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TOKAMAK EXPERIMENTS
- Summary -

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1. Introduction

One hundred and thirty papers were presented in this Conference from more than thirty tokamak machines, among which two (JET, TFTR) were operated with DT fuels and three (Tore Supra, TRIAM-1M, HT-7) employed super-conductor toroidal coils. Extensive progress was made in tokamak performance and understandings of tokamak plasmas have been improved significantly on these tokamak experiments in the last two years since the 16th IAEA Fusion Energy Conference in Montreal. These achievements have made significant contributions to the development of the next step devices such as ITER and to the concept development of future fusion power plants.

Highlights of tokamak plasma performance reported in this Conference were the achievements of high fusion power up to 16.1 MW in JET and of the equivalent energy gain Q^{eq}=1.25 in JT-60U. Many experimental attempts have been made for the advanced tokamak scenarios such as reversed shear configurations. Achievement and long sustainment of high performance plasmas were the main subject of the advanced tokamak experiments. Divertor configurations were investigated for optimization of heat and particle exhaust and impurity shielding. Understandings of tokamak plasmas were improved in confinement, divertor or boundary plasmas, and MHD stabilities. Control of plasmas was made progress in the current profiles, MHD and plasma-wall interactions. The innovative approach with a low aspect ratio spherical tokamak concept was successful in achieving a high beta up to $\langle\beta\rangle$=0.4.

2. DT Experiments and High Energy Particle Physics

2-1. D-T Fusion Power Production

A high fusion power of P_f=16.1 MW was obtained transiently in the JET ELM-free H-mode D-T discharge at I_p=4.2 MA and B_t=3.6 T with combined heating of 22.3 MW of NB and 3.1 MW of hydrogen minority ICRF. The fusion gain $Q_{in}=P_f/P_{in}=0.62$ and $Q_{tot}=P_f/(P_{in}-dW/dt-P_\alpha)=0.94$, and the fusion triple product of $n_{DT}\,\tau_E\,T_i(0)=9\times10^{20}$ m^{-3} s keV with $T_i(0)$=28 keV and $T_e(0)$=14 keV. No alpha particle-driven instability was observed even in the presence of P_α greater than 10 % of P_{in}. In the ELMy H-mode D-T discharge of JET at I_p=3.8 MA and B_t=3.8 T, $Q(=P_f/P_{in})=0.17$ was maintained for about 4 s.

2-2. Alpha Heating

Alpha particle heating was clearly observed in JET discharges carried out with various D-T mixtures. The ELM-free H-mode confinement was not or less dependent on isotope, while the plasma energy as well as the global confinement showed the peak around 50:50 D-T mixture and the electron temperature was highest at a 60 % of tritium discharge. It was concluded that the alpha heating was classical in the absence of alpha-driven TAE modes.

Ripple loss of alpha particles was significant in the TFTR reversed shear configuration more than the monotonic shear one. Careful ripple reduction will likely be important in the advanced (reversed shear) operation of a future tokamak reactor. Most types of MHD activity enhanced alpha loss in TFTR.

2-3. TAE Modes

TAE modes were not observed in JET, while weak n=2-5 TAE modes were observed in specific TFTR DT discharges; weak magnetic shear and q(0)>1 after switching off the beam injection. Radiative damping was likely the dominant damping mechanism and no measurable alpha loss associated with the TAEs was observed on TFTR.

In an ICRF hydrogen minority heating of deuterium plasmas in TFTR, it was observed that frequency sweeping modes (chirping modes) caused a substantial fast ion loss while stationary modes (TAE modes) less influenced the outward flow of fast ions.

TAE modes were excited at a fast ion beta $<\beta_h>\geq 0.1\%$ and burst modes and chirping modes were excited in addition to the TAEs at $<\beta_h>\geq 0.2\%$ in the negative-ion neutral beam (330 – 360 keV) injection experiments on JT-60U. Fast ion loss caused by these beam driven modes was rather small in these experiments.

3. Advanced Tokamak Operation

3-1. High Fusion Performance and Its Sustainment

Improvement in fusion performance and its sustainment were greatly progressed in the past two years. In DT experiments, JET achieved a fusion power of 16.1 MW in a hot-ion H-mode and a fusion product of $n_{DT}\ \tau_E\ T_i(0) = 1.1 x 10^{21}\ m^{-3}$ s keV in an optimized shear discharge. In DD experiments, JT-60U recorded the highest Q_{DT}^{eq} value of 1.25 in a reversed shear discharge. While these results were obtained transiently, many emphases were also placed upon the sustainment of high fusion performance. In a quasi-steady state with a time scale longer than the energy confinement time, high fusion performance was successfully sustained as Q_{DT}=0.22 for ~3 s with ICRF in ELMy H-mode in JET and Q_{DT}^{eq}=0.16 for ~4.5 s with NBI in high-β_p H-mode in JT-60U.

3-2. Internal Transport Barriers in Weak-Negative Magnetic Shear Configulations

The advanced operation was widely explored in many tokamaks such as JT-60U, JET, TFTR, DIII-D, ASDEX-U, C-MOD, Tore Supra, FT-U and HT-7. These advanced operations were closely associated with the current profiles with a weak or reversed magnetic shear and the formation of internal transport barriers.

Double transport barrier discharges with an internal transport barrier in combination with an edge transport barrier associated with H-mode were also observed in JT-60U, JET, DIII-D and ASDEX-U. In JT-60U, double transport barriers were clearly observed for both T_e and T_i in the high-β_p H-mode. Stationary H-mode discharges with ITB were sustained sufficiently longer than energy confinement times with a high figure of merit of $\beta_N H$ such as: $\beta_N H_{89P}$>5 for a duration over 12τ_E or ~2.8 s at q_{95}=3.3 with the high-β_p H-mode in JT-60U, $\beta_N H_{89P}$ >5 for a duration over 25τ_E or ~3 s with the ELMy H-mode in DIII-D, and $\beta_N H_{89P}$ ~4.8 for a duration over 40τ_E or ~6 s with the ITB+ELMy H-mode in ASDEX-U.

The internal transport barrier was also observed in various heating schemes as reported with NBI from JT-60U, DIII-D, TFTR and ASDEX-U, NBI+IC from JET, LH from TRIAM-1M, ECRH from FT-U and RTP, and OH from TUMAN-3M. The ITBs were sustained for several seconds as: ~6 s with LH+NB or NB alone in JT-60U for reversed shear, ~5 s with NBI in DIII-D for weak shear and L-mode edge, ~6 s with NB for ITB and H-mode edge, and 1 min. with LH in TRIAM-1M for non-reversed shear, respectively.

4. Confinement Physics

A wide range of confinement studies has been developed mainly for H−mode discharges in respect to L/H transition, confinement scaling, confinement degradation at high density, edge pedestal structure, non-dimensional transport study, and transport modelings. Core transport was extensively interested with regard to the internal transport barrier formation. Strong radiative mantle demonstrated a remarkable confinement improvement at a high density.

4-1. L/H Transition

The L/H transition study in the JET DT experiments showed that the H-mode threshold power decreased with increasing isotope mass as A^{-1} and could be scaled as P_{th} $n_e^{0.75}BR^2A^{-1}$. It was suggested in Alcator C-Mod that the LH transition could occur with electromagnetic suppression of edge turbulence via self-generated ExB flow combined with Bx B diamagnetic flow.

4-2. H-mode Scaling

The H-mode confinement scalings have been intensively investigated in the ITER physics activities. Pedestal and core plasma energies were separately studied in JET. The pedestal energy scales with mass ($A^{0.5}$) as if edge is at ideal ballooning limit and the core energy scales as gyro-Bohm mass ($A^{-0.2}$). Consequently the overall confinement scaling has no mass dependence in JET.

4-3. Confinement at High Density

The density limit is one of critical features of tokamak operation. The DIII-D experiments indicated that the MARFE instability condition yielded an edge density limit scaling which was very similar to the Greenwald scaling. It was also predicted that higher edge temperatures in future devices could allow access to densities well above the Greenwald limit.

Pellet fueling with divertor pumping allowed discharges with a ELM-free H-mode confinement ($\sim 1.2 \mathrm{x} \tau^{ITER93H}$) at $n_e \geq 1.5 \mathrm{x} n_{GW}$ in DIII-D. Strong degradations of H-mode confinement were observed at $n_e/n_{GW}>0.7\text{-}0.8$ in JET, ASDEX-U and JFT-2M, while high-field-side pellet injection recovered H-mode confinement at $n_e/n_{GW}>1$ but only with low heating power (ASDEX-U).

A radiative improved (RI) mode was demonstrated and sustained stationarily in TEXTOR, where the core plasma was surrounded by a cold radiative mantle formed by introducing neon during the auxiliary heating phase and resulted in a highly peaked density profile. The confinement time of radiative mode discharges was improved linearly above the H-mode confinement with an electron density up to $1.1 \mathrm{x} n_{GW}$. It was shown in DIII-D that neon or argon injection resulted in dramatic reduction in high wave number density fluctuations and thus improved confinement due to the reduction of edge pressure gradient as well as of edge bootstrap current.

4-4. Edge Pedestal

Edge pedestal structure affects both core confinement and divertor performance. The edge pedestal pressure could determine the confinement performance as $H \ (p_e^{PED})^{2/3}$ in DIII-D. The p_e^{PED} could be larger by at least a factor two than the high-n ballooning limit because the bootstrap current opens up a ballooning second stability region and rather be limited by lower-n MHD modes. The pedestal width Δ was scaled as a linear proportion of poloidal larmor radius ρ_{pi} of thermal or superthermal ions (JT-60, JET) or the 0.4 powered of pedestal poloidal beta $(\beta_p^{ped})^{0.4}$ (DIII-D, C-MOD). The higher triangularity was effective to get a wider width Δ with a higher pressure p_e^{PED} probably due to the improved pedestal stability against ballooning or low-n kink modes.

4-5. Core Transport

It was confirmed in many tokamaks that the E_r shear stabilization of drift turbulences played the key role for the confinement improvement of core plasmas in H-mode, high-β_p mode, super shot and reversed shear operations. The ion (and electron in some experiments) thermal diffusivities in the core region were usually reduced to a level of neoclassical diffusivity in reversed shear configurations, while a flat profile of core plasmas insides the internal transport barriers was caused by a low E_r shear (JT-60U) or by a finite ion orbit effect (ASDEX-U). The internal or edge transport barrier dynamics which induced confinement bifurcations were investigated in JT-60U, TFTR and DIII-D. The ExB shear played a dominant role in the barrier formation in these experiments and the internal transport barrier in the JT-60U negative magnetic shear configuration developed rapidly as the ExB shearing rate ω_{ExB} approached the linear growth rate of drift turbulence γ_L. The density fluctuation amplitudes with a broad band spectrum increased in the formation phase of JT-60U internal transport barriers with a high shear and were turned to decrease after its formation with increase of fluctuation coherence.

The formation of electron heat transport barriers was less understood. The RTP tokamak investigated the electron heat transport by using a localized power deposition with electron cyclotron heating (ECH) and observed the formation of sequential thin barriers for electron heat transport located close to (half) integer values of q. In the meanwhile the ECH experiments in the DIII-D negative magnetic shear configuration was not successful in forming electron heat transport barriers inside the plasma but the FIR scattering measurements suggested that high k turbulence might be affecting electron transport. The internal barrier for electron heat transport was clearly formed around the shear transition layer.

4-6. Non-Dimensional Transport Experiments and Transport Modeling

Non-dimensional transport experiments and transport modeling have been conducted widely to improve the confinement expectations of ITER ELMy H-mode plasmas. Transport of ELMy H-mode plasmas has been found to scale close to gyro-Bohm from the ITER-relevant experiments with dimensionless parameters; plasma beta β, collisionality ν^*, safety factor q, and normalized gyro-radius ρ^*, while this invariance principle has not been always fulfilled at high density.

Several theoretical and (semi)empirical transport models have been bench-marked with the world-wide ITER profile database. Sufficient theoretical understanding of tokamak confinement physics requires further investigation.

5. Divertor and Boundary Physics

5-1. Semi-Closed Divertor

Optimization of divertor geometry constitutes a key element for improvement of divertor performance. Semi-closed divertor geometries have been introduced in several tokamaks. Systematic investigations of divertor geometry have been conducted on JET, reporting Mark IIGB divertor in this conference. More closed divertor geometry increased the particle exhaust rate and consequently the neutral compression factor (JET, JT-60U, ASDEX-U, JFT-2M), while the plasma purity did not depend on the configuration, probably because carbon chemically sputtered at the wall contributed significantly to the core carbon concentration. DIII-D and JT-60U experiments showed that the plasma flow induced by puff and pump was really effective for enhancing impurity exhaust, resulting in improving the main plasma purity. Efficient helium exhaust of $\tau_{He}^*/\tau_E \sim 4$ by this puff and pump scheme was reported from JT-60U and ASDEX-U. More closed geometry also enhanced radiative cooling through onset of volumetric recombination and enhanced hydrogen radiation, resulting in a reduction by a factor two of target heat load in ASDEX-U.

5-2. Long Pulse Control

Sustainment of a long discharge was pioneered by TRIAM-1M, demonstrating 2 hour long discharges in 1995, albeit modest levels of density and power (1.5×10^{18} m^{-3}, 20 kW). In this conference, TRIAM-1M reported 1 min. long discharges at 2×10^{18} m^{-3} and 100 kW, whereby control of wall saturation by particles (>20 s) and heat load was crucial. An ELMy H-mode discharge (H_{89}^{L}~1.7, Q_{DT} ~ 0.1) was sustained for 9 s in JT-60U and the RI-mode with H_{93}^{H}~ 1 for 7 s in TEXTOR, at $n_e \sim n_{GW}$ with P_{rad} ~ 0.6 P_{in} (neon seeded) at P_{in} = 2.3 MW, demonstrating compatibility of these enhanced confinement modes with a long pulse.

5-3. Density Limit

Density limit remains an important element in considering the operation regime of tokamaks, and its explanation is still open to debate. The Greenwald density limit was explained in terms of thermal stability of the edge plasma in DIII-D. The density limit was reduced after divertor closure and for higher isotope mass in JET, and central fueling by pellet injection enabled to exceed the Greenwald density in ASDEX-U and DIII-D. High-field-side pellet injection exhibited fueling efficiencies superior to low-field-side injection in ASDEX-U.

5-4. Detachment Physics

Volumetric recombination is an essential mechanism for divertor detachment. In the JET divertor after closure, the onset of volumetric recombination and detachment occurred at a lower main plasma density due to increased carbon impurity production. From intensities of recombination line, particle sink of recombination was estimated to be around 0.7 of ionization source (C-MOD, DIII-D, JET). Heat flow to the divertor target was dominated by convection in the SOL transport of detached plasmas, while conduction became the major transport process in attached plasmas (DIII-D).

5-5. SOL Flow

It was demonstrated in DIII-D, JT-60,C-MOD and ASDEX-U that the SOL flow enhanced the impurities enrichment and thus high radiation loss in the divertor plasmas with sustaining low core dilution and high core plasma confinement. Mach probe measurements showed flow reversal at midplane with ion grad-B drift toward divertor in JT-60U, and also in private region in JT-60U and DIII-D. Measurements of deuterium and impurity flows were supported by UEDGE modeling in DIII-D.

5-6. Edge Database

A large-scale multi-machine database has been compiled and analysed for the first time by the ITER Divertor Modelling and Database Expert Group. This database enabled scaling studies for ITER extrapolation, which suggests that target heat load is below the engineering limit during OH, L- and H-mode between ELMs, but type-I ELM power load is more than 5 times the target ablation threshold.

5-7. Erosion and Codeposition

Erosion and codeposition remain an important issue in a reactor design from the viewpoint of divertor lifetime and tritium inventory. Measurements of erosion and codeposition were reported in ASDEX-U and DIII-D, and the results were compared with code estimation. Characteristics of chemical sputtering yield were presented by ASDEX-U and JET groups. JET experiments indicated that about 17 % of co-deposited tritium was unrecoverable, which poses a serious issue for ITER.

6. MHD and Disruptions

To establish the scientific basis for a steady-state fusion power plant with high-β and high bootstrap current, the MHD stability of advanced mode discharges has been studied extensively in a large number of tokamaks, where the profile and shape control is essential for improving stability. Extended databases of the MHD stability boundaries and disruptions have provided appropriate design guidelines for ITER.

6-1. MHD Activities around Transport Barriers

(1) Internal Transport Barriers (ITBs)

The ITBs formed in the reversed magnetic shear mode or the weak magnetic shear mode have been studied extensively in recent tokamak experiments. A large pressure gradient in the ITB layer destabilized low-n kink-ballooning modes and induced a beta collapse at q_{min}=2 in JT-60U and ASDEX-U. The beta limit was β_N =2 at q_{min}=2 in JT-60U.

The n=1 modes localized at the ITB were observed in JT-60U and TFTR even at a medium β. Appearance of the instability depends on the local pressure gradient, the local magnetic shear and the proximity of rational surfaces. In order to avoid excessive pressure gradients which provoke global n=1 ideal pressure driven kink modes, the heating power was controlled by a real time feedback on the neutron rate in JET and JT-60U and stable operational paths to high performances were achieved.

Double tearing modes were systematically observed in the EC heated reversed shear discharges in FTU when the q_{min} crossed the q=2 value. On the other hand, in the weak shear modes, the sustainable β was usually limited by the neoclassical tearing modes (NTMs). In the reversed shear discharges with the ITB located outside the q_{min} radius, the NTMs destabilized at the outer resonant q surface limited the β values in DIII-D.

(2) Edge Transport Barrier

The edge stability is closely related with the occurrence of ELMs, which influence core confinement and divertor performance. It was observed in DIII-D that the edge pressure was limited by the intermediate- to low-n modes, rather than by the high n-ballooning modes.

This was because the self-consistent bootstrap current provided the second stability access at the plasma edge. The high performance terminated usually by low to medium n ideal instabilities at the edge associated with a large ELM. The squareness of a plasma shape was a useful tool for controlling the edge ballooning stability by eliminating the second stability access, and at large squareness the ELM frequency was increased and the amplitude reduced.

In JT-60U, the edge pedestal pressure limit increased with triangularity in the type I ELMy H-mode. At high q_{95} and high triangularity, the type I ELMs disappeared and the type II grassy ELMs appeared.

6-2. Neoclassical Tearing Modes (NTMs)

Sustainable β in long pulse discharges has been limited basically by the NTMs, and thus the NTM onset condition is the critical factor of determining β in tokamak discharges. In ASDEX-U and JET, the critical β_p or β_N increased with the ion gyro-radius and was less dependent on the collisionality for collisionless plasmas: $\beta_{p,onset} \sim \rho_p^{*1.02}\ \nu^{-0.02}$ in ASDEX-U and $\beta_t^{crit} \sim \rho_p^{*0.5}\nu^{-0.1}$ in JET. Collisionalities have a stabilizing effect in collisional plasmas (ASDEX-U, DIII-D). The seed islands could be triggered by MHD instabilities such as sawtooth crashes, fishbones and ELMs. Sawtooth-free discharges allowed higher β than sawtoothing discharges (DIII-D, ASDEX-U, JT-60U). Stabilization of NTMs was demonstrated by ECCD (ASDEX-U) and by off-axis LHCD (COMPAS-D).

6-3. Resistive Wall Modes

Stabilization of low-n kink modes with a real finite conductivity wall is crucial for high β, steady state tokamak scenarios. DIII-D employed non-axisymmetric external coils and a new array of saddle loop detectors to stabilize the resistive wall mode, typically m=3 and n=1. Experiments were successful in sustaining β greater than the no-wall limit for durations much longer than the resistive wall penetration time.

6-4. Error Field Instability

Small levels of non-axisymmetric error field could initiate locked mode instabilities leading to disruptions. COMPASS-D, DIII-D and JET identified the scaling of the critical error field, enabling prediction of the allowable level in future large machines such as ITER. COMPASS-D and DIII-D data showed that the side band harmonics to the (m,n)=(2,1) component played an important role. These results suggested the error field thresholds of $\delta B/B = 2\mathrm{x}10^{-5} \sim 10^{-4}$ for ITER. A correction coil system for ITER with flexibility to correct sidebands, possibly assisted by beam rotation, was proposed.

6-5. Halo Current

Observations of the halo current strength in large tokamaks, JT-60U and JET, provided the substantial basis for the design of next step device such as ITER. In these devices, the ratio of the maximum halo current to the initial plasma current I_{hmax}/I_{p0} was less than 0.3 and the product I_h/I_{p0} x TPF (Toroidal Peaking Factor) was less than about 0.5. A discharge termination technique without generation of halo current was developed in JT-60U, in which the plasma column remained at a stationary position near the neutral point free from the vertical displacement events (VDE). Intense neon gas puffing during the VDE was also effective to reduce the magnitude of halo current in JT-60U.

7. Current Drive and Heatings

7-1. Non-Inductive Current Sustainment

Many attempts were made to sustain improved confinement discharges with non-inductive current drive. In JT-60U, reversed shear configurations with ITBs were sustained in an almost full non-inductive condition for 4.7 s by off-axis LHCD. Off-axis LHCD in JET helped to keep the optimized shear profile during a high performance phase for several seconds. The hot ion temperature discharge was sustained for 1 min by LHCD in TRIAM-1M. No degradation of the LH current drive efficiency was observed even at a high density ($\sim 1\times10^{20}$ m^{-3}) in FTU. Favorable scaling of the LH current drive efficiency with the volume average electron temperature was found. In a view of progress of antenna technology, 6.5 MW (ICRH+LHCD) were successfully coupled to an I_P = 1 MA plasma for 25 s in TORE SUPRA. Remote coupling of LH launchers to the H-mode edges was successful with a gap of 14 cm in JT-60U. Quasi steady-state H-mode discharges were achieved with off-axis LHCD and LHH in HT-6M

7-2. Current Profile Control (NBCD, ECCD, LHCD, IBW)

Current profile control is the key element to improve tokamak plasma performance. Central current drive was confirmed in 360 keV negative-ion-based NBI (N-NBI) experiments on JT-60U. The observed current profile agreed well with the theoretical calculation. It was confirmed that the current drive efficiency η_{CD} increased with the central electron temperature.

Hollow current profile control by off-axis ECCD was investigated on DIII-D. The theoretically predicted fall off in normalized current drive efficiency η_{CD}/T_e was not observed, provably because trapping of the heated electrons was much weaker than the theoretical predictions. Central electron temperature up to 14 keV was obtained with ECRH on the current ramp phase (hollow current profile) in FTU. On the other hand, with off-axis EC heating in TdeV, reduced particle transport was observed inside the power deposition zone, while no energy transport barrier was obtained.

Sawtooth suppression was achieved by counter-directed ECCD in WT-3 and disruption prevention was observed with ECH in T-10. Localized current drive/heating on neoclassical islands were effective in suppression of NTMs with modulated ECCD/ECRH in ASDEX-U. Optimized combination of ECRH and off-axis LHCD was effective in suppressing the large m=2, n=1 NTMs in COMPASS-D. Improved confinement was observed with current profile modification at edge by LHCD in HT-7 and with LHCD assisted E_r formation in HL-1M.

Sheared poloidal flow with direct-launch IBW (~ 360 kW) was observed in TFTR, but its magnitude was not sufficient to trigger transport barrier formation.

7-3. ICRH in DT Plasmas

ICRF heating scenarios for D-T plasmas were tested in JET. The fundamental resonance heating of D-minority(9%) in tritium plasmas produced 1.66 MW of fusion power at Q=0.22 with 6 MW of RF power and sustained a steady-state discharge. Second harmonic tritium heating of a 50:50 D:T plasma resulted in the dominant coupling of the RF power to electrons, and strong bulk ion heating was observed with addition of 5-10 % of ^{3}He. .

8. Concept Innovation -Spherical Tokamaks

8-1. High Beta Achieved on ST

The spherical tokamak with a small tight aspect ratio scheme has been receiving an increasing interest. The START device has an aspect ratio of 1.2 and the natural elongation of 2. The beta value in START recorded 0.4, which was over 3 times more than the highest value ever achieved in a conventional tokamak. A confinement improvement beyond L-mode was predicted in the experiments and further investigation was expected for confirmation.

8-2. Growing ST Families

The success in START has encouraged new initiatives in the world and more than ten of ST programs have been initiated or have been in considerations. These new machines will explore the key features of ST plasmas such as extensive operation regime of high elongation, high beta and longer discharge assisted with non-inductive current drive, improved confinement, and resilience to major disruptions.

9. Conclusions

Typical aspects of tokamak research progress can be listed as; the extension of plasma performance on large tokamak devices, extensive approach to the advanced tokamak operations with internal and/or edge transport barriers, optimization of divertor geometries and performance, suppression of MHD activities with profile control, and finally the success in the ST configuration to achieve a very high beta.

(1) JET achieved a high fusion power of 16.1 MW in an ELM-free hot ion H-mode DT plasma and the equivalent fusion energy gain of a DD plasma in JT-60 reached 1.25 with a reversed shear configuration.

(2) Advanced tokamak operation with internal and/or edge transport barriers was explored in many tokamak experiments with monotonic weak magnetic shear and negative shear configurations.

(3) Divertor geometry optimization has been one of the critical issues of tokamak research and various attempts have been done to improve the divertor performance. Further progress would be expected for satisfactory divertor functions.

(4) MHD activities have been investigated in advanced tokamak configurations. Current and pressure profile control as well as the shape control is substantial to stabilize these activities.

(5) Remarkable achievement of beta value up to 40 % was achieved in the START experiments, which explored a new trend of tokamak concept innovation.

The present tokamak research has shown a significant progress toward the next generation program such as the reduced size ITER and the future tokamak fusion power plant. The experimental database achieved till now are important but further progress should be expected to sustained high performance burning plasmas for a long time for a stable power production.