Stability, Divertors and Innovative Concepts

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Abstract: This paper contains a short resume of the sections on "Stability, Divertors and Innovative Concepts" presented at the 19th IAEA Fusion Energy Conference. The main conclusions are: (1) the problem of type I ELMs in tokamaks seems to be not so dramatic; (2) it was demonstrated that the working pulse length of large thermonuclear devices can achieve 100 s and more; (3) the problem of tritium retention seems to be not so dramatic now; probable approaches of its solution are visible; (4) active methods of plasma instabilities suppression (NTM, RWM, sawteeth, external MHD) work successfully; (5) new methods of mitigation of the disruption consequences were offered. New technological ideas and new ideas on magnetic confinement were presented.

Introduction

About 70 reports were submitted in the frame of "Stability, Divertors and Innovative Concepts". A significant part of them was directly or indirectly devoted to the ITER project. Part of the reports contained offers of its improvement. Thus the stimulating and combining role of ITER is obvious.

Divertor and stability problems are the most critical issues of the ITER project. These important issues are common for tokamaks, stellarators, reverse field pinches (RFP) and in some aspects even for opened magnetic traps.

For example, the L-H transition or the storage energy saturation generated by the growth of m=2/n=3 and m=2/n=2 modes in stellarators (Toi, EX/S3-2, LHD) look similar to the corresponding events in tokamaks. But the spectrum of Alfven eigenmodes (AE) seems more rich in stellarators compared to present tokamaks and maybe can be used for predictions of the AE behavior in tokamak reactors. On the other hand, the methods of tokamak wall conditioning and gas fuelling may be used for stellarators and RFP, too.

The first problem of the "Divertors" section, which calls today common interest, is a conflict between the basic ITER scenario, which supposes an H regime with type I ELMs, and acceptable thermal loads on the divertor plates. Type I ELMs are normally characterized by the best performing H modes, but also by pulsed heat flows, which could exceed the ablation limit and in certain cases drastically reduce the divertor lifetime in ITER to an unacceptable level. The H ELMing problem was the topic of 10 reports.

15 reports were devoted to the problem closely related to the first one, namely boundary plasma, transport in the scrape-off layer (SOL) and biasing.

11 reports were dedicated to the problems of the first wall and divertor plates, fuelling, particle balance and tritium retention. It should be noted that these problems are common for all stationary reactor projects.

The reports of the "Stability" section can be divided into three groups:

1. MHD stability and characterization (neoclassical tearing modes-NTM, resistive wall modes-RWM and other MHD modes): - 14 reports.
2. MHD control and feed back stabilization: - 8 reports.
3. Tokamak disruptions (characterization and mitigation): - 4 reports.

And finally, the "Innovation Concepts": - 13 reports.

This separation is very rough, but it permits to make a primary classification of the presented reports and a short summary of the last results.

Divertors

ELM Characterization

It is known, that type I ELMs have a character of protuberances, pulled out from the plasma boundary. Their radial speed may achieve 500 m/s, while the displacement from the boundary may be up to 20 cm (Counsell, EX/D1-2, MAST). The appropriate magnetic perturbations are of helical geometry with $m/n \approx q(a) \approx 3$. High level harmonics with $n > 1$ (Koslovski, EX/P1-14, JET) and observations of the density collapse (Oyama, EX/S1-1, JT-60) allows to assume, that they are localized along the torus mainly at the low-field side mid-plane at least at the initial phase of their development.

The total plasma power losses ΔW during the development of type I ELMs may achieve 25% of the pedestal energy (W_{ped}) being reduced in accordance with the growth rate ν^* (i.e. in accordance with the growth of the plasma density) up to 1% approximately as $\nu^{*(-0.5)}$. Usually this decrease is followed by plasma confinement degradation. The power flux initiated by type I ELMs on the Plasma Facing Components (PFC) and the divertor plates, has a mixed character: convective and conductive (connected with an electronic thermal conduction along the torus; Labombard, EX/D2-1, Alcator C-Mod, and Leonard, EX/P3-06, DIII-D). The inconsistent character of the ratio between them in different tokamaks might reflect the different conditions in the experiments, in particular, the different ratio between the radial displacement of the temperature perturbation during ELMs and the length of the magnetic force line up to the divertor.

The nature of this instability is not clear yet. Most probably, it is connected with the density or pressure gradient. It could also be the conventional ideal kink instability. However, it was demonstrated that it begins more deeply in the pedestal zone than in the n_e gradient (Leonard, EX/P3-06, DIII-D, and Loarte, EX/P1-08, JET). This instability might be initiated also by the bootstrap current. As theory predicts (Snyder, TH/3-1), the development of ideal kink modes with $n > 1$ (peeling mode) can only be possible if they arise close to the pedestal boundary. It can be supposed today, that the most probable mechanism of the type I ELM development in tokamaks should be a two-folded effect of ballooning and high-n peeling modes.

It is important to note that the development of type I ELMs precedes the clearly visible broad bend oscillations and the low-frequency helical precursors of different duration. In two cases, (DIII-D, ASDEX-U) it was possible to realize an ELM-less quiescent H mode operation regime with quasi-stationary, low-frequency perturbations at the plasma boundary. This operation mode demonstrates high quality plasma confinement but, also, as it usually happens in tokamaks, high impurity penetration to the center. This phenomenon requires further analysis. We expect that these investigations will allow further clarification of the physical ELM origin.

The absolute thermal loads during type I ELM developments can achieve from 100 up to 500 MW/m^2 in JET (Matthews, EX/D1-1, JET). In ITER they can be several times higher. The expected ELM energy fluxes on the ITER divertor are close to marginal values for an acceptable divertor lifetime (Loarte, EX/P1-08, JET). However, a model for the ELM energy loss that allows a fully physics-based extrapolation of present tokamak results to ITER is absent. Therefore, the development and validation of theoretical or semi-empirical models for the ELM energy losses, which can be used with confidence to extrapolate present experimental results to ITER, are very important issues of future investigations.

The ELMs Mitigation Problem

The first way for a solution is the search for a new ITER baseline scenario without H ELMing regimes. Two alternative ideas were suggested (Luce, EX/P3-13, DIII-D, and Wade, EX/P3-16, DIII-D).

The first one assumes $q_{95} > 4$ and $q_{min} > 1$. Under these conditions, $\beta_N H_{98}=7$ has been achieved in long discharges of DIII-D (8 sec $=35\tau_E$) without visible H ELMing.

The second one offers a scenario with negative magnetic shear ($q(0) > q_{min}$) which was also realized in DIII-D with $\beta_N=4.1$ and $H_{89}=3$. Though the observed regime duration was only about 0.6 s, the authors are convinced, that with electron cyclotron current drive (ECCD) they will achieve stationary conditions.

The disadvantage of both scenarios is a rather small practical experience of their realization in tokamaks.

The advantage is that they can be tested in ITER without major modifications of the basic installations. As a result, they can be easily included in the ITER experimental program.

The second way of ELM-problem solution is the weakening of their consequences in the frame of baseline scenarios by the transition from type I ELMs to type II ELMs or to type III ELMs with softer peak power loads but without significant confinement degradation of the main plasma.

Appropriate experiments were performed practically in all ITER-like tokamaks. For example, ASDEX-U offers “benign” type II ELM regimes close to double-null configuration with $\beta_N=3.5$ and $H_{89\text{-}P}=1.3$ in steady state regimes at plasma densities of 90 % of the Greenwald limit n_G and $q_{95} > 3.5$.

Several methods of ELM softening were tested in different tokamak experiments. In particular:

- by impurity seeding (Rapp, EX/P1-09, JET),
- by frequent pellet injection (Gruber, EX/C2-1, ASDEX-U)
- by triangularity increase (Kamada, EX/P2-04, JT-60U)
- by application of different methods of heating,
- and, at least, by the variation of special parameters such as “the closeness of the active and passive separatrices which have to be kept well bellow 2 cm in the torus mid-plane”. (Gruber, EX/C2-1, ASDEX-U)

As a result, the ELM power load can be reduced to a level of less than 5 MW/m^2 without significant confinement degradation.

SOL Investigations

Scrape-off layer (SOL) investigations present a traditional issue of scientific activities for a number of years. At the conference several new interesting experimental results were presented.

In the report of Labombard (EX/D2-1), performed on Alcator C-Mod, it was obviously demonstrated that as soon as the electron density n_e approaches n_G, the convective energy flux across the separatrix becomes dominating, while the conductive energy flux to the divertor becomes small. This transition, probably, underlies a density limitation.

Thus it was manifested, that in the far SOL the transport fluxes exhibit clear "non-diffusive events". These intermittent features (so called "blobs") were investigated not only in Alcator C-Mod, but in DIII-D and NSTX, too (Terry, EX/P5-10). The first attempt of their simulation has met difficulties connected to the necessity of a transition to a 3D model.

The following bright phenomenon – "flow reversal", the SOL flow away from the divertor – was observed both at the high field side (HFS) and the low field side (LFS) near the separatrix in JT-60U (Asakura, EX/D1-3). In a geometry with the ion grad-B drift direction towards the divertor, net particle fluxes to the LFS and HFS divertors were observed, but the SOL flow away from the divertor ("flow reversal") has been generally observed at the LFS SOL near the separatrix. This event was modeled by using the UEDGE code and should be taken into account during the particle drift calculation.

Active Control of SOL Processes

A significant group of reports, presented at the conference, was devoted to questions of active effects on turbulent processes in the SOL (Gunn, EX/P1-06; Stöckel, EX/P1-11, CASTOR; Khorshid, EX/P1-07 and CT-6Rb; Silva, EX/P1-10; ISTTOK; Xu Guoshen, EX/P3-08, HT-7; and Ivanov, EX/P5-12, GDT).

For example, the edge turbulence in the CASTOR tokamak was modified by switching on the resulting sheared radial electrical field. As biasing (f < 20KHz) was applied to a poloidal ring of 32 electrodes, the turbulent flux was reduced due to a strong reduction of the cross-phase between the density and the poloidal electric field fluctuations. The alternating voltage (AV) biasing was six (!) times more efficient than applying a static electrical field.

The modification of the boundary plasma behavior by ion Bernstein wave heating (IBWH) was performed on the HT-7 tokamak. (Xu Guoshen, EX/P3-08). The electrostatic fluctuations were nearly completely de-correlated in the high frequency region.

The biasing in the GDT mirror device (EX/P5-12) provided an almost 2 times increase of the total plasma energy.

An attempt to control the processes in the SOL by creating a stationary magnetic island (m=1/n=1) at the plasma boundary was undertaken in the LHD stellarator. The formation of a radial electrical field shear at the boundary of the magnetic island and the reduction of the heat transport inside the island were discovered. This information can be used in the process of island divertor creation.

The island divertor was used successfully in the W7-AS stellarator (Brakel, EX/C5-4). Compared with the previous limiter configuration, the operational space of the device has been substantially extended with respect to pulse length, density and heating power.

Thus, the ongoing attempts of SOL process control were successfully continued. They gradually expand the spectrum of our opportunities.

Steady State Regimes

The development of steady state fusion reactor concepts attracts a very wide circle of researchers. These problems will arise immediately after solving the ignition problem. First of all, it will be the D-T-fuelling problem and tritium retention.

Today only two large fusion devices, LHD and Tore Supra, can work more than 100 seconds in practically stationary conditions.

What new results were presented at the Conference? First of all, a new way of molecular hydrogen injection by supersonic molecular jets (Mitteau, EX/D1-5, Tore Supra and Yao Lianghua, EX/P4-08, HL-1M) should be mentioned. The injection speed achieved exceeds 1 km/s and can compete with pellet injection. But it is necessary to take into account that some parts of the injected atoms act as clusters in the plasma containing up to 1000 atoms of hydrogen with unit electrical charge. In future they might be accelerated by static electrical fields and be competitive to neutral beam injection (NBI). Probably we are at present at the birth of a new fuelling technique.

Tritium capture by products of the first wall erosion is a problem, which calls more and more attention in accordance with the motion to a steady state reactor.

The experiments on ASDEX-U with tungsten walls (Rohde, EX/D1-4) look very promising. It is important that electron cyclotron (EC) heating of the plasma center according to neoclassical predictions prevent tungsten penetration to the center. It might be possible, that tungsten as divertor plate material will be one of the solutions of the tritium capture problem.

Another possible solution of this problem may be the use of lithium as plasma facing component (PFC). As has been shown recently by our colleagues from Japan and USA (15th Plasma Surface Interaction-PSI Conference 2002), lithium heating up to 500^0C only will be enough for hydrogen isotopes release from lithium. The first successful lithium test as tokamak limiter material was made in USA and Russia (Kaita, EX/P4-19, CDXU; Evtikhin, EX/P1-17, T-11M). A pleasant surprise was that down to a temperature of 500^0C, lithium does not demonstrate any anomalous mechanisms of erosion except usual ion sputtering by hydrogen or lithium plasmas.

Stability

MHD- Stability, Characterization

Experimental research of the plasma stability has received additional impulses during the last two years after the start-up of LHD (Motojima, OV/1-6, and Toi, EX/S3-2), the National Spherical Torus Experiment NSTX (Synakowski, OV/2-2, and Menard, EX/S1-5) and the Mega Ampere Spherical Torus MAST (Lloyd, OV/2-3, and Buttery, EX/S1-6). The magnetic geometry of these systems remains topologically similar to tokamaks, however they differ essentially from them. Stellarators, for example, allow basically to divide pressure driven modes from the current driven modes. The comparison between tokamaks and these systems is only initiated, but already today in LHD, for example, it is possible to observe the development of the well known tokamak helical perturbations m=2/n=1 or m=3/n=2, and also the toroidal Alfven eigenmodes (TAE) excited during NBI heating.

On another hand, spherical tori (ST) configurations differ from conventional tokamaks by high q close to the plasma boundary and a wide area of q(r) = 1 localization inside the plasma column. In particular, the sawtooth dynamics have a clear visible distinction from usual tokamaks. For example, it was observed that a q=1 snake in MAST can exist through a train of sawteeth, indicating, that full Kadomtsev reconnection is not triggered at the sawtooth collapse (Buttery, EX/S1-6, MAST).

The Reversed Field Pinch (RFP) community concentrated strong efforts on the task dedicated to the control of magnetic turbulences. The self organized quasi single helicity (QSH) states with dominant $m=1/n=n_D$ mode and improved plasma confinement have been found and investigated in several experiments (Martin, EX/C1-2, RFX, MST and TPE-RX). It might be made parallel between QSH states in the RFP, tokamak and ST states with long-living m=1/n=1 “hot snakes” (Buttery, EX/S1-6, for example).

Besides high β_n, the spherical torus allows to observe more precisely than in usual tokamaks the neo-classical tearing modes (NTMs) development and their seeding by sawteeth. NTMs were identified in MAST and NSTX. It is expected, that the development of these investigations will expand our present knowledge of the nature of MHD instabilities.

The dynamics of NTMs development remain one of the most important practical issues for tokamaks. At the conference, new results (Hender, EX/S1-2, JET) about stabilization and destabilization of the sawteeth-activities by NBI, ion cyclotron resonance heating (ICRH) and ion cyclotron current drive (ICCD) were presented. The last results can be very important for ITER. It was found that sawteeth are destabilized when ECCD is driven exactly in and outside of the sawtooth inversion radius. The sawteeth producing the NTM seed islands are reduced and the β_n threshold for m=3/n=2 NTMs is increased. Similar results on sawteeth activity control by ECRH and ECCD were submitted by the TCV and ASDEX-U (Goodman, EX/P5-11) groups.

A new NTM regime, called FIR (Frequently Interrupted Regime) was observed in ASDEX-U (Guenter, EX/S1-4). The nonlinear coupling of the m/n NTM to a m+1/n+1 via m=1/n=1 mode activity results in an NTM – a small disruption and a clear visible increase of β_n. That is a very important example of an MHD modes interplay that can be used for tokamak optimization.

MHD Control and Feed Back Stabilization

In the tokamaks ASDEX-U (Guenter, EX/S1-4) and DIII-D (La Haye, EX/S1-3), the known experiments on active suppression of NTM modes by ECH and ECCD were continued, with the result of an appreciable β_n growth. It has been shown in ASDEX-U that non-modulated current drive is as effective as the modulated one. But in ITER, where the critical seed islands size should be much smaller than in present experiments, complete NTM stabilization could require modulated current drive (EX/S1-4). That is an important prediction for ITER.

Significant successes were reached recently by feed back stabilization of resistive wall modes (RWMs). First of all, that is DIII-D team success (Strait, EX/S2-1). Direct feedback control of RWMs by the use of internal sensors (magnetic probes placed in the chamber) and external correction coils permitted to improve the plasma stability in DIII-D. The ideal wall stabilized beta limit has been reached at approximately 1.5-2 of the free-boundary limit. These results open a new regime of tokamak operation scenarios above the free-boundary stability limit.

To the RWM problem closely adjoin: 1) the problem of error fields compensation and
2) the problem of locked mode (LM) avoidance.

Apart from DIII-D, these problems were investigated in NSTX (Sabbagh, EX/S2-2), MAST (Buttery, EX/S1-6), JET (Hender, EX/S1-2) and T-10 (Ivanov, EX/S2-3).

Basic ways of LM avoidance are removal of the error field origin, error field compensation or creation of shear of plasma rotation.

For example, the removal of some sources of error fields permitted an almost twofold extension of the pulse length in NSTX (Sabbagh, EX/S2-2).

The new method of LM control by creation of halo currents between the rail limiter and the wall was offered and realized on T-10 (Ivanov, EX/S2-3). It appeared that alternating currents (AC) can drive the frequency of m=2/n=1 mode (effect of "capture of frequency"). It seems to be possible, that this effect will further be used for monitoring MHD activities near the plasma boundary.

Tokamak Disruptions: Characterization and Mitigation

The final aims of disruption investigations are their prediction and mitigation of their consequences.

Very important results from JET were presented at the conference (Pamela, OV/1-4). Recent analyses show that only a few percent of the thermal and the thermal plus magnetic energy can be accounted for in the JET divertor and it is spread rather uniformly. If the same applies in ITER, then the energy density within the divertor would be below the level at which carbon ablation or tungsten melt layer formation are expected.

The report of Mirnov, EX/P1-15, was concerned to the physical mechanism of disruptions. The last experimental results – loss of helical symmetry of MHD perturbations during major disruptions and rapid impurity penetration to the plasma center – give the crucial argument pro a fast convective mechanism of disruptions, it could be, for example, the helical instability generated by the Kadomtsev-Zakharov model. The reason of the driving forces, expanded by primary MHD perturbations, could be their displacement relative to the resonance magnetic surfaces. This mechanism might be reversed in principal for the injection of a poloidal magnetic flux to the plasma column (reversed disruption). It should be a good test for a disruption model.

The direct observations of runaway electrons generation near to the X point of a magnetic island was seen during sawtooth crashes (Savrukhin, EX/P1-16) in T-10. Probably they are a group of primary electrons, which can be a seed for further runaway avalanches followed by major disruptions (Helander, TH/8-1).

Disruptions generate large mechanical stresses on tokamak components. The problem of halo currents, which are the major cause of mechanical stress on the machine during disruptions was analyzed in two reports, Pautasso, EX/P4-14, and Nakamura, EX/P4-13. The main practical recommendation is to first choose a magnetic configuration with a neutral point, which is known as the advantageous vertical plasma position for VDE (vertical displacement events) avoidance, and secondly, the injection of impurity gas or of impurity pellets during VDE developments.

Very interesting and perspective methods of disruption mitigation using high pressure noble gas injection were suggested and tested in DIII-D (Whyte, EX/S2-4). As a result, the plasma energy is dissipated uniformly by UV radiation from the injected impurity species to the entire wall. The conducted heat flux to the divertor was found to be only 2%-3% of the storage kinetic energy of the plasma compared to 20%-40% found for non-mitigated disruptions. The runaway electrons were suppressed in contrast to mitigation attempts with argon cryogenic pellet injection. The mechanical loading due to halo current was greatly reduced, too. This method can be used in ITER and future tokamak reactors.

Innovative Concepts

The papers, presented at this section, can be divided in five themes:

1. Improving of stellarator concepts: Nührenberg, IC/P-06; Lyon, IC/P-05; Okamura, IC/P-07; and Yokoyama, IC/P-08. All suggestions are interesting, but all of them require experimental testing.
2. The next theme was devoted to different ideas of poloidal magnetic flux injection. These are: Sinman, IC/ P-01; Jarboe, IC/P-10; and Farengo, IC/P-14. The main issue of all suggestions is very actual. However the present ideas are based on magnetic reconnection, which contradicts with the concept of high plasma confinement. Due to this disadvantage we consider these ideas as interesting physical suggestions only.
3. The idea to use stochastic magnetic fields for the control of plasma transport is not very new, but it seems to be very perspective. Two papers were devoted to this problem: Abdulaev, IC/P-02 and Grossman, IC/P-03.
4. Kotschenreuter suggested the idea of a bundle-like poloidal divertor. This idea needs careful technical analysis.
5. Other reports of this section deal with particular questions of physical experiments.

Conclusions

A lot of new information was presented at the conference making the idea of a fusion reactor more and more realistic:

- The problem of type I ELMs in tokamaks seems not to be so dramatic.
- It was demonstrated that the working pulse of large thermonuclear devices can achieve 100 s and more.
- The problem of tritium retention seems not to be so dramatic now. Probable approaches of its solution are visible.
- Active methods of plasma instabilities suppression (NTMs, RWMs, sawteeth, external MHD) work successfully.
- New methods of mitigation of the disruption consequences are offered.
- New technological ideas and new ideas on magnetic confinement have appeared.

Thus, the 19th IAEA Fusion Energy Conference has continued the tradition of the previous conferences on plasma physics and nuclear fusion in improving the understanding of thermonuclear plasmas.