Conference Summary: Experiments in confinement & Plasma-wall Interaction and Innovative Confinement Concept

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Abstract: This paper summarizes the results presented at the 20th IAEA Fusion Energy Conference 2004 in the sessions of confinement, plasma-wall interaction and innovative confinement concept. The highlights of the presentations are as follows. Long pulse operation with high beta and high bootstrap fraction much longer than the current diffusion time has been achieved. The discharge scenario optimization and its extrapolation towards ITER have progressed remarkably. Significant progress has been made in understanding of global confinement and transport physics.

1. Introduction

In the 20th IAEA Fusion Energy Conference, 91 papers on confinement (EX-C), 22 papers on plasma-wall interaction (EX-D) and 21 papers on innovative confinement concept (IC) were presented. This summary report highlights the new progress from the last IAEA Conference. The topics covered in the following sections are: 2) Tokamak Regimes Extended towards ITER, 3) Scenario Optimization, 4) Global Confinement Physics, 5) Transport Physics, 6) Plasma-wall Interaction and 7) Innovative Confinement Concepts. The section 8 summarizes conclusions.

2. Tokamak Regimes Extended towards ITER

One of the highlights of this conference is the tokamak regime extended successfully towards ITER. In particular, significant progress has been made in long pulse operation. There are four characteristic times in plasma, i.e., the energy confinement time ($\tau_E$), the particle replacement time ($\tau_p^*$), the current diffusion time ($\tau_R$) and the wall saturation time ($\tau_W$). Demonstration of plasma sustainment much longer than these characteristics times is very important to confirm that the good plasma performance can be maintained, and the sustainment of plasma much longer than $\tau_E$ and $\tau_p^*$ has been obtained. In this conference, it was reported that high values of normalized beta ($\beta_N$) and the fusion gain parameter ($G = H_{89p} \beta_N / q_{95}^2$) are maintained much longer than $\tau_R$ in JT-60U and DIII-D. Here, $G \sim 0.4$ corresponds to the ITER standard ELMy H-mode scenario with $Q = 10$ and $G \sim 0.3$ corresponds to the ITER steady state scenario with $Q = 5$.

Figure 1 shows the JT-60U result, where $\beta_N = 2.3$ and $G > 0.4$ were maintained for 22.3 s by optimizing high $\beta_p$ ELMy H-mode plasma [1, 2]. This duration corresponds to $13\tau_R$. The time evolution of $\beta_N$ was carefully optimized in order to reach $\beta_N$ as high as possible but avoiding the neo-classical tearing mode (NTM). DIII-D has demonstrated a discharge corresponding to the ITER baseline scenario as shown in Fig. 2 [3, 4]. The fusion gain parameter of 0.58 and $<\beta>$ of 4% was sustained for 9.5 s, which is $9.2\tau_R$. In this discharge, the normalized pressure $\beta_N = 2.6$.
was also maintained by feedback control of the neutral beam injection (NBI) power. In JET, the reversed shear plasma with \( H_{89p} = 2 \), \( \beta_N \sim 1.66 \) and a bootstrap fraction \( (f_{BS}) \) of about 33\% was maintained for 20 s (Fig. 3) [5].

Another important issue in long pulse operation is the extension of discharge duration and increase in input energy, which enables us to study particle recycling and high heat handling. Figure 4 shows the experiments of JET, JT-60U, Tore Supra, LHD, HT-7 and TRIAM-1M. More than 320 MJ has been injected to the diverted plasmas of JET and JT-60U without any problem. Wall saturation was observed in 30 s heating plasmas of JT-60U. The discharge duration has been extended to up to 6 min and energy of 1.07 GJ was injected in Tore Supra [6]. The discharge duration has been extended to 115 s in LHD [7], 4 minutes in HT-7 [8] and 5 hours 16 minutes in TRIAM-1M [9]. Further increase in discharge duration and injected energy in diverted plasmas is important.

3. Scenario Optimization

The discharge scenario optimization and its extrapolation towards ITER have progressed remarkably. As for the ITER baseline scenario, long sustainment of \( G \) has been demonstrated in DIII-D and an integrated exhaust scenario has been demonstrated in ASDEX Upgrade and JET. As for the steady-state and hybrid operation scenario, the sustainment of full current drive with high \( f_{BS} \) was reported in JT-60U, DIII-D and JET. Long sustainment of the weak magnetic shear plasmas and high integrated performance were reported in ASDEX Upgrade, DIII-D, JET and JT-60U. In addition, the ranges of density and radiation power...
have been extended successfully in DIII-D, JET and JT-60U. In many confinement regimes and magnetic configurations, the improved confinement regimes have been extended either in their performance or their duration.

3.1.1 ITER baseline scenario

DIII-D has demonstrated $G$ which meets or exceeds the value required in ITER: $G = 0.42$ and $t_{\text{dur}}/\tau_R > 2$ under stationary conditions as shown in Fig. 5 [4]. This result provides a basis for confidence that the core plasma in the ITER can meet the performance goals of the project. As for the particle control, ASDEX-Upgrade demonstrated the integrated exhaust scenario, including the divertor temperature control by Ar seeding and ELM control with pellet injection (Fig. 6) [10]. JET obtained $HH_{98y2} \sim 1$ with $n_e/n_G$ from 1 to 1.1, corresponding to an absolute density $\gtrsim 9 \times 10^{19}$ m$^{-3}$ in ELMy H-mode (Fig. 7) [11]. They also demonstrated high performance and high density plasmas using impurity seeding [12]. We can conclude from these results that the feasibility of ITER reaching the target performance has been confirmed.

![Fig. 5. Normalized fusion performance vs. duration normalized to current diffusion time (DIII-D). Filled squares are ITER baseline scenario discharges, filled circles are hybrid scenario discharges, open circles are other types.](image)

![Fig. 6. Demonstration of an integrated exhaust scenario in ASDEX Upgrade: a) heating (4.5 MW NBI, 4 MW ICRH) and main chamber radiated power, b) gas valve fluxes, c) and d) target and measured values of divertor temperature and neutral gas density e) $f_N$ and line averaged $Z_{\text{eff}}$, f) tungsten concentration in the outer central plasma, g) Greenwald fraction and H-factor, f) peak power density in the outer divertor.](image)

3.1.2 Steady state/hybrid scenario

JT-60U, DIII-D and JET have demonstrated long sustainment of the nearly fully non-inductively driven plasma with a high $f_{\text{BS}}$. Figure 8 shows the result of JT-60U [13]. In a weak shear plasma regime, nearly fully non-inductively driven plasma with $f_{\text{BS}} \sim 50\%$ was sustained for $\sim 2.3 \, \text{s} \, (\sim 3\tau_R)$ in stationary condition. The confinement enhancement factor ($HH_{98y2}$) was $\sim 1.0$ under the condition of $T_e \sim T_i$. In reversed shear plasma regime, plasma with $f_{\text{BS}} \sim 75\%$
and HH_{98y2} \sim 1.7 has been sustained for 7.4 s under nearly full non-inductive current drive condition. In DIII-D, nearly fully non-inductively driven plasmas have been sustained for 1 s (\sim \tau_p) with good current drive alignment guided by integrated modeling, where $\beta \sim 3.6\%, \beta_N \sim 3.4$ and H_89P \sim 2.3 [14, 15].

As for the more advanced operations, the results of weak shear plasmas were reported from JET, ASDEX Upgrade, DIII-D and JT-60U. The weak magnetic shear regime is also called as high-$\beta_p$, hybrid or advanced hybrid. This regime is characterized with a flat or nearly flat safety factor inside $r/a = 0.5$ and $q(0) = 1 - 1.5$. Figure 9 shows identity experiments performed between JET and ASDEX Upgrade to confirm the existence of the hybrid regime closer to ITER parameters (as low as $\rho^* = 3.3 \times 10^{-3}$ or $2\rho^*_{\text{ITER}}$ and $v^*_{\text{e}} \sim 0.06$ or $3 v^*_{\text{e,ITER}}$) [16]. DIII-D, ASDEX Upgrade and other devices demonstrated simultaneous achievement of high values of $G$ and $f_{\text{BS}}$ as shown in Fig. 10 [3, 17]. JT-60U has demonstrated the integrated performance suitable for ITER under almost fully non-inductive condition as shown in Fig. 11 [1, 13]. The results of these weak shear operations are approaching the ITER plasma regimes in terms of $\rho^*$ and $v^*$.

The density and radiation fraction were also increased in DIII-D, JET and JT-60U with keeping favorable confinement properties. DIII-D showed favorable values of $G$ exceeding the ITER baseline up to 70% of the Greenwald density (Fig.12) [3, 4]. JET demonstrated high core temperature inside ITB in
reversed shear discharges (Fig.13) [18]. In JT-60U, with impurity seeding, the density exceeded $n_G$ and radiation reached 90 to 100% in both the reversed shear and weak shear regimes (Fig. 14) [19]. Integration of core and divertor plasmas is a remaining issue.

### 3.1.3 Extension of Improved Regimes

The H-mode confinement in small and no ELM regimes has been improved in ASDEX Upgrade, Alcator C-Mod [20], DIII-D, JET, JFT-2M [21] and JT-60U. Improved regimes with pellet injection and high $\ell_i$ were also reported in FTU [22] and HT-7 [8].

The parameter space of good confinement regime was also expanded in spherical tokamaks. In MAST, Co-NBI/Counter-NBI ITB comparison discharges have been examined and the increase in $H_{98y2}$ with toroidal rotation frequency was obtained [23, 24]. In NSTX, progress of high non-inductive fraction and high $\beta_T$ was obtained as shown in Fig.15 [25]. A strong $B_T$ dependence of confinement was also observed in contrast to conventional aspect ratio tokamaks and $H_{98y2}$ up to 1.4 was obtained at the highest $B_T$.

Good energy and particle confinements with H-mode were also reported from helical devices (CHS [26], Heliotron-J [27] and TJ-II [28]). The parameter space of ITB regimes has also been expanded.

Regarding mirror machines, HANBIT obtained a stable high-density mode [29]. GOL-3 has increased its temperature twice due to recent conversion of the facility into complete...
multi-mirror magnetic system [30]. GAMMA-10 increased ion-confining potential up to 2.1 kV and verified their proposed physics scaling of potential formation [31].

4. Global Confinement Physics

The studies of global confinement, L/H transition and internal transport barrier (ITB) have been continued in tokamaks, stellarators and reversed field pinches to investigate its physics and contribute to the design of next generation devices.

4.1 Scaling studies of global confinement

In tokamaks, dependence of energy confinement on dimensionless parameters is still an important subject and is studied in dedicated experiments. JET and DIII-D reported a $\beta$ scan with fixed $\rho^*$ and $\nu^*$ in ELMy H-mode [32]. The result shows that the energy confinement does not degrade with increasing $\beta$, in contrast to the multi-machine confinement scaling IPB98(y,2). This favorable $\beta$ dependence would predict improved confinement for high $\beta$ operation in ITER as shown in Fig. 16, while the confirmation by multi-machine experiments is required. This also may be related to improved confinement observed in so-called “hybrid scenario” plasmas.

In stellarators, previous scaling law (ISS95) has been revised by incorporating new data from LHD, TJ-II, Heliotron J and so on [33]. As a result, a new gyro-Bhom scaling has been extracted (Fig. 17). The systematic offset depending on magnetic configuration has been found and is ascribed to effective helical ripple. The scaling law for RFP plasmas was also reported [34].

4.2 L-H transition

Although the understanding of the L-H transition has been progressed, the quantitative prediction of the L-H transition still has some uncertainties. The effect of small variation in magnetic configuration near the double-null on the threshold power has been investigated in Alcator C-Mod, NSTX, MAST and ASDEX Upgrade. In Alcator C-Mod, low threshold power was found in lower-single-null configuration with grad-B drift towards the X-point, where the counter toroidal rotation becomes small as shown in Fig. 18 [20]. The inter-machine
comparisons in NSTX, MAST and ASDEX Upgrade with respect to the influence of the magnetic topology on the power threshold have been made and revealed a reduction of the power threshold in true double null (C-DN) configuration as shown in Fig. 19, where a larger electric field is generated [35,36]. The other major topic is the biasing technique to trigger an H-mode (TCABR [37], ISTTOK [38], and Tohoku University Heliac [39]). In particular, Tohoku University Heliac shows a clear bifurcation phenomenon, where negative resistance features were observed between the electrode current and the bias voltage (Fig. 20). The ion viscous damping force is estimated from the driving force for the $J \times B$ poloidal rotation, and the effect of the local maxima on the ion viscosity is investigated. In addition, NSTX shows that HFS gas puffing reduces $P_{LH}$ (less momentum drag of HFS neutral) and Heliotron J reported H-mode with edge iota windows.

4.3 ITB

The understanding of ITB is still a major issue and many results were reported in this conference. The ITB has proven to be a common phenomenon across many machines, including small and large tokamaks and stellarators, which suggests that many devices can contribute to the progress of understanding of ITB physics. In addition, electron ITB and ITB with no or small momentum input has recently drawn attention due to their relevance to ITER.

The results reported in the conference are summarized in Fig. 21. In spherical tokamaks, the thermal transport is usually dominated by electron channel so that the improvement of electron confinement or formation of electron ITB is crucial for confinement improvement. Electron ITB was formed in MAST [36] and NSTX [40], with a large toroidal rotation in MAST.
and with a negative shear scenario in NSTX. The profile of density inside the ITB is strongly peaked while those of electron and ion temperatures are flat as shown in Fig. 22, which is a very interesting phenomenon to investigate. The density range of electron ITB has been extended in FTU with combined heating of LHCD and ECRH [22]. In TCV tokamak [41] and TJ-II stellarator [42], the importance of magnetic shear and rational surface was demonstrated. In JET, ion ITB was obtained with dominant RF heating where momentum input and ExB shear are small [43].

5. Transport Physics

In this conference, the efforts to understand the physics in toroidal plasma common across the magnetic configurations have been made and many interesting results have been obtained. The highlights include zonal flow, electron transport, particle transport, momentum transport and radial electric field. Many results of turbulence analysis and its suppression were reported from many devices but they are too numerous to be included in this summary.

5.1 Zonal flow

Recent progress of diagnostics enabled us to investigate electrostatic and magnetic turbulence, especially zonal flow and Reynolds stress. Zonal flows are known to be driven by electrostatic and specifically in high beta plasma by electromagnetic Reynolds stress, and also found to be affected by so called Geodesic-Acoustic-Mode (GAM) oscillation. Extrap-T2R and HT-7 have succeeded in measuring the radial profiles of electrostatic and magnetic Reynolds stress using arrays of probes (Fig. 23) [44, 45]. The results show that the radial gradient of

Fig. 22. \( n_e, T_e \) and \( T_i \) profiles (top), rotation profile (Mach number), ExB shearing rate, \( \omega_{SE} \) and ITG growth rate, \( \gamma_m \) (bottom) for a counter-NBI internal transport barrier in MAST.

Fig. 23. Reynolds stress observed in Extrap-T2R and HT-7.
electrostatic Reynolds stress changes sign across the last closed flux surface (LCFS).

In JFT-2M and T-10, GAM was observed and an impressive measurement of low frequency zonal flow was reported from CHS \[46, 47, 48\]. JFT-2M shows the relation between the amplitude of potential fluctuation and that of density fluctuation due to GAM. The envelope of ambient density fluctuation and the potential fluctuation have a significant coherence at the GAM frequency as shown in Fig. 24. Intermittent particle flux just inside the separatrix is also correlated with GAM. CHS provided two HIBPs located approximately 90 degrees apart in the toroidal direction and observed both toroidal and poloidal mode numbers of fluctuations. As a result, they succeeded in observing two branches of zonal flow, i.e., the low frequency zonal flow and GAM as shown in Fig. 25. This is the first identification of the low frequency zonal flow in a toroidal plasma. The experimental observation of zonal flows has confirmed the theoretical prediction.

5.2 Electron transport

The mechanism of electron transport is yet an open question. The existence of critical gradient $T_e$ is related to the temperature profile stiffness. JET and JT-60U show the existence of critical $VT_e$ and non-linearity of $\chi_e \sim (VT_e)^nT_e^{\prime\prime}$ as shown in Fig. 26 \[49, 50\], while DIII-D shows no critical $VT_e$ nor non-linearity \[51\]. LHD also shows no critical $VT_e$ and $\chi_e \sim T_e^{\prime\prime}$ as shown in Fig. 27 \[50\]. The similarity of electron transport is observed in JT-60U reversed shear plasmas and LHD (reversed shear), suggesting possibility of electron transport physics common across toroidal plasma. Effect of plasma shape and shear on $\chi_e$ was investigated in TCV \[52\].
5.3 Particle transport

In Trace Tritium experiments in JET thermal tritium transport properties have been investigated [53]. Thermal tritium particle transport coefficients \( (D_T, v_T) \) were found to exceed neo-classical values in all regimes except in ELMy H-modes at high density, and in the region of ITB in reversed shear plasmas (Fig. 28). In ELMy H-mode dimensionless parameter scans, T particle transport scaled in a Gyro-Bohm manner in the inner plasma \( (r/a<0.4) \), whilst the outer plasma particle transport behaved more like Bohm scaling. Dimensionless parameter scans showed contrasting behaviors for particle confinement (increases with collisionality \( \nu^\ast \) and \( \beta \)) and energy confinement (decreases with \( \nu^\ast \) and independent of \( \beta \)). Comparing regimes (ELMy H-mode, ITB plasma, and Hybrid scenarios) outside the central plasma region \( (0.65 < r/a < 0.85) \), normalized tritium diffusion \( (D_T/\psi_p) \) scaled with normalized poloidal Larmor radius \( (\rho_0^\ast = (R/a)q_0^\ast) \) in a manner close to Gyro-Bohm \( (\sim \rho_0^3) \), with an added inverse \( \beta \) dependence.

Electron density profile strongly affects fusion power of ITER, so that the study of anomalous particle transport is important. Impressive results have been obtained in Tore Supra, FTU, ASDEX Upgrade and JET. The existence of anomalous particle pinch has been unambiguously proven in fully non-inductive discharges of Tore Supra [54] and FTU [55]. Figure 29 shows the dependence of \( \nabla n_e/n_e \) upon \( \nabla q/q \), where \( \nabla n_e/n_e \) increases linearly as a function of \( \nabla q/q \) with a positive slope, indicating the inward pinch. The pinch driven by \( \nabla T_e/T_e \) is outward, but its effect on the density profile is weak. On the other hand, the density profile is observed to peak with decreasing collisionality in H-modes of ASDEX Upgrade [56] and JET [57] as shown in Fig 30. These two results could lead to higher
fusion power in ITER, though confirmation to ITER requires further experiments.

5.4 Momentum transport

Rotation is important for transport and stability. In this conference rotation without torque input was reported from Alcator C-Mod, DIII-D, TEXTOR and Tore Supra. In Alcator C-Mod, strong rotation produced in the absence of an external torque was observed and the rotation velocity changes with divertor configuration as shown in Fig. 31 [58]. Both core and SOL values show similar, extreme sensitivity to this parameter, suggesting a common origin. In DIII-D, counter-rotation is observed with ECH [59]. Figure 32 shows toroidal rotation profiles, which are hollow, with co-rotation in the outer region and reduced, or counter-rotation in the core. This is in contrast to relatively flat profiles in ohmically heated H-modes but its driving mechanism is not clear yet. TEXTOR has demonstrated rotation control by 3/1 Dynamic Ergodic Divertor [60]. Further understanding in this area will contribute to the projection of the performance of a future reactor, where little torque input is expected.
5.5 Radial electric field

Various studies of radial electric field (E_r) are made in different magnetic configurations such as ISTTOK, TJ-II, LHD, HSX and GAMMA-10.

In ISTTOK [38], biasing experiments have been performed with both a moving limiter and an emissive electrode. Large currents (>15 A) can be drawn at negative applied voltage by both localized limiter and emissive electrode bias, leading to significant modifications in the edge plasma potential profile and to an improvement in particle confinement. Furthermore, its use leads to the formation of stronger radial electric fields and consequently to a much larger improvement in particle confinement as shown in Fig. 33.

In TJ-II stellarator, the influence of plasma density and edge gradients in the development of perpendicular sheared flows and radial electric fields, has been investigated in the plasma edge region [61]. The results provide the first direct experimental evidence of coupling between sheared flow development and increase in the level of edge turbulence. In general, the radial electric field in stellarators can be controlled by changing the collisionality, and positive or negative electric field have been obtained by decreasing or increasing the electron density. However, a new method is required to produce stronger radial electric field shear. In LHD, the radial electric field shear was demonstrated to be controlled by the modification of the helical ripple associated with the shift of the magnetic axis [62]. Measurements of plasma flow damping have been conducted in the HSX stellarator [63]. A fast-switching biased electrode system is used to impulsively generate plasma flow. Two time-scales in the flow evolution were observed for both the bias turn-on and turn-off and the results are compared to neoclassical modeling.

GAMMA-10 shows a mirror machine can also contribute in this field. High ion-confining potential up to 2.1 kV shown in sub-section 3.3 produces a strong E_r shear, suppressing vortex-like turbulent fluctuations and resulting in radial-confine ment improvement as shown in Fig. 34 [31].

Fig. 33. Profile of drift wave, turbulence, vorticity and Er shear in both weak and strong Er shear cases in GAMMA-10.

Fig. 34. Time evolution of bias current, density/H\_e and E\_r in ISTTOK.
6. Plasma-wall Interaction

The following investigations are progressed: active control of edge plasma, recycling/wall retention corresponding to extension of discharge duration, compatibility of tungsten wall with plasma and carbon migration/T retention.

6.1 Active control of edge plasma

Control of edge plasma is important for core plasma performance. Several ideas for active control of the edge plasma were tried. In LHD, a local island divertor (LID) has been installed [64]. The LID is a kind of pumped limiter inserted inside the n/m=1/1 magnetic island produced at the plasma edge. Since the LID head was inserted to the middle of the O-point of the magnetic island, the core plasma did not touch the limiter and the outward heat and particle fluxes did not directly go to the front of the limiter but flowed to the backside of the LID limiter along the field lines across the separatrix. High confinement was obtained by LID, where sharp edge with large $T_e$ gradient was observed (Fig. 35). In TEXTOR, Dynamic Ergodic Divertor was tried. Effect of DED was shown by the onset of 2/1 and 3/1 tearing modes [60]. At the onset of 2/1 tearing mode, reduction of the edge poloidal velocity was observed (Fig. 36).

6.2 Recycling/wall retention

Wall pumping is not expected in long pulse discharges. Variation of wall retention with long time scale should be studied. TRIAM-1M experiments with metallic wall and limiter have demonstrated 5 hour operation without wall saturation (Fig. 37) [65]. The co-deposition of hydrogen with the metal (i.e. Mo) plays a substantial role in the wall pumping as well as the carbon. Tore Supra experiments have also demonstrated no wall saturation during 6 min (Fig. 38) [66]. On the other hand, long pulse experiments in JT-60U ELMy H-mode discharge have shown wall saturation [67], as shown in Fig. 39. The net number of particles absorbed by the first walls/divertor plates gradually decreases, and subsequently, it becomes zero. Under the condition of the wall saturation, the plasma density continues to increase. Retention areas wider than the area directly interacted with plasma were observed in many devices. Further study is necessary to understand the mechanism of the wall inventory.
6.3 Tungsten

Tungsten wall is one of the candidates of first wall in a fusion reactor. In ASDEX Upgrade, 65% of PFC is coated with tungsten as shown in Fig. 40 [68]. The only two major components that are not yet coated with tungsten are the strike-point region of the lower divertor and the limiters at the low field side. It was shown that tungsten wall is compatible with high performance discharge. In addition, they show that tungsten concentration is controllable with central electron heating and pellet triggering of ELMs. On the other hand, experiments in Nagoya University show blisters and bubbles are formed on the surface of tungsten irradiated with low energy (~100 eV) H beam (Fig. 41) [69]. Further studies, especially with high power heating and long pulse, are necessary to confirm the feasibility of using tungsten.

Fig. 37. Time evolution of wall inventory in TRIAM-1M.

Fig. 38. Wall retention rate calculated from particle balance during 3 consecutive long discharges in Tore Supra.

Fig. 39. Time evolution of number of injected particles, pumped particles and retained in the first walls/divertor plates in JT-60U long pulse ELMy H-mode.

Fig. 40. (a) Cross section of ASDEX Upgrade and location of W. (b) Time traces for the USN discharge #19424 (I_p = 800 kA, B_t = 20 T) with a continuous rise of the auxiliary heating power. The discharge is terminated by a safety interlock to prevent from overloading the upper W target-plates. (c) Ratio of central and edge tungsten concentration vs. density peaking for improved H-mode discharges.
6.4 Carbon migration and T retention

Carbon migration is important from the viewpoint of T retention, which is a serious concern in ITER. The results of C migration and D and T retention were reported from JET [70], ASDEX Upgrade [71], DIII-D [3], JT-60U [72] and Tore Supra [73]. Carbon 13 injection experiments were performed in JET and DIII-D, indicating C migration towards the inner divertor target probably due to the SOL flow (Fig. 42). Carbon migration on the divertor plate was also studied in ASDEX Upgrade and JT-60U using SIMS, SEM, TDS, etc. These results show that the inner divertor is mostly deposition-dominated, while the outer divertor is erosion-dominated (Fig.3). Table 1 shows the summary of H, D and T retention in these devices. The fuel retention fraction was 10 % in JET with horizontal target (MKIIA) and 3 % with vertical target (MKIIGB), which suggests the importance of target geometry. D/C ratio is 0.4-1.0 in JET and ASDEX Upgrade, while it is lower than 3 % in JT-60U. The dust is 1 kg in JET and 7 g in JT-60U. The difference could be ascribable to better alignment of targets (no protruding edges) and higher temperature in JT-60U.

Table 1  Summary of D/C and T retention

<table>
<thead>
<tr>
<th>Device</th>
<th>JET</th>
<th>ASDEX Upgrade</th>
<th>JT-60U</th>
<th>DIII-D</th>
<th>TEXTOR</th>
</tr>
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<tbody>
<tr>
<td>D /C, (H+D)/C</td>
<td>0.4-1.0</td>
<td>0.4-1.0</td>
<td>&lt;0.02-0.04</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel retention fraction (%)</td>
<td>10.5 (MKIIA)</td>
<td>3 (MKIIGB)</td>
<td>3</td>
<td>3</td>
<td>8</td>
</tr>
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</table>

7. Innovative Confinement Concept

In this session, experiments, numerical studies and new ideas were reported. Because the covered areas are very wide, just a few examples are illustrated here.

As for the experiments, Min-RT levitated super-conducting internal ring in ECH plasma and investigated potential control and flow generation [74]. Axial flow shear was measured in
Zap flow Z-pinches [75]. Current drive by Helicity injection has been investigated in HIT-II & HIT-SI and its results are reflected to NSTX [76]. The following results are also reported: FRC plasmas produced and sustained by the rotating magnetic field (FRX-L) [77], supersonic rotation with centrifugal confinement (MCX) [78], etc.

As for the numerical studies, the following results are reported: nonlinear evolution of MHD instability in FRC [79], design of magnetic measurement for 3D equilibrium and model of ambipolar plasma flow for NCSX [80, 81], optimization of quasi-poloidal stellarator [82], etc. New concepts include a burning spherical tokamak by pulsed high-power heating of magnetic reconnection [83], selective heating with LH for He ash removal [84], etc.

8. Conclusion

In this conference, a significant progress has been reported in the following areas compared with the 19th IAEA Fusion Energy Conference:

- Long pulse operation with high beta and high bootstrap fraction much longer than the current diffusion time.
- Extension of parameter regimes for ITER baseline scenario and steady state/hybrid scenarios and their sustainment
- Understanding of global confinement and transport physics, especially, non-dimensional scaling, zonal flow and particle transport
- Material and retention/migration studies

The discharge scenario optimization and its extrapolation towards ITER have progressed remarkably, by which the feasibility of ITER reaching the baseline target performance has been confirmed. Results of $\beta$ scan and particle transport could lead to higher fusion power in ITER. The efforts to understand the physics in toroidal plasma common across the magnetic configurations have been made and many interesting results have been obtained. These results will greatly contribute to ITER and the ultimate goal of fusion power.
References

[22] Gormezano C, “Overview of the FTU Results”, Paper OV/4-6, This conference.
[32] McDonald D.C, et al., “Particle and Energy Transport in Dedicated $\rho^*$, $\beta$ and $v^*$ Scans in JET ELMy H-modes”, Paper EX/6-6, This conference.
[34] Sakakita H, et al., “Characteristics of the TPE Reversed-Field Pinch Plasmas in Conventional and Improved Confinement Regimes”, Paper EX/P2-17, This conference.


[56] Peeters A.G, et al., “Understanding of the density profile shape, electron heat transport and internal transport barriers observed in ASDEX Upgrade”, Paper EX/P3-10, This conference.


[58] Rice J.E, et al., “The Dependence of Core Rotation on Magnetic Configuration and the Relation to the H-mode Power Threshold in Alcator C-Mod Plasmas with No Momentum Input”, EX/6-4Ra, This conference.


[62] Ida K, et al., “Control of the radial electric field shear by modification of the magnetic field configuration in LHD”, Paper EX/P6-6, This conference.


[70] Philipps V, et al., “Overview of recent work on material erosion, migration and long-term fuel retention in the EU-fusion programme and conclusions for ITER”, Paper EX/10-1, This conference.