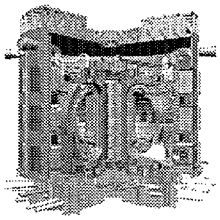

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ITER CO-ORDINATED TECHNICAL ACTIVITIES

As agreed upon between the ITER Engineering Design Activities (EDA) Parties at their Meeting in Tokyo in December 2000, "Co-ordinated Technical Activities" (CTA) means technical activities which are deemed necessary to maintain the integrity of the international project, so as to prepare for ITER Joint Implementation.

Such activities will build on the results of the ITER EDA and be conducted considering specific conditions of the site(s) offered. They will result in technical documents in support of negotiations towards the ITER Joint Implementation, to enable the project to move smoothly towards construction.

Participants in the CTA are the Parties who had been involved in the ITER Engineering Design Activities (EDA), as well as the governments of other countries that have presented to the Parties a specific construction site offer. The invitation to join this latest phase of ITER was made by the IAEA Director General Dr. ElBaradei earlier this year. Canada, the European Union, Japan, and the Russian Federation have all accepted this invitation and confirmed their participation. At this time, only Canada has offered a site to Host ITER. Other interested countries which possess relevant specific capabilities and which can contribute significantly to ITER Joint Implementation may also join under terms to be unanimously approved by the Participants.

The Participants will conduct the CTA in a co-ordinated manner, under the auspices of the International Atomic Energy Agency (IAEA) till the end of 2002. The scope of the activities includes:

- Design adaptation to the specific site(s) conditions,
- Safety analysis and licensing preparation that are based on specific site offer(s),
- Evaluation of cost and construction schedules,
- Preparation of procurement documents, and
- Other technical issues raised by the Negotiators collectively, whilst assuring the coherence of the ITER project including design control.

The organizational structure for the CTA consists of the Project Board, the International Team and the Participants' Teams.

The Project Board has executive functions to the CTA and is co-ordinating the Participants' contributions to the activities, facilitating liaison within the organizational structure. The Board consists of the Leader of the International Team and the Leader of each Participant Team. The Chair of the Board is Academician E. Velikhov. The results of the activities will be reported by the Chair to the Negotiators. In addition, the Board will respond to issues raised by the Negotiators collectively. The Board will be assisted by the International Team dedicated to the co-ordination of the technical activities, with particular emphasis on assuring the coherence of the ITER project.

Each CTA Participant will contribute staff to the International Team which, collectively, should have the capability of assisting the Project Board in the oversight of the CTA. The Team will use the work sites Garching (near Munich) and Naka (north of Tokyo).

Each Participant will establish its own Team in order to undertake activities described above as entrusted to it by the Project Board. The Team of a Participant offering a specific construction site will play a leading role in developing details of the design adaptation and in licensing preparations for its specific site.

The IAEA has played an important role in starting the CTA. As confirmed by Dr. ElBaradei, IAEA Director General, the Agency is in agreement with the CTA Terms of Reference and will provide the auspices for the CTA as well as services of the same nature as during the ITER EDA. These services essentially include the following:

- Providing meeting space for CTA Meetings held in Vienna together with the required support;
- Publishing and distributing the CTA documents and reports;
- Editing, printing and distributing the CTA Newsletter.

To support these services, the IAEA is maintaining the adequately staffed and equipped ITER Office at its Headquarters at the Vienna International Center.

MEETING OF THE ITER CTA PROJECT BOARD

A preparatory meeting of the Co-ordinated Technical Activities (CTA) Project Board took place in Vienna on 16 July 2001. The Board Members of Canada, EU, Japan, RF and of the CTA International Team participated in the Meeting, which was chaired by Acad. E. Velikhov.

The major item on the Meeting Agenda was the discussion of the scope of the CTA. In this discussion the following comments were expressed:

- One of the prime objectives during the CTA is to develop technical specifications for procurement of critical items (magnets, vacuum vessel, and buildings). It was noted that the discussions with potential suppliers should confirm manufacturing processes in details in order to explore possible schedule reduction strategies.

- Safety analysis and licensing preparation should proceed on all proposed sites up to the preferred site designation, to ensure the overall implementation schedule is minimized and to resolve major technical issues needed for licensing.

- Several R&D issues remain to be further developed during the CTA. Special attention should be given by the Participants to two areas: Diagnostics; Heating and Current Drive Systems.

- Arrangements for continuation of the ITER Physics Expert Groups activities should be provided. To this end a new framework, called International Tokamak Physics Activity, is being planned. The Board encouraged the Participants' Representatives in the Co-ordinating committee of this activity to support the preparation for urgent Topical Group Meetings.

The Board agreed that the Design Authority will be invested in the International Team and that proposals for site specific design changes should be agreed upon by the International Team Leader before being studied in detail.

The Meeting agreed on some arrangements which will remain from the EDA, namely the ITER EDA Council Office in Moscow as Office of the PB Chair, and the ITER Office located at the IAEA in Vienna as agreed by the IAEA. The Board recommended that effective August 2001 the ITER Newsletter will be published in the same way as the ITER EDA Newsletter had been done.

It has been agreed that, as a rule, the Project Board will meet every two to three months for one or two days before the Negotiators' meetings, at the same place. In line with this decision, the next meeting will be held in November in Toronto, Canada. For that meeting the International Team would prepare a preliminary Work Plan for the CTA and all Participants' Teams would provide their organization charts.

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MEETING OF THE EXPERT GROUP ON MHD, DISRUPTIONS AND PLASMA CONTROL

by Dr. M. Shimada, ITER Naka Joint Work Site

The Expert Group Meeting on MHD, Disruption and Plasma Control was held on 25–26 June 2001 in Funchal, Madeira, Portugal. The meeting followed the 28th EPS Conference on Plasma Physics and Controlled Fusion.

The main objectives of the meeting were:

- stabilization of ß limiting MHD modes in conventional and advanced tokamak scenarios (NTMs, RWMs);
- tolerable ELMs with good confinement alternate to type I ELMs;
- disruptions at $q \approx 3$; prediction, avoidance and mitigation of disruptions; related time scales;
- control issues in conventional and advanced scenarios including current drive/profile control.

A review of high priority research issues for burning plasma experiments was done at the end of the meeting.

A brief summary of the discussions at the meeting is given below.

Neoclassical Tearing Modes (NTMs)

NTMs limit plasma performance in conventional and advanced scenarios with peaked to flat current density profiles. A high confinement regime at high values of β has been found on ASDEX Upgrade ($\beta_N \geq 2.3$) and JET ($\beta_N \geq 2.1$) in spite of existence of a (3,2) NTM. The reason for confinement improvement is frequent amplitude drops due to non-linear coupling to an ideal (m+1, n+1) mode. High confinement regime with $\beta_N \geq 2$ can be also expected for ITER.

One of the typical characteristics of NTMs is the hysteresis observed between the β value at which the mode is triggered (β_{onset}) and the β value at which the mode is stabilised or below which the mode is unconditionally stable ($\beta_{crit} > \beta_{onset}$). Both values scale with rho*ion gyration radius/plasma radius, and depend only weakly on collisionality. The value of β_{onset} depends on the perturbation (size of island) triggering the NTM. Recent JET experiments have shown a direct relation between the sawtooth period and β_{onset}. Ion cyclotron current drive was used both to stabilise or destabilise sawteeth, leading to longer sawtooth period and hence lower β_{onset} in the first case, and vice versa. Ion cyclotron or electron cyclotron current drives could also be used on ITER to destabilise sawteeth. The value of β_{crit} depends only on plasma parameters, and can be established by a slow power ramp-down in presence of an excited NTM until the mode vanishes. This study of β_{crit} has been started at ASDEX Upgrade and JET. Comparisons with other machines would be very helpful for predicting β_{crit} for ITER.

Experiments for NTM stabilisation by external non-resonant helical fields have been carried out on JET and DIII-D, supporting the hypothesis that NTMs with different helicity can not exist simultaneously in ITER. The applied helical fields, however, lead to a strong drop in plasma rotation, and often to a reduced plasma pressure. This role of poloidal and toroidal plasma rotation through the polarization term on the NTM was further investigated on JT-60U.

Very similar instability behaviour is observed for m=2,n=1 NTMs in joint experiments on JET and DIII-D. Similarity experiments using closely matched non-dimensional plasma parameters (plasma shape, aspect ratio, q, ρ^*, collisionality) show similar β_{onset} values. First results from spherical tokamak MAST contribute to the database both for 3/2 and 2/1 NTMS.

Resistive Wall Modes

The feedback stabilization of the n=1 resistive wall mode (RWM) at DIII-D prolongs the duration of a state with $\beta \approx 2\beta_{no\ wall}$ for many wall times (more than 50). Thereby the role of direct stabilization of the RWM and the effect of a quasi-ideal wall by sustained toroidal rotation via reduced error fields was a matter of debate. A negative effect of ELMs, whose amplitudes and periods increase with RWM feedback, was observed, as the control coils ("C-Coils") respond on each ELM despite the quite different frequencies involved deteriorating the RWM stabilization.

Edge Localized Modes

Type II ELMy H-mode operation in ASDEX Upgrade was achieved at high plasma triangularity (δ = 0.4) close to a double null configuration. It allows to reduce and to smooth the power flux to the target plates significantly as compared to type I ELMs, even at Greenwald density and without reducing the confinement compared with ITER-H98P scaling. The operational regime for pure type II ELMy H-modes has been extended downwards q = 3.5.

Disruptions

Improved experimental characterisation of disruptions, namely quench time scales and halo current amplitudes and toroidal distribution should contribute to the establishment of adequate margins for structural integrity of ITER FEAT in extreme cases (e.g. in 17 MA scenario).

Previous JET data (presented in the ITER Physics Basis) on current quench rates during disruption is confirmed by analysis of more recent pulses. Neither the minimum current quench rate, nor the halo current fraction, seems to show any obvious change for $q_{95} < 3$. The JT-60 disruption data were also re-examined. The minimum current quench time seems to be independent of both the current density and the safety factor. The minimum current quench time in a machine with a metal wall was not significantly different from that with a graphite wall.

The product of toroidal peaking factor (TPF) and I_h/I_{po} is decreasing with the size of machine and with the actual normalised plasma current during the current quench I_p/I_{po}. Here I_p and I_h are the actual plasma and maximum halo currents, I_{po} is the plasma current before the current quench. For JET and JT-60U the product $I_h/I_p \times \mathrm{TPF} \leq 0.4$ is reduced compared with smaller devices $I_h/I_p \times \mathrm{TPF} \leq 0.6$. Influence of smaller elongations and closeness to the neutral point (verified in JT-60, AcatorC-MOD, ASDEX Upgrade, not seen in TCV) in this comparison has still to be established.

The experimental observations from several tokamaks suggest that the n = 1 halo current asymmetry, producing the TPF, originates on the q = 1 surface. This is probably the manifestation of a common MHD phenomenon during the disruption. A clear m = 1 n = 1 perturbation of the halo current is seen in many ASDEX Upgrade disruptions; which is clearly correlated with q_{cyl} dropping below 2 and with an MHD event. The local degree of asymmetry of the halo current can be relatively large (ftp = 3.5). There is clear indication that avoiding peaking of the current profile after thermal quench and reducing the region inside of the q = 1 surface, suppresses the (1,1) asymmetry.

Analysis of the DIII-D disruption database for 1998, 1999 and 2000, in current flat top only, shows an average disruptivity of 13% and is not distinctly increasing when q_{edge} approaches 3. This does not include "forced" disruptions, power supply failures, control errors etc. Comparable rates were reported from other devices during the last years.

Runaway discharges have been produced in JET by inducing disruptions in low elongation plasmas by argon injection. These data show similar lower limits in q_{edge} and B_t for runaway production as observed on JT-60U. This runaway production can be avoided by quenching the discharge by massive helium puffs.

Plasma control and current drive

Inductive 15 MA and steady-state 9 MA ITER scenarios were studied with DINA code. The code simulates feedback and feedforward plasma current, position and shape control provided by the PF system simultaneous with solution of the 1D plasma transport equations. The study demonstrated the capability of ITER PF system to support these scenarios. For the problem of AC-loss minimisation in ITER a model was developed and validated at CRPPL including effects from the controller design. There are still open points which have to be adressed in the next meetings, as location of the breakdown region compared with the region having minimum stray field or plasma disturbances in scenarios with weak negative shear.

Large ELMs can significantly perturb the vertical position observer and lead to bad control of plasma position. The vertical position observer is usually a weighted sum of the magnetic field on pick-up coils around the plasma. It has been shown on TCV and JET that slight modifications of the coefficients can avoid or strongly limit the perturbations due to large ELMs, while maintaining an accurate measure of the plasma position, thus avoiding changes in the control algorithm.

The next expert group meeting will be under the auspices of International Tokamak Physics Activities. It was proposed to be held in combination with the IEA Large Tokamak Agreement Workshop W49 (Real Time Control of ITB Discharges Approaching Steady State) and the US-Japan MHD workshop FP2-1 at JAERI, Naka, Japan, in February or March 2002. The organizers of the present meeting were complimented for their arrangements. We are especially indebted to Mrs. Maria Fernanda Pinto of IST Lisboa, Portugal, for her excellent assistance.

Participants in the meeting

List of participants

M. Shimada	ITER, Naka
Y. Gribov	ITER, Naka
Takahisa Ozeki	JAERI
D. Campbell	EFDA Garching
K. Lackner	EFDA Garching
S. Guenter	IPP Garching
O. Gruber	IPP Garching
T. Hender	UKEA Culham
J.B. Lister	ECFPL Lausanne
O. Sauter	ECFPL Lausanne
G. Pautasso	IPP Garching
N. Ivanov	Kurchatov Moscow
V. Lukash	Kurchatov Moscow
S. Mirnov	TRINITY
V. Pustovitov	Kurchatov Moscow
R. LaHaye	GA San Diego
J. Manickam	PPPL Princeton
J. Snipes	MIT Boston

Items to be considered for inclusion in the ITER CTA Newsletter should be submitted to B. Kuvshinnikov, ITER Office, IAEA, Wagramer Strasse 5, P.O. Box 100, A-1400 Vienna, Austria, or Facsimile: +43 1 2633832, or e-mail: c.basaldella@iaea.org (phone +43 1 260026392).

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