Conference Report

Summary of the 27th IAEA Fusion Energy Conference in the categories of EX/W, EX/D, and ICC

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Abstract

This is a summary paper of the 27th IAEA Fusion Energy Conference which was held from 22–27 October 2018 in Gandhinagar, India. The results in the categories of EX/W (wave–plasma interactions, current drive, heating, energetic particles), EX/D (plasma–material interactions, divertors, limiters, scrape-off layer (SOL)), and ICC (innovative confinement concepts) in magnetic confinement experiments are summarized. In total, 121 papers have been contributed to these categories. Interesting results on the coupling between energetic particles, magnetohydrodynamics (MHD), wave–particle interaction, turbulence, SOL physics, and divertors are presented at this conference. For example, control of energetic particle driven MHD by electron cyclotron heating and resonance magnetic perturbation, mitigation of disruption by energetic particle driven MHD, control of decay length by SOL turbulence, and wave scattering by SOL density fluctuations were discussed in this conference. Deeper understanding of these couplings will be essential for the sustainment of high performance steady-state plasma in ITER.

Keywords: IAEA Fusion Energy Conference, current drive, heating, energetic particles, divertors, scrape-off layer, innovative confinement concepts

(Some figures may appear in colour only in the online journal)

1. Introduction

The 27th International Atomic Energy Agency (IAEA) Fusion Energy Conference (FEC) was held from 22–27 October 2018, in Gandhinagar, India. This conference aims to provide a forum for the discussion of key physics and technology issues as well as innovative concepts of direct relevance to the use of nuclear fusion as a source of energy. The scientific scope of the FEC 2018 is, therefore, intended to reflect the priorities of this new era in fusion energy research. The conference aims to serve as a platform for sharing the results of research and development efforts in both national and international fusion experiments that have been shaped by these new priorities, and to thereby help in pinpointing worldwide advances in fusion theory, experiments, technology, engineering, safety and socio-economics. Furthermore, the conference will also set these results against the backdrop of the requirements for a net energy producing fusion device and a fusion power plant in general, and will thus help in defining the way forward.

The FEC 2018 consists of four keynote presentations, 24 overview talks, 78 regular talks, 17 rapporteur papers, 11 overview poster presentations, 518 regular poster presentations, and five summary talks over six days. The total number of participants was close to 1000. The topics of this conference are (1) magnetic confinement experiments (EX), (2)...
magnetic confinement theory and modelling (TH), (3) plasma overall performance and control) (PPC), (4) inertial fusion experiments and theory (IFC), (5) innovative confinement concepts (ICC), (6) fusion engineering, integration and power plant design (FIP), (7) fusion nuclear physics and technology (FNS), (8) materials physics and technology (MPT), and (9) safety, environmental and economic aspects of fusion (SEE). There are four subtopics in the topics of magnetic confinement experiments (EX) and magnetic confinement theory and modelling (TH), these are (1) confinement (C), (2) stability (S), (3) wave–plasma interactions, current drive, heating, energetic particles (W), and (4) plasma–material interactions, divertors, limiters, scrape-off layer (SOL) (D). The results presented on these topics in this conference were summarized on the last day of the conference by the following five speakers: Dr David Campbell (topics EX/C, EX/S & PPC), Dr Katsumi Ida (topics EX/D, EX/W & ICC), Professor Ricardo Galvao (topic TH), Dr Radha Bahukutumbi (topic IFC), and Dr Takashi Inoue (topics FIP, FNS, MPT & SEE).

In this paper, results in the categories of EX/W (wave–plasma interactions, current drive, heating, energetic particles), EX/D (plasma–material interactions, divertors, limiters, SOL), and ICC in magnetic confinement experiments are summarized. In total, 121 papers have been contributed to these categories. Interesting results on the coupling between energetic particle, MHD, wave–particle interaction, turbulence, SOL physics, and divertors are presented at this conference. For example, control of energetic particle driven MHD by electron cyclotron heating and resonance magnetic perturbation, mitigation of disruption by energetic particle driven MHD, control of decay length by SOL turbulence, and wave scattering by SOL density fluctuations are discussed in this conference. Deeper understanding of these couplings would be essential for the sustainment of high performance steady-state plasma in ITER.

2. Wave–plasma interactions, current drive, heating, energetic particles

2.1. Energetic particle (EP) driven instability

Figure 1 shows a diagram of physics elements in the field of wave–plasma interactions, current drive, heating and EPs. The electron cyclotron heating (ECH) and resonant
magnetic perturbation (RMP) field have a significant impact on the energetic ion driven instability. The EP-driven geodesic acoustic mode (EGAM), beta-induced Alfvén eigenmode (BAE), toroidal Alfvén eigenmode (TAE), compressional Alfvén eigenmode, energetic particle mode (EPM), EP-driven resistive InterChange mode (EIC), global Alfvén eigenmode (GAE), and fishbone are in this group. The other group is energetic electron (runaway electron) driven instabilities such as whistler waves and kinetic instabilities. The suppression of the energetic ion driven instability is needed for the improvement of fast-ion confinement and the resulting heating efficiency. Because the abrupt loss of thermal kinetic energy is triggered by these instabilities, the suppression of the energetic ion driven instability contributes to the avoidance of plasma collapse. These important findings are the fact that various EP-driven instabilities appear simultaneously when the beam pressure is comparable to the bulk pressure. The reason the community is interested in these scenarios with large $\beta_{EP}$ is that various non-linear saturation mechanisms appear at large $\beta_{EP}$ that need to be understood by theory, captured by numerical models, validated by code–experiment comparisons and then used for the prediction of burning plasmas.

Observation of the various EP-driven instabilities has been reported from ASDEX Upgrade. Figure 2 shows various EP-driven instabilities simultaneously observed in ASDEX Upgrade: EGAMs, BAEs, reversed shear Alfvén eigenmodes (RSAEs) and TAEs that are modulated by transient $q = 2$ sawtooth-like crashes, the neoclassical tearing mode (NTM) and the edge localized mode (ELM) [1]. It was reported that one EP-driven instability triggers the other EP-driven instability. For example, TAE bursts are observed to trigger the onset of EGAMs which indicates coupling of these modes via the velocity space (EP avalanches) and via mode–mode coupling processes. It is an important finding that several EP-driven instabilities can co-exist in the plasma when the EP pressure generated by a neutral beam is comparable to the background plasma pressure. A crucial point of this paper was that $E_{\text{beam}}/T_i, \epsilon_{\text{thermal}} \sim 100$, as in a burning plasma. This was reached via keeping low background temperatures (core tungsten accumulation) despite 5 MW neutral beam injection (NBI) heating. In the plasma with a high fraction of EP pressure, interactions among EP instabilities become more significant due to the non-linear coupling between the different EP-driven instabilities. Bicoherence analysis reveals the non-linear coupling processes between different frequency bands.

Observation of various high-frequency modes driven by energetic anisotropic fast ions or runaway electrons are reported from DIII-D, which is summarized in figure 3 [2–5]. High-frequency modes observed in the DIII-D tokamak have been identified as EP instabilities driven by anisotropic fast ions or runaway electrons. Doppler-shifted cyclotron resonant CAEs with frequencies $f$ less than the main ion cyclotron frequency $f_{ci}$ are measured and many aspects of observation are consistent with CAE theory. CAEs are excited on DIII-D when the beam ions are near-Alfvénic, with onset frequencies of $\sim 0.6f_{ci}$. ICE is readily excited across a wide region of operational space by kinetic instability at harmonics of the main ion $f_{ci}$. ICE is strongest in neutral-beam-heated plasma with a clear dependence on beam geometry, with the highest emission levels with counter-current beams. Whistler waves with $f \gg f_{ci}$ excited by multi-MeV runaway electrons, have been observed in a low-density ($n_e \sim 10^{19} \text{ m}^{-3}$) tokamak plasma [6]. The waves occur in multiple discrete frequency bands in the 100–200 MHz range, with the measured whistler frequencies scaling with magnetic field strength, as expected from the whistler dispersion relation. Another electron-drive mode with $f \sim 0.1f_{ci}$ has recently been observed that correlates with runaway losses and formation suppression of a runaway beam. The observation of whistler waves excited by energetic electrons (runaway electron) is reported in tokamak plasmas for the first time.

2.2. ECH effect on EP-driven instabilities

Many observations that suggest the effective suppression of EP-driven instabilities by ECH have been reported at this conference. Various EP-driven instabilities disappear when the ECH is applied in both helical and tokamak plasma. It is confirmed that the on-axis ECH is quite effective for suppressing the EIC in the experiments in deuterium plasma as seen in the Large Helical Device (LHD) in figure 4 [7]. ECH is changing the stability properties of the EIC modes (interchange physics) and thus the ECH indirectly increases the
Suppression of EP-driven instability by ECH is also observed in the various experiments in tokamaks. The stabilization of \( m/n = 1/1 \) ion fishbone activities by ECH was reported from HL-2A [9]. By scanning the ECH configuration parameters, it was found that the fishbone stability depends not only on the injected power, but also on the radial deposition position of ECRH, which is similar to finding that suppression of EP-driven instability depends on both the ECH power and ECH deposition location in the Heliotron. The fishbone can be completely suppressed when the injected ECH power level exceeds a certain threshold. Figure 6 shows the time evolution of magnetic field fluctuation spectra at the ECH power level \( (P_{\text{ECH}} \sim 1 \text{ MW}) \) above the threshold. When the ECRH power is deposited on-axis \( (\rho \sim 0) \), the observed mode frequency obviously decreases, but the mode amplitude is only weakly reduced. When the power is deposited at \( \rho \sim 0.42 \) outside of the \( q = 1 \) rational surface \( (\rho \sim 0.38) \), the mode is fully stabilized. However, the mode is partially stabilized when the power is deposited off-axis \( (\rho \sim 0.66) \). This experiment clearly shows that the control of the location of the ECH deposition is quite important to suppress the EP-driven instability effectively.

The correlation of TAEs with ECH injection was reported from KSTAR [10]. Figure 7 shows the total neutron rate, stored energy, and \( \beta \) poloidal, for the normal \( (18597) \) and high \( \beta_p \) H-modes \( (18602) \) and the time evolution of magnetic field fluctuation spectra in the discharge with ECH. During the ECH off phase \( (t = 14-16 \text{ s}) \), \( n = 1 \) \( (80 \text{ kHz}) \) and \( n = 3 \) \( (120 \text{ kHz}) \) TAEs, which are suppressed during the ECH phase, appear. In this phase, the total neutron emission rate dominated by the beam–target D–D reaction decreases associated with the appearance of TAEs, which suggests the significant decrease of fast ion pressure rather than bulk plasma pressure similar to the experiment in the LHD. The decrease of stored energy and \( \beta_p \) values is also observed during the ECH off phase and this implies the degradation of confinement due to TAEs.

Several experiments in helical and tokamak plasmas provide common results, which are summarized below. (1) EP-driven instabilities have a significant effect on fast-ion confinement (reducing the fast-ion pressure) rather than bulk confinement. (2) The EP-driven instabilities can be suppressed by applying ECH with the power above the critical value. (3) The location of the deposition of ECH is important to suppress the EP-driven instabilities efficiently. Although the mechanism of suppression of EP-driven modes by ECH has been studied in several experiments [11], more detailed investigation for the mechanism is necessary for the development of more efficient techniques for EP-driven instability suppression.

2.3. Effect of RMP on EP-driven instability

RMP has been used for the H-mode discharge to suppress ELMy H-modes (ELMs). Recently, the RMP is also applied to suppress the EP-driven instability both in helical and tokamak plasma.

An example of suppression of EP-driven instability (EIC) at \( m/n = 1/1 \) RMP was reported from the LHD [7], which is summarized in figure 8. After the additional perpendicular

![Figure 4](image-url)
NBI (#4), repeated bursts of magnetic fluctuation (EIC) are observed. Associated with the MHD burst, the neutron emission rate drops, which indicates the loss of trapped fast ions of the beam. Although the drop of stored energy is observed associated with the instability, the drop of stored energy is smaller than the drop of the neutron rate, which is mainly

Figure 5. Dependence of (a) EPMs and (b) GAE amplitude of ECH injection power $P_{\text{ECH}}$ in NBI-heated Heliotron J plasma. Reproduced courtesy of IAEA. Figure adapted from [8]. Copyright 2018 IAEA.

Figure 6. Effects of electron cyclotron resonance heating (ECRH) power and the deposition position on the stabilization of ion fishbone activities in HL-2A in dependence of ECH radial deposition location $\rho$. Reproduced courtesy of IAEA. Figure adapted from [9]. Copyright 2019 IAEA.

Figure 7. (a) Time evolution of total neutron rate, stored energy, and $\beta$ poloidal, for the normal (18597) and high $\beta_p$ H-modes (18602) and (b) the time evolution of magnetic field fluctuation spectra in the discharge with ECH in high $\beta_p$ H-modes (18602) in KSTAR. The ECH is turned off only for $t = 14–16$ s. Reproduced courtesy of IAEA. Figure adapted from [10]. Copyright 2018 IAEA.
provided by the beam–plasma reaction. In the discharge with RMP field, no EP-driven instability and no drop of neutron emission rate associated with the instability are observed. The small decrease of the neutron emission rate in the discharge with RMP is due to the NBI gap in the background measurement of charge exchange spectroscopy. This suppression is observed when the electron density is not too low (line averaged electron density \( n_e \approx 0.9 \times 10^{19} \text{ m}^{-3} \)). Because the magnitude of the perturbation of the magnetic field is too small to affect the orbit of EPs, the suppression mechanism of EP-driven instability by RMP is not well understood. The stabilization of the seed interchange mode and decrease of the gradient of energetic fast ion pressure are possible candidates for the suppression mechanism.

Suppression of EP-driven instability of the RMP was also reported from ASDEX Upgrade (AUG) [12, 13]. The NBI driven TAEs are mitigated or suppressed by RMP. Figure 9 shows the TAE instability in the frequency range of 70–120kHz measured with a magnetic pick-up coil in AUG. A clear modulation of the TAE instability is observed with the application of externally applied RMP fields. Timing of the applied RMPs is highlighted in the bottom panel with vertical grey stripes. In the bottom panel, the temporal evolution of the root-mean-square amplitude of the TAE is plotted, together with that of the of the measured fast-ion losses. As the discharge evolves over time, the overall TAE activity becomes weaker due to the change in \( q \)-profile. RMP blips are clearly modulating and eventually suppressing the TAE instability completely. Fast-ion losses measured with one of the FILDs show a clear correlation between applied RMP fields and TAE instability.

It was also found that the phase of RMP is important for the effective suppression of TAE instability by the RMP. Then, two different shots were carried out in which the MPs were applied intermittently with two differential phases corresponding to the maximum and minimum level of fast-ion losses measured by FILD. The RMP phases of \( \Delta \phi = 100^\circ \) and \( \Delta \phi = 50^\circ \) correspond to the maximum and minimum levels of fast-ion losses, respectively. The modulation of the fast-ion loss level measured by FILD is larger in the discharge with \( \Delta \phi = 100^\circ \) than that in the discharge \( \Delta \phi = 50^\circ \). This experiment clearly demonstrates that the suppression of EP-driven instability is quite sensitive to the phase of RMP. This is because the fast ion profile, especially the given pitch angle at the edge, is significantly modified depending on the phase of RMP.

2.4. Energetic particle transport simulation

The validation of the transport model for EPs owing to the EP-driven instability was reported from DIII-D. The transport model for EPs due to the EP-driven instability is validated by comparing the experimental data of neutron rate and fast ion \( D_n \) (FIDA) signal. As the EP-driven transport model, the kick model has been designed to fit within the Monte Carlo implementation of NUBEAM. The wave–particle interaction process is distilled into the kick probability matrix through orbit modeling and the effect of EP transport by instability is included into the tokamak transport code TRANSP [14–17]. Figure 10 shows the time evolution of neutron rate and that from TRANSP modeling and the radial profile of the FIDA signal measured for co- and counter- passing fast ions in DIII-D discharge with a large 2/1 NTM oscillation amplitude. The deviation between the measured neutron rate and classical simulation indicates substantial fast ion transport. The neutron rate calculated by using the measured magnetic island width (predictive) and adjusting NTM amplitude (interactive) has a good agreement with the measurements. Although there is no velocity space resolution in the neutron rate, FIDA signals provide the information of velocity space. Therefore, comparison of FIDA signals between the measurements and reproduced by the kick model provides a more definite validation of the model. Deviations between the measured FIDA signal and classical simulation results are much more significant in the counter-passing particle than co-passing particles, which indicate the loss of counter-passing fast ions. The reproduced FIDA signals for the kick model gave a good agreement with the measurements of both the co-passing and counter-passing fast ions.
The kick model has also recovered the measured TAE spectrum of NSTX-U and DIII-D and has reproduced details of fast-ion diagnostics data (neutron rate and FIDA signal). Figure 11 shows the comparison of the measured neutron rate and the radial profile of fast ions between the measurements and kick model in DIII-D and NSTX-U discharge. [15, 17, 18]. A significant loss of fast ions is observed in the core region in the plasma, both in DIII-D and NSTX-U, as seen in the radial profile of fast ions. In the discharge of NSTX-U, the RSAEs are observed for $t < 400$ ms, fishbone is observed for $t = 400$–$500$ ms, and the TAEs/RSAEs are observed for $t > 500$ ms. The deviation of the measured neutron rate and the kick model from the classical model appear for $t > 400$ ms, which is when the fishbone appears and becomes
more significant for $t > 500\,\text{ms}$, when the TAEs appear. These observations show that the fast ion loss by TAEs is larger than that by fishbone and RSAEs, and the good agreement between the measurements and kick model simulation demonstrates the validation of the model is quite good.

2.5. Heating scenario by ion cyclotron range of frequencies (ICRF) heating and ECRH

A novel heating scenario using ICRF heating and ECRH has been developed in tokamaks and stellarators. A three-ion species ion cyclotron resonance heating (ICRH) scheme on JET and AUG was reported [19–23]. A D–(3He)–H three-ion ICRH scenario (minority heating of 3He ions in H–D plasma) was found to be an efficient technique for heating H–D mixed plasma in JET. As seen in figure 12(a), a large increase of stored energy where $\Delta W_p/\Delta P_{\text{ICRH}} > 0.15\,\text{MJ MW}^{-1}$ is obtained in the plasma with 0.1%–1% 3He density and 75%–82% H density. Note that circles represent pulses in which more than 2 MW of ICRH power was coupled with $+\pi/2$ phasing, while squares correspond to pulses in which all RF power was coupled with the dipole phasing. The grey area corresponds to the region where more than 80% of incoming RF power was computed to be absorbed by 3He minority ions. Energetic species, such as injected NBI ions and fusion products, can also play the role of the ‘third’ species. An example of the performance of the three-ion D–(DNBI)–H scheme on JET-ILW with fast D-NBI ions was demonstrated. Figure 12(b) shows an overview of JET pulse #91256 (2.9 T/2 MA, H–D $\sim 85\%$–15%), where the neutron rate increased by a factor of $10$–$15$ when 2.5 MW of ICRH power ($f = 25\,\text{MHz}$, dipole phasing) was applied simultaneously to 3.5 MW of D-NBI. The scenario was tuned such that D-NBI ions with an injection energy of 100 keV absorbed most of the launched ICRH power in the vicinity of the ion–ion hybrid layer and were accelerated to much higher energies using ICRH. The presence of a population of energetic D ions with energies $\sim 1$–2 MeV during the combined ICRH + NBI phase was confirmed by neutron spectroscopy and $\gamma$-ray measurements as seen in the bottom panel in figure 12(b).

A high-density plasma scenario with O2-ECRH and pellet injection was reported from Wendelstein 7-X (W7-X) [24]. As shown in figure 13, the plasma start-up and the first density ramp was performed by four ECRH beams in X-mode. During the density ramp-up phase of 2 s, the polarization of these beams had been changed from X2-mode
to O2-mode and the density was increased by pellet injection to above the X2 cut-off. The second row shows the chance of polarizer position for the three ‘start-up’ gyrotron beams, when their polarization was switched from X2 to O2. High densities of up to $n_e(0) = 1.4 \times 10^{20} \text{ m}^{-3}$ were achieved by the 141 GHz second harmonic O-mode (cut off density = $2.4 \times 10^{20} \text{ m}^{-3}$) by multiple pellet injections. Although the electron temperature measured with ECE agrees with that measured with Thomson scattering before $t = 3.0 \text{ s}$, ECE signal intensity drops to almost zero as the line-integrated density exceeds the critical value. This is clear evidence that the electron density exceeds the X2 cutoff density ($1.2 \times 10^{20} \text{ m}^{-3}$).

It was demonstrated that the combination of the switching of ECRH from the X2-mode to the O2-mode and multiple pellet injections was a good scenario to achieve high density ECRH plasma in W7-X.

3. Plasma–material interactions, divertors, limiters and scrape-off layer (SOL)

Figure 14 shows physics elements in the field of plasma–material interactions, divertors, limiters and SOL. Turbulence spreading from the plasma core to the SOL and filaments generated at the divertor leg has been recognized to play an important role in determining the SOL decay length at the divertor plate and hence divertor erosion. Turbulence in the SOL is strongly affected by the radial electric field ($E_r$) shear at the last closed flux surface (LCFS) and the turbulence generated at the pedestal region inside the LCFS. The SOL decay length can be controlled by flux expansion, magnetic perturbation and turbulence in the SOL and filaments in the divertor leg. Turbulence and filaments also affect the lower-hybrid current drive efficiency.

3.1. Power decay length in the SOL

In the SOL region in a tokamak, the power fall-off length is typically small at the mid-plane and this short power fall-off length causes localized heat flux and hence the severe erosion of the divertor plate. The localization of the heat flux at the divertor can be reduced by expanding the magnetic flux surface, which is called flux expansion. In order to distinguish the geometric change due to the flux expansion and change in SOL transport, the upstream-remapped power decay length ($\lambda_p$) is evaluated.

Results of flux expansion experiment were reported from TCV. Figure 15 shows the ratio of power fluxes to inner- and outer-divertor targets, SOL power decay length and spreading factor, which describes the transport scale length in the divertor SOL, as a function of the flux expansion $f_x$ [25, 26]. Here, the flux expansion is defined as $f_x = (B_p/B_{tot})_{MP}(B_p/B_{tot})_{SP}^{-1}$, where $B_p$ is the poloidal magnetic field, and $B_{tot}$ is the total magnetic field, both evaluated at the mid-plane separatrix ($R_{SP}$) and at the strike point ($R_{SP}$). An increase in both divertor-leg length and flux expansion has the effect of reducing the asymmetry in the power load at the inner and outer targets. Their ratio increases to nearly unity at the largest values of flux expansion. However, the increase of the connection length by flux expansion has a much weaker effect on the power decay length. Although the spreading factor at the inner target plate is almost unchanged in the wide range of flux expansion, the spreading factor at the outer target plate decreases almost proportionally to the inverse of flux expansion as $S_u \propto f_x^{-0.97}$. This variation is attributed to a decrease in the outer conductance.

The dynamic control of the heat flux profile at the divertor target plate using the perturbation magnetic field of RMP was reported from DIII-D [27, 28] and KSTAR [29]. Figure 16(a) shows measured heat flux profiles during pure $n = 3$ RMP, and additional $n = 2$ RMP with a phase at $\phi = 0^\circ$ and $\phi = 180^\circ$ in DIII-D. Prominent heat flux splitting on the outer divertor has been observed using the infrared camera during ELM suppression by $n = 3$ RMP in this low collisionality regime. Variation of heat flux splitting pattern of different $n = 2$ phases in these mixed toroidal harmonic cases are observed. These
characteristics agree well with modeling of field penetration depth using TOP2D with the plasma response modeled by MARS-F. A similar experimental observation is also reported in KSTAR. Figure 16(a) shows the radial profile of heat flux in the plasma with three-row \( n = 1 \) RMP with a phasing of \((-90^\circ, 90^\circ), (-85^\circ, 85^\circ), (-80^\circ, 80^\circ), \) and \((-75^\circ, 75^\circ)\) in KSTAR. The radial profile of heat flux with RMP is broader than that without RMP. Variation of the heat flux pattern at different phasing is also reported in KSTAR.

Turbulence in the SOL and filaments in the divertor leg are expected to contribute to the broadening of the radial profile of heat flux and increase of power decay length. The measurements of the fall-off length of power, density and temperature during the L-mode, I-mode and H-mode transition were reported from the AUG tokamak \([30, 31]\) (see figure 17). The fall-off lengths of power both in the outer and the inner divertors decrease after the transition from the L-mode to the I-mode phase and a further decrease is observed after the transition to the H-mode. As seen in the change in the fall-off length of the density and temperature, the decrease of the power fall-off length at the transition from the L-mode to the I-mode is mainly due to the decrease of the fall-off length of temperature \(\lambda_T\). In contrast, the decrease of the power fall-off length at the transition to the H-mode is mainly due to the decrease of fall-off length of density \(\lambda_n\). These observations suggest that suppression of temperature fluctuation in the I-mode causes the decrease of \(\lambda_T\) and \(\lambda_q\) and the suppression of the density fluctuation in the H-mode causes the decrease of \(\lambda_n\) and \(\lambda_q\).
3.2. Turbulence/filaments and flow in the SOL and divertor leg

In the transport study, the spreading turbulence is usually ignored and the turbulence observed is assumed to be locally driven. However, in the region where the pressure gradient is small, such as inside a magnetic island and the SOL, the spreading turbulence can be dominant. In fact, experimental data suggest the spreading turbulence has been reported in magnetic islands and the SOL. Turbulence spreading into the O-point from the X-point of the magnetic island was experimentally identified from the hysteresis relation between the temperature and the turbulence in the heat pulse propagation experiment in DIII-D [32].

In the SOL, turbulence which is spreading from the plasma inside the LCFS could be important, because the locally driven turbulence is relatively weak in the SOL region where the temperature and density gradients are small. The novel analysis of turbulence to distinguish the origin of the turbulence with the modulation of bias was reported from TJ-II [33]. In this analysis, the rate of turbulence drive and the rate of turbulence spreading are defined as

\[ \omega_D = -2 \left( \partial n / \partial r \right) \langle V_r \tilde{n} \rangle / \langle \tilde{n}^2 \rangle \]

and

\[ \omega_S = \partial \langle V_r \tilde{n}^2 \rangle / \partial r / \langle \tilde{n}^2 \rangle. \]

The angular brackets (...) indicate a running average over a time interval much larger than the turbulence correlation time. The impact of edge-biasing on the local turbulence drive (\( \omega_D \)) and the turbulence spreading (\( \omega_S \)) can be seen in figure 18. When the bias is turned on, locally driven turbulence inside LCFS (\( r - r_0 < 0 \)) is substantially reduced due to the edge radial electric field shear. The impact of the edge radial electric field shear on the locally driven turbulence outside the LCFS (\( r - r_0 > 0 \)) is small. In contrast, spreading turbulence inside the LCFS (\( r - r_0 < 0 \)) is almost unchanged during the bias modulation. Spreading turbulence outside the LCFS (\( r - r_0 > 0 \)) is significantly reduced due to

![Figure 17](image1.png)

**Figure 17.** (a) Power fall-off length (\( \lambda_q \)) mapped to the outer midplane and (b) corresponding SOL density fall-off length (\( \lambda_n \)) and temperature fall-off length (\( \lambda_T \)) during the L-mode, I-mode and H-mode transition on the AUG tokamak. Adapted from [31]. CC BY-NC-ND 4.0.

**Figure 18.** Contour of rate of turbulence drive (\( \omega_D \)) and turbulence spreading (\( \omega_S \)) in time and space with modulation of limiter bias in TJ-II. Reproduced courtesy of IAEA. Figure adapted from [33]. © EURATOM 2018.

![Figure 19](image2.png)

**Figure 19.** Schematic of the different contributions to intermittent cross-field transport in the divertor observed in MAST. Reproduced courtesy of IAEA. Figure adapted from [34]. © EURATOM 2019.
the edge radial electric field shear. Thus, when the shear of the $E \times B$ flow reaches a value comparable to the inverse of the turbulence correlation time, the level of turbulence spreading is affected and, as a consequence, the level of edge–SOL coupling also is affected. Therefore, the shearing rate of edge radial electric fields may be an important tool to suppress turbulence and also to decouple the edge and SOL regions, since it reduces the level of non-local effects (turbulence spreading).

Recent development of high-speed cameras reveals the turbulence and filament structure in the SOL region and divertor leg. Observation of filaments at the SOL and divertor-leg was recorded from MAST. Several regions where the characteristics of the observed filaments are qualitatively different are observed using the high-speed imaging of the lower divertor in MAST [34, 35], as seen in figure 19. High frequency filaments with a small size ($\sim 1$ cm) are observed close to the separatrix of the outer leg away from the X-point and in the private flux region. Filaments generated upstream (red) are shared by the X-point and aligned along the magnetic field at the divertor leg near the target plate. In contrast, the filaments locally generated (blue) at the divertor leg are not field aligned. Filaments at the private flux region (green) are small high frequency filaments.

Another observation of filaments at the SOL and divertor-leg was reported from NSTX-U. In NSTX-U, images of C III emissions are measured using a high speed imaging camera with 9 $\mu$s exposure [36, 37]. The typical structure of divertor filaments is shown in 20(a) with a rendering of flux tubes in the NSTX-U divertor corresponding to divertor localized fluctuations. Figure 20(b) shows filaments on the inner and outer divertor legs with an image of the lower divertor. Filaments were approximately field aligned, connected to the divertor target plate and localized to the bad curvature region on both the inner

Figure 20. (a) Rendering of the NSTX-U divertor with flux tubes corresponding to divertor-localized fluctuations; (b) divertor image in C III emission ($9 \mu$s exposure) after high pass filtering. Separatrix (red), inner (green) and outer (blue) divertor legs, and inner (cyan) and outer (orange) strike points are overlaid. Reproduced courtesy of IAEA. Figure adapted from [36]. Copyright 2018 IAEA.

(i.e. in the private flux region) and the outer divertor legs (i.e. in the common flux region). Measurements of poloidal and radial correlation length using a high speed imaging camera suggest the filaments are locally generated. The experimental observations with a high speed camera reveal the complexity of the turbulence structure in the SOL region. The filament generated at the upstream SOL (mid-plane) and the filament generated at the divertor leg co-exist in the divertor leg. Although the filament generated at the upstream SOL is expected to be elongated along the magnetic field (field aligned), the filament locally generated at the divertor leg has a small structure. The enhancement of turbulence and filaments of the SOL and divertor leg provides another facet in the control of the power decay length.

The measurements of parallel flow velocity were reported from DIII-D and the LHD. The structure of the plasma flow parallel to the magnetic field is measured by the Doppler shift of the impurity emission in the SOL and the divertor region in order to identify the parallel convective transport towards the target plate. The coherence imaging spectroscopy diagnostic [38] provides a tangential image of the C III intensity (465 nm) and flow velocity along the line-of-sight. Figure 21 shows the image of tangential views of C$^{2+}$ flow that have been inverted onto the poloidal plane in the density where the C$^{2+}$ velocity is well coupled to the background deuterium velocity [39]. The flow images suggest a parallel flow towards the target throughout the divertor in the range of 15–30 km s$^{-1}$. Flow towards the outboard target plate is observed in the wide region near the outer leg, while the parallel flow towards the inboard plate is observed in the private region. These flow velocities are roughly equal to the ion sound speed for the $T \sim 3–5$ eV characteristic of the divertor plasma. Imaging of the plasma flow implies a parallel plasma flow towards the target in the divertor near the sound speed, $M \sim 1$.

Figure 22 shows vertical profiles at the bottom edge of the ergodic layer of ion temperature, and flow velocity derived from the Doppler profile of the second order of C IV line emission in the LHD [40]. The measured flow velocity ($v_R$) is the
projection of the flow along the observation chord which can be approximately considered to be the direction of the plasma major radius. Assuming that the flow measured is simply the projection of flow parallel to the magnetic field lines to the line-of-sight of the measurements, the value of parallel flow can be evaluated by

\[ v_\parallel = (B_\phi / B_R)v_R \]

The flow velocity toward the outboard direction develops clearly with the maximum value at \( Z = -480 \) mm, which is a location close to the outermost region of the ergodic layer in the H plasma. The location of the LCFS is \( Z = -460 \) mm and the peak of the flow velocity is located 20 mm outside the LCFS. The ion temperature has no clear change between the H plasma and the D plasma, and \( \sim 30 \) eV at the peak of plasma flow. Flow velocity of \( v_R \) is larger (11.3 km s\(^{-1}\)) for H plasma than that for D plasma (8.0 km s\(^{-1}\)). The difference of the estimated values between D and H plasma is caused by the \( m_1^{1/2} \) dependence of the parallel velocity when the friction force term is dominant in the force...
balance. In the LHD, the parallel plasma flow towards the out-
board divertor target plate at the upstream of SOL is one-half
of the sound speed, $M \sim 0.5$. The Doppler shift measurements
with impurity emission in DIII-D and the LHD confirm that
the parallel plasma flow in the SOL and divertor leg is towards
the divertor target plate and near the ion sound velocity (the
Mach number of the parallel flow velocity is 0.5–1).

3.3. Impact of SOL turbulence on lower hybrid (LH) current
 drive efficiency

The novel measurements of LH electric field vector were
reported from Alcator C-Mod. The LH electric field vector,
$E_{\text{LH}}$, was directly measured in the edge region of Alcator
C-Mod using dynamic Stark spectroscopy (DSS) by fitting
the Schrödinger equation to the shape of $\pi$ and $\sigma$ polarized
spectra of $D_3$ simultaneously [41, 42]. Figure 23(a) shows the
example of $\pi$ and $\sigma$ polarized spectra of $D_3$ with fitted lines
by the Schrödinger equation where the magnitude of $E_{\text{LH}}$ is
fitting parameters. Experimentally, $E_{\text{LH}}$ was found to be com-
posed of nearly equal poloidal and radial components. In the
simulation, $E_{\text{LH}}$ was found to be nearly radial. Figure 23(b)
shows the direction of $E_{\text{LH}}$ averaged over LH power as a
function of vertical locations. The direction of $E_{\text{LH}}$ is nearly
70°–80° in the simulation without density fluctuation. When
the density fluctuation is added into the simulation, the direc-
tion of $E_{\text{LH}}$ becomes 40°–50°, which is much more consistent
with the direction measured with DSS. These results indicate
that SOL density fluctuations cause severe scattering of the
LH wave in $k_\perp$ space and make the magnitude of the poloidal
component equal to that of the radial component. It is con-
cluded that diffraction and scattering by turbulence driven
SOL density fluctuations is the responsible mechanism behind
the experimentally observed modification in the direction of
$E_{\text{LH}}$. These results further support the proposition that edge
turbulence is an important mechanism behind the anomalous
drop in LH current drive efficiency for diverted plasma in the
high density regime.

Efficient LH current drive at densities up to $\bar{n}_e \sim 1.5 \times
10^{20} \text{ m}^{-3}$ was demonstrated in the Alcator C-Mod tokamak.
Figure 24 summarizes the non-thermal bremsstrahlung emission
rates as a function of line-averaged density [43, 44]. At
fixed line-averaged density, the count rates measured at low
Greenwald fractions ($I_p = 1.2$ and 1.4 MA) are increased by
more than two orders of magnitude compared to those at the
lowest current (0.55 MA). This result suggests that a recov-
ergy of efficient LH current drive correlates with the change in
the edge/SOL density condition, rather than with the temper-
taxture. The recovery of current drive efficiency to a level con-
sistent with engineering scalings is related to a reduction in
density shoulders and turbulence levels in the far SOL. The
experiment in Alcator C-Mod demonstrated that the density fluctuations in SOL are an important parameter in determining the LH current drive efficiency because it causes severe scattering of the LH wave and changes the direction of the LH electric field vector. The reduction of SOL density fluctuation is desirable for high LH current density efficiency in a high density plasma. This feature is compatible with the requirement for the long power decay length in the SOL, because the enhancement of SOL density fluctuation is preferable to increase the power decay length in the SOL and to reduce the heat load on the divertor plate.

3.4. Detachment

Detachment experiments have been investigated as another approach to reduce the heat load on the divertor plate. Long pulse discharge was demonstrated in the W7-X stellarator with a magnetic island divertor in the detached condition \[45-47\]. As seen in figure 25, the transition to detachment occurred at \( t = 2 \) s during a density ramp with the heating power held approximately constant around 2.7 MW, and detachment lasted until the programmed end of the discharge at \( t = 4 \) s. A local heat flux to the divertor target is reduced by about one order of magnitude during detachment. The detached condition was also applied to demonstrate the long pulse discharge over 30 s with the central electron density of \( n_e \sim 8 \times 10^{19} \) cm\(^{-3}\), central electron temperature of 2.4 keV and global energy confinement time of 125 ms.

Effects of magnetic island produced by an RMP field at the plasma edge on detachment was reported in the LHD. In helical plasma, the electron density is limited by the radiation collapse because there is no disruption in high density. The density limit, \( n_{\text{sudo}} \), was found to be determined as \( 0.25 PB^2/a^2/R^{0.5} \), where \( P \), \( B \), \( a \) and \( R \) are heating power (MW), magnetic field (T), minor radius (m) and major radius (m), respectively. As seen in figure 26, the radiation increases sharply as the plasma density is approaching the critical density so-called Sudo density limit, \( n_{\text{sudo}} \) in the plasma without an RMP field \[48\]. In this experiment, the heating power is 7–8 MW and the density limit \( n_{\text{sudo}} \) is \( n_e/n_{\text{sudo}} \sim 0.45 \). After the detachment, the increase of radiation power is gradually increased even near the Sudo density limit. Therefore, the magnetic island produced by the RMP field enhances the transition to the detachment phase and results in the access to the higher density. It was found that the RMP application leads to easier control of
detached plasma in the LHD. The detached experiments in W7-X and the LHD demonstrated that the magnetic islands near the plasma boundary are quite beneficial for controlling the detached plasma and accessing the steady-state long pulse operation with a high density.

The transition from an attached condition to a strongly detached condition at the high density was demonstrated in tokamaks. As seen in figure 27(a), the electron temperature measured with divertor Thomson scattering in front of the low field side (LFS) of the target plate shows the abrupt drop when the upstream separatrix electron density exceeds a critical value [49, 50]. The critical value depends on the direction of magnetic field. The critical density is higher in the case of forward $BT$ ($\nabla B$-drift towards the X-point and a lower power threshold for the H-mode) than in the case of reversed $BT$ ($\nabla B$-drift away from the X-point and a higher power threshold for the H-mode). Figure 27(b) shows the comparison of measured electron temperature along a flux tube in the LFS SOL to that predicted by the UEDGE simulation code in the attached condition (black) and detached condition (red). The UEDGE simulation includes full cross-field drift and SOL current and reproduces the experimentally observed step-like inset of the LFS divertor detachment with increasing plasma density in the case of forward $BT$. Discharges with the small angle slot (SAS) divertor also show less degradation in the detached plasma in the LHD. The detached experiments in W7-X and the LHD demonstrated that the magnetic islands near the plasma boundary are quite beneficial for controlling the detached plasma and accessing the steady-state long pulse operation with a high density.

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H-mode confinement factor with increasing edge pedestal density than that with the open lower single null divertor or a more closed upper single null divertor for similar conditions.

A new concept of divertor was reported from DIII-D. A prototype SAS divertor, which consists of progressive opening and a small angle target, has been installed on DIII-D as illustrated in figure 28(a) [14]. The SAS divertor was designed to both enhance neutral trapping and optimize neutral distribution along the target to achieve strong plasma cooling across the divertor at a lower upstream plasma density [51]. As seen in figure 28(b), the critical density for the transition from the attached to the detached condition indicated by the abrupt drop of electron temperature is $\bar{n}_e/n_G = 0.37$ for the SAS divertor, which is lower than that ($\bar{n}_e/n_G = 0.5$) for the conventional divertor (closed upper single null divertor).

### 3.4.1 Impurity seeding

The 3D coupled plasma fluid and kinetic neutral edge transport Monte Carlo code EMC3-EIRENE is commonly used for predictive modeling and the interpretation of experimental results in the domain of plasma–wall interactions and SOL transport [52, 53]. Calculations of EMC3-EIRENE for the first attempt in preparation of seeding experiments were reported from W7-X [54, 55]. Impurity seeding in the island divertor was simulated assuming either sourcing scaled with the main recycling flux or sourced from the puff location only. Figure 29 shows the main results of a first comparison of the features of radiative power exhaust with neon (Ne) and nitrogen (N$_2$) seeding within the modeling and its impact on divertor heat flux. Significant reduction of the heat flux on the target plate is predicted both for Ne and N$_2$ seeding. In the case of Ne seeding, the impact of recycling on the reduction of heat flux is studied. The heat fluxes are slightly less reduced in the case of Ne and sourced from the puff location in the island center without recycling. By including the recycling effect, a further reduction of heat flux due to impurity seeding is predicted. 3D modeling using EMC3-EIRENE confirms the main features of the power exhaust experiments in W7-X.

The effect of magnetic geometry on SOL transport and detachment behaviour was also reported from TCV [25, 56]. As seen in figure 30, a significant reduction of ion saturation current $j_{sat}$ at the outer target in the vicinity of the separatrix ($\rho_s = 1$) is observed with nitrogen seeding. The strong peak at the separatrix observed before the nitrogen seeding vanishes after it, which is a clear indication of the partial detachment. After the nitrogen seeding, the $j_{sat}$ profile shows small double
peaks across the separatrix. Edge transport simulation in TCV shows that divertor dynamics are modified by the existence of significant $E \times B$ drifts. The $E \times B$ drifts are one of the candidates for splitting of the heat flux during the nitrogen seeding. The simulation including $E \times B$ drifts predict the double peaked heat flux profile at the inner strike point in TCV [57]. However, the double peaked heat flux profile is not predicted in the simulation with impurity seeding experiment in W7-X.

The impurity seeding experiment demonstrates a significant reduction of heat flux on the target plate, which can reduce the erosion of the divertor plate significantly. The impact of impurity seeding is not just the decrease of the magnitude of the heat flux. The heat flux profile is modified by impurity seeding as seen in the splitting of the sharp peak of the heat flux into small two peaks at the separatrix. Impurity seeding can be a candidate for the mitigation of the localized heat flux at the target plate in the H-mode plasma, where the power decay length at the SOL is shortest due to the reduction of turbulence in the SOL, as discussed in the previous section.

3.5. Erosion and sputtering

Detailed analysis of tungsten erosion and sputtering was reported from JET. The outer strike point (OSP) was located on a part of the bulk-W divertor (tile 5, stack C) which could be correlated to the exposure in this period of plasma operation before tile removal. The electron temperature and density at the OSP are $35 \text{ eV}$ and $6 \times 10^{19} \text{ m}^{-3}$, respectively. A comparison of W source spectroscopy in the intra- and inter-ELM phases with post-mortem analysis (PMA) provides gross and net W erosion. W gross erosion determined by optical emission spectroscopy is $40–60 \text{ g}$. In contrast, net erosion determined by the PMA of deposition of the divertor tile after removal is $2.4–4.8 \text{ g}$ over the 900 s considering the interactions are in the circumference of the outer divertor. This corresponds to a higher deposition fraction of more than 94% averaged over both phases. The plasma–material interaction and transport code (ERO) coupled with improved PIC modelling for the complex re-deposition in both phases was applied to simulate a comparable experimental case [58]. Figure 31 shows the erosion and deposition distribution along the outer target plate in the conditions at the OSP. The ERO code can reproduce the large gross erosion, the low net erosion, and thus the local high re-deposition on tile 5 with more than 94% in average and at 99% for the sputtering dominating intra-ELM phase [59].

The erosion of tungsten (W) by plasma isotope species (H, D and T) and intrinsic (Be) and extrinsic impurity (N,
Ne or Ar) sputtering determines both the lifetime of divertor components and the impact on plasma performance. Binary collision approximation SDTrimSP simulations [60] allow prediction of the potential isotope dependences for the physical sputtering and thus give an insight into how sputtering may impact the integration of the main D–T scenarios with the W/Be wall. Figure 32 shows the sputtering yields of both Be and W by bulk ions (H, D and T) as a function of impact energy which is strongly dependent on the plasma conditions, such as the L-mode, inter-ELM, hybrid scenario and intra-ELM [23]. The sputtering yield of Be has a peak at ∼100 eV which corresponds to the plasma condition for the inter-ELM and hybrid scenario. In this energy range, the sputtering yields of W are negligible. Sputtering yields of W start to increase when the incident ion energy exceeds a few hundred eV and become significant above 1 keV, which corresponds to the plasma condition for intra-ELM. The isotope effect on the sputtering yields is clearly observed. The sputtering yields of Be by T and by D are 2.5 and 2 times the sputtering yields by H. In contrast, the sputtering yields of W by D and T are five to ten times that of H in the energy range for intra-ELMs. This simulation implies that the erosion of W during intra-ELMs will be a serious problem in D–T plasma mixtures in future.

4. Innovative confinement concepts

Experimental results of the newly constructed C-2W experimental device (also called ‘Norman’) were reported. It is the world’s largest compact-toroid device, which has several key upgrades from the preceding C-2U device, such as a higher input power and a longer pulse duration of the NBI system as well as the installation of inner divertors with upgraded electrode biasing systems. Figures 33(a) and (b) shows field-reversed-configuration magnetic topology and density contours for the outer-divertor and inner-divertor operating modes [61]. Figure 33 shows the typical plasma discharges under three different machine configurations. One discharge is without inner-divertor magnetic field flaring but with outer-divertor edge-biasing, which is a C-2U-like configuration. In this configuration, plasma was sustained for more than 8 ms. The other configuration is that with field flaring but without edge biasing at all. The plasma was sustained for ∼6 ms. The last configuration is that with field flaring and outer-divertor biasing. Although the duration of the plasma discharge is only ∼4 ms, the highest electron temperature (150–200 eV) was obtained.

The experimental results of high-power reconnection heating of ST plasma in TS-3, TS-4 and MAST were reported. The reconnection heating has been recognized to be efficient enough to realize a significant increase of ion temperature as the reconnecting magnetic field \( B_{\text{rec}} \) increases. As shown in figure 34, when the two STs are merged an X-point is formed at their contact point, followed by a new high-beta ST [62]. The increase of the ion temperature (\( \Delta T_i \)) due to the reconnection heating is proportional to the reconnecting magnetic field \( B_{\text{rec}}^2 \) as seen in the database of merging experiments in TS-3, TS-4 and MAST in figure 34(c). Here the electron density is \( n_e \sim 1.5 \times 10^{19} \text{ m}^{-3} \). Although the increase of ion temperature is in the range of 10–100 eV in TS-3, it was extended to 1.2 keV in MAST.

Although the plasma parameter obtained in the innovative confinement concepts experiment is far from the parameters required for the fusion reaction, the heating mechanism using the magnetic field is interesting and the knowledge obtained in these experiments can contribute to deep understanding of the heating mechanisms of solar flares.

5. Concluding remarks

Interesting results on the coupling between EPs, MHD, wave–particle interaction, turbulence, SOL physics and divertors are presented in this conference. Although EP-driven instability has little impact on bulk confinement, it has a significant...
impact on fast ion confinement and hence causes the degradation of heating efficiency. The suppression of EP-driven instabilities by ECH or RMP has been demonstrated in several tokamaks (ASDEX Upgrade, HL-2A, KSTAR) and helical systems (LHD, Heliotron-J). Energetic particle transport simulation has been developed and validated with energetic particle diagnostics (neutron rate and FIDA signal) in DIII-D and NSTX-U. Novel heating scenarios using ICRF heating and ECRH have been developed in tokamaks and stellarators. ICRF using three-ion species has been developed in JET, Alcator C-Mod and AUG and dynamic control of polarization of ECRH was developed to achieve high density in W7-X.

In tokamaks with high magnetic fields such as ITER, the power decay length in the SOL is predicted to be short and to cause severe erosion at the divertor plate due to the highly localized heat flux at the strike point at the separatrix. Several approaches to control the power decay length have been developed. The flux expansion was found to be inefficient for increasing the upstream power decay length in TCV. The modification of the heat flux profile of RMP has been demonstrated in DIII-D, EAST, and KSTAR. It is found in ASDEX that the power decay length becomes shorter in the H-mode plasma than in the L-mode plasma. This is speculated to be due to the reduction of turbulence in the SOL associated with the suppression of turbulence spreading observed in TJ-II. The complicated structure of turbulence, filaments and plasma flow in the SOL and divertor leg has been investigated using imaging measurements recently developed in MAST, NSTX-U, DIII-D and the LHD. Although turbulence in the SOL is beneficial for increasing the power decay length, turbulence reduces the LH current drive efficiency due to the scattering of the LH waves in $k_L$. The control of the power decay length by SOL turbulence may not be a good approach because the turbulence in SOL reduces the LH current drive efficiency, which is demonstrated in Alcator C-Mod experiments. Therefore, the detachment has been recognized to be a solution to reduce the heat load at the strike point. The experiment for the transition from an attached condition to a detached condition has been made in tokamaks (DIII-D, Alcator C-Mod) and helical plasma (the LHD and W7-X). Impurity seeding experiments performed in W7-X and TCV demonstrate that it is a useful tool to enhance the detachment as well as to reduce the heat flux. The isotope effect on the sputtering has been investigated in JET. It was found that the tungsten sputtering is recognized to be a serious problem in the D–T mixture plasma such as ITER.

Several interesting results on the coupling between the energetic particle and MHD instability, EP-driven turbulence and fast ion confinement, SOL turbulence and current drive efficiency, and SOL turbulence and heat load on the divertor plate have been presented in this conference. Deeper understanding of these couplings will be essential for the sustainment of high performance steady-state plasma in ITER.

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