. Also, the amount of residual deuterium rapidly decreased with the increase of the exposure temperature.
For comparison, the amount of residual deuterium in the pebbles after vacuum conservation at 373K was also measured. The residual gas for the vacuum-conserved pebbles was similar to that with helium exposure at 373K. These results suggest that helium gas exposure was not effective on the removal of deuterium in the present conditions.
The effects on helium gas exposure with a little hydrogen gas at different temperature will be also presented.

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Ion-beam induced luminescence of Er_2O_3

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Er_2O_3 electric insulating coating is being developed for reduction of the MHD pressure drop in Li/V-alloy blanket systems [1]. The Er_2O_3 coating is a promising candidate also for tritium permeation barrier of blanket systems made of other breeder and coolant such as Flibe and Li-Pb [2]. However, neutron damages of the coating in fusion reactors are big issues. Ion-beams can be used to introduce defects in sample surfaces to simulate the neutron damages. Changes in crystallinity of the irradiated sample may be inferred from ion-beam induced luminescence spectra. Tanaka et al. [3] measured the visible luminescence of Er_2O_3 coating samples by 100 keV H^+ and Ar^+ ion-beams irradiation at Osaka Univ. It was demonstrated that the ion-beam induced luminescence was potentially useful to evaluate changes in crystallinity of irradiated samples.
In the present study, the similar measurements of sintered Er_2O_3 samples were performed using an apparatus in NIFS for a better understanding about correlations between ion-beam energy deposition and luminescence spectra. The experimental apparatus consists of an ion-beam source, a collision chamber, and a CCD spectrometer. The ion source is a part of medium current ion implantor (ULVAC IM-200MH-FB) used for semiconductor production (Freeman-type). We could use primary Ar^+ beam of about 0.1 - several μA in current at 33, 50, and 70 keV in kinetic energy. In the collision chamber, a movable stage made of stainless steels was installed to set the sample disk (25 mmφ × 1 mm). In the present measurements, three luminescence-bands were observed in 490-520, 540-580, and 640-690 nm, respectively. Stark components of the band in 640-690 nm are clearly identified as ${}^4F_{9/2} \rightarrow {}^4I_{15/2}$ transitions of trivalent Er ions in the Er_2O_3 crystal. Intensity of the luminescence-band in 640-690 nm increased with ion current density, while intensities of the other bands decreased as the ion current density increased. Measurements on stopping power dependence of the spectra are also being planned.

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Electroplating of erbium on steel surface in $ErCl_3$ doped LiCl-KCl

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The functional layers such as a tritium permeation barrier and an electrical insulation layer coated on structural materials are the essential technology for liquid breeder blankets of fusion reactors. One of the candidates is the coating of erbium oxide (Er_2O_3) layer. The chemical stability of the oxide was extremely high, and this can allow a long-term application in reductive liquid breeders. Then, the oxide coating on the steel surface has been fabricated with some coating method, and the hydrogen permeation and the compatibility with Li have been investigated. An important issue for the layers is the improvement of interfacial adhesion strength between the layer and the substrate. The thermal stress was induced by the difference of the thermal expansion ratio of the oxide layer and the steel substrate, when the temperature increased or decreased. The stress destroyed the interface between the layer and the substrate. Then, the peeling off of the layer from the substrate was promoted. This phenomenon can be improved when the layer has a gradient composition with the substrate. An electrical plating method can make the gradient composition rather than the direct coating method of the oxide on the steel substrate. The oxide layer is formed by oxidation of the surface after the plating of Er metal on the substrate. The metal layer can work as self-healing layer by the in situ oxidation if some part of the oxide layer was damaged and peeled off from the metal layer in the flowing liquid breeders. The plating technology for the metal, which has extremely high affinity for the oxidation, was not established so far. The purpose of the present study is to make clear the feasibility of the erbium- metal plating on the steel surface by an electroplating method.

The electroplating experiments were carried out in $ErCl_3$ doped LiCl-KCl bath. Three electrodes of a reference electrode, a counter electrode and a working electrode were placed in the molten salt. The reference electrode and the counter electrode were made of Al-Li and glassy carbon, respectively. The rectangular shape specimen of 304 type austenitic steel was used as the working electrode. The temperature of the bath was 450oC. The constant electrical potential of 0.3V (Li_+/Li) was applied on the working electrode. The plating was performed for 1 hour. After the plating, the specimen was taken out from the molten salt. The surface was analyzed using XRD. The analysis results indicated that the formation of Er metal layer on the specimen surface. The metal layer was formed due to the electrical precipitation of Er^{3+}, which was made by the dissolution of $ErCl_3$ in the salt, on the working electrode in the molten salt.

Microstructure and Mechanical Properties on Oxide Insulator Coating before and after Thermal Cycling Test

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The oxide coating process is one of the attractive methods of the electrical insulator to restrain of the for the Magneto-Hydrodynamic (MHD) pressure drop in liquid breeding blanket systems such as Li and Li-Pb. It is well known that Erbium oxide (Er2O3) was shown to be a high potential candidate material for electrical insulator coating. Furthermore, Er2O3 coating is also effective to reduce hydrogen permeation in molten-salt blanket. Er2O3 oxide coating is one of the important processes to realize an advanced breeding blanket system. We investigated to develop to form insulator layer on the complicated shape ducts and large-area walls of breeding blanket components. Recently, we succeeded to form Er2O3 coating layer into large interior surface are of metal pipe using Metal Organic Chemical Vapor Deposition (MOCVD) process.
In this paper, we investigated the microstructure and mechanical property of Er2O3 coating layer on stainless steel 316 (SUS 316) plate before and after heat cycling test with hydrogen permeation. We found that the permeation reduction factor (PFR) of MOCVD coating layer showed high performance. From the results of TEM observations, we confirmed that Er2O3 coating layer with 700 nm thickness was formed on the SUS 316 plate and this layer was identified to poly-crystal phase because the diffraction fleck which was arranged like a ring was observed in the selected electron diffraction pattern. No macroscopic defects such as crack and peeling in Er2O3 coating layer were observed before and after permeation test. The change of crystallinity and mechanical properties of the Er2O3 coating layer on before and after heat cycling test will be reported in detail.

Fabrication of the hydrogen recovery unit in the molten salt loop Orosh^{2}i-1: Operational Recovery Of Separated Hydrogen and Heat Inquiry-1

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Liquid breeder blanket is promising for DEMO and commercial fusion reactors. Molten salt, such as Flibe ($LiF + BeF_2$) and Flinak (LiF + NaF + KF), is attractive liquid breeder materials because of (1) non-reactivity with the air and water, (2) low vapor pressure leading to low operation pressure, (3) low tritium inventory and (4) low MHD resistance. In the liquid breeder blanket system, the breeder is circulated between the reactor core and the outside as a coolant. The fuel tritium and heat generated in the blanket are transferred by the molten salt itself, and then extracted by a tritium recovery system and a heat recovery system located outside the reactor core. In order to keep high thermal efficiency for the heat recovery system, the tritium recovery system should be operated at high temperature, i. e. 550 ^{o}C of blanket outlet temperature.

In order to examine the compatibility between the hydrogen recovery and the heat recovery, an experiment with a molten salt loop system, Operational Recovery Of Separated Hydrogen and Heat Inquiry – 1 (Orosh^{2}i-1), has been initiated in National Institute for Fusion Science. In the present study, a hydrogen recovery unit for Orosh^{2}i-1 was fabricated. The hydrogen recovery unit consists of a hydrogen permeation tube made from pure Ni. Prior to installation to Orosh^{2}i-1, hydrogen permeation rate was evaluated for the hydrogen recovery unit filled with helium-hydrogen gas mixture.
The feasibility of advanced hydrogen permeation materials, such as pure V, Nb, Ta and their alloys, for the hydrogen recovery unit is also discussed.

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Hydrogen Permeation in Stainless Steel Modified with Surface Nitride by Electrochemical Technique

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Molten fluoride salt such as Flibe and Flinak has high potential for liquid blanket in nuclear fusion reactor [1]. Fluoride salt, however, corrodes structural materials such as steel at high operation temperature over 500°C. The structural materials require compatibility with fluoride salt. And leakage of the tritium, which is bred in the blanket system, has to be prevented. In addition, the main materials for heat exchanger have to be as high in thermal conductivity as steel. To begin with, compatibility with fluoride salt was examined on several materials. It was revealed that nitride has compatibility with fluoride salt [2]. And a surface modification method with nitride on metal surface using an electrochemical technique in molten fluoride salt was developed. The surface modification, which consists of graded composition structure, is expected to be more robust than such as ceramics coating film against defects such as peeling, crack and pinhole. Considering in-situ maintenance of structural components in the blanket system, nitriding treatment in molten salts themselves might be more advantageous than other gas phase nitriding treatments such as ion nitriding and radical nitriding. Stainless steel 316 (SS316) was successfully nitrided [3]. A SS316 specimen in LiF-KF-Li_3N (49mol% LiF, 49mol% KF, 2mol% Li_3N) was treated by potentiostatic treatment for 100 min at 600 °C and 1.0V vs. Li^+/Li. Formation of multilayered structure near the surface was examined by EPMA, SEM, XPS and XRD. CrN was formed from the surface to the depth of 2μm and condensed in the layer. The surface structure changed drastically in composition and crystal structure with nitrogen diffusion. α-$Fe_{x(x>8)}N$ was also formed as solid solution that nitrogen atoms are inserted between iron atoms. It was formed from the surface to a depth of 35μm as an inside diffusion layer. Considering tritium permeability, hydrogen permeability is being measured by a build-up method. With surface modification of nitride on SS316 surface, its hydrogen permeability will be discussed.

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The evaluation of heat transfer for the sphere packed pipe by using a molten salt under high heat flux

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The first wall of the helical type fusion reactor FFHR is supposed to be imposed by high heat flux of 1 MW/m^2 at most[1] and to be cooled by a molten salt Flibe. With several attractive features, however, the Flibe is high Prandtl number (Pr) fluid and has a problem of poor heat transfer properties. In order to overcome this disadvantage, heat transfer enhancement by a sphere-packed pipe (SPP) has been proposed and tested for several conditions. Its basic heat transfer characteristics were evaluated[2][3][4]. For the case of high heat flux above 1 MW/m^2 is to determine empirical correlations between Nu and Re numbers, however, the validity of these correlations should be verified since there are large temperature change near the heated wall to induce large changes in the thermal properties like viscosity. In this study, therefore, heat removal experiment is conducted under the condition of high heat flux of 1 MW/m^2 in order to propose a heat transfer correlation equation available under the condition of large viscosity change. Moreover, fin effect is also evaluated. In the experiment, Tohoku-NIFS Thermofluid loop (TNT loop) is used to circulate a high temperature molten salt (HTS). Heat flux can be imposed up to about 1 MW/m^2 at test section by electrical heating. The test section is a pipe with diameter of D=14 mm, 20 mm in length and 0.5 mm in thickness filled with pebbles of D/2 diameter made of aluminum or SS304 whose thermal conductivities is 230 or 18 W/(m·K), respectively. From the experimental results, there is no significant difference in the heat transfer characteristics between aluminum and SS304 pebble cases. This means that the fin effect does not appear even in the high heat flux case. In comparison of different heat flux conditions, there is an apparent difference in the Nu number. It can be considered that this difference is attributed to the change in viscosity due to the temperature change between the heated wall and bulk fluid. By introducing the ratio of viscosity at the bulk temperature to that at the heated wall (μ_b/μ_w), the heat transfer correlation equation is modified based on the experimental data.

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Effects of Argon and Water Plasma Treatments on Philippine Coconut Fibers

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Philippine Coconut fibers (coirs) are organic materials which are naturally hydrophilic in nature. Low temperature plasma treatment was used to modify the surface and physical properties of coirs. Samples were treated with argon and water plasma generated using a Plasma Enhanced Chemical Vapor Deposition (PECVD) facility. The effect of varied discharge conditions was investigated. Scanning Electron Microscopy (SEM) and Fourier Transform Infrared Spectroscopy (FTIR) were used to determine the physical and chemical changes between untreated and treated samples. It was observed that argon plasma treatment lower the hydrophilitic nature of the coirs and eventually convert it to hydrophobic fibers.

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P2-10

Anti-bacterial property of oxygen-ion treated cotton gauze

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The anti-bacterial property of oxygen-ion treated cotton gauze is determined using contact angle measurements, scanning electron microscopy (SEM), and Escherichia coli (E. coli) adhesion tests. Cotton gauze samples are irradiated using oxygen low-energy gas discharges using a gas discharge ion source (GDIS) facility. Ion energies varied by changing the plasma discharge current (Id) from 1mA to 3mA with 1mA intervals. Results show oxygen plasma treatments modify the gauze surface in morphology. Oxygen treatment shows that as the Id increases, the gauze becomes more hydrophilic. The oxygen treated gauze samples exhibited reduced E. coli attachment on the samples.

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P2-11

Sterilization of polyethylene terephthalate (PET) using low-pressure glow discharge H_2O_2 and O_2 plasmas

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Low-pressure glow discharge hydrogen peroxide(H_2O_2) and oxygen (O_2) plasmas were used to sterilize the *Bacillus subtilis* and other spores artificially inoculated on polyethylene terephthalate (PET) sheets. A plasma enhanced chemical vapor deposition (PECVD) facility developed at the Plasma Physics Laboratory, National Institute of Physics (NIP) was used to treat the samples. The effect of varied discharge conditions on the decimal reduction value (RDV), which is the time required to inactivate or destroy 90% of the original population of microorganisms, was investigated. The results showed that the sterilization efficiency was closely related to the plasma exposure time, gas flow rate and sample temperature. Furthermore, it was found out that placing a negatively biased copper shield around the test substrates, which confined the chemically active species in the plasma that are believed to be responsible for the inactivation of the bacteria, enhances the effectiveness of the plasma sterilization.

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Wettability and Antimicrobial Property of Plasma Treated Polyethylene Terephthalate (PET)

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Polyethylene terephthalate (PET) sheets were treated using a plasma enhanced chemical vapor deposition (PECVD) to enhance the wettability properties and to obtain antibacterial surface properties. Bacterial adhesion performance of PET materials was tested using contact angle measurements, scanning electron microscopy (SEM), Fourier transform infrared spectroscopy (FTIR), and adhesion tests. The quantity of Bacillus subtilis adhered onto the different PET sheets was quantitatively determined using colony forming units (CFU) plate counting in vitro. The results showed that changes in surface morphology and chemical composition caused by plasma polymerization and etching mechanism enhanced the bacterial repellence of the polymer surfaces.

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P2-13

I_c analysis of Nb_3Sn strand cable-in-conduit conductor under the electromagnetic force by the structural mechanics

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The Cable-In-Conduit Conductor (CICC) with Nb_3Sn superconducting strand is the most popular conductor for high-field magnet of fusion devices. The CICC is composed of assembled twisted strands and has over hundreds of strands which get tangled up surrounded by conduit. Although the mechanical strength of the CICC is high enough against very large electromagnetic force (EMF) during energizing the magnets, the superconducting characteristic such as I_c is sensitive to the bending distortion and degradation of the performance seems unavoidable. Our approach to analyze the conductor performance is using structural mechanics based on measured strand traces inside the conduit. We simulate the strand traces under EMF using beam model which uses measured traces as initial state. The results indicates the detail information about the degradation of I_c under the large EMF, which is not reported ever before.

Analysis of contact lengths of strands with Cu blocks in CICC joints

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It is observed that measured critical current of Cable-in-Conduit-Conductor (CICC) for ITER TF coil becomes lower than expected one. This comes from unbalanced current distribution in the CICC. If a lot of strands carry different currents each other, some strands reach the critical current faster than others during charging and discharging and can cause degradation of the CICC. Unbalanced current distribution at steady state is caused through contacting resistance between strands and Copper (Cu) blocks, saddles or sleeves at CICC joints. In order to evaluate the contacting resistances, we measure 3 dimensional positions of all strands in the CICC for LHD OV coil, and then we measure contact parameters such as number and lengths of strands which appear on surface of the cable contact with the Cu blocks, which are inversely proportional to the contacting resistance. Then, we developed the numerical code which simulates strand positions in the CICC, and compare the analyzed contact parameters with measured ones. It is found that the both results are in good agreement and hence the numerical code is available for evaluating the contact parameters. We apply the code to various CIC conductor joints, and thereby obtain optimal joint parameters.

P2-15

Measurement Technique of Stress-Strain Properties for Cryogenic Materials by Neutron Diffraction

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Strain of superconductive wire for magnets should be controlled for a good performance of the magnet system because properties of many superconductive materials show deterioration by strain on wire. Accordingly, strain measurements for superconductive wire are required. As one of the method of strain estimation on superconductive wire, *in-situ* neutron diffraction measurement has been considered by taking an advantage of its possibility to measure the superconductive material through sheath layer of wire. On the other hand, structural materials for superconductive magnet are required high strength at cryogenic temperature. Therefore, estimation of stress-strain properties of the structural materials at cryogenic temperature is of the same importance as that of superconductive wires. In this study, we developed uniaxial tensile machine for cryogenic temperature test applicable to neutron diffraction measurements in order to low-temperature tension test for superconductive wire of magnets including low-temperature structural materials [1]. Stress-strain relation of a structural material and a superconductive wire at low-temperature obtained using the tensile machine will be introduced for a demonstration of the in-situ neutron diffraction measurement method.

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Long-Term Monitoring of Hydraulic Characteristics of LHD Poloidal Coils

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We present fourteen-year data summary of hydraulic characteristics of the large helical device (LHD) poloidal coils. The superconductors of the poloidal coils are cable-in-conduit conductors (CICC) cooled by circulated supercritical helium. Because the CICC has narrow cooling channels, impurities in helium, such as a metal piece, ice of impurity gas and oil from helium compressors, might obstruct the circulated helium flow in the conductor. The flow obstruction results in degradation of stability of the conductor. Therefore, we have continuously monitored a pressure drop between the inlet and the outlet of each coil since 1998 when the LHD operation started. The observed pressure drops were converted into dimensionless friction factors to eliminate the effects of a flow rate and a configuration.

The coils experienced a cool-down, an experimental campaign and a warm-up in every fiscal year. The mass flow rate of supercritical helium was approximately 50-60 g/s per coil and the Reynolds number was 2000-3000 during the experimental campaign. The long-term monitoring showed discontinuous decreases in friction factor with each experimental campaign over the first seven years (the second to the eighth campaigns). That is a change for the better. The friction factors then remained approximately constant for the next six years (the ninth to the fourteenth campaigns). The long-term operation of the LHD demonstrates that the initial hydraulic characteristics of coils with CICC can be maintained for years. Although the cooling system has operated smoothly in the most recent campaign (the fifteenth campaign), the observations indicate an apparent increase in the friction factor.

In the poloidal coils, fine mesh filters were installed at the inlet to trap particles of impurities. The filters actually trapped impurities during cool-down of the coils, which were confirmed by monitoring the pressure drop of the filters. A clogged filter were warmed up and evacuated to clear the impurities. During the past fourteen years, the filter clogged in the first three years and the most recent campaign. Therefore, particles of impurities that passed through the filter during cool-down probably influenced the friction factor in the coil.

P2-17

Discussion of heat transfer to liquid helium on surface orientation dependence

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For the stability of pool boiling superconducting magnets, heat transfer characteristics of conductor surfaces are essential. Heat transfer surface orientation may be varied in a magnet. Helical coils of the Large Helical Device (LHD) are an example. It is well-known that heat transfer characteristics depend on the orientation.[1] To date, a relatively large copper surface was prepared, and the dependence of liquid helium heat transfer on surface orientation has been studied experimentally for the stability analysis of the helical coil.[2] This paper discusses that the experimental date compares to theory and empirical laws.

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Design Study of a 15 T Test Facility for High-current Superconductors

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Future fusion power plants need larger scale and higher field superconducting magnets than ITER magnets, the highest field and conductor current of which are 11.8 T and 68 kA for TF coils and 13.0 T and 45 kA for CS coils. It is well known that the sufficient length of testing area is necessary to examine the superconducting properties of large conductors. Therefore, the existing 9 T test facility in National Institute for Fusion Science [1] is planned to be upgraded to 15 T solenoid coils with the large cold bore of 0.7 m. The existing 9 T split coils are replaced to a set of solenoid coils. A sufficient testing area longer than 1.5 m can be obtained by adopting conductor samples with coil shape even in the case of one turn coil. The 15 T solenoid coils are designed to be divided into two parts, which are the inner coil and the outer coil. The larger coil sample can be tested at 7 T with only the outer coil.

High coil current density of 70-90 A/mm^2 is needed to attain 15 T at sample position, the radius of 0.30 m, under the restriction of the outer diameter less than 1.3 m. The highest field of 7 T is suitable for the outer coil made of NbTi conductors to increase the average current density. As a reference design, the magnetic field at sample position is 14.7 T when the center magnetic field and lengths of inner/outer magnets are 14.0 T and 1.2 m/1.0 m, respectively. The highest field in the coils is 14.99 T. When only the outer coil is installed, 7.4 T can be generated at the radius of 0.44 m. The total weight of the coils is estimated at 8.0 tons.

Rutherford cables are candidates to attain the high current density. An external protective resistance is used for the coil protection. In order to suppress the discharge voltage less than 800 V, the operating current is set at 6 kA. The copper current density of the conductor is set at 112 A/mm^2 to suppress the hotspot temperature below 200 K at the discharge time constant of 16 s. In these design criteria, the expected current densities are 65 A/mm^2 for the inner coil and 81 A/mm^2 for the outer coil, by increasing the load factors, Iop/Ic (the operation current / the critical current) to 88% and 81%, respectively. These high factors seem marginal for the high field magnets. The thickness of inside coil cases is 10 mm including the ground insulation, and that of outside case is 30 mm to withstand the strong magnetic force. The detailed mechanical analysis is a future work. The highest field in the outer coil can be reduced by adopting longer inner coil than the outer coil. In the case that the length of inner coil is elongated from 1.0 to 1.2 m, the highest field is reduced from 7.7 T to 7.08 T and the load factor on the loading line of the outer coil is reduced from 87% to 81%. In this paper the optimization of the coil figure is discussed.

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Directory on Recent Information about Operational Experiences on Large Superconducting System and Cryogenic Facility at Fusion and Particle Accelerator Research Organizations

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Large superconducting and cryogenic technologies become dominantly applied in the area of high energy particle accelerators and magnetic fusion experimental devices. The purpose of the study is to accumulate and construct an extensive and beneficial directory about their operative experiences from the proceedings of international conferences held during past 20 years. With a preliminary investigation more than 600 papers were collected and each experience as well as device specifications was tabulated as a data card in the form of Microsoft Excel format which yielded an extensive directory (database) about 91 devices. These are ready for query for key words with FileMaker Pro. The report talk about the several result obtained so far but the work must keep continued to expand the database from time to time and expects any collaboration of the relevant researches and engineers to the authors. It is intended to be open before public.

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Thermal hydraulic analysis on supercritical helium in cable-in-conduit conductor of helical coils for FFHR-d1

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A design study on the helical-type fusion reactor (FFHR) has been promoted in National Institute for Fusion Science (NIFS) [1]. In the latest design for the helical DEMO reactor FFHR-d1, the maximum nuclear heating in the helical coils is estimated to be 0.6 mW/cc due to the restriction of the blanket space. In the case of the continuous helical coils, it has become a major issue from the aspect of cooling. In the present study, a one-dimensional thermal hydraulic analysis in the longitudinal direction of the superconductor was conducted in order to evaluate the temperature rise, and the temperature and pressure of the supercritical helium in the 480-meter-long cable-in-conduit conductor (CICC) of the innermost layer of the helical coils with the largest heat load were obtained under the supposed operational condition.

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P2-22

Conceptual design of dc power supplies for FFHR superconducting magnet

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FFHR is a design study of a fusion power plant, and it will use three sets of superconducting coils to make the magnetic field to confine the plasma.
The FFHR is not a fusion plasma experimental device, and the requrements to the power system for superconducting coils are different with the LHD case. For example, the magnet field configulation will be optimaized and it doesn't require the ability of modification.
At the sametime, the stored energy of FFHR magnet system may be much larger than the LHD case, so the protection system must be designed to supress the coil terminal voltage or unbulance of eletro-magnetic force while the current dumping.
Under these requirements, a conceptual design of the power system was studied.
In this design, all of coils are connected in series and only one dc power supply is used for excitation.
With this circuit configuration, the current ratio between coils is kept constant and the electro-magnetic force is balanced eventhou the coils are under the excitation or dumping.

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Radial build design in the Helical DEMO reactor FFHR-d1

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FFHR-d1 is a conceptual design of the helical DEMO reactor being developed at the National Institute for Fusion Science. It has many advantages such as the steady-state nature and lack of plasma current operation [1]. In the LHD-type configuration, the space between the plasma boundary and the nearest region of the superconducting helical coil is small especially at the inboard of the torus. Many components have to be installed in the radial build, and geometrical positions of the components have to be guaranteed during the operation. They are the first wall, the tritium breeding blanket, the neutron radiation shield, the vacuum vessel, the thermal shield with the vacuum gap, and the coil can for the superconducting helical coil. An estimation of the distance for the FFHR-d1 is 89 cm which includes the blanket space of 70 cm [2, 3]. Each component changes its position by the surrounding conditions, such as the mechanical support and electromagnetic forces, etc. For example, the temperature difference at the superconducting magnet between the construction phase and the normal operation phase is approximately 300 K, which causes a thermal contraction of 4 cm. Furthermore, deformation of the magnet by the electromagnetic force can be 1-2 cm [4]. To guarantee the geometrical position at the normal operation phase, the following design criteria have been proposed. (1) The breeding blanket and the radiation shield are divided into several sectors in the poloidal cross-section to prevent hoop deformation. (2) Thin vacuum vessel sector is attached to the blanket sector and each sector is connected to each other by using bellows. (3) The blanket is connected to the fixed support nearby the vacuum vessel so that the displacement does not affects the gap. (4) The thermal shield is attached to the outer surface of the vacuum vessel. (5) The coil position at the construction phase is set taking into account the thermal contraction and the electromagnetic hoop deformation. The detailed geometry and the support method for the radial build components will be introduced in the presentation.

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P2-24

Feasibility study on applying cryogenic oscillating heat pipes to fusion magnet

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Improvement of superconducting magnets for high magnetic field and high heat load is an important subject to achieve the early realization of nuclear fusion power generation. Application of high-temperature superconductors (HTS) to magnets has widely been studied, because HTS magnets may achieve higher magnetic fields with less operation cost and higher stability against a coil quench compared to low-temperature superconducting magnets. However, it is difficult to remove the local heat generated in an HTS magnet because the thermal diffusivity decreases as the operating temperature increases. Therefore, large temperature gradients are easily produced in magnets, which could cause degradation of superconducting properties and mechanical damages by thermal stresses.
A new method of including cryogenic oscillating heat pipes (OHPs) in the HTS coil windings as a heat transfer device has been suggested [1]. The OHP is a highly effective two-phase heat transfer device which can transport several orders of magnitude greater heat flux than the heat conduction of solid metals. In our previous work [2], the performance of cryogenic OHPs were intensively examined. The typical effective thermal conductivity was observed to be > 10 $kWm^{-1}K^{-1}$ at 20 K. The results indicates the possibility of dramatically improving the performance of HTS magnets by using cryogenic OHPs.
Encouraged by these successful experiments, the feasibility of applying cryogenic OHPs to a fusion magnet is examined with a preliminary design of coil windings having a rectangular cross-section (1.8 m wide and 0.9 m high). In the design, HTS superconductors are stacked in layers and cooled by heat conduction from the cooling panels which contain channels for refrigerant and maintain the temperature at 20 K. Using this structural model, the temperature distribution with nuclear heating of 1 kWm^{-3} has been calculated by a finite element thermal model (ANSYS Release 12.0). In the result, the temperature rise has been limited to 3.1 K when the cooling panel includes OHPs, which is lower than 6.3 K without OHPs. It is concluded that cryogenic OHPs enable to eliminate steady-state heating more effectively. Further investigation on thermal analysis including the transient thermal phenomenon like a magnet quench is also discussed.

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Potential Use of HT-SC Wires in Fusion Reactor Application

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As a next heliotron-type fusion reactor, FFHR-2m2 is now designing under various preconditions, in which a severe condition of maximum magnetic filed of 12 T with stored energy of 160 GJ is subjected to the magnet technology. In order to realize this large scale fusion reactor, advanced magnet technology is inevitably required. For maintaining stable and homogeneous current distribution under operation, a number of twisted SC wires are braided and placed into the metallic outer envelope, which acts as strengthening member and stabilizer. Thus the conductor experiences complicated stresses during fabrication process and under operation. Due to achieving the higher performance, several options are proposed by selecting the SC material, the conductor type and the cooling method. The use of HT-SC wires generates advantage for designing the fusion reactor. It is possible to operate the magnet with high operation current of 100 kA at a relatively higher temperature at 20 K.
As candidate of practical HT-SC wires, REBCO coated conductor (RE = Y, Gd and Sm) and BSCCO multi-filamentary tape might be considered In the first part of the present paper, it is attempted to review their general performance on the engineering critical current and its magnetic field and stress dependences and to discuss the applicability to the high performance fusion reactor.
The more important performance relates to the mechanical- electromagnetic properties of wire and conductor. In the second part, our recent results on REBCO and BSCCO wires are presented on the mechanical properties and their temperature dependence, the critical current and their magnetic field and strain dependences. Further the local strain exerted on the SC element and its external stress and temperature dependences are reported, which have been directly measured by means of quantum beam techniques. Based on the experimental results, we discuss the intrinsic strain effect, the reversible strain limit on critical current under various temperature and magnetic field conditions and elucidate the influence from the thermal strain and the fabrication condition. As a summary, we point out the key technology to accelerate future progress of their performance and the potentiality to apply the large scale fusion reactor.

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HTS Current Lead Units Prepared by the TFA-MOD Processed YBCO Tapes

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High temperature superconducting current leads have been prepared by the TFA-MOD processed YBCO tapes. The YBCO tapes are 5 mm in width, about 130 μm in the overall thickness and 190 mm in length cut from the long tape. The YBCO superconducting layer with 1.5 μm in thickness is formed on oxide buffer layers/Hastelloy substrate tape. An Ag layer with 26 μm is deposited on the YBCO layer to improve the stability and protect the YBCO layer against the moisture of open air. The critical current Ic of the tapes ranges from 122 A to 152 A at 77 K in self-field, the mean Ic being 140 A.
A current lead unit is composed of twenty YBCO tapes soldered to Cu caps at both of ends, a GFRP board and a pair of stainless steel boards as a shunt. The twenty tapes are arrayed in five rows of stacked four tapes in parallel. The transport current of ten current lead units ranges from 1.6 kA to 2.5 kA at 77 K, the mean current being 2.2 kA. The voltage at Cu end caps is around 0.4 mV at transport current of 2 kA, which corresponds to small resistance of 0.2 μΩ between YBCO tapes and Cu caps.
The heat leakage of the current lead unit with 150 mm in length between 77 K and 4.2 K is estimated to be 273 mW. Therefore, the heat load at 2 kA corresponds to 137 W/kA, which is around one order of magnitude smaller than that of conventional current lead (1.2 W/kA). The present YBCO current leads with large transport current and small heat load are promising for superconducting magnet systems.

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Accurate and Stable Numerical Method for Analyzing Shielding Current Density in High-Temperature Superconducting Film Containing Cracks

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Although several methods have been so far proposed for analyzing the shielding current density in an high-temperature superconducting (HTS) film, they can be applicable only to an HTS film containing no cracks [1], [2]. In other words, the method for calculating the shielding current density has been well-established only for the case where an HTS film does not contain any cracks. On the other hand, it has remained developing for the case where an HTS film containing cracks.

In the present study, a numerical method is developed for analyzing the shielding current density in an HTS film. When an HTS film contains a crack, an additional boundary condition is imposed on the crack surface and it can be easily incorporated into the weak form. Although the weak form can be numerically solved with the essential boundary conditions, the resulting solution does not exactly satisfy Faraday’s law on the crack surface. In order to resolve this problem, the following method is proposed: *a virtual voltage* is applied around the crack so as to make Faraday’s law satisfied numerically.

A numerical code for analyzing the shielding current density is developed on the basis of the proposed method and, by means of the code, the permanent magnet method [3], [4] is investigated numerically. Especially, the influence of a film edge or a crack on accuracy is assessed. The results of computations show that, if the magnet is positioned in relation to a film edge or a crack at a distance equal to three times its radius or less, the accuracy of the permanent magnet method will be degraded remarkably.

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Microstructure of V3Ga Superconducting Wire Using Cu/V with High Ga Contents

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Our co-worker, Hishinuma et. al. has established a new route Powder-In-Tube (PIT) process using a high Ga content Cu-Ga compound in order to improve the superconducting property of the V_3Ga compound wire. In this study, we investigated microstructure of this high Ga content Cu-Ga/V composite superconducting wire. The different contrasts of matrix, V-Ga phase and Cu-Ga core were observed by SEM observation in cross section of 19 multifilamentary wire. And V-Ga phase was confirmed by SEM mapping. The area fraction of V-Ga phase increased when Ga content increased from 30% to 50%. Thin film sample with V-Ga phase for TEM was fabricated by FIB and observed by TEM in detail. Selected area diffraction pattern was obtained for V matrix, V-Ga phase and Cu-Ga core. The ratio of V to Ga for V-Ga phase was probably V_3Ga according to the EDS result. There was a linear interface between V matrix and V-Ga phase, while the interface between Cu-Ga core and V-Ga phase was not linear. On the other hand, there were some granular grains observed in V-Ga phase wear Cu-Ga core.

P2-29

Microstructure and Superconductive Property of MgB2/Al Based Composite Materials

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MgB_2 has the higher critical temperature of superconducting transition (T_C : 39K) among the intermetallic compound superconductive materials, however, MgB_2 is hard for practical use because of its unworkable and lower critical current density (J_C) in a high magnetic field than Nb-based superconductive materials. We have developed the original method of three-dimensional penetration casting (3DPC) to fabricate the MgB_2/Al composite materials. In the composite material we made, MgB_2 particles dispersed to the matrix uniformly. Thus, these composite materials can be processed by machining, extrusion and rolling. The T_C was determined by electrical resistivity and magnetization to be about 37～39K. In this work, we made composite material with ground MgB_2 particle with the purpose of extruding thinner wires of composite material, successfully produced 1mm wire and changed the matrix from pure Al to Al-In alloy. J_C of composite materials with the matrix of Al-In alloy was calculated from the width of the magnetic hysteresis based on the extended Bean model. The result was better than that of MgB_2/Al composite material without Indium.

Fundamental evaluation of joint resistance in mechanical butt joint of a stacked GdBCO conductor

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We proposed a remountable high-temperature superconducting magnet where segments of the magnet can be mounted and demounted repeatedly with mechanical joints, since fusion reactors usually needs superconducting magnet with complex geometry as seen in the helical system [1, 2]. In our latest study [3], we achieved $1.05 \times 10^{-12}\ \Omega \cdot$m of joint resistance for mechanical butt joint of a stacked BSCCO 2223 conductor. In this study, as a next stage, we experimentally evaluated joint resistance of a stacked GdBCO conductor in a strong magnetic field. The GdBCO conductor consisted of 4 GdBCO tapes and copper jacket. For comparison, we carried out the testing of direct joint and indium film compliant layer joint using an indium film of 10 μm thick. The experimental result showed that joint resistance decreases with an increase in joint stress, and the minimum joint resistances were 1.5 μΩ and 1.0 μΩ for the direct joint and the indium film compliant layer joint. However, the joint resistance of the stacked GdBCO conductor was significantly larger than that of the stacked BSCCO 2223 conductor obtained in the previous study [3]. The reason could be explained by smallness of cross-sectional area of conductive metal layer in the GdBCO in comparison with that in BSCCO 2223 tapes since the current may flow only through the metal layer at the joint region. The effect of the thickness of the metal layer on the joint resistance was evaluated by current distribution analysis for a butt joint model of the GdBCO tape. In the model, the tape consists of GdBCO layer, silver layer and copper layer. In addition, a virtual material region is introduced in the joint section. The electrical resistivity of the virtual material was decided based on the normalized joint resistance of the stacked BSCCO 2223 conductor. The current distribution analysis was performed using ANSYS ver. 8.1 and the joint resistance was evaluated by joule loss of the model. The thickness of the copper layer is chosen as a parameter in the analysis. The calculated result showed that the joint resistance decreases with an increase in the thickness of the copper layer from 0 μm to 200 μm, and the joint resistance can be reduced to 20 % of the joint resistance without the copper layer. Based on the above discussion, we plan to carry out experiments with the lager conductor.

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Measurement of joint resistance of large-current YBCO conductors

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Feasibility studies on applying high-temperature superconductors (HTS) to the heliotron-type fusion energy reactor FFHR are being carried out [1]. Because the HTS conductor has high cryogenic stability at elevated temperature operations (e.g. 20 K), we consider that refrigeration power of FFHR can accept Joule heating dissipation in segment-fabricated HTS coils of FFHR.
To examine this scenario, we are carrying out R&D tests with 10 kA-class conductors using YBCO tapes. In order to connect two conductors, we used two methods: soldered lap joint and mechanical lap joint. The soldered lap joint has a merit of assuring low joint resistance. The mechanical lap joint of conductors may further ease the construction process of FFHR.
In both methods, we first made two separate conductors having seven layers and two lines of stacked YBCO tapes (4.3 mm width and 0.19 mm thickness) imbedded in a copper jacket (7 mm × 10 mm) using solder. We applied a current to the conductor and measured the joint resistance. From the measured result, the joint resistance is evaluated. For the soldered lap joint, it is determined to be 19.1 nΩ and the contact resistivity is 103.5 nΩ-cm^2. Because the joint resistance of the single YBCO tapes that we measured beforehand was 42.5 nΩ-cm^2, the present result with a 14-tape conductor is about 2.5 times larger than the expected value.
For the mechanical lap joint, we presently applied clamps to make a connection. The joint resistance was measured by changing the compressive load. We observed that the joint resistance decreased as the load increased, however, the minimum value so far was 406 nΩ (4.9 μΩ-cm^2), which is about 45 times larger than that measured for a soldered lap joint. From these results, we presently consider that the soldered lap joint is more feasible than the mechanical lap joint.

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P2-32

Discussion of structural design issues on a remountable high-temperature superconducting magnet

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A remountable (demountable) high-temperature superconducting (HTS) magnet was proposed for a helical reactor having huge and complex superconducting magnet in the long term [1] and for a component-testing machine of tokamak which require good access in the near and intermediate term [2]. The proposed magnet consists of some segments that can be assembled and disassembled repeatedly with mechanical joints (electrical joints). The superconducting material of the magnet is HTS having high critical current and high heat capacity at relatively high operating temperature (>20 K), which could afford mechanical joints. Refrigeration power for the remountable HTS magnet could be lower than conventional low-temperature superconducting magnet due to the high operating temperature though there exists resistive loss at joint section. The design can contribute to simplify the fabrication of a complex and huge magnet like helical coils and to repair or replace damaged components such as by neutron irradiation. This concept is also included in the HTS coil option for the LHD-type fusion DEMO reactor FFHR designed by National Institute for Fusion Science [3, 4].
In previous studies [5, 6], R&D of a mechanical joint of a HTS conductor have been performed successfully. Conceptual design of a remountable HTS magnet using structural analysis is also another important issue. In an actual magnet, electromagnetic forces and thermal strain can influence joint condition. In this study, we performed structural analysis using a model of a remountable circular coil as a first step of the design. In the analysis, we evaluated contact condition at the joint section with a change in radial electromagnetic force for several shapes of joint section. Based on the results, we discuss about structural component of the remountable magnet. Then design issues are considered based on the discussion.

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P2-33

Beam-ion losses due to magnetic field ripple under various plasma parameter ranges on the Large Helical Device

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Magnetically confined fusion devices such as tokamaks and helical devices/stellarators have been significantly developed, approaching the conditions of nuclear fusion reactor. Compared with tokamak plasmas, helical/stellarator plasmas have intrinsically a great advantage in its easy controllability. One of the important issues to achieve the high plasma parameter is how well fast ions would be confined. Obviously, good confinement of D-T produced alphas is essential to sustain the fusion plasma. In addition, auxiliary heating produced by fast ions such as a neutral beam injection (NBI) is also needed to ignite the fusion plasmas. If they are not confined well, then their energy is lost from the plasma, meaning that the efficiency of heating is lowered. Also, concentrated fast-ion losses might cause localized damage of plasma facing components. For these reasons, it is indispensable to understand the confinement/loss characteristics of fast ions.
Beam-ion losses have been measured with scintillator-based lost-fast ion probe installed on the outboard side of the Large Helical Device. The loss flux reaching the probe with information of their pitch angle (χ and the energy (E) distributions were investigated in various plasma parameters. In relatively low-field experiments (<1 T), not only the losses having a transition orbit (the range of E and χ ?are about up to 180 keV and 55 degrees), but also the losses having a passing orbit (the region of E and χ ??are about up to 180 keV and 30 to 40 degrees) are regularly observed because of the large deviation of the orbit from the flux surfaces. Those loss fluxes are significantly suppressed as the toroidal field magnetic strength (B_t) is increased. Losses having the transition orbit are still remained on relatively-high-field experiments (B_t/R_{ax}=1.375 T (CCW)/3.85 m) with no strong instability. The loss flux is proportionate to fast-ion density ~ $P_{NB}\times\tau_{se}$, where P_{NB}, τ_{se}, R_{ax}indicate the deposition power of NBIs, a slowing-down time of beam ions by electrons, and a magnetic axis position in vacuum, respectively. It is clearly shown that collisional loss coming to the probe depends on the fast-ion density.

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P2-34

Evaluation of the non-axisymmetry of the central cell plasma by using a segmented limiter on GAMMA 10

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In the GAMMA10 tandem mirror, electromagnetic waves of two different frequencies in the ion-cyclotron range of frequency (ICRF) are used for plasma production and ion heating. For the magneto-hydrodynamic (MHD) stabilization of the whole plasma, ICRF waves are also used at the anchor cells, which are composed of minimum-B field and located at both ends of the central cell. Waves in the electron-cyclotron resonance (ECR) frequency are used for the electron heating and the formation of the plasma potential along the magnetic field line. A segmented limiter which is divided into 8 equal parts in the azimuthal direction is installed near the midplane of the central cell. The diameter of the limiter is 36 cm. The segmented elements are electrically-insulated from the ground. Thus, the floating potentials and its azimuthal distribution in the peripheral region can be measured. By connecting all segmented elements electrically, we can change the boundary conditions in the peripheral region. Two types of low-frequency fluctuations which degrade the plasma parameters are detected in GAMMA10. These are drift-type and flute-type fluctuations. It is clearly observed that the amplitude of the drift-type fluctuation with low-azimuthal mode numbers becomes small with the connection.

The azimuthal distribution of the floating potential is a good indication of non-axisymmetry of the plasmas. Non-axisymmetry of the plasmas in a simple mirror configuration will cause diffusion in the radial direction and degrade plasma parameters. In this paper, non-axisymmetry of the plasmas in the central cell of GAMMA 10 is evaluated with the azimuthal distribution of the floating potentials on each element of the segmented limiter. When ICRF and ECR are used for the experiments, the azimuthal distribution of the floating potentials depends on their injecting powers. We measure the influence of ICRF and ECR on non-axisymmetry of the plasma. The mode numbers of the floating potential in the azimuthal direction are analyzed. Drive mechanisms of non-axisymmetry of the GAMMA 10 plasmas are investigated.

P2-35

Magnetic Diagnostics for Magnetic Islands

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Generating and keeping clear flux surfaces are an aim of magnetic confinement researches, because magnetic islands and stochasticity of magnetic field leads the degradation of the confinement connecting and overlapping filed lines. In tokamaks, the degradation of the confinement due to generating islands like the locked mode and neoclassical tearing mode (NTM) were observed and studied. In addition, same degradation is also observed in helical system. Thus, understanding and controlling of island dynamics are urgent and critical issues to aim the fusion reactor.

A method to identify magnetic islands is the magnetic diagnostics. Since the magnetic diagnostics detects the change of magnetic flux directly but the diagnostics must be installed appropriately to detect perturbed field of islands. In this study, we design the magnetic diagnostics to study island dynamics in LHD plasmas. To design the magnetic diagnostics, we apply 3D MHD modeling codes. A special attention is the identification of magnetic islands with m=3 and 4 modes.

P2-36

Time-dependence of Balmer series emission intensity during pulse plasma flow in divertor simulator TPD-SheetIV

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The research on dynamic behavior of plasma during pulse plasma flow is a topic in space plasma, plasma processes and nuclear fusion. In particular, transient behavior from recombination to ionization processes during pulse plasma flow has become important for divertor region of nuclear fusion device. The transient behavior during pulse plasma flow has been studied by observing the short double minimum (negative) spike in D_α emission intensity from the plasma[1]. This response in D_α emission is explained by electron temperature increase associated with bursts of heat and particles along the magnetic field on pulse plasma flow. In order to understanding of effects on transient behavior during pulse plasma flow, however, it is necessary to take into account of high energy electrons in plasma flow.
We have carried out the experimental observation and modeling of time-dependence of Balmer series emission intensity in hydrogen recombination plasma in a linear plasma device, TPD-SheetIV[2]. Pulse plasma flow generated by controlling the TPD-type plasma source were introduced into hydrogen recombination plasma in the experimental region. The pulse plasma flow was generated by the switching circuit controlled the electric potential of the next floating electrode of the anode in plasma source. The duration of the pulse plasma was 0.3ms and the frequency of cycle of the plasma was 50Hz. The time-dependence of electron density n_e, electron temperature T_e, electron energy distribution function $f_e(E)$ were measured using Langmuir probe. The ionization and recombination events are discussed by Collisional-Radiative model, taking into account of high energy electrons.
After the pulse plasma flow in the recombination plasma, the high energy electron was generated in the electron energy distribution function, immediately. Also, electron temperature T_e and electronic density n_e have increased. At the same time, double minimum (negative) spike were observed in H_α emission intensity from the plasma.

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P2-37

Calculation of geometry matrices for IRVBs for application to 3D tomography of radiative phenomena in LHD

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InfraRed imaging Video Bolometers (IRVBs) can provide hundreds of channels of bolometric data forming an image of the plasma radiation [1]. By calculating the geometry matrix (or response matrix) of the detector field of view (FoV) with respect to a predefined three dimensional (3D) plasma grid of plasma voxels these geometry matrices can be used in two ways. The first of these is to compare the 3D carbon radiation results of an impurity transport code such as EMC3-EIRENE [2,3], which are translated via the geometry matrix into the field of view of the IRVB, with the experimental results of the IRVB. This type of analysis is called a synthetic instrument (or synthetic diagnostic). The second technique is to divide the LHD plasma into half field periods (18° in toroidal angle) and assume that the plasma reproduces itself every half field period. This assumption is called periodic helical symmetry. Then by combining the FOV of multiple imaging bolometers with different views of the plasma (top, bottom, tangential, semi-tangential, radial, etc.), one large geometry matrix can be derived relating thousands of IRVB channels to thousands of plasma voxels. Then by applying tomographic techniques the geometry matrix can be inverted and applied to the measured data to yield the 3D radiation distribution, which can be directly compared with the 3D carbon radiation distribution from the EMC3-EIRENE code. The advantage of the synthetic diagnostic technique is that it does not require the assumption of periodic helical symmetry and can be applied to nonsymmetric phenomena such as those involving the m/n =1/1 magnetic island.

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Design considerations for an Infrared Imaging Video Bolometer for observation of 3D radiation structures of detached LHD plasmas

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Plasma detachment is an important operational regime for future diverted magnetic fusion devices where the power load on the plasma facing components (PFCs) is expected to be tremendously high. The detachment of the plasma will help by temporarily reducing the peak power loads on the PFCs and thus will facilitate in maintaining the power loads within the engineering design limits. Plasma detachment in LHD is aided by the external addition of an m/n=1/1 magnetic perturbation [1]. During detachment the radiation structures are found to be localized near the X- points of the magnetic islands [1,2]. Infrared Imaging Video Bolometers (IRVB) are successfully being used to study the 2D impurity radiation profiles from the LHD plasma [2,3]. IRVBs can serve as a promising diagnostic for studying the radiation structures of detached plasmas in LHD and hence a comparison can be established with theoretical models. A new IRVB system is being designed for the bottom port 6.5-L for better access to the magnetic X – points and thus studies of the 3D radiation structures. This presentation gives a design overview of this new IRVB system for the bottom view. The design includes magnetic and neutron shielding for the IR camera, IR optical design, spatial resolution and field of view of the IRVB, sensitivity and signal to noise estimates. The design configurations and parameters will be reviewed and discussed.

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Design and Development of Infrared Imaging Video Bolometer for ADITYA Tokamak

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Infrared Imaging Video Bolometer (IRVB) is one of the modern plasma diagnostic which provides the measurement of the temporally resolved two & three dimensional (2-D / 3-D) power profile radiated from the plasma devices [1-4]. It utilizes free standing ultra thin large area metal foil, which absorbs radiated power from high temperature magnetically confined plasma through pinhole camera geometry in wide spectral range (SXR to UV). This absorbed power alters the temperature profile on the foil area which is optically (by infrared radiation) imaged and measured by the Infrared Camera seating out side the vacuum vessel through Infrared Transmitting vacuum view port. Using suitable analysis of 2-D temperature profile, total radiated power profile from plasma can be determined. Since the IRVB is optically transmits the signals (Infrared Radiation) it is immune to EM noise and also can work under reactor relevant conditions, hence it is more suitable for the steady state high temperature plasma devices and future fusion devices [3-5].

The techniques is being first time utilize for the medium size tokamak Aditya (R=75cm, a=25cm, Ip=75 kA, Te(0)~350 eV, BT= 0.7T), where the plasma shot duration is ~100ms and total radiated power 40kW to 80kW typically during current flattop[6]. IRVB system was design, developed and installed on the Aditya tokamak having 8 mm diameter pinhole, 2.5 micron thick Platinum foil (Size: 64mmx64mm), MWIR-camera (320x240 pixels, 200Hz frame rate, NETD~20mK), 8x8 Bolometer pixel array, 10 ms optimal time resolution. This paper describes the design, calibration experiments and initial total radiated power measurement results from the Aditya Tokamak.

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Calibration of IR imaging bolometer foil by laser irradiation

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An IR imaging video bolometer (IRVB) [1] is a measurement instrument for plasma radiation, and useful for the measurement of both radiation intensity and spatial distribution. A bolometer foil with a carbon coating is employed in the IRVB system. The temperature distribution on the foil formed by the incident radiation is measured, and then the radiation profile is determinedfrom the temperature distribution by using the calibration data for the foil.
Thus, the thermal response of the foil to the incident power has to be examined and the radiation power has to be calibrated with the temperature rise. The temperature distribution depends on the properties of the foil such as thickness and emissivity. The foil properties are often non-uniform across the foil. This non-uniformity introduces errorin the radiation measurement. The carbon coating also influencesthe temperature distribution.
In the present study, the bolometer foil was irradiated with a He-Ne laser while changing the irradiation position by moving the foil in two dimensions [2]. The spatial distribution of the temperature was measured by using an IR camera. The temperature distribution measured was compared with that calculated by a Finite Element Method (FEM) analysis. In the FEM analysis, the thickness [3] and the emissivity at each position were repeatedly changed until the FEM profile converged to the measured profile. The FEM profile converged to the temperature profilemeasuredafter only two iterations. In the present study, it is found that the FEM profile agreed wellwith the temperature profile measured, so that the calibration between the radiation power and the temperature profile can be suitably conducted by using the FEM analysis.

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P2-41

Development of an Array System of Ultra-Soft X-ray Detectors with Large Sensitive Area on the Large Helical Device

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A new 17-channel soft X-ray diagnostic system was developed for study of magnetohydrodynamics (MHD) fluctuations and installed on the Large Helical Device (LHD). The Absolute X-ray Ultraviolet Photodiodes (AXUV diode made by IRD) with the large sensitivity area 10 mm &x 10mm was adopted as the detectors. The sightlines were designed to cover the whole plasma with 3.8 cm space separation and the expected radial resolution was 10cm. The toroidal elongated pin hole (12mm &x 36mm) was designed to increase the signal to noise ratio and a Be foil with 15 μm thickness was used to shut the visible light. The detector array was placed inside the LHD vacuum vessel, being shielded by an aluminum box. Electromagnetic noises were effectively reduced by this arrangement.

In the experimental campaign of LHD, this fiscal year 2011, various kinds of MHD fluctuations excited in core and edge plasma regions have clearly been detected by this newly installed diagnostic system. The internal disruption, which would be induced by resistive interchange mode with m=2/n=1 (m, n: poloidal and toroidal mode numbers, respectively), was observed. The sawtooth–like crash occurs around the q=2 rational surface.

Tangential and vertical Soft-X ray imaging for three-dimensional structural studies in the RFP

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Three-dimensional (3-D) effects on MHD phenomena in axisymmetric systems such as reversed field pinches (RFPs) have attracted much attention. In the RFP, for example, recent progress have shown the importance of helically deformed RFP configuration where single helical magnetic axis state is self-organized. Since the toroidal mode number n of the dominant m=1 mode is much larger than 1, development of imaging diagnostics for 3-D structure is required for the detailed study on dynamics of the self-organizing process. In a low-A RFP machine RELAX (R = 0.51 m/a = 0.25 m (A = 2)), a quasi-periodic transition to quasi-single helicity (QSH) state has been observed. We have applied a soft-X ray (SXR) pin-hole camera and a ICCD camera to take tangential SXR snapshots of the RFP during the QSH state, identifying characteristic helical SXR structures which suggest hot or dense helical core.[1] Moreover, we have constructed a fast successive SXR imaging system where SXR camera and high-speed camera. As a preliminary experiment, we have taken tangential SXR pin-hole pictures with time resolution of 10 micro sec, to identify time evolution of helix structure in RELAX plasmas.[2,3] As a next step, we have been developing a SXR imaging diagnostic system, which uses multiple SXR cameras together with high-speed cameras to take time evolution of SXR images from tangential and vertical directions simultaneously for the study of dynamic structures of 3-D SXR emissivity, through which we expect to discuss 3-D dynamics of MHD instabilities associated with the QSH state. Initial results will be reported, together with some discussion on 3-D reconstruction.

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Identification of spurious modes of high power 77 GHz gyrotron for collective Thomson scattering in LHD

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Collective Thomson scattering (CTS) diagnostic in the Large Helical Device (LHD) has been developed as a technique for measuring bulk and fast ion velocity distribution functions in high density and high temperature plasmas. For the bulk and tail ion temperature measurements in the LHD, the CTS diagnostic has been performed using existing electron cyclotron resonance heating (ECRH) system with high power mega-watt gyrotrons with the frequency 77 GHz [1, 2]. In order to improve the signal to noise ratio for CTS measurement, the power of the CTS probe beam is modulated to subtract the background electron cyclotron emission (ECE) and system noise from received signals. During the turing on/off phase of the modulation, the modulation of gyrotron power excites spurious mode radiations. They exist in the measured CTS spectrum. They even with small power result in the saturation of the IF amplifiers if the frequency is in the sensitive band but outside of the notched one.
As the first step, we measured the radiation frequency of a gyrotron with the nominal frequency of 77 GHz using a heterodyne receiver and a fast sampling oscilloscope. We found that spurious mode frequency was about 74.7 GHz when the anode voltage of the gyrotron started up. The spurious mode is considered to be the $TE_{17,6}$ from the resonator structure of the gyrotron. The starting current calculation code [3] shows that the $TE_{17,6}$ mode is relatively easy to start oscillation as compared with the other modes under the measured condition. The method to suppress or reduce the radiation of the spurious mode will be discussed.

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Coaxial multiple laser beam combiner for the LHD Thomson scattering system

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Thomson scattering (TS) system is a basic diagnostic of LHD, because the electron temperature and density profiles can be simultaneously measured. The LHD TS system has been designed to handle multiple Nd:YAG lasers for improvement of time resolution or accuracy of measurement by packing mirror method [1]. This method works well in the focal point of near center of plasma but outside this region the laser beams gradually separate. As the result sampled scattering light from separated laser beam line have the different information of plasma volume.
A coaxial multiple laser beam combiner for LHD TS system have been developed to solve this problem. Multiple beams from different aperture of laser gain mediums are combined to a coaxial beam line in time series by a combination of the cubic polarizer and an electro optical (EO) device [2]. One cubic polarizer and a half wave plate are needed for two beam combing. A horizontal polarized laser beam passes through the cubic polarizer. On the other hand laser beam polarization from the second laser is rotated to vertical by half wave plate and reflected by cubic polarizer. After the refraction and the transmission of cubic polarizer, two different laser beams are aligned at coaxial beam line. For additional multiple lasers, EO device is added after the first cubic polarizer for preparing the polarization for use of the next cubic polarizer.
We have installed this beam combiner for LHD Thomson scattering system from 15th experiment campaign and confirmed the improvement of data quality. In the presentation, we will discuss the detailed design and performance of polarization based laser combining.

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P2-45

The Thomson Scattering Diagnostic System on the KSTAR tokamak

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The Thomson scattering system of the KSTAR tokamak is described. The KSTAR Thomson scattering system has been developed and tested in the KSTAR 3rd campaign, 2010. Through this commissioning, we could measure the Thomson scattering signal successfully in the KSTAR 4th campaign, 2011. Last year we found that the main reflection mirror had an errors near the Thomson signal wavelength region thus before the KSTAR 4th campaign we changed main mirrors in the core, edge collection optic module. In this campaign, we installed the 1000μm diameter single core silica/silica type optical fibers for transfer the Thomson signals and Charge-to -Digital conversion type VME modules for measuring core and edge region plasma parameters. To suppress the noise from the signal, we installed an uninterruptible power supply (UPS) system. From this system we measure the electron temperature and density profiles on the KSTAR plasma. This paper reports the experimental results obtained from KSTAR Thomson scattering diagnostics system and discuss the future works.

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The comparison of electron temperature and its density calculation methods in Thomson scattering diagnostics

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The Korea Superconducting Tokamak Advanced Research (KSTAR) tokamak [1] achieved the first plasma on the last day of June 2008 at the National Fusion Research Institute (NFRI), Daejeon, Korea. Thomson scattering has become an important diagnostic for measuring electron temperature and its density profiles in most tokamaks [2] [3] [4].

In Thomson scattering diagnostics, electron temperature and its density are indirectly derived through calculations of scattered light signals.[5] The signal noise is influenced by the laser power fluctuation, and the final signals get out after couples of signal conversions such as a conversion of optical signal to electrical signal or internal electrical signal conversions. The noise can cause great effects on the calculations. In the calculations, χ^2 method is mostly used. Here the robustness of χ^2 method to the noise is examined and an alternative method is considered.

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Demonstration of an in-situ relative calibration method for a Thomson scattering diagnostics on TST-2

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Development of in-situ calibration methods for Thomson scattering diagnostics is an essential task. However, intense radiation (i.e., neutron and γ-ray) from high performance plasmas changes spectral transmissivities of optical components and degrades accuracies of electron temperature (T_e). In order to determine T_e with unknown or degraded sensitivities (transmissivities), a method using double pass scattering system, where two scattering signals with different scattering angles (θ) can be measured, has been proposed [1]. Since scattered spectra depend on θ and T_e, the ratio of the scattering signals at the same wavelength becomes a function of T_e. Here, the efficiencies of each scattering in the double pass is assumed to be the same. Therefore, the ratio is independent from the sensitivities. Once T_e is decided, the relative sensitivities among different wavelength channels can also be calculated, because the spectral waveforms are already fixed by the known T_e and θ. To demonstrate the method, a Thomson scattering diagnostic on TST-2 [2] utilizing a double-pass scattering system was used.

As the first step of the demonstration, T_e measured from the ratio and T_e from the standard method (i.e., wavelength spectrum) were compared. In TST-2, however, the effective scattering length (L_s) and laser energy (E) were different between the two scattering in the double pass. Thus, the measured ratio depends not only on T_e , but also on the decay of $L_s \times E$. The decay of $L_s \times E$ at the second pass can be estimated by assuming that electron density does not change between the two passes. With this assumption, T_e (100 – 300 eV) measured from the ratio agrees with T_e from the standard method with the accuracies of about 2 – 15 %. When we have more than two wavelength channels, we obtain more than two ratios. In this case, we can calculate two (fitting) parameters T_e and the decay of $L_s \times E$. The agreement between T_e s was not so clear due to the lack of the number of spectral channels with high SN ratio. The error evaluations in relative sensitivities determined from T_e measured by the ratio will be also discussed.

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Off-axis temperature anisotropy measurement by a double-pass Thomson scattering diagnostic system on TST-2

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Multi-pass Thomson scattering scheme is attractive from the viewpoints of low density plasma measurements and electron temperature anisotropy measurements. Especially, direct measurement scheme of temperature anisotropy has not been established, while temperature anisotropy has important effects on plasma equilibrium. As a pilot experiment for the temperature anisotropy measurement, a double-pass Thomson scattering system was constructed on the Tokyo Spherical Tokamak 2 (TST-2) device. In this system, a laser pulse makes a round trip through the plasma. Combining the double-pass configuration with a fast detection system, the backward scattering and the forward scattering can be measured almost simultaneously. In the double-pass configuration, electron temperature perpendicular and parallel to the toroidal field can be measured from the backward scattering and the forward scattering, respectively. Using this system, electron temperature and electron density have been measured around the center of plasma, in which region no temperature anisotropy was detected within error bars.
A multi-point Thomson scattering system has been also prepared on TST-2. In this system, Thomson scattering measurement can be performed at six points in the plasma simultaneously. When the plasma is in a banana region, $T_{e\perp}$ can be higher than $T_{e//}$ in the high field side, and $T_{e//}$ can be higher than $T_{e\perp}$ in the low field side. Therefore, temperature anisotropy profile is expected in this case. Test experiments have been performed to search for the off-axis temperature anisotropy by the double-pass and the multi-point Thomson scattering system. Preliminary results of the measurements will be presented.

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The method to measure of poloidal plasma rotation by Heavy Ion Beam Probe

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The paper proposes a method to measure poloidal rotation velocity of toroidal plasma using Heavy Ion Beam Probe (HIBP) with a multi-slit energy analyser. The method is based on calculation of phase shift between broadband density turbulence measured simultaneously in two poloidally separated sample volumes. Oscillatory component of HIBP beam current is used as a density turbulence characteristic. HIBP is capable to provide the temporal evolution of the v_{pol} in a fixed radial position and also the evolution of the v_{pol} profile by periodic radial scan. Method was verified in real plasma experiment in ECRH discharges on the TJ-II stellarator. Result shows that in low density discharges ($n_e \approx 0.3\text{-}0.5\times10^{19}m^{-3}$) absolute values of local v_{pol} is about 6-4 km/s. The plasma rotation velocity is oriented in the ion diamagnetic drift direction. When HIBP operates for radial scans, it is conventionally measuring the plasma potential profile, and so provide the radial electric field E_r and velocity of ExB drift (v_{ExB}) at the same time as plasma rotation. Experimental data shows that in low density ECRH plasma the rotation velocity coincides with ExB velocity within achieved experimental accuracy. When the density is increasing, both v_{ExB} and v_{pol} tends to decrease and then change the sign at $n_e \approx 0.6\text{-}0.7\times10^{19}m^{-3}$.With this new proposed technique HIBP becomes the new effective tool to study plasma rotation and turbulence characteristics in toroidal plasmas. [*]

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Development of a multichannel far-infrared interferometer for the Experimental Advanced Superconducting Tokamak

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A new vertical five-channel far-infrared (FIR) deuterium cyanide (DCN) laser interferometer has been developed successfully and will be operational in the next Experimental Advanced Superconducting Tokamak (EAST) campaign. The system is based on the three-channel FIR HCN laser interferometer. The FIR beams with a wavelength of 195 μm, at which the high-power source availability makes this wavelength ideal for EAST plasma, is produced by DCN laser. The laser is designed to have a full remote control capability using the Programmable Logic Controller (PLC). It can improve the stabilization of laser power in long-term operation automatically. The technical details of the new FIR system will be described.

The first high-confinement mode (H-mode) with type-III edge localized modes has been obtained during last EAST campaign. The electron density limit and effects of triangularity on confinement of H-modes from the old FIR laser interferometer have been studied. The ‘H-mode’ limit is found at 0.65 n_e/n_G (the ordering parameter to describe the confinement with respect to the ITER scaling, n_G is the Greenwald density). It is also observed that the higher triangularity discharges have better confinement for a given density, and meanwhile, the confinement degrades with density for a given triangularity.

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On Plasma Diagnostics Required for Heliotron-type DEMO Reactor

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Conceptual design studies of heliotron-type DEMO reactor FFHR-d1 have been initiated lately in National Institute for Fusion Science [1]. In existing fusion devices, many superior plasma diagnostics are routinely operated for extensive physics studies. Note that necessary and/or feasible diagnostics depend on the stage of experiment. Diagnostic instruments and/or methods have been greatly developed as plasma parameter rises according to practicability and/or requirement. Because existing critical physics issues will be solved until the start of DEMO operation, measurements in DEMO are thought necessary only for burning control and performance optimization. Reliable long-term operation of diagnostics in DEMO must be challenging because instruments have to function in hostile thermal and radiation environments. The implementation of diagnostics is also challenging since the diagnostic port and space that can be utilized must be fairly limited since the reactor plasma has to be surrounded by a massive blanket. This line of thinking will reach the minimum diagnostic set and the least implementation to the DEMO machine. In this paper, required plasma parameters that must be diagnosed to support stable, steady-state operation of heliotron-type DEMO are going to be discussed. In addition, realistic issues and possible solution of diagnostic set will be also discussed.

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P2-52

Conceptual design of electron density measurement system for DEMO-relevant helical plasmas

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Electron density measurement is still indispensable to control fueling on a DEMO reactor. Since operation of the DEMO reactor will be steady-state, the density measurement should be highly reliable and accurate. On the other hand, interferometers used for the continuous density measurement on present fusion devices have to suppress the measurement error due to mechanical vibrations and sometimes suffer from "fringe jump errors". Two-color measurement, which uses two different wavelengths, can compensate the phase component due to the mechanical vibrations and a short-wavelength laser can reduce risk of the fringe jump errors. However, introduction of different schemes in principle is necessary to establish highly-reliable density measurement enough for steady-state operation.
A dispersion interferometer [1] is less sensitive to the mechanical vibrations. A mixed beam of the fundamental and the second harmonic components is used as a probe beam. The second harmonics is generated with a nonlinear crystal. After passing through a plasma, another second harmonic component is generated from the fundamental again. The fundamental is cut with a filter and then an interference signal between two second harmonic components is detected. While the phase shifts due to mechanical vibrations are the same between the second harmonic components, those due to the plasma are different because of the dispersion. Hence the phase shifts due to the vibrations are canceled optically and those due to the plasma only remain. It also becomes free from fringe jump errors principally by selecting a short enough wavelength, whose phase shift due to a plasma is smaller than 1 fringe. Hence it is suitable for the density measurement on steady-state fusion reactors. Dispersion interferometers with a 10.6 μm laser source [2,3] are operated at present. A wavelength around 1 μm is one of candidates for DEMO-relevant plasmas from viewpoints of a variety of optical components, efficiency of the frequency doubling and the phase shift due to plasmas. High power and stable continuous-wave lasers such as a YAG laser with a wavelength of 1.064 μm are commercially available. Periodically poled crystals such as PPMgSLT and PPKTP will generate enough power of the second harmonics enough for detection. In case that the phase shift is difficult to be smaller than 1 fringe, compensation with a polarimeter which measures the Faraday rotation will be effective. The appropriate wavelength will be several μm to avoid coupling with the Cotton-Mouton effect which causes difficulty of conversion to the electron density from the measured rotation angle. Although the polarimeter is less accurate than the interferometer because of the small rotation angle and weight by the magnetic field, the fringe shift of the dispersion interferometer can be corrected easily since the fringe shift is only 3-4 fringes at most. The conceptual design of the density measurement system will be presented, taking FFHR-d1 as an example. Some challenges particular to DEMO reactors (e.g., degradation of first mirrors, availability of optical fibers) are also discussed.

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Developments of Pulsed Terahertz Wave Diagnostics for Fusion Plasma

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In the future burning plasma experiment, the electron density will be quite high and up to the order of 10^{22} m^{-3}. In such high density plasmas, the conventional microwave diagnostic technique will be still expected to measure the density profile, its fluctuation, etc. However, the demanding and utilizing frequency is getting into terahertz regime (0.1 THz-10 THz). Recently, the terahertz wave technologies have been much interest in world wide. There are several techniques for own applications. Among them, THz Time-domain Spectroscopy (THz-TDS) on which many of the researches have focused has several features, because it has a capability of getting broadband spectral information covering from sub 100 GHz to a few THz. At present, however, the high power source and high resolution detector in terahertz regime have not been enough to apply plasma measurements.

Currently, we heve been constructing a typical THz-TDS system. The pumping femto-second fiber laser light with 780 nm is focused to the photoconductive antenna. THz optical pulse of which the width is less than 1 pico-second is generated by exciting a bow-tie type of photoconductive antenna. THz beam is collected and focused on a detector which is activated only for a sub-pico-second period by the optical pulse branched from the excitation one. By scanning the time-difference between the excitation and detection pulses, the time-domain trace is recorded as in sampling oscilloscopes. The amplitude and phase spectra were obtained by the Fourier transformation of the time domain trace.

At the conference, we will present the system performance and discuss the future plan of developments.

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Assessment of multi pulse laser damage threshold of metallic mirrors for laser diagnostics in fusion devices

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In ITER, laser transmission mirrors will be used for Thomson Scattering Diagnostics. For the laser transmission mirrors, which transmit laser pulses to plasma, high reflectivity at 1064 nm, high multi pulse LIDT(Laser induced damage threshold) and high durability for sputtering by plasma are needed. Furthermore, for laser transmission mirrors it is required to maintain optical properties even after 10^7 laser pulses irradiation. If the surface of mirror is damaged by laser irradiation, reflectivity of the mirror decreases seriously. However, most of the experimantal data about multi pulse LIDT of copper mirror is about 10^2-10^3 pulses and few investigation has been done about silver mirror. Therefore, in this study, the effects of the multi pulse LIDT on OFHC(Oxygen - free high thermal conductivity)-copper(KUGLER Co.) are experimentally investigated up to 7.3×10^4 pulses. The copper mirror was irradiated with pulsed Nd:YAG laser (1064 nm, 5 ns). Single shot LIDT was 4.8 J/cm^2 and multi pulse LIDT was 1.5 J/cm^2 for 7.3×10^4 pulses. Experimental results perdicted that the expected multi pulse LIDT for 10^7 pulse is ~0.5 J/cm^2. This value was about twice as high as the predicted value in ref [1].The surface roughness of the KUGLER mirror was better than 3nm and the good surface property could have raised the LIDT.

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Confinement Analysis of Spherical Tokamak-Stellarator Hybrid Configurations with Simple Shaped Coils

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One of the critical issues for the achievement of attractive fusion reactors is to reduce reactor construction cost and to get good economic efficiency. To solve this problem, it is desired that innovative magnetic configurations with compactness and high capability of confining fusion plasmas are researched. In our laboratory a new concept of magnetic configuration called TOKASTAR[1] was proposed. This configuration is a tokamak-stellarator hybrid with a few spherical helical coils. In the previous study of TOKASTAR one type of coils shape was analyzed. Analyses of different types of coils are needed to get deep knowledge of TOKASTAR configurations and improved capability of plasma confinement.
In this study, we proposed 5 new types of TOKASTAR configurations in addition to previous N=2 TOKASTAR; N=1, N=4, N=2 with central helix, N=2 with crescent cross-sectional configuration, and N=2 cylindrical coil system. Here N is the toroidal mode number of spherical helical coils. We carried out comparative computational analysis among these configurations to improve TOKASTAR's capability of confinement.
We studied the magnetic flux surface, the single particle orbit, the equilibrium, and the Mercier mode stability of TOKASTAR configurations. The HSD code was used for the analysis of magnetic flux surface and single particle orbit, and the fixed boundary VMEC code was utilized for the analysis of equilibrium and Mercier mode stability.
In the magnetic flux surface analysis, the aspect ratio of magnetic flux surface, rotational transform and magnetic well were analyzed for 6 types of TOKASTAR configuration. These configurations produce low values of aspect ratio from 1.1 to 3.1, low values of last-flux-surface-averaged rotational transform from 0.01 to 0.04, deep well depth from 30% to 50%. Especially, N=2 center helical configuration produces relatively high value of rotational transform about 0.04 in comparison with other TOKASTAR and deep well of 50%. In this configuration Mercier mode can be stabilized due to the effect of well. In the analysis of equilibrium, equilibrium beta limit is analyzed. The results indicated that high equilibrium beta is achieved in the configurations with high ellipticity, triangularity.
Separately from computational analysis, we are going to make experiments of TOKASTAR. We also mention about present status of experiments in the conference.

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Experiments of tokamak discharge and simulation of magnetic field configuration in tokamak-helical hybrid device TOKASTAR-2

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TOKASTAR configuration [1] is one of compact tokamak-helical hybrid confinement systems. The small device named “TOKASTAR-2” (the major radius is ~0.1m and the magnetic filed strength is ~1kG) was designed and constructed to generate both tokamk and stellarator configurations independently. One of main purposes of TOKASTAR-2 is to evaluate the effect of helical field application on tokamak plasmas.

We produced pre-ionized plasma using toroidal field (TF) coil system and 2.45GHz RF wave injection, and induced plasma current through pre-ionized plasma using ohmic heating (OH) and vertical field (VF) coil systems. About 4V one-turn voltage was applied to the pre-ionized plasma. However, we could not obtain more than 0.1kA plasma current within the limitation of DC power supply for VF coil system and induced eddy current on the vacuum chamber. Photographs taken by fast camera showed that the plasma moved vertically upward when one-turn voltage was applied. It was considered this movement is one of factors preventing the increase in plasma current.

Then, we installed the conducting shell in TOKASTAR-2 and plasma current of shell-plasma was about 40A larger than that of shell-less plasma. Although plasma current increased, it was unclear that the conducting shell restrains vertical movement of plasma. Therefore, we plan to investigate that effect using the fast camera. Additionally we have been making magnetic probes to measure magnetic field configuration of tokamak plasma and its movement.

Other purpose of TOKASTAR-2 experiments is to generate current-less helical configuration. By magnetic field line tracing analysis, it was revealed the vacuum magnetic surface can not be formed because present TOKASTAR-2 has only outer helical field (HF) coil system. Installing additional helical field (AHF) coil system is able to generate the vacuum surface. For example, magnetic field tracing analysis clarified the formation of last closed surface in the case that coil currents of TF (I_{TF}), VF(I_{VF}), HF(I_{HF}) and AHF (I_{AHF}) are I_{TF}=6122AT, I_{VF}=17596AT, I_{HF}=6250AT and I_{AHF}=7187.5AT respectively.

In the poster, the detailed experimental results of tokamak plasma and the simulation analysis of helical configurations will be presented.

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Improved Data Acquisition Methods for Uninterruptible Signal Monitoring and Ultra Fast Plasma Diagnostics in LHD

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Since the beginning of LHD's steady state experiments, long pulse plasma diagnostic data have been acquired according to a simple concept that partitions a long pulse into a consecutive series of 10-second "subshots" [1-2]. It can deal with not only 100 MB/s high-speed real-time (RT) data acquisitions (DAQs), but also non-RT batch processing ones using conventional CAMAC digitizers. The newest digitizers applying the high speed PCI-Express technology, however, can output GB/s nonstop data stream from each DAQ frontend. They would produce too huge data for a "subshot" chunk if 10-second rule were just applied.

Uninterruptible monitoring for environments and device healthiness will be more required for steady state fusion plants. Device healthiness signals and environmental radiological dosage had been acquired by their embedded engineering computer systems, respectively, for the past ten years in LHD. They had done their term of service and have been renewed as a part of the standard DAQ framework of LHD, namely, LABCOM system.

In order to cope with both the uninterruptible monitoring and ultra-fast diagnostics, the subshot length fixed on 10 seconds has been modified to be able to have variable length. The former adopts dual subshot lengths of 10 minutes and 1 day. Longer subshots contain 1 Hz sparse data thinned from the original 1 kHz sampling ones. The latter uses shorter variable intervals less than 10 seconds, according to each use case. In this study, the design modification for variable subshot lengths has been implemented and then verified its practical effectiveness at LHD.

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Study of Performance Improvement of Real-Time Mapping of Thomson Scattering Data to Flux Coordinates in LHD

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More than 100 diagnostic devices are attached to the vacuum vessel of the LHD (Large Helical Device), and they are measuring the various aspects of the plasma physics. Because the shape of the LHD plasma is not symmetric, each diagnostics obtains the physical values in different cross section. For example, Thomson scattering measures the electron temperature profile in the horizontally elongated cross section and the laser interferometer measures the line integrated electron density profile in the vertically elongated cross section. In order to analyze the data obtained by different diagnostics, their measurement positions must be mapped to the unified coordinate system, or the flux coordinate system. Therefore, the authors have been building the database to map the physical coordinate to the flux coordinate. Using the database, a mapping system of the electron temperature profile to the flux coordinates, which is called TSMAP, has been developed.

The profiles calculated by TSMAP are fundamental data to analyze the plasma physics during the experiment. Therefore, they are required as soon as possible. However, the execution of TSMAP needs the computational power, and the performance of a typical personal computer is not high enough to catch up with the 3 minutes’ plasma discharge cycle. In order to increase the performance, the authors use multiple personal computers. Because the calculation of each plasma discharge is independent from each other, they can be executed at the same time. Therefore, the authors use multiple physical and virtual machines to execute the calculation of several plasma discharges simultaneously. Using this approach, the calculation manages to catch up with the experiment cycle.

THE INITIAL DESIGN OF PROCESS CONTROL SYSTEM BASED ON EPICS FOR HL-2A & HL-2M

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A new TOKAMAK named 'HL-2M' will be built in next four years at SWIP. The objectives of HL-2M is to elucidate important research issues such as the mechanism of plasma confinement and transport, the physics of high energy particles, MHD stability, the interaction between first wall and plasma and so on in physical and engineering related to ITER and future fusion reactor. Many conceptual designs for the operation of the new device may be validated on HL-2A which is current running at SWIP.

In addition to the real time plasma control system, the process control is also very important to the operation of the Tokamak. The process control system will aim at the normal control of plant systems, the information sharing, the schema of Tokamak operation, the alarm handling and the error tracing. For those purposes, we select EPICS as the basic platform for the process control system for its performance, reliability, expansibility and maintainability. Some user interface, data management and development tools are adopted such as control system studio(CSS), MS SQL Server, Matlab, LabVIEW, VC++,Java, etc. For the initial design, we will take several subsystems such as the operator interface (OPI), the data acquisition system, the ECRH system, the fueling system, the power system etc. into the platform and will handle about 2000 process variables (PV). The paper will focus on the initial conceptual design for the process control system for HL-2A/HL -2M.

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Operation planning of tritium recovery system based on investigation results of LHD exhaust system

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In order to realize the planned deuterium plasma experiments using the Large Helical Device (LHD), the National Institute for Fusion Science (NIFS) is planning to install a system for tritium recovery from exhaust gas.
In the LHD, following two types of tritium recovery systems are planned. One system will be used to recovery tritium generated during the plasma experiments and contained in vacuum pumping gas (processing capacity: 10Nm3/h) and the other to recovery tritium remained in the vacuum vessel and contained in purge gas of the vessel during inspection and maintenance of the vessel (processing capacity: 300Nm3/h).
It is necessary to understand the treatment conditions at the connecting point between the LHD vacuum pumping system and the tritium recovery system, such as, the hydrogen gas concentration, the impurity concentration, and the gas flow rate to decide fabrication specifications and the operation method of the two types of tritium recovery systems. Then, the measurement survey in a present vacuum pumping system was carried out.
Both the hydrogen concentration and the gas flow rate were tried to measure simultaneously by using gas sampling lines installed before and behind the exhaust blower. Two types of hydrogen meters (gaseous heat conduction type and flammable gas combustion type) were set up in the sampling line at the blower exit part and the hydrogen concentration was measured. On the other hand, the pure He gas is added to the sampling line at the blower entrance part in the constant flow rate. The increment in concentration by the addition of the He gas was measured from the difference of the measurement values of the flammable gas combustion type hydrogen meter (only hydrogen can be detected) and the gaseous thermal conducting hydrogen meter (both hydrogen and helium can be detected). The flowing rate of the exhaust was evaluated from the dilution rate of the pure He gas.
The reconstruction plan of the LHD exhaust system and the best operation plan of the two types of tritium recovery systems for the deuterium plasma experiments will be discussed based on the investigation result of the present exhaust system.

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Electromagnetic Fields Measurement and Safety Consideration in Magnetic Confinement Fusion Test Facilities

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Occupational electromagnetic fields were measured around the Large Helical Device (LHD), which is a large magnetic confinement nuclear fusion test facility located in Japan. The LHD equipped with a superconducting magnet coils system and high-power plasma heating systems like Neutral Beam Injection (NBI), Electron Cyclotron resonance Heating (ECH) and Ion Cyclotron Range of Frequencies (ICRF) heating. The leakage of the static magnetic field from LHD was almost less than 0.1 mT, although it varied according to the superconductive coils system operation conditions. The extremely low frequency electromagnetic field was measured around the power supply boxes for the coil system, and higher magnetic strength which exceeds the guideline levels was observed. Leakage of high frequency electromagnetic fields from the ICRF devices of 25-100 MHz were observed in bursts according to the experimental plasma shots. The measured leakage was less than the occupational guide line levels. As results, it revealed that the electromagnetic fields in the magnetic confinement fusion test facility should monitor in more precisely and in long term from the viewpoint of the occupational safety and management for workers. Also because the electromagnetic fields were generated in burst like and statistically varying mode, we have developed and established the continuous measurement system.

Analysis on Tritium Management in FLiBe Blanket for FFHR2

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In FFHR2 design, FLiBe has been selected as a self-cooling tritium breeder for low reactivity with oxygen and water and lower conductivity. Considering the fugacity of the tritium, particular care and adequate mitigation measures should be applied for the effectively extract tritium from breeder and control the tritium release to the environment. In this paper, a tritium analysis model of the FLiBe blanket system was developed and the preliminary analysis on tritium permeation and extraction for FLiBe blanket system were done. The factors which affected tritium extraction and permeation were calculated and evaluated, such as the heat exchanger material, tritium permeation reduction factor (TPRF) in blanket, proportion of FLiBe flow in tritium recover system (TRS) and efficiency of TRS etc. The results of the analysis showed that further R&D efforts were required for FFHR2 tritium system to guarantee the tritium self-sufficient and safety, for example reasonable quality of tritium permeation barriers on blanket, requirement for the TRS and fabrication technology of the heat exchanger etc..

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Hydrogen Isotopes Recovery from Liquid Li with Y Hot Trap

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Materials to construct fusion reactor chamber are subjected to neutron irradiation. It is necessary to analyze how it affects the material in International Fusion Materials Irradiated Facility (IFMIF). The material is irradiated by the neutron beam which is generated by D-Li stripping reaction. Tritium(T) generated by the reaction need to be recovered from liquid Li for safety. T recovery by an yttrium(Y) hot trap is one of the ways to recover T from liquid Li in IFMIF. Therefore, it is necessary to analyze the behavior of T in the liquid Li. We've already traced the results under the conditions of static liquid Li and dynamic liquid Li. We concluded those hydrogen absorption rates of Y from liquid Li depended on the condition of surface no matter it was static or dynamic. In this study, we measured H_2 absorption rates in stirred conditions of liquid Li and analyzed the effect of hydrogen absorption rate of Y by elevating temperature from 250°C to 400°C to establish the way to recover T from liquid Li with the Y hot trap in IFMIF.

In the experiment, we prepared a Y plate of 99.9% in purity, 0.25mm in thickness and the rectangular size of 25mm × 25mm. Oxide films on the Y surface were removed by HF treatment because the films prevent Y from absorbing H. The Y plate was put in a Mo crucible and Li was put on the plate. The Li was stirred by a propeller under conditions of the rotation rates of 0-100rpm. When a constant concentration of H_2/Ar mixture was supplied into the Y-Li system, the H_2 effluent concentration was determined by a gas chromatograph. H absorption rates in the Y-Li system were compared with numerical calculations. We compared those H absorption rates in the temperature range of 250°C -400°C, various rotation rates and activation conditions of Y. Judging from the comparison, we considered the rate-determining step was H diffusion in Y. Mass transfer coefficients were determined by fitting among the results of the calculation and the experiment.

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Analysis of simultaneous H and D permeation through lithium-lead

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Lithium-lead eutectic alloy (Li-Pb) is a promising liquid blanket of an advanced fusion reactor because of the potential abilities for less radiation damage, high T breeding ratio, easy maintenance, and possible use as a coolant. For a fusion reactor to be realized, it is necessary to recover T from a blanket loop highly efficiently and to realize low T leak from fusion reactor. For establishing a T recovery system and safety confinement, mass-transfer data of permeability, diffusivity and solubility of hydrogen isotopes in Li-Pb are important issues. We have experimentally determined the mass-transfer properties by an unsteady permeation method. When recovering T from Li-Pb, isotopic exchange reaction is proposed to improve T recovery ratio, which means that there are multi-component hydrogen isotopes in Li-Pb. We have presented mass -transfer data of a single component of H_2 or D_2 in Li-Pb. In order to study an influence of permeation of multi-component hydrogen isotopes in Li-Pb, we have experimented on simultaneous H and D permeation through Li-Pb in the two component (H+D) system. T data can be predicted based on isotope effects between H and D. In our experiment Li-Pb was put on a α-Fe plate in a permeation pot, which side wall was made of SUS304. The following several kinds of H -D composition ratios were tested ; H2 : D2=100:0, 90:10, 80:20, 66:34, 44:56, 34:66 and 0:100 in the temperature range from 673K to 973K. Permeability, diffusivity and solubility were obtained by comparing with analytical equations. We have simulated two-dimensional permeation analysis through Li-Pb. By this simulation, part of hydrogen leaks in the side direction was estimated. The effect is quantitatively taken into consideration. As a result, it was found that H and D leaks with a low rate through side wall and permeate independently regardless of the H/D component ratio. While permeability and diffusivity of H were around 1.4 times larger than that of D, solubility of H was close to that of D in the experimental temperature. The ratio of the isotope effect is considered to be equal to the square root of mass ratio of H and D, so T permeability and diffusivity can be predicted as around 1.7 times of H in the temperature range from 673K to 973K.

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Optical emission and mass spectra observations during hydrogen combustion processes in atmospheric pressure microwave plasma

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Recovery of leakage tritium in a nuclear fusion reactor building is one of the important issues. So far, the tritium recovery system removes tritium from a gas by cracking the tritium-containing components on a heated precious metal catalyst. The tritium combines with oxygen in the air stream to form tritiated water. Then the tritiated water contained in the air stream is removed by a molecular sieve bed. Although this system has suitable performance efficiency, there are some problems for high-pressure drop, the utilization of a large amount of precious metals and heating efficiency etc., when the processing throughput is quite huge. In order to resolve the issues, we have proposed a hydrogen isotope oxidation process by atmospheric pressure plasma. This method have the advantages as follows, low pressure drop, without noble metals such as platinum and palladium, hydrogen and oxygen radicals are easily generated by high energy electron and ion impact etc.

Experimental studies on hydrogen isotope oxidation by an atmospheric pressure plasma generated by 2.45 GHz microwave discharge have been done. Small amount of hydrogen and oxygen were mixed in the operational Argon gas during the discharge. The constituents of combustion-processed gas were observed by a quadruple mass spectrometer. To clarify the detail of the combustion process, optical emission measurement has been done. Notice that hydrogen was used as simulated gas of tritium in this experiment.

The degree of hydrogen oxidation, so called conversion rate, is increased as increasing the input microwave power. Maximum hydrogen conversion rate was reached to 85.5% in our experimental condition. During the discharge, optical emission spectra from OH radicals were observed. These results indicate OH radical might play important role in the hydrogen combustion process.

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Nonlinear Gyrokinetic Simulation Study on ITG Turbulence and Zonal Flow in LHD Discharge

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For ion temperature gradient (ITG) turbulent transport and zonal flows in a Large Helical device (LHD) discharge, nonlinear gyrokinetic simulations are applied by means of the GKV-X code [1]. In recent LHD experiment with the high ion temperatures, spatial profiles of the density fluctuation were measured precisely by two-dimensional phase contrast imaging [2]. The measured fluctuations show the characteristics of the ITG turbulence, i.e., propagating fluctuation along the ion diamagnetic direction in the plasma frame, and increase of the amplitude due to the growth of the ion temperature gradient. In order to investigate the turbulent transport driven by the ITG modes, we perform gyrokinetic simulations for the equilibrium field configuration of the LHD discharge. While the linear ITG modes are most unstable in radial and wavenumber regions where the fluctuation of the experimental measurement peaks [3], nonlinear simulation results give the turbulence transport levels which are comparable to the experiment results of the anomalous transport, and reasonable agreements with the experiment are found in the wavenumber spectra of the fluctuations. Furthermore, we obtain a clear relationship between the turbulence, zonal flow, and the transport, including the case with optimized LHD configurations.

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Effects of ion-temperature-gradient driven turbulence on magnetic islands

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Effects of ion-temperature-gradient driven turbulence on magnetic islands are investigated by means of numerical simulations of a reduced set of two-fluid equations which include not only electron diamagnetism but the ion diamagnetism in slab geometry. Simulations are carried out in the island fixed frame, where the width and poloidal location of magnetic island do not change. Uniform ExB flow is applied to the island, and the drive as well as the drag force acting on the island are calculated as a function of the externally applied flow velocity. The turbulent fluctuations enhance the momentum exchange across the sepratrix of island and thus enhance the drag force acting on the island. The zonal flow produced by the turbulence makes the island propagation velocity deviate strongly from the one without the turbulence.

A two-scale model for zonal flow and turbulence in non-axisymmetric systems

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Interactions of zonal flows and turbulence and resultant turbulent transport reduction have long been an important issue in theoretical, numerical, and experimental studies of the anomalous transport in magnetically-confined plasmas. In numerical simulations, zonal flows are supposed to be n=0 modes with low m mode numbers because of the long scale-length in poloidal and toroidal directions, while the micro-turbulence, such as ITG mode, is regarded as high n modes with small spatial scales. The toroidal scale separation has naturally been assumed in the flux tube or in the wedge torus models of toroidal plasma confinement. In non-axisymmetric systems such as LHD, however, the conventional scale-separation is not trivial, as a n=0 mode perturbation may couple with the non-axisymmetric component of the confinement field with the field-line-label dependence. The toroidal coupling of zonal flows and the non-axisymmetry is closely associated with the zonal flow enhancement in case with the equilibrium-scale radial electric field (that is, a macro-scale poloidal ExB flow) [1-3]. Therefore, we consider a two-scale approach to this issue. The new theoretical model is applied to simulation of the zonal flow response enhancement in the LHD type configuration with the macro-scale poloidal ExB flow.

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Neoclassical Tearing Mode Analysis in Tokamak Plasmas

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For the achievement of high beta value in tokamak fusion reactors, it is important to control magnetic islands produced by the neoclassical tearing mode (NTM) and to suppress incidental plasma confinement degradation [1].

In this work, plasma parameter change due to NTM is analyzed and the NTM island control by the electron cyclotron current drive (ECCD) was studied using 1.5-dimensional transport code TOTAL. The time variation of magnetic island is described by the modified Rutherford equation. The NTM control by ECCD is affected by the stabilization efficiency of the EC current. The stabilization efficiency of the EC current localization changes by EC injection phase, position and width. However, how much the efficiency of the EC current change by these values is not clarified.

The EC control efficiency η_{ec} decreases by the lag in the EC injection phase from O-point of magnetic island $\Delta\alpha_c$. In the case of narrow EC injection width case, a large value η_{ec} can be achieved when $\Delta\alpha;_c$ is nearly equal to zero. However, η_{ec} decreases more suddenly by increasing $\Delta\alpha_c$ than that in the wide poloidal width case.

When EC current I_{ec} is small, η_{ec} for complete stabilization of NTM becomes a larger value. So NTM cannot be completely stabilized when $\Delta\alpha_c$ is larger.

The change of the EC control efficiency by the lag in the EC injection phase from O-point of magnetic island and the temporal evolution analysis of magnetic island width will also be reported in the conference.

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Nonlinear Simulation of Energetic Particle Modes in High-Beta Tokamak Plasma

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The global hybrid code MEGA [1,2] is used to study the dynamics of energetic particle modes (EPM) [3,4] in parameter regimes close to the onset of ideal magnetohydrodynamic (MHD) ballooning instability. This work is motivated by observations of energetic-particle-driven modes in the wall-stabilized regime (so called EWM) in JT-60U [5,6], the physics of which have not been fully understood yet. The present simulations exclude several important ingredients required for a complete description of EWM, such as a free plasma boundary and kinetic thermal ion compression, and focuses only on the effect of high bulk plasma beta, $\beta_{\rm bulk}$, on the stability and nonlinear evolution of EPM. In the linear regime, it is found that the EPM growth rate increases as $\beta_{\rm bulk}$ increases, which may readily be explained in terms of a reduction in ideal MHD potential energy δW_f. Simulations are currently underway to analyze the effect of high $\beta_{\rm bulk}$ on the nonlinear saturation level of the mode and the energetic ion transport. New results will be reported as they become available.

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Simulation Study of Energetic Particle Driven Geodesic Acoustic Mode in LHD Plasma

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The energetic particle driven geodesic acoustic modes (GAMs) in LHD plasma were simulated using a hybrid simulation code for magnetohydrodynamics (MHD) and energetic particles[1]. The energetic particle distribution employed in the simulation is anisotropic in velocity space. The GAM frequencies in the simulation results are 38kHz for 5.0keV electron temperature and 20kHz for 1.3keV electron temperature. The frequencies agree reasonably with both the experimental observation[2] and the prediction from MHD theory. The poloidal velocity v_θ with mode number m=0 and perturbed pressure and density with mode number m=1 are observed. The GAM is excited around the normalized minor radius 0.2, and it propagates in radial direction. Both inward propagation and outward propagation are observed in the simulation. This is the first simulation of the inward propagation of the energetic particle induced GAM. Furthermore, the mode intensity burst is also simulated. In addition, the spatial profiles of the perturbed pressure and density rotate in poloidal direction. After the saturation of the instability, redistribution of energetic particles takes place in pitch-angle space both for the lower pitch-angle part and for the higher pitch-angel part.

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Effect of Nuclear Plus Interference Scattering on Fast-Ion Slowing-Down Distribution Functions in Thermonuclear Plasmas

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It is well known that fast ions slow down via Coulomb and/or non-Coulombic, i.e. nuclear plus interference (NI)[1], scattering and create knock-on tails in fuel-ion distribution functions. Effect of the knock-on tail formation on plasma burning characteristics [2] and its application to plasma diagnostics [3] have been examined. The NI scattering accelerates the slowing-down process of fast ions [4], and it reduces the fraction of energetic component to total distribution function. The reduction of energetic-ion population in steady-state plasma may influence the fast-ion loss and plasma heating efficiency [5]. It is important to quantitatively evaluate the effect of the NI scattering on the fast-ion slowing-down distribution.

In this paper, on the basis of the Boltzmann-Fokker-Planck (BFP) simulation [2], effect of the NI scattering on fast-ion slowing-down distribution function is evaluated. As a fast ion, alpha-particle produced by the $T(d,n)^4He$ reaction, proton by the $^3He(d,p)^4He$ reaction or deuterium beam injected into plasma is considered. A noticeable reduction in the fast ion population in confined plasma is shown. The effect of the reduction on fast-ion loss process is discussed.

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Simulation study of NBI beam ion distributions and heat depositions in the time development plasma of LHD

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High ion temperature plasma is obtained after the rapid decay of plasma density due to the pellet injection in the LHD plasma. The increase of the plasma density by the pellet injection enhances the ionization of neutral beams, and following rapid density decay increases the heating power per particle. In order to analyze the transport property of this time developing plasma we have to use the beam deposition analysis code including the effect of the plasma time development. In this paper we improve the GNET code[1] taking into account the time development of the plasma density and temperature. We perform the simulation in the time development plasma of LHD and estimate the time development of the beam ion distribution and heat depositions. Also we compare the results with that of the previous version of GNET.

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Study of Plasma Blob Dynamics with Particle Simulation

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Recently, it is observed that coherent structures along magnetic field lines in scrape-off layer (SOL) of magnetic confinement fusion devices propagate from the edge of core plasma to the first wall [1]. These structures, blobs, are thought to transport a plasma into the far SOL across magnetic field lines. Then many theoretical and numerical investigations of blob dynamics have been performed on the basis of two-dimensional reduced fluid models [1]. However, in this kind of fluid model, kinetic effects, such as sheath formation between plasma and divertor plate and velocity difference between electrons and ions are treated under some assumptions and parameterization. In this work, for the purpose of investigating blob dynamics including such kinetic (microscopic) effects, we have developed a three dimensional electrostatic particle simulation code with particle absorbing boundaries. Results of preliminary simulations show that blobs move to the first wall across the magnetic field lines. In the case of periodic boundary condition in toroidal direction (that is, the effect of sheath is neglected), it is observed that propagating blobs are collapsed by some instabilities. This feature is consistent with the expectation with the fluid theory. Furthermore, the effect of sheath on blob propagation is discussed.

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Analysis of ECRH Preionization for Plasma Startup in JT-60SA

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Tokamaks using superconducting coils have been developed towards a steady state operation. In a superconducting tokamak, the generated loop voltage for breakdown is generally lower than that of normal conducting tokamaks due to limitation in the coil voltage. This may cause a difficulty of plasma startup especially when wall conditions are not ideal. To solve this problem, preionization using Electron Cyclotron Resonance Heating (ECRH) has been proposed. ECRH preionization has been experimentally investigated at low loop voltage in normal conducting tokamaks such as JT-60U and DIII-D. In the Korean Superconducting Tokamak Advanced Research (KSTAR) tokamak, it was found that ECRH assist is helpful for a plasma production. Since the JT-60SA superconducting tokamak (R_0 = 2.97 m, a = 1.18 m, B_T = 2.25 T), which is now under construction in JAEA [1], has a limitation of the toroidal electric field up to 0.5 Vm^{-1}, we need to clarify conditions for robust plasma initiation. In the initial JT-60SA operation phase, application of the second harmonic 110 GHz ECRH is under consideration for reliable plasma startup. In this paper, we study the plasma startup assisted by ECRH in JT-60SA using a zero-dimensional (0-D) model developed by Lloyd [2] and Bae [3]. The main purposes are to examine the conditions for reliable startup and to clarify the dominant physics process. Five temporal equations are solved for a spatially uniform plasma, that is, the electron and neutral density conservation equation, the electron and ion energy density equations, and the electric circuit equation. We evaluate the ECRH power required to produce plasma and determine the effects of the neutral density and the carbon and oxygen impurity ions. It is found that the ECRH power of about 200 kW is required to start up the plasma under the assumptions that the injected ECRH power is fully absorbed and the initial neutral density is $n_0 = 0.3\times10^{19}$ m^{-3}. In addition, as the initial neutral density and impurities density increase, the more ECRH power is required. The carbon impurity effect is a little larger than that of oxygen. When the carbon impurity is more than 0.5% of the electron density, the required ECRH power is more than 400 kW.

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Nonlinear collision effect on energetic particle confinement in LHD plasmas

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Confinement of energetic particles is one of the important issues in the
fusion reactor research.
Energetic α-particles are produced by D-T fusion reactions and the
energy of α-particle is indispensable to sustain a high temperature fusion plasma.
Also the lost energetic α-particles might damage the first wall.
Therefore,
it is important to confine the energetic particles until the energy
slow-down to thermal energy.
Particularly,
in helical systems, energetic particle trajectory is complicated in a
three dimensional magnetic configuration.
Thus the confinement of energetic particles is one of the critical
issues in designing helical reactor.
Additionally,
collisions between the energetic particles could enhance the pitch angle scattering[1]
and deteriorate the confinement.
Thus the analysis including the both complicated orbit and nonlinear
collision effects are necessary to make clear the energetic particle
confinement in LHD plasmas.

In this paper,
we study the energetic particle confinement including the collisions
with various plasma species such as electrons, ions, and
energetic particle.
We have improved the GNET (Global NEeoclassical Transport) code[2] to
take into account the nonlinear collision effect in LHD.
We analyze the real and velocity space distributions and the energy and
particle loss rate changing the background plasma parameters,
and verify the effects of collision between energetic particles on the
energetic particle confinement.

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Plasma domains and development of operation scenarios in JT-60SA

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JT-60SA (Super Advanced) is a superconducting (all the toroidal and poloidal coils) tokamak updated from JT-60U [1]. The objective of the project is to support researches on ITER and develop physics and engineering basis towards DEMO reactor. One of the most important objectives is to establish a plasma operational scenario of high normalized beta (β_N) and high bootstrap current fraction (f_{BS}) towards realization of a steady-state tokamak DEMO reactor. For this establishment integrated achievement and sustainment of high confinement, high β_N stability, fully non-inductive current drive and heat and particle control are required. In other words, high integration of control in wide extent is indispensable. In JT-60SA, various kinds of control actuators/schemes are prepared such as, both passive (with the stabilization conducting plate) and active (by the saddle coils) control for the resistive wall mode (RWM), NB lines with a variety of both acceleration energy (up to 500keV) and injection angles for non-inductive current drive and toroidal rotation control, and an ITER-like divertor with gas fueling, impurity seeding and pumping for heat and particle control. Towards ITER and DEMO relevant regime, operation at a higher current is desired. JT-60SA is designed to operate up to 5.5 MA with with q_{95}~3 to study plasmas at the lower normalized collisionality and the lower normalized poloidal gyroradius domain. Achievable performance in these plasma domains and required operational scenarios have been investigated using various codes. And an advanced plasma of high f_{BS} of >60%, a high β_N of ~4.3 is expected to be sustained fully non-inductively at 2.3 MA, for example. For the achievement of these plasmas, operation scenario and control at the current ramp phase is important. These are evaluated also utilizing various codes. In this presentation, expected plasma domain and assessment of plasma performance in JT-60SA with emphasis on development of operational scenarios, including the ramp-up phase, will be discussed.

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Current profile control for high bootstrap current operation in ITER

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For the achievement of steady-state reactor, non-inductive current-drive plasma operation should be maintained in tokamak fusion reactor. Total non-inductive current is a summation of a bootstrap current proportional to the pressure gradient of plasma and an externally driven current, such as neutral beam current-drive. But in order to establish a commercial reactor, it is necessary to reduce the amount of external current drive and to cover the majority of the plasma current with bootstrap current. Burning plasma has high autonomy, so we need to consider the current density profile with little disturbance including changes in particle and heat transports.
In this study we conducted an analysis of time evolution of the current density profile for burning plasmas in the ITER machine by using 2.0-dimensional equilibrium, 1.5-dimensional-transport code (TOTAL code [1]). Here we used current-diffusive ballooning mode model as a heat transport model.
First we performed iteration considering the difference between trial value and target value of the total plasma current, such that the current profile becomes consistent with the given profile of the safety factor. After this plasma equilibrium iteration analysis, we evaluate the bootstrap current ratio and the total current density profile during steady state which is obtained from profiles of the external driving current and the bootstrap current given in the end.
We performed the simulation driving a current at only the center of plasma and both the center and the normalized minor radius r/a=0.5 of plasma.
In the case of driving a current only at the plasma center, the ratio of the bootstrap current I_{bs} and the total plasma current I_p, I_{bs}/I_p is 0.981. However, electron density cannot be maintained and nuclear fusion reactions become lower. In the case of driving a current at both the center and the half normalized minor radius, the ratio I_{bs}/I_p is 0.901 and the profiles of temperature and electron density can be maintained throughout more than 200 second.
From this we concluded that to maintain the steady-state temperature and electron density profile of ITER in high bootstrap current ratio in the present model, external current-drive is required at both the center and periphery of the plasma.

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MHD simulation on pellet injection in torus plasmas

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Injecting small pellets of frozen hydrogen into torus plasmas is a proven method of fueling [1]. Since a plasmoid induced by pellet ablation drifts to the lower field side, pellet fueling to make the plasmoid approach the core plasma is successful when a pellet is injected from the higher field side in a tokamak. On the other hand, such good performance has not been obtained yet in experiments in the planar axis heliotron; the Large Helical Device (LHD), even when a pellet was injected from the higher field side [2]. The purpose of this study is to explain the difference between the motion of a plasmoid in tokamak and helical plasmas, and make a suggestion to obtain the good performance on fueling in the LHD. To investigate plasmoid motion, a three-dimensional MHD code has been developed by extending the pellet ablation code (CAP) [3]. In the LHD, the drift motion depends on the initial location of the plasmoid, whereas in a tokamak, the plasmoid always drifts in the direction opposite to that of the magnetic curvature vector. The plasmoid motion is mainly determined by 1/R force due to toroidal field and the force due to a dipole field induced by the plasmoid. The former force implies a drift in the direction opposite to the magnetic curvature vector, which is dominant in a tokamak. The latter force implies the oscillation due to the field tension, which is dominant in the case that the plasmoid is located inside the torus in the LHD. Therefore, the plasmoid motions in the LHD are differ from those in a tokamak. It is also verified that the connection length determines the force that dominates the plasmoid motion. In addition, the behavior of the plasmoid induced by ablation from a moving pellet has been investigated. The pellet velocity dependence on the plasmoid behavior will be discussed.

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Modeling of SMBI experiments based on Monte-Carlo simulation in GAMMA 10

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Gas fueling control is one of the most important issues to obtain good performance plasmas. Fueling control enables the profile control of the core plasma density and reduction of neutral particles in the peripheral area. Supersonic molecular beam injection (SMBI) technique, which has been developed by L.Yao et al.[1, 2], is a new method of gas fueling. SMBI can inject neutral particle deeper into the core plasma compared to the gas puffing.

SMBI system is installed in GAMMA 10.The first results of SMBI showed that SMBI achieved higher density plasmas at the core region than the conventional gas puffing case. A fast camera has been installed at the central-cell in order to observe plasma behavior. The camera system has two lines of sight in the horizontal and vertical direction of the cross-section by using dual branch optical fiber bundles. It is observed that the penetration depth of SMBI determined from the emission brightness was longer than that of gas puffing.

In order to interpret above observation results, DEGAS three-dimensional Monte-Carlo code [3, 4] for neutral transport simulation has been applied to GAMMA 10 [5]. In a whole area of the central-cell vacuum vessel, a detailed 3-D mesh structure for the simulation was constructed. Transport of neutral particles from SMBI can be simulated by using the DEGAS code. In this paper, we will describe the detail of the simulation geometry for SMBI and the simulation results carried out under conditions of various energies of neutral particles and of various positions of SMBI. Effective operation method of SMBI is also discussed based on the 3-D simulation.

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Self-consistent Modeling of Plasma Production with Radio-frequency Heating

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In stellarator type machines, besides the electron-cyclotron method, the plasma production in the ion-cyclotron range of frequencies is practiced. The self-consistent model of the radio-frequency (RF) plasma production in stellarators is described in this work. With this model of plasma production, one can perform calculations for different antenna systems. The self-consistent model includes the system of the particle and energy balance equations and the boundary problem for the Maxwell's equations. The balance of the electron energy includes the RF heating, the energy losses for the excitation and the ionization of atoms by the electron impact, energy exchange with ions vie Coulomb collisions and the losses caused by the heat transport. The balance of the charged particles includes the particle supply owing to ionization and the diffusion particle losses. In the model, it is assumed that the neutral gas is uniformly distributed throughout the vacuum chamber volume, including the plasma column. Besides plasma build-up inside the confinement volume, the RF field produces plasma outside it. The losses of the charged particles in this zone have a direct character because the particles of plasma escape to the wall along lines of force of the magnetic field. This process is accounted in the model in tau-approximation. The RF power density is calculated from the solution of the boundary problem for the Maxwell's equations. The collisional and Landau wave damping are accounted as mechanisms of the RF field dumping. The Maxwell's equations are solved each time moment for the current plasma density and temperature distributions. The calculated value of the local RF power, deposited to the electron component of plasma, is used in the energy balance equation. This value influences on the electron temperature and, in this way, on the ionization rate which determines the evolution of plasma density. The model for the stellarator plasma column is the plasma cylinder with identical ends. The plasma is assumed to be azimuthally symmetrical and uniformly distributed along plasma column. The Crank-Nicholson method is used for solving the system of the balance equations. The Maxwell's equations are solved in 1D using the Fourier series in the azimuthal and the longitudinal coordinates. Using this self-consistent model, the plasma density ramp-up with four-strap π-phased antenna is modelled. Antenna is fed with the frequency below ion-cyclotron and, plasma production in the Alfven resonance heating regime is realized.

Wavelet-based analysis of Lower Hybrid full-wave fields

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In this paper, we introduce the use of Continuous-Wavelet-Transform (CWT) to postprocess full-wave fields [1] of Lower Hybrid waves which have been generated using the LHEAF code. Compared to the Fourier transform, the CWT has the appealing property of yielding information as to the spatial location of spectral modes. Using a complex-Morlet CWT, the complicated full-wave field pattern is decomposed into its spectral components parallel to the static magnetic field ($n_{||}$). In general, the CWT of the LHEAF full-wave fields shows that the local wave spectrum broadens as the waves propagate through the plasma and after reflection off the low density cutoff or the vacuum vessel walls. According to beam tracing theory, this result is to be attributed to diffraction effects taking place close to caustic surfaces [2]. The eventual goal of this analysis is to provide a tool that can be used to assess the range of validity of ray-tracing calculations. Also, the importance of full-wave effects on the transformation of the wave $n_{||}$ spectrum and on the LH power absorption and driven currents can be studied. Concerning this subject, particular attention is given to LHEAF simulations of Alcator C-Mod LHCD discharges at high densities, in a regime where an anomalous loss of LHCD has been reported [3]. LHEAF simulations using a realistic 2D SOL model were found to reproduce qualitatively well the experimental drop of HXR emission that is observed for increasing plasma density. This result has been attributed to the combined effect of an $n_{||}$ upshift and radial diffusion of fast electrons [4]. The full-wave origin of such spectral broadening needs to be confirmed by the CWT analysis hereby introduced.

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Reconstruction of a toroidal flow profile of a field-reversed configuration

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Strong toroidal flow is generated in a field-reversed configuration (FRC) plasma. It is continuously accelerated during and after the formation phase. The centrifugal force resulting from the toroidal flow causes deformation of the toroidal cross section with toroidal mode number n = 2. The deformation eventually brings a contact between the plasma surface and the chamber wall and then the configuration is terminated.
The radial profile of this self-generated toroidal flow has been recognized as rigid rotor (RR) i.e. uniform angle velocity profile. However, the toroidal flow profile measured in the field-reversal theta pinch FRC on NUCTE (Nihon University Compact Torus Experiment) -Ⅲ facility has not agreed with the RR profile. The toroidal flow profile measured by ion Doppler spectroscopy (IDS) technique has a peak at the vicinity of the separatrix and falls gradually in the scrape-off region. The employed IDS is basically line-integrated measurement. Therefore, in this work, measured distribution of line emission on wavelength and radial position has been reconstructed accurately by applying Abel transform. The reconstructed profile has been discussed comparing with two analytical equilibrium profile model i.e. widespread RR profile and newly presented two point equilibrium (2PE) model [2].

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Dependence of the quasi linear heating term model on the ECCD in helical plasmas

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Electron cyclotron current drive (ECCD) is one of the reliable methods of driving a plasma current. ECCD can control current profile locally and has been applied for toroidal devices to keep the current profile, to stabilize MHD instabilities and to cancel the bootstrap current in helical systems. In order to study the ECCD physics related to trapped electrons, ECCD experiment has been performed in Heliotron-J and the ECCD dependence on magnetic configurations has been investigated[1]. It was found that EC current depended on the magnetic configuration and the current direction was reversed in high bumpiness configuration compared with the other two magnetic configurations.

In this paper, we study the ECCD on helical plasmas by GNET code[2] in order to make clear the role of trapped electrons in ECCD. GNET can evaluate a steady state solution of distribution function in 5D phase-space using the Monte Carlo method. GNET has been developed for the study of high energy electron transport in helical systems and is applied to ECCD analysis.

We analyze ECCD assuming magnetic configurations, plasma parameters and heating conditions similar to the experimental ones. Here, electron cyclotron heating is taken into account through quasi linear heating term. In the previous study[3], we approximated the heating term as a point heating term using the delta function in the velocity space for simplicity. Simulation results with this heating term showed that the current direction reversed between high and low bumpiness configurations and these directions are determined by the balance between the Fisch-Boozer effect and the Ohkawa effect. These tendencies agrees well with the experimental ones. However, the obtained values of ECCD current is several times larger than that of the experimental ones. We modify the heating term to the realistic one, which has the distribution in the velocity space.

In GNET simulation, we follows the test particle orbits to solve the drift kinetic equation using the linear Monte Carlo collision operator (Boozer and Kuo-Petravic model[4]). This operator does not conserve the momentum and energy between test particles and field particles. We develop the momentum conserved collision operator for GNET in order to simulate the current drive. We present the development process of this operator and simulation results.

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Effect of Radiation Power Loss due to Ne Impurity Gas Puff for Ergodic Layer

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Reducing high heat load on a divertor plate is an issue to prevent serious damage to divertor plates in future fusion devices such as the ITER, a DEMO, and a helical-type reactor (FFHR) [1]. Gas injection to a divertor plasma is one of possible ideas to reduce heat load on divertor plates, since impurity gas causes radiative power loss and decreases electron temperature, expected to result in plasma detachment. The effect of impurity radiation loss was examined analytically for the ITER divertor [2] and neon was found to be the optimum candidate. A helical-type device has different characteristics of edge magnetic field line structure from a tokamak. Plasma in the Large Helical Device (LHD) at National Institute for Fusion Science has ergodic layer surrounding a core plasma and the field connection lengths are distributed from orders of 1 m up to 10^5 m [3]. Charged particle flux onto divertor plates in LHD is not uniformly distributed and the heat load becomes very high at a small region of the divertor plates. High particle flux points correspond to points with longer connection lengths. Some plasma experiments with Ne gas puff into the edge region in the LHD were performed during the 14th experimental campaign in 2010-2011. It was found that the Ne gas puff reduced electron temperature in the outer region and the particle fluxes onto the divertor plate were decreased much more at higher flux points. These results imply that the radiation power loss due to Ne gas plays an important roll at the ergodic layer. In order to examine the effect of Ne gas radiation loss for the ergodic layer, we study a simplified 1 dimensional fluid model for magnetic flux tube with different lengths. In a one-zone model we found the Ne gas radiation loss can reduce electron temperature less than 1 eV and we expect plasma detachment [4]. In this work we examine how the Ne radiation power loss affects the plasma with spatial distribution along a flux tube and condition for plasma detachment.

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A Pilot Plant as the Next Step toward an MFE Demo

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With ITER now well underway, the next steps toward commercial magnetic fusion energy (MFE) are under consideration worldwide. Substantial science and technology development is required beyond ITER, for example in energy and tritium extraction, rapid replacement of internal components, and steady-state plasma control with minimal recirculating power. An intermediate integration device between ITER and the first prototype commercial power plant, or Demo, may be required to reduce the risks of developing a reliable fusion system with increased availability. This motivated an examination of possible pilot plants: facilities that would substantially narrow the technical gap to a prototype commercial system in a next step. The Pilot Plant mission is to: 1) test internal components and tritium breeding in a steady-state fusion environment, 2) prototype a maintainable design and maintenance scheme for a power plant, and 3) generate net electricity. Such a facility would integrate the science and technology of a fusion power plant, demonstrate overall system efficiency, and convincingly demonstrate fusion's potential as an energy source.

Preconceptual designs based on the advanced tokamak (AT), spherical tokamak (ST), and compact stellarator (CS) have been developed in order to compare their relative merits as fusion systems. [1] Any of them would take a large step toward Demo in key performance metrics, e.g. engineering gain Q_{ENG} (≥1), neutron wall load (> 1 MW/m^2), tritium breeding ratio (> 1), pulse length (10^6 - 10^7 s), blanket lifetime fluence (≥ 3 MW yr./m^2), plant lifetime (6 20 MW-yr./m^2), and availability (10-30%). Pilot plant facility designs differ from traditional physics research devices in important ways. Magnet systems must have large openings to accommodate rapid replacement of large blanket and shield modules for high availability. First wall openings for diagnostics and heating systems needed for plasma control must be minimized since these compete with blanket coverage needed for tritium breeding. Plant systems, especially those for plasma heating and thermal conversion, must achieve a high level of energy efficiency in order to produce net electricity.

A description of pilot plant design characteristics and an assessment of pilot plants against Demo prerequisites will be presented.

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Comparative study of cost models for tokamak DEMO fusion reactors

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To prove the feasibility of the nuclear fusion power plant, various designs of tokamak-typed demonstration reactor DEMO have been proposed [1, 2]. One of the tasks of DEMO includes establishment of the economical perspective after realization of fusion power plants. Therefore, an attractive design of DEMO requires the assessment of the economical evaluation. Especially, appropriate choice of the cost model in the studies of the system codes for the optimization of the reactor design is necessary to show the economical validity of the DEMO or commercial reactors.
In the present study, we investigate the economical evaluation methods for previously proposed DEMO reactor designs. The concept of both design and cost evaluation will be compared among each design. The amount of materials used in the fusion island is employed as fundamental data of cost evaluation. The cost of the balance of plant will be considered corresponding to the thermal output power. We construct the database of the amount of materials using account number classification for each design concept.
In the international comparison of cost models, it is important to consider the temporal variation of price of commodities or rate of exchange from the time when each cost evaluation was performed. The methodology for the development of more generalized international cost comparison should be discussed based on the past examples of cost comparison.
As a future study, development of a scaling law for the cost calculation is planned to clarify the effect of each design parameter on the cost and to calculate the cost conveniently using representative design parameters. It will be useful for the parameter survey study to optimize the cost using the system code for the reactor design [3].

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Economic evaluation of D-T, D-^{3}He, and catalyzed D-D fusion reactors

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D-T fusion reactor system can be realized at low plasma temperatures in the first; however it generates 14MeV neutrons that cause engineering problems and it needs tritium breeding system. Since D-^{3}He reaction generates no neutron and D-D reaction has abundant fuel resource, these reactions are expected as advanced fuel fusion reactions. Not only the engineering problems but also the economics are important in order to achieve the advanced fuel fusion reactor as a commercial plant. Therefore, we estimate and compare the cost of electricity (COE) of D-T, D-^{3}He, and catalyzed D-D fusion reactors using the Physics-Engineering-Cost (PEC) system code [1].

Since the reaction rates of D-^{3}He reaction and D-D reaction are small, D-^{3}He and D-D reactor need to have high efficient confinement properties and be operated at high temperature. Furthermore, power density of D-^{3}He and D-D reactor are smaller than that of D-T reactor, so that D-^{3}He and D-D reactors require large plasma volume. Consequently, COE increases owing to large size of these reactors.

Because of neutron damage to the first walls, D-T reactor requires several replacements of blanket within plant lifetime. In contrast, D-^{3}He reactor has low neutron wall lord, and no need to replace blanket. Thereby, D-^{3}He reactor has no replacement cost. In addition, D-^{3}He reactor can omit tritium breeding system, except ^{3}He gas should be explored in the moon.

Assuming high ion temperature (=70keV) and high normalized beta value (=8), COE of D-^{3}He reactor is expected to be similar to that of D-T reactor. According to analysis, pure ignited D-D reactor has no feasibility as commercial plant. In terms of cost, catalyzed D-D might be disadvantage compared with D-^{3}He and D-T reactor.

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Economical and Life-Cycle Energy Assessment of Magnetic Fusion Power Reactors

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Fusion reactors might require enormous amount of construction costs and rare and valuable materials. To search for an optimized power plant, we analyzed several types of fusion reactors (tokamak TR, spherical tokamak ST, and helical HR reactors) using physics, engineering and cost (PEC) code [1], which evaluates economical and life-cycle energy amount. We compared the cost of electricity (COE) and the energy payback ratio (EPR) of each fusion reactor to the fission reactor [2]. In addition, we compared several blanket and shield designs used rare and valuable materials; such as silicon carbide (SiC), vanadium alloy (V), and ferritic steel (FS).

Reference fusion reactors in this study using PEC code have 1GW electric output, 75% availability factor, and 30 year operating period, which corresponds to the reference of the fission reactor. The EPR is defined as a ratio of output /input total system energy. As for electric power generation plants, input energy is defined by the total energy relevant to fuel production, fusion power plant construction, management and operation (M&O), replacement and decommission of reactor equipment. Output energy is net total electricity from the fusion reactor during the plant life period.

In the present study we found that the COE of TR is the lowest than those of other fusion reactors, since the capital cost of the tokamak reactor TR is lower than those of HR and ST. However, the COE of TR is higher than that of the fission reactor. The EPR of TR with the SiC blanket is highest among several fusion reactor designs because of high thermal efficiency, and higher than that of the fission reactor.

In the conference, we will also present the assessment results of the inertial confinement fusion reactor IR.

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P2-94

Design Window Analyses for the Helical DEMO Reactor FFHR-d1

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A conceptual design activity for the helical DEMO reactor FFHR-d1 was started at the beginning of the last fiscal year. As the first step, design window analyses using a system design code are carried out to fix main design parameters (major radius, magnetic field strength and fusion output). A candidate design point is narrowed and examined by considering consistency of engineering and physics constraints.

According to the steady-going progress in the LHD experiment [1], an LHD-type helical fusion reactor is now foreseeable. In past design studies, a design with a commercially-attractive concept has been considered and several design concepts, represented by the latest design FFHR-2m2 [2], were proposed. From the last fiscal year, a new conceptual design for a helical DEMO reactor, FFHR-d1, was started by utilizing the knowledge of the past design studies. As the next step reactor, FFHR-d1 aims at an early demonstration of maintainability, tritium self-sufficiency and net electric power production. To determine main design parameters of FFHR-d1, design window analyses are carried out using the system design code HELIOSCOPE [3]. In the evaluation of the core plasma performance, the direct profile extrapolation (DPE) method [4], which directly extrapolates the radial profiles observed in the LHD experiments to the fusion reactor condition assuming a gyro-Bohm type parameter dependence, is used to enhance the reliability.

It was found that a reduction in minimum blanket thickness can moderate the requirement on the confinement improvement with keeping the stored magnetic energy of the coil system, which is one of the key parameters in the engineering design. In the past design study, the stored magnetic energy of 160 GJ has been considered as an achievable maximum value with the extension of the ITER technical basis. If the minimum blanket thickness of ~ 70 cm is accepted, a steady-state, self -ignition plasma can be designed with the core plasma parameters which can be reasonably extrapolated from the present experimental results under the constraint of the stored magnetic energy of $\leq$ 160 GJ. To examine engineering design feasibility of the thinned blanket system, an evaluation of a tritium breeding ratio (TBR) and fast neutron shielding performance of the blanket system is carried out by neutronics calculations which reflect a consideration on the radial-build of the helical coil, vacuum vessel and the blanket system.

The details of the result of the design window analyses and the selected main design parameters will be described in the presentation.

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P2-95

Neutronics Investigations for Helical DEMO Reactor FFHR-d1

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The conceptual design activity for the helical DEMO reactor FFHR-d1 has been started in National Institute for Fusion Science. For an early demonstration of a steady state helical reactor, a design window analysis has been conducted by balancing three parameters of (1) a confinement improvement factor of a core plasma, (2) a stored magnetic energy of a superconducting magnet system and (3) a blanket space for installation of a tritium breeding blanket and a radiation shield. The decision of the minimum required blanket space which has a direct impact to the reactor size is the most major neutronics issue at this first stage.
Since a position of a magnetic axis shifting to the inboard side of the torus has been selected, the available blanket space will be smallest at the inboard side. For the average neutron wall loading of 1.3 MW/m^2 tentatively assumed for FFHR-d1, the minimum blanket space to provide an adequate radiation shielding performance for protection of a superconducting magnet has been investigated by neutron transport calculations using MCNP5 code with a simple torus model. The investigation showed that a combination of a 15 cm thick Flibe+Be/ferritic steel blanket and 55 cm thick WC radiation shield followed by a 5 cm thick vacuum vessel could suppress radiations in a superconducting magnet to acceptable conditions, i.e., a fast neutron flux of <~3 x 10^{10} n/cm^2/s and a nuclear heating of <~0.5 mW/cm^3 . The minimum blanket space has been set to 70 cm.
The ratio of the inboard region where the blanket space is limited to 70 cm (15 cm thick breeding blanket and 55 cm thick radiation shield) was estimated to be only ~15 % of the torus from a calculated magnetic field distribution. In contrast to the inboard region, a sufficiently thick breeding blanket and radiation shield could be installed with the total thickness of >150 cm in the outboard region. Fully covered tritium breeding ratios (TBRs) considering no neutron leakage have been evaluated with a torus calculation model simulating a 15 cm thick breeding blanket at the inboard side and a 60 cm thick breeding blanket at the outboard side. The fully covered TBR of >1.3 was obtained by the Flibe+Be/ferritic steel blanket (^{6}Li enrichment: 90 %). The result indicates that the tritium self-sufficiency could be expected in the FFHR-d1 concept even considering neutron leakage through divertor ports and injection ports for plasma heating.
The neutronics investigation for FFHR-d1 has proceeded into three dimensional transport calculations by simulating detailed non-axisymmetric helical blanket shapes. Results of the three-dimensional evaluation of the neutronics performances will be discussed.

P2-96

Conceptual Design of Heating Devices for Heliotron-type DEMO Reactor

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The FFHR-d1 has been proposed as a conceptual heliotron-type DEMO reactor in National Institute for Fusion Science [1]. The design study has started on the FFHR-d1 heating system including ECH, NBI, and ICRH, whose total injection power is 80 MW. The FFHR-d1 plasma is surrounded with the twisted blanket and divertor, and arrangement of those heating devices is one of the key issues to reduce the neutrons heat the FFHR-d1 plasma efficiently. The ECH is applicable in low-density operation with normal electromagnetic modes, and even in high-density operation with the electron Bernstein wave (EBW). The ECH system with EBW has more accessibility to the core plasma than the other heating devices. The waveguide outlet can be shielded from the neutron flux by applying a tilt-angle mechanism for microwave direction, and the mechanism is also applied to adjust the heating position of the core plasma. The NBI beamline will be installed to the FFHR-d1 tangentially, and the maximum beam-energy is set at 1.5 – 2.0 MeV for high-density operations. The beam is not required to sustain the plasma current as in tokomak reactors, and the injection port can be closed after achieving self-ignition. The R&D of the plasma- or photo-neutralizer is required to improve the efficiency of the port-through power to the electric input. Present technology of ICRH is applicable for the DEMO-class reactor. A key issue of the ICRH is the damages of the conventional ICRH antenna or the launcher exposed with high fluence of high-energy neutrons and alpha particles. A blanket imbedded antenna is investigated for an ICRF heating interface.
In this article, we report the specifications of the heating devices required for the FFHR-d1 reactor. The arrangements of the devices reducing the neutron flux with the shield and blanket is also going to be discussed, and the advantage of each device is compared.

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P2-97

Fuel Particle Balance Study in FFHR DEMO Reactor

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The establishment of the fuel cycle system for a fusion DEMO reactor is one of key issues. The first step to consider the fuel cycle system would be estimated the fuel particle balance and flow in the fusion reactor system. The scenario of steady state fuel particle balance is as follows: the fuel particles are supplied into vacuum vessel via the fueling system such as pellet injection, gas puff, etc. A part of supplied fuel particles is ionized and burned in the core plasma, and the rest of fuel particles transports to the edge plasma region. Then, although the fuel particles and the helium ash are exhausted from the vacuum vessel, various interactions between plasma, neutral particle and material are occurred in the edge and the diverter plasma region. As the results, a small amount of the fuel particles is trapped in the wall by the retention and it is the loss of fuel particles. The exhausted fuel particles and helium ash are separated and purified by fuel processing system. On the other hand, tritium from blanket system and external deuterium are supplied for fuel cycle system to replenish burned fuel particles. Then, the fuel particles from both the fuel processing system and the blanket system are temporarily stored in the fuel storage bed and supplied to fuel injection system according to demand. In the fuel cycle loop, however, a part of the fuel particles are loss from the wall by permeation. These fuel particle losses are considered to be affected on the fuel balance. As an inevitable consequence, tritium loss causes the requirement of higher tritium breeding ratio in the blanket system.
To consider the fuel balance in fusion reactor system, hence, the fuelling rate or burning rate in core plasma, the tritium breeding ratio (TBR) and the tritium loss rates both the permeation from the wall and the retention in the vacuum vessel would be critical parameter. In previous studies, the analytical model of fuel balance have been reported with regard to the efficiency of a tritium processing system, the fueling scenario, the effect of tritium loss, and TBR, etc [1-3]. In this study, we focus on the particle balance of tritium and deuterium in FFHR DEMO reactor with regard to fueling efficiency into core plasma. Then, it is found that the fueling efficiency is one of critical parameters to realize the fuel particle balance. The detail analytical model and calculation results will be presented in the conference.

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P2-98

Studies on Investigation of Behavior of Traveling Wave Direct Energy Converter Using a Simulator Installed on GAMMA 10 Tandem Mirror with One Side Plugging

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The large part of produced energy in D-^{3}He fusion is released as the kinetic energy of protons. The energy is directly converted into electricity by a Traveling Wave Direct Energy Converter (TWDEC). Experimental verification has not been studied well, although the energy broadening of flux influences significantly on the conversion efficiency. The authors have investigated behavior of TWDEC for flux with wide energy spread by application of a TWDEC simulator to end-loss flux of GAMMA 10 tandem mirror [1]. As one method for those studies, end-loss flux of one side plugging operation of GAMMA 10 has been employed for higher energy flux. This report will treat that method.

The TWDEC simulator was installed on one of the end cells of GAMMA 10. The TWDEC simulator consists of a modulator and a decelerator. The RF voltages, the frequency of which is 7MHz, are supplied to these electrodes. Two synchronized RF power is supplied to the modulator and the decelerator, respectively, and relative phase difference between them ($\Delta\varphi$)can be set up arbitrarily.

During one side plugging operation, end-loss flux of the plugging side is significantly reduced due to confinement of plasma, while reduction of flux of no plugging side is not so large. To examine the dependence on relative phase difference, the variation of energy distribution function due to TWDEC operation has been investigated. The peak of function appears in the lower energy side in the case of $\Delta\varphi=\pi$, while it appears in the higher energy side in the case of $\Delta\varphi=0$. Those variations correspond to effects of deceleration and acceleration, respectively. In both cases, incident energy components remain around peak of incident energy. To investigate in more detail, one dimensional orbit calculation has been performed. In the calculation with change of $\Delta\varphi$, the effects of both deceleration and acceleration have been found, and the similar results with experiments have been obtained. Although some incident energy components have remained, the shapes of the peak are more gradual compared with experiments. To investigate the cause of this difference, the numerical calculation with varied parameter is planned.

The experiments using one side plugging have been continued to increase the number of data by improvement of measurement system. More experimental and numerical results will be shown in the conference.

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Studies on Application of Cusp-Type Particle Separation and Two-Stage Deceleration to Dense Plasma for Direct Energy Conversion

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In the D-^{3}He fusion power generation, high plant efficiency can be expected by applying the direct energy conversion because the majority of the produced energy is kinetic energies of charged particles. Moreover, harmful high energy neutrons are hardly generated, so the D-^{3}He fusion power generation is expected as a new excellent power generation method in the economical and environmental aspects. For efficient recovery, particles escaping from the reactor should be separated from each other in energy and electrical polarity. For this purpose, use of a cusp-type direct energy converter (Cusp DEC) is expected. Because of the difference of Larmor radius, ions go to point cusp and are separated from electrons going to line cusp.

Working characteristics of Cusp DEC on plasma density is an important subject. Not only charge separation, but also deflection of particle's orbit due to space potential becomes a problem. When the plasma density increases, ions stagnating ahead of the ion collector create the high potential area there. This is a barrier for the low energy ions, and they are reflected before it reaches the ion collector. To improve this problem, an additional energy conversion scheme with setting the second ion collector can be applied. The two-stage direct energy conversion has already been reported[1], but its experimental verification is not enough. This paper treats those subjects on working characteristics of Cusp DEC for dense plasma.

Using a multi-hole extraction electrode for the plasma source, 3-4 times increment of plasma density has been achieved. Further increment of density is expected by enhancement of magnetic field of plasma source or applying much further input power. As for two-stage direct energy conversion, systematic experiments on the ion collector's location and size have been performed by simulating the ion space potential due to bias voltage of the point cusp ion collector. As a result, it has been shown in the experiment that the collected low-energy ion increases by appropriately setting of the location of the point cusp ion collector. Moreover, it also has been shown in the experiment that the electric power obtained by the second ion collector increases by enlarging the size of the collector. Details of the behavior of the particles due to the change of these ion collectors have been understood by using a numerical analysis.

Not only two-stage direct energy conversion, but also separation experiments for higher density plasma are in progress and the detail will be presented in the conference.

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P2-100

Internal Strain Measurement in Superconducting Composites by Neutron Diffraction

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It is well known that superconducting materials are very brittle and therefore they are needed to be composites with generally metals to increase their strengths and/or ductilities. These facts lead to the prediction that phase strains are always generated in the superconducting composites, after the manufacturing and also during their uses, due to differences on thermal expansion and deformability among the constituent phases. In contrast, it has been reported elsewhere that internal strain state has high influences on superconducting properties, e.g. the presence of internal strain degrades critical current densities. The internal strain states in superconducting composites depend on compositions of composites, thermo-mechanical treatment during manufacturing, superconducting conditions during uses. Clarifying internal strains behaviors accurately at various conditions will help us to develop superconducting composites having optimum superconducting performances during uses.
However, the internal strains in superconducting composites have not been measured so far because of the composites configurations. Moreover, in the case of Cable-In-Conduit Conductors for ITER cables, superconducting phases are located in a thick jacket. Neutrons that have high penetration ability into metals may be a powerful tool to measure internal strains in the superconducting composites by the diffraction technique. Here, we introduce internal strain measurement techniques, our engineering materials diffractometer at J-PARC facility and our trials on measuring internal strains in several kinds of superconducting strands and tapes with various conditions, and a CICC for ITER TF coils.

Numerical Investigation on Accuracy and Resolution of Contactless Methods for Measuring j_C in High-Temperature Superconducting Film: Inductive Method and Permanent Magnet Method

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A critical current density j_C is one of the most important parameters for engineering applications of high-temperature superconductors (HTSs). The standard four-probe method has been generally employed for measuring j_C. However, it may lead to the destruction of a sample surface or to the degradation of superconducting characteristics. For this reason, contactless methods have been so far desired for measuring j_C.

For the purpose of contactlessly measuring j_C in an HTS film, Claassen et al. have proposed the inductive method for measuring the critical current density j_C [1]. By applying an ac current to a small coil placed just above an HTS film, they monitored a harmonic voltage induced in the coil. They found that, only when a coil current exceeds a threshold current, the third-harmonic voltage develops suddenly. They conclude that j_C can be evaluated from the threshold current.

Ohshima et al. proposed another contactless method for measuring j_C [2]. In the method, while moving a permanent magnet above an HTS film, the electromagnetic force acting on the film is measured. As a result, they found that the maximum repulsive force F_M is roughly proportional to j_C. This means that j_C is estimated from the measured value of F_M. This is called the permanent magnet method.

In order to simulate a numerical code was developed for analyzing the time evolution of a shielding current density in a non-axisymmetric HTS film [3]. By using the code, the inductive method/ permanent magnet method was reproduced. The results of computations showed that, the accuracy of the two contactless methods degrade remarkably when the coil or the magnet is placed near the film edge.

The purpose of the present study is to simulate two contactless methods and to investigate the accuracy and the resolution of the methods numerically.

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Adsorption Behavior of Lithium from Seawater using Manganses Oxide Adsorbent

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The deuterium-tritium (D-T) fusion reactor is expected to be a system to provide the main electricity in the future without any serous release of hazardous products such as a radioisotope of tritium, and it is the easiest fusion reaction to achieve. Lithium will be required in amounts dependent on the reactor design concept. When liquid lithium is used as a tritium breeder and a coolant, lithium inventories are large [1]. Lithium is now recovered from the mines and from salt lakes which contains about 14 millions tons of lithium. Although the amount of lithium in those resources is quite insufficient at this point, alternative resources should be found to satisfy lithium inventories for nuclear fusion plants and the increasing demand for battery and so on in the near future. Seawater, which contains 2300 hundred million tons of lithium in total, has thus recently become an attractive source of this element and the separation and recovery of lithium from seawater by co-precipitation, solvent extraction, adsorption, etc. have been investigated. Among these techniques, the adsorption method is suitable for recovery of lithium from seawater because certain inorganic ion-exchange materials show extremely high selectively for lithium ion only. Among the inorganic adsorbents, spinel-type manganese oxides are interesting materials because of their extremely high affinity toward lithium ions only.
In this study, we prepared the lithium adsorbent by elution of spinel-type lithium di-manganese-tetra-oxide ($LiMn_2O_4$), and the kinetics of the adsorbent for lithium ions in seawater was examined using pseudo-second-order kinetic model.
The adsorbent,which can be recover lithium ion from seawater, can be prepared from $LiOH{\cdot}H_2O$ and Mn_3O_4. With increasing temperature of solution, the kinetics of adsorption becomes faster and the amount of lithium adsorbed on the adsorbent increases. The thermodynamic values, ΔG^0, ΔH^0 and ΔS^0, indicated that adsorption was an endothermic and spontaneous process.

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Application of Virtual Reality System to Fusion Science

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Virtual reality (VR) technology can give you a deep absorption into the VR world, and enables you to analyze complex structures in a really three-dimensional space. This scientific visualization is a powerful and useful tool in an analysis of simulation data and development of experimental devices. National Institute for Fusion Science (NIFS), Japan installed VR System “CompleXcope” based on CAVE system [1] in 1997 as a scientific analysis instrument for the simulation results. The system produces a stereo, immersive and interactive view. In this paper, we report recent development of an application of VR system to Fusion Science.

In the previous Toki Conference, we reported an integrated scientific visualization of plasma simulation data and experimental device data in one VR world [2,3]. In the integrated visualization, an isosurface of plasma pressure, a stream line of a magnetic-field line and a trajectory of drift particle calculated from the MHD equilibrium simulation data [4,5] were interactively visualized in the Large Helical Device (LHD) vessel data with an objective description (that is, a realistic device mechanical vessel) in the VR space [6,7]. Recently, we modified the software for visualizing the simulation results to visualize the trajectories of multiple particles simultaneously. Trapped and un-trapped particles can be investigated at the same time. This success has opened up a lot of attractive possibilities for intuitively understanding the physics of plasma, for aiding in the design and arrangement of the devices, and for confirming the field of vision from the observation port in VR space.

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P2-104

Localization of Mobile Client using Wireless LAN and Natural Features of Camera-Image Sequences for Indoor Structure

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Most nuclear fusion reactor is generally large and its buildings are also large and multi-storied. Data on the physical position of mobile agents, for example, workers with personal electronic devices or mobile robots, are important for achieving the surveillance and maintenance task in a nuclear fusion reactor system. By integrating the data and stored information on objects and places, workers and robots can be provided with information on the location of the devices. Many kinds of localization services have been presented for mobile agents.

We introduce a method for localization of mobile client using wireless LAN and natural features of the camera-image sequences for the indoor structure. A Global Positioning System (GPS) is widely used for localization [1]; the GPS is not suitable for localization in the indoor environment and the area that is surrounded by many buildings. We use the stochastic localization system for wireless mobile client using the signal strength from wireless LAN access points based on a sparse Bayesian learning [2][3]. The estimated location of the client is used for searching the database of natural feature points obtained by the camera-image sequences [4]. The sensory data fusion improves the localization accuracy. The presentation consists of two parts; (1) construction of the estimation system based on the signal strength and the database of the natural feature points, (2) localization and acquisition of the information based on the sensing data at the other time.

Our goal is to realize the localization method for surveillance or maintenance tasks in the real nuclear fusion reactor system. The experimental testbed is a laboratory building; the building is three-storied, and there are several stairwells and open-ceiling spaces in the building. Experimental results show that the proposed method is feasible.

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P2-106

Behavior of core plasma potential and fluctuation in end-diverter simulation experiments on the tandem mirror GAMMA10

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Recently, diverter simulation experiments have been performed by using a mirror end of the tandem mirror GAMMA10, where the plasma is initiated by plasma guns and sustained using ion cyclotron heating (ICH). The axial confinement potentials are produced using electron cyclotron heating (ECH) at both end mirrors of GAMMA10. In this study, we have measured and analyzed potential, electric field and those fluctuations of the core plasma at the central cell using a gold neutral beam probe (GNBP) to investigate influence of the insert of the diverter plate to the end region on the core plasma. GNBP can measure both potential and density fluctuation simultaneously. Then it is useful for analysis of the cross spectra between them and the particle fluxes induced by them. Recently, GNBP has been developed to obtain local electric field in a single plasma shot with simultaneous multipoint plasma potential measurement[1]. The error of the potential measurement is about ±10 V. The time and spatial resolutions are estimated to be about 3 μs and 10 mm (i.e. a beam diameter), respectively. In the plasma with ICH and ECH, the potential fluctuation of the core plasma occurred in the low frequency region (1-20kHz). When the diverter plate was inserted to the mirror end in the (ICH+ECH) plasma, the fluctuation greatly increased and the mode number of the fluctuation changed from m=+2 (the direction of electronic diamagnetic drift) to m=-3. The potential also increased due to the insert of the diverter plate. The increase in the potential and its fluctuation occurred in the whole radial region of the plasma, which was beyond the region at the central cell projected along the magnetic field from the diverter plate. On the other hand, in the plasma with ICH only, the fluctuation was not observed in the low frequency region, and the potential did not change even though the diverter plate was inserted.

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Measurement of LHD divertor heat flux response of plasma detachment with the Hydride Directional Langmuire Probe

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In the design of fusion reactors like International Tokamak Experimental Reactor (ITER), vast heat flux (> 10 [MW/m^2]) is expected to flow onto divertor target plates. In order to reduce this heat load, enhancement of plasma detachment with cooling gas puff has been proposed. The concept of thermal probe method has been proposed to measure heat flux directly in the reactive plasma and divertor plasma.[1-3] Since present fusion plasma devices have shorter discharge duration than heat diffusion time of the probe tip, however, heat flux response of plasma condition change such as detachment has not been reported yet.

The Hydride Directional Langmuire Probe (HDLP) recently equipped in Large Helical Device (LHD)[4] has the heat diffusion time with the same order as plasma discharge. Heat flux must be determined with the solution of heat conduction problem in probe tips and the data of Thermo Couple (TC) embedded in the tip. In order to deal with the change of plasma heat flux, time dependent heat flux is modeled as the summation of step-like heat flux with both positive and negative amplitude. The size of each step is determined so as that the summation of their temperature response reproduces the observed temperature variation data. If a general mathematical method such as singular value decomposition or Marquart method is applied to this fitting procedure, reconstructed heat flux shows unreasonable large positive and negative value. We consider the casualties of the heat conduction problem and develop a new iterative optimization method to determine each component step-like flux amplitude.

Present modeling was applied for the first time to the calibration experiment of HDLP with Nd:YAG laser and reasonable heat flux variation could be reproduced. Plasma heat flux analysis for many discharges is also done successfully. By using Neon Gas puffing, plasma detachment can be controlled in LHD plasma. TC data obtained by HDLP set on the divertor leg shows the increase after plasma discharge. If plasma detachment occurs, this increment becomes small. By using present modeling, the onset of heat flux decrease was confirmed to agree well with those of ion saturation current.

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Plenary / Invited / Oral Sessions

(Wednesday 30th November)

Challenges on the Path to DEMO and Activities in the Power Plant Physics and Technology Department under EFDA

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The technical basis for designing a next-step DT burning plasma experiment has greatly expanded during the last two decades thanks mainly to remarkable improvements in plasma performance and control in today's machines and advances in various areas of physics and engineering. Integrating and extending these advances toward long pulsed or steady state burning plasmas is now the focus of international tokamak research, which is proceeding with the construction of ITER in the south of France and JT60-SA in Japan.

However, beyond ITER there are still several physics and technology issues, which must be addressed and resolved for a Demonstration Power Fusion Reactor (DEMO). The most important technology problems to be solved include the qualification of resilient materials for in-vessel components, the development of sound technological solutions for the divertor and of optimised remote maintenance schemes for high machine availability, the achievement of adequate thermal efficiency and tritium breeding, and the reliability and efficiency of heating and current drive systems. Among the physics questions, the divertor power exhaust, the definition of a reliable modes operation, the need to guarantee plasma performance at high density, the avoidance and mitigation of disruptions and ELMs, which can damage the in-vessel components, are the most important.

A Power Plant Physics and Technology Department (PPPT) has been recently established under EFDA, with the purpose to begin a coordinated effort in Europe (building on efforts done in the past) to quantify the key physics and technology prerequisites for DEMO and to address the remaining outstanding physics and engineering problems that need to be solved to confirm our ability to design a device that meet the requirements. In particular, it shall (i) define a set of technical characteristics for DEMO and, subsequently, carry out the design work necessary to establish its conceptual design; (ii) define future research and development needs and to draw up cost, and schedule estimates; and (iii) carry out, in a coordinated manner, specific validating R&D supportive of the conceptual design activities;

This talk will describe the main technical challenges on the path to DEMO and the organization of the EFDA PPPT activities with emphasis on the elements of the work programme 2011 and those planned for 2012.

Progress of ITER Test Blanket Module Development in Japan

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ITER Test Blanket Module (TBM) program is the first module scale breeding blanket performance test in the real DT fusion reaction environment which will be realized in ITER. Therefore, it is regarded as one of the most important development step of breeding blanket for DEMO.
For the TBM program, Japan has the plan to test Water Cooled Ceramic Breeder (WCCB) TBM as the primary candidate breeding blanket. Japan also has intention to participate and perform liquid breeder TBM testing liquid breeder TBM testing, based on the firm achievements and activities which Japan has on the development of liquid breeder blanket technologies. ITER construction activity is showing progress toward the first plasma. Keeping consistency with ITER construction activities, TBM design and development activities are also being performed. This paper overviews the TBM testing program and the recent achievements of the development of the module fabrication technology and tritium production technology of the WCCB TBM in Japan.
Fabrication technology development of the module structure is one of key technologies. In Japan, fabrication of a real scale first wall, side walls, a breeder pebble bed box and assembling of the first wall and side walls have succeeded, based on the elementary technology development for various joining technologies, such as welding and Hot Isostatic Pressing (HIP) joining. Recently, the real scale partial mockup of the back wall was fabricated clarify the fabrication procedure of the back wall, whose thickness is up to 90 mm. Important key technologies are almost clarified for the fabrication of the real scale TBM module mockup. From the view point of testing and evaluation of solid breeder TBM tritium production performance in fusion environment, development of the technology of the blanket tritium recovery, development of advanced breeder and multiplier pebbles and the development of the blanket neutronics measurement technology are also performed. Also, tritium production and recovery test using D-T neutron in the Fusion Neutronics Source (FNS) facility has been started as the verification test of tritium production performance.

A part of research on tritium performance of TBM was supported by KAKENHI (Grant-in-Aid for Scientific Research) on Priority Areas "Tritium for fusion" from the Ministry of Education, Culture, Sports, Science and Technology of Japan.

Effect of lithium purity on in-situ formation of Er_2O_3 oxide layer on V-4Ti-4Cr alloy

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The *in-situ* formation of Er_2O_3 oxide layers on the surface of V-alloys pre-charged with oxygen in the liquid Li doped by Er has been recognized as a promising technology to form the viable and functional insulator coatings for V-Li blanket concept. Most of the works were carried out in "pure" Li. However in the real system the contamination of the liquid lithium by non-metallic impurities is inevitable. The aim of this work is to determine the influence of N dissolved in Li on the formation of Er_2O_3 oxide coating on the surface of V-4Ti-4Cr alloy.
The samples of V-4Ti-4Cr alloy (NIFS HEAT) were pre-charged by oxygen in Ar-7%O_2 atmosphere (oxidation at 700°C for 3 h followed by the vacuum annealing at 800°C for 34 h). Weight gain of samples averaged 0.35 mg/cm^2, while the depth of the hardened zone (~ 680 kg/mm^2) reached 100 μm. The formation of Ti–O net structure in the hardened zone was confirmed by TEM. Then samples were exposed to the N-contaminated ($C_{N[Li]} \leq 0.5$ wt%) liquid Li doped with Er at 700°C for 100 h. The post-test examinations (weight change, hardness, XRD, SEM/EDX, OM) showed that samples lost the oxygen while the nitride phases (TiN, $VN_{0.2}$) were formed on the surface at the same time. The weight of the Er sample does not change, i.e. Er was not dissolved in Li. It was supposed that formation of ErN on the surface of Er sample prevented it dissolution since from the thermodynamic point of view the affinity of Er to N is higher than that of Li. As a result the Er-oxide coating was not formed on the surface of V-alloy.
The next experiment was carried out in Li pre-cleaned by getter material (Zr, Nb-Ti alloy, V) at 700°C for 24 h. Then pure Li was transferred into the reaction zone in which samples of V alloys and Er were placed. The test was carried out at 650°C for 100 h. Contrary to the N-containing Li the Er_2O_3 oxide film was formed on the surface of the V-alloy successfylly. Samples of V-alloy showed weight gain while Er sample demonstrated weight loss indicating dissolution of Er in Li.
It was concluded that *in-situ* formation of Er_2O_3 oxide coating on the surface of V-alloys depends strongly on the purity level of Li with regard to the N. The cleaning of Li by means of getters is a sine qua non of successful mass transfer in Li[Er]-V,Ti[O] system.

Silicon Carbide Nanopillar Fabrication on Silicon Carbide Substrate by Direct CF4 Etching Using the Gas Discharge Ion Source

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This paper reports the fabrication of silicon carbide nanopillars on a silicon carbide substrate. The samples were etched with low energy CF_4 ion shower produced by a gas discharge ion source. The flow rate and discharge current were varied to determine the effects on the growth of the silicon nanostructures. SEM images of the treated samples were investigated. Furthermore, the topography of the sample surfaces were analysed through AFM.

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I-12

Assessment and comparison of pulsed and steady-state tokamak power plants

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The current visions for steady-state tokamak plants are mostly based on operation with the reversed-shear mode in order to achieve a high bootstrap current fraction at a high plasma β. This mode of operation has been demonstrated in current experiments only in transient fashion mainly due to the lack of sufficient mid-plasma current drive power. Furthermore, it appears that reversed -shear mode require stringent control on plasma current profile and may require stabilization of resistive wall modes. As such, there is a wide-spread belief that steady-state tokamak power plants require major extrapolations from current data base. However, the first vision of steady-state tokamak power plant, introduced around 1990 simultaneously in US (ARIES-I) and in Japan (SSTR), did not utilize revered-shear mode and operated with a monotonic q profile. In fact, ARIES-I proposed plasma profiles are very similar to those of the "hybrid" mode in tokamak experiments.

In this paper, we compare the operation and needs of such a "conservative" steady-state tokamak plant with a pulsed-plasma version. Plasma physics constraints on both systems are identified and compared, the major difference is the need for ~100-150 MW of current-drive power for steady-state system and the limited control on "optimizing" plasma performance in a pulsed-plasma device. Most of the paper, however, is devoted to the impact of pulsed-plasma operation on components: thermal energy storage, issues related to superconducting magnets (cyclic fatigue, larger PF coils, rapid PF ramp rates, large and expensive PF power supplies, large joule losses in cryogenic structures), reduced performance in fusion power technologies (first wall, blanket, and divertor), and reliability of complex components under cyclic operation. Overall, we find that the additional costs associated with the pulsed-plasma operation far exceed savings due to the elimination of the current-drive power.

Design and R&D activities for the LHD-type heliotron DEMO

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Conceptual design studies on the LHD-type heliotron DEMO reactor FFHR-d1 are being conducted at NIFS through domestic and international collaborations. Intensive R&D activities are being simultaneously carried out for various components. The key requirements are the safety, net electric power production, tritium self-sufficiency and maintainability. The design parameters are examined by a system design code incorporating the confinement scaling deduced from the LHD experiments. The 3D transport of neutrons and gamma rays shows that the distance between the helical coils and the plasma at the inboard side of the torus should be >0.8 m with advanced shielding materials used in the blanket. The tritium breeding ratio must be supported by thicker breeder blankets at the outboard side. The present design gives the major radius 15.6 m, toroidal magnetic field 4.7 T and stored magnetic energy 160 GJ. The low neutron wall loading assures long lifetime blankets. The structural integrity of the in-vessel components including the built-in helical divertors is examined. The start-up scenario with external heating (NBI, ECH, ICRF) is discussed in accordance with the fueling scheme.

The superconducting magnet system is being designed along with development of 100 kA, 13 T conductors. Cable-in-conduit conductors are supposed to be the primary option as an extension of the ITER technology using low-Tc superconductors (LTS). To reduce the degradation of critical currents due to strains, Niobium Aluminum is considered for strands. To simplify the winding structure, an indirectly-cooled magnet concept seems attractive and a solid LTS conductor with an aluminum-alloy jacket is being developed. The winding process of the continuous helical coils using LTS conductors is being investigated in detail. High-Tc superconductors (HTS) using YBCO tapes could be a counter option that proposes an innovative winding method by joining half -pitch conductors on site. The supporting structure is being designed using a 3D FEM to withstand the large electromagnetic stress while supplying large maintenance ports.

The liquid breeder blanket using molten salt or liquid metal is being designed with emphasis on safety. A circulation testing facility using Flinak and liquid lithium has been constructed and corrosion tests are being carried out. Suppression of tritium permeation and reduction of MHD effects in liquid blankets are important subjects. Large-area ceramic coatings with Erbium oxide are being tested. The operation temperature of blankets can be enhanced by suppressing thermal creep deformation of reduced activation ferritic/martensitic steels and vanadium alloys by oxide dispersion strengthening.

Critical Issues of Burning Plasma, Engineering, Economic and Environmental Assessments in Steady-State Fusion Reactors

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In order to search for attractive steady-state tokamak fusion reactors, burning plasma studies and engineering system design analyses were carried out focusing on advanced plasma operations with internal transport barrier (ITB) and high bootstrap current (BSC) fraction, high field and high neutron wall loading limits, high economic efficiency without frequent blanket exchanges and environmental assessments related to global warming gas emission.
Assessments have been done using integrated toroidal transport linkage analysis code (TOTAL) with ITB [1], high BSC [2], impurity and NTM [3] effects, and global Physics-Engineering-Cost system code (PEC) [1] focusing on steady-state operations of D-T tokamak fusion reactors. Analyses are extended to spherical tokamak, helical, inertial and advanced fuel reactors [4].
Plasma simulation models clarified the requirement of deep penetration of pellet fueling to realize steady-state advanced burning plasma operation and appropriate non-inductive current drive profile adjustment for ITB with high BSC fraction (>90%) [2] in steady-state tokamak reactors.
Engineering assessment shows the relationship among the maximum magnetic field strength, neutron wall load, blanket thickness, thermal efficiency, operational period and reactor system availability. Economic and environmental system analysis has been carried out [5-6], and comparative studies among other conventional electric power plants are done, such as oil, coal, water, solar, wind, fission power plants, with respect to cost of electricity (COE), CO_2 emission amounts and energy payback ratio (EPR). The assessment shows that fusion power plants have an advantage in energy payback ratio (EPR), but have disadvantages in COE and CO_2 emission, in comparison with fission reactors.

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O-16

Assessment of the physics and technology requirements for a fusion DEMO

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In 2010 the German fusion laboratories formed a closer collaboration to assess the physics and technology requirements of a fusion power plant. The activity is organized in six physics topics (long pulse / steady state / high beta operation, high density operation, plasma wall interface, disruptions, plasma diagnostics and integrated control, combination of first principles and reactor codes) and seven technology topics (heating and current drive, structural and functional materials, in-vessel components design and integration, fuel cycle, magnet design, maintenance and remote handling, safety and licensing). In a first step the interdependencies between the topics have been analyzed and critical areas have been identified. Examples are (i) the heat flux limits to the divertor targets, assuming tungsten as a plasma facing material and He-cooling, (ii) current drive efficiencies to achieve steady state operation in tokamaks, (iii) the understanding of the density limits in tokamaks and stellarators to maximize the fusion reaction rate and ease the requirements for pumping neutral gas in the divertor chamber. In a next step these requirements are implemented in three essentially 0-D power plant models which are presently under development: (1) A pulsed tokamak which does not rely on a large current drive fraction. Such a tokamak will have a pulse length of 10-15 hours, but with the possibility of reduced confinement (H_H ~ 0,85). (2) An advanced tokamak aiming at steady state operation, decreasing the size, but requiring improved confinement and significant power for current drive, and reducing the wall life time. (3) An optimized stellarator which is basically a further development of the Wendelstein 7-X concept. Recent engineering studies suggest that coils of the size of ITER and with the ITER superconductor technology can be used. The paper will report on the main conclusions from the assessment of the interdependencies of the different physics and technology requirements and give an initial comparison of the power plant models.

Plenary / Invited / Oral Sessions

(Thursday 1st December)

History and prospects of applied superconductivity technology for fusion magnets

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One of the most fascinating and challenging large scale applications of superconductivity are magnets for fusion reactors. Up to now, many tokamak and a few stellarator facilities using superconducting magnets have been built or are under construction with the most prominent being the ITER project currently realized in Cadarache, France. The magnetic confinement in fusion devices places heavy requirements on the different magnets. In general, currents of more than 10 kA have to be handled in magnetic fields of more than 10 T resulting together with the large dimensions in huge forces and high energies. As a consequence, the investment in superconducting magnets and cryogenics takes a significant fraction of the overall cost for a fusion device.

This contribution summarizes the history, the state-of-the-art and the future developments of the superconductivity technology for fusion magnets. Major highlights of the significant progress in this field are shown and the magnet technology for large scale fusion magnets e.g. Tore Supra, LCT, TFMC, ITER, LHD and W7-X is presented. As far as possible, material aspects, conductor technology as well as magnet design and manufacturing issues are covered.

Finally, the prospects for high temperature superconducting materials in fusion magnets are summarized because these materials have a great potential to decrease magnet cost, increase reliability and efficiency of fusion power plants and simplify the construction of a fusion power plant.

What is the nature of the critical current density in superconducting materials under strain in high magnetic fields?

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Understanding the magnetic field (B), temperature (T), strain (ε) and angular dependence (θ) of the critical current density (Jc) in practical superconducting wires/tapes is critical for optimizing magnet design using superconducting technology. We review critical current measurements reported on both LTS and HTS superconductors. For the LTS materials, we present the evidence for the well-established scaling law for Jc, that can be best understood in terms of flux pinning. For HTS materials, we present recent measurements that make it possible to derive a scaling relation for Jc(B, T, ε, θ) and discuss whether in contrast to LTS materials, wave-like properties provide the best description of the mechanism that limits Jc .

Neutron Diffraction Study of Internal Strain in Nb_3Sn Cable-In-Conduit Conductors

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Internal strain in Cable-In-Conduit Conductors (CICC) is caused by differences in the coefficients of thermal expansion between Nb_3Sn strands and the stainless steel jacket over a temperature range of 5 - 923 K. In addition, transverse electromagnetic loading of 800 kN is generated by a current of 68 kA and a magnetic field of 11.8 T in the case of ITER TF coils. The superconducting performances of Nb_3Sn strands change significantly, depending on the presence of strain in Nb_3Sn. The presence of internal strain in Nb_3Sn cables is important to evaluate the superconducting performance. However, the strain of strands in the conductor has not been measured so far because of the cabling configuration and their location in a jacket. Internal strain can be determined by neutron diffraction measurement using Takumi of J-PARC[1]. Neutron diffraction measurement becomes a strong tool for evaluating directly the internal strain of Nb_3Sn in CICC. The neutron diffraction measurement will be applied to evaluate the large bending due to the buckling of Nb_3Sn strands in CICC samples, especially the slippage of the superconducting cable from low field zone to high field zone along the short CICC sample. In this presentation, test results of high field zone in a cut CICC sample using neutron diffraction procedure and the future plan of the neutron diffraction study will be presented and discussed.

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O-18

Internal strain states of high strength Nb_3Sn wires and cables

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For high field superconducting magnet applications, it is necessary to develop the high strength superconducting wires and cables. Rutherford and cable-in-conduit conductors are typical conductors of Nb_3Sn. However, internal strain states of Nb_3Sn phase in the strands and cables are complicated but very important. We evaluated the internal strain of the high strength CuNb/Nb_3Sn strands and twisted cables with stainless steel under tensile stress by neutron diffraction. We found that 3-dimensional strain states of Nb_3Sn in the strands under tensile stress are determined by the elastic-plastic deformation of the constituents of the composite wires, by the detailed analysis of the each strain of the composed materials of the strand. Therefore, the mechanical deformation history such as pre-bending and pre-loading changes the internal strain state, which is closely related to the superconducting properties. In addition, the strain states of the cables, which consist of many strands, are more complicated. We also evaluated internal strain properties of the twisted cable under the tensile stress which consists of 3 CuNb/Nb_3Sn and 4 stainless steel strands. The reduction of the internal strain of the Nb_3Sn under tensile stress due to the reinforcement could be seen directly in the twisted cable. This suggests that a reinforcement effect by the stainless steel is effective for the twisted cable.

I-15

Particle Simulations on GPU supercomputing systems

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Graphical processing units (GPUs) have become a strong candidate for high performance supercomputing this decade. GPUs have evolved from specialized devices for graphics to general purpose computing. One attractive feature of GPUs is that the cost of floating point operations per watt is substantially lower than with traditional processors. Although GPUs appear to be an exotic and specialized architecture, they are constructed from familiar parts: a hierarchical collection of SIMD (vector) processors. As a result, one can develop an abstract view of the hardware and programming languages that support this view (such as OpenCL and CUDA). This abstract machine not only represents GPUs, but also other devices such as Intel multi-core processors. As a result, there is hope in developing portable, scientific applications that will support emerging supercomputing architectures.

At UCLA, we have recently installed a 96 node cluster with 288 NVIDIA GPUs called Dawson2. The system ranked 148 in the most recent compilation of the top500 computers in the world, and has 129,024 GPU processing cores and 1152 cpu cores. In preparation for this machine, we developed algorithms for Particle-in-Cell (PIC) codes which run on a single GPU[1]. This algorithm reorders particles every time step and makes use of a sorting cell that contains a small number of grids. The algorithm has a small number of parameters that allow one to adapt the code to different hardware architectures. We obtained speedups of around 50-60 compared to traditional CPUs. We are now designing algorithms that will allow multiple GPUs to work together using a hybrid MPI-CUDA domain decomposition. Our design goal is to create a PIC code with interchangeable parts that could be adapted to emerging exascale supercomputers.

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Kinetic Simulations of Neoclassical and Anomalous Transport Processes in Helical Systems

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Quantitative predictions of transport fluxes of particles, momentum, and heat in magnetically confined plasmas are a critical issue for the design of fusion reactors. In this paper, recent results from kinetic simulations of neoclassical and anomalous transport processes in helical systems such as the Large Helical Device (LHD) are reported. In contrast to axisymmetric systems such as tokamaks, the radial electric field in helical systems can be determined by the condition of ambipolar neoclassical radial particle fluxes. Using the delta f particle simulation code, FORTEC-3D [1], the radial profiles of the neoclassical transport fluxes and the radial electric field in the LHD are calculated and compared with experimental results. It is observed that the radial electric field profile is well predicted by the ambipolar neoclassical particle flux condition even though the anomalous transport is dominant over the neoclassical one in LHD plasmas. Using the gyrokinetic Vlasov simulation code, GKV-X [2], simulations of the ion temperature gradient (ITG) turbulence are done including precise equilibrium conditions corresponding to the LHD experiments. Comparisons between the profiles of the turbulent ion heat flux obtained from the GKV-X simulations and from the high ion temperature LHD experiment show a good agreement. Besides, the simulations show that generation of the zonal flows becomes more evident when the rotational transform increases or the inward shifted plasma configuration is realized. It is theoretically predicted that, in helical systems, the macroscopic or background radial electric field can enhance generation of zonal flows leading to the further reduction of turbulent transport and the favorable isotope effect of the ion mass [3]. These effects of the background electric field are confirmed by the poloidally global simulations of the ITG turbulence using the extended gyrokinetic Vlasov simulation code [4].

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High Performance Integrated Simulation of Fusion Plasmas

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A high fidelity integrated simulation requires the component codes to be as first-principles as possible. The most fundamental first-principles component codes available today are full-function gyro-kinetic codes for microturbulence, neoclassical, RF-plasma interaction, neutral beam particles, and neutral transport physics; and MHD/fluid codes for fast and large scale instabilities. Such a simulation must be performed on leadership-class high performance computers (HPCs). Due to the large scale data I/Os from large number of processors, the elevated fault tolerance problem, and the heterogeneous computing methods on different grids, compilers, and numerical libraries between codes, there are numerous issues we need to understand and resolve in order to perform integrated simulations on leadership-class HPCs. To solve these problems, we have developed the EFFIS (End-to-end Framework for Fusion Integrated Simulation) in service oriented architecture. EFFIS utilizes a new adaptive I/O method, ADIOS, for simultaneous in-situ coupling through memory-to-memory and files. DataSpace method will allow optimal in-HPC coupling and in-situ data-compression and analysis. For the internal coupling workflow, an extended ADIOS technology with DataSpace is used. For the external coupling workflow, together with the data storage, the provenance information, the external analysis and the visualization, the old Kepler technology is utilized, in cooperation with a fast data movement technology. Web-based eSiMon dashboard is used for remote in-situ data analysis and visualization on personal computers. New integrated scientific discoveries which were achieved using the EFFIS technology on HPCs will be presented, which include the global core-edge full-f gyrokinetic simulation, kinetic RMP penetration, and pedestal-buildup and ELM cycle. Utilization the 10-100 peta flop HPCs, and beyond, will also be discussed.

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Theory of driven magnetic islands

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Electric currents flowing in coils located outside magnetically confined plasma give rise to magnetic islands. Such externally driven islands are of growing importance as a means to control the edge conditions in stellarators and tokamaks alike. They are experimentally observed to affect the profiles of density, temperature, and electric field, as well as to modify the properties of the turbulence in their vicinity. In many respects, these islands are analogous to airfoils in wind tunnels. In particular, two parameters play a central role in the description of their interaction with the plasma: the local drive, and the electromagnetic force acting on the plasma. These parameters are loosely analogous to the lift and the drag acting on an airfoil. The central problem for the theory of driven islands has been to determine the dependence of these two parameters on conditions such as the strength of the external drive and the scale-lengths of the equilibrium fields such as density, pressure, and plasma rotation velocity.

Several regimes can be distinguished according to the size of the island and the collision frequency. The relevant dimensionless measure of the collision frequency is the ratio of the skin depth at the diamagnetic frequency and the ion Larmor radius. In large machines with a hot edge, this parameter is small and the effects of diamagnetic drifts are important. The island is then analogous to a sailboat, with the electrons playing the role of the wind and the ions that of the water. Models for the plasma response that assume quiescent conditions find that in all regimes, the electrons inside the separatrix of the island are trapped or frozen into the island. This circumstance, together with the continuity of the ion velocity, determines the electric field inside the island in terms of the other plasma parameters. The density and temperature for electrons and ions, by contrast, are determined by the competition between transport rates parallel to and across the magnetic field. The presence of turbulence significantly alters the picture, giving rise to island amplification by correlated eddies and a turbulent drag. Gyrokinetic simulations for large islands show that the drag force reverses when the electric field is such that the ions outside the separatrix are almost at rest with respect to the island, in contrast with small islands where the reversal occurs when the outside electrons are at rest. Forcing the plasma to flow past the island gives rise to significant steepening of the density and temperature profiles, an effect that is only observed for much smaller islands in quiescent plasma. Lastly, sufficiently strong flows are found to unlock a magnetic island from the coils that generated it. The island will then heal as it rotates, leaving in its place a much smaller, “suppressed” island.

Cross-cutting Ideas for a Fusion DEMO Plant with Current and Generation IV Nuclear Power Plants

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Ever since the beginning of the ancient civilization, human beings have obtained energy from natural resources to enhance their welfare and sustain economic growth. To successfully realize fusion energy that is to be the first energy not from natural resources but from technology, there still are technology barriers to break through and the risks associated with investing to the technologies that have not been fully developed. The easiest way of mitigating the risks is to be postponement of developing fusion energy until the technologies will be matured. However, as the economic benefits that an early mover of fusion energy is to take are so huge [1], it is worthwhile taking due risks associated with developing it [2].

A practical way of mitigating the risks is to be cross-cutting of fusion DEMO development making use of the commonalities between a fusion DEMO systems and existing systems [3]. Among the existing systems, current and generation IV nuclear power plants (NPPs) could have many areas of commonalities that could be utilized for the of development of fusion DEMO systems. The cross-cutting ideas for the development of fusion DEMO systems, based on these commonalities, may include the following areas:

- Regulatory requirements and licensing processes;
- Codes and standards;
- Safety analysis methods;
- Design methods;
- Computational codes and Thermo-hydraulic test facilities;
- Material development and irradiation test facilities;
- Radioactive waste management and environmental protection;
- Program lifecycle management skills.

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Research platforms of laser and ion beam induced ablation plasmas as interdisciplinary reactor studies

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The extreme states of density ranging from vacuum to solid density, temperature eV or above are created in both magnetic and laser fusion reactors due to disruption, ELM, explosions of fusion reactions etc. These states can be studied using laser and ion beam induced ablation schemes as experimental platforms. Considering this laser irradiation as a heat pulse source, the ablated plasma plumes can be studied in various way.
Our recent study focuses on plasma shielding for material under extreme conditions. In orthogonally oriented two targets are placed in a vacuum chamber where two laser pulses can irradiate. The pulse laser creates two plasma plumes crossing each other at the densities 10^{11} to 10^{13}/c.c. at temperature at around eV [1, 2]. In case of Carbon targets are used, the crossing points show stagnation up to microsecond, considerably longer than the laser pulse width 6 nsec. Further processes include the molecular formation leading to the formation of Carbon Fullerene and Carbon nano tubes. These collision processes is observed to absorb the kinetic energy of incoming plasma plumes up to 50 % at the laser energy density 10 J/cm^2. This may be considered as a energy absorption mechanism of incoming plasma flux when another plasma plume is prepared in advance in front of the material to be protected.
Magnetized Coaxial Plasma Gun has been upgraded at the University of Hyogo. The plan is to reach 2MJ/m^2 ion beam irradiation energy density that is expected in the Type-1 ELM at ITER. The upgrade included two categories: (1) Increase of capacitance from 1mF to 2.9 mF and (2) Installation of tapered drift tube to increase the ion beam flux. The test operation has been conducted to confirm the designed output energy density.

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Particle and heat control for steady state burning plasma in helical reactor

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Particle and heat control is essential for steady state burning plasma. The control scenario in helical reactor has been studied on the basis of knowledge obtained by the Large Helical Device (LHD) experiment. In this presentation, the scenario in the helical reactor, FFHR-d1, which is under conceptual design, will be shown.
Fueling has to be well done, and helium ash and unreacted deuterium and tritium have to be pumped to sustain steady state burning plasma. In FFHR-d1, fuelling will be conducted mainly with ice-pellet injection. The fuelling efficiency is estimated to be 0.18%, and necessary injection amount of DT pair is $6x10^{23}$ DT/s for 3 GW fusion-power operation in which about $1.1x10^{21}$ reactions occur during 1s. The helical divertor which is intrinsically equipped in heliotron-type magnetic configuration will be utilized for the compressing and pumping of residual fueled particles and helium ash. The required pumping flux will be the order of 1000 $Pa \cdot m^3/s$.
Heat load to plasma facing components has to be less than that which causes serious damages of the components especially divertor plates. In this presentation, the heat control scenario will be discussed mainly from the point of view of plasma operation. One of the characteristics of the helical divertor is non-uniform profiles of the particle and the heat flux on the divertor both in the toroidal and poloidal directions. Studies of the particle and the heat loads on the helical divertor have been conducted in LHD, and the understandings obtained from them can be applied to the estimation of the divertor heat load in FFHR-d1. To reduce the divertor heat load, impurity gas-puffing is planned. It can enhance radiation in the scrape-off layer, and reduce plasma temperature.