TOROIDAL FIELD MODEL COIL PROJECT *

by R. Thome, N. Mitchell (ITER Joint Central Team) and E. Salpietro, R. Maix (EU Home Team)

Coil Systems

The ITER coil systems provide the magnetic field intensity and field geometry to contain and control the plasma during various phases of pulsed operation. During these pulses the toroidal field (TF) coils operate with a constant current. The central solenoid (CS) and poloidal field (PF) coils, on the other hand, are independently powered, and each follows a current versus time scenario selected such that the combined action of the fields and flux from the CS and PF coil set is consistent with the conditions necessary for plasma growth control, burn and shutdown. Model coils are being built to develop fabrication technology for the CS and TF coils. The model coil programme supporting the CS design has already been described in the IAEA ITER Newsletter, March 1997.

ITER TF Coil System

This system will create a stationary toroidal magnetic field (5.7 T at 8.14 m major radius). The TF coil system comprises 20 coils, each consisting of a winding pack in a vacuum tight case, about 17 m tall and 12 m wide. The winding pack is pancake wound and uses a circular conductor wrapped in insulation and located in grooves in radial plates. Each turn is insulated from the enclosing plate, and the plates are insulated from each other. These radial plates transfer the forces acting on each conductor directly to the casings without accumulation of forces on the turn insulation. The winding pack is wrapped in a ground insulation, the outer surface of which is matched without bonding to the case. This allows predictable gaps to open during cooldown and to reduce winding pack shear in operation. The in-plane hoop forces of the coil are carried by the combined structure of plates and case, and the case provides most of the bending and torsional rigidity against out-of-plane forces. The coil operating conditions are summarised in Table. 1.

Table 1. TF Full Size Coil and TF Model Coil Operating Conditions

	Full Size Coil	Model Coil
Maximum Field (T)	12.8	8.5*
Operating Current (kA)	60	70
Number of Turns	192	98
Dimensions (m)	12x17	2.7x3.8
Peak Tresca stress in case (Mpa)	560	620
Peak Tresca stress in jacket (MPA)	432	495
Peak interplate shear stress (Mpa)	35	35

* Combined 7.4 T self field and field from the EURATOM LCT coil at 16 kA conductor current. For details of the LCT coil see the article by D.S. Beard, et al. "The IEA Large Coil Task", Fusion Engineering and Design, Vol. 7, September 1988.

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* This is the fifth article in a series describing the Seven Large ITER R&D Projects. For the previous articles in the series see Newsletter Vol. 5, Nos. 8 and 9, and Vol. 6, Nos. 2 and 3.

Conductor for the TF Coils

The conductor will be a Nb_3Sn cable in conduit type, force-flow-cooled with supercritical helium, and a maximum current of 60 kA. The cable, consisting of Nb_3Sn and copper strands surrounded by helium, is contained in a thin Incoloy 908 jacket.

The conductor is designed to tolerate a range of disturbances (in particular, plasma disruptions) without initiating a quench. However, in the event that a quench occurs, detection is provided by voltage measurements between the ends of a coil layer or pancake, with voltage balancing between adjacent conductors to allow inductive voltages to be eliminated. Resistive voltages can then be detected and a rapid discharge can be initiated. A backup for the primary quench detection system is provided by coolant flow meters or pressure sensors within each cooling channel, which require a few seconds longer to clearly identify the helium expulsion associated with a quench.

The copper in the cable provides a limited degree of thermal protection, but to prevent local overheating and possible coil damage, the coil affected must be discharged with a time constant of about 20 s. The voltage requirements for the insulation are driven entirely by the need for a fast discharge in the event of a quench.

Two Model Coil Programmes

The construction and test of the two (the CS and TF) model coils are necessary because of the significantly different conductor, structural support, manufacturing methods, and operational conditions for the two full-scale systems.

The CSMC (see IAEA ITER Newsletter, March 1997) and TFMC programmes will drive conductor development and demonstrate coil manufacturing feasibility based on the full-scale coil requirements. Conductor performance at full-scale operating conditions will be demonstrated with insert coils to be tested within the bore of the CSMC. In addition to the mainstream fabrication activities, there are full-scale coil component and manufacturing R&D tasks underway that are associated with determination of conductor and joint performance, insulation properties, and properties of structural material. The overall relationship of the Model Coil Projects and the associated R&D is shown in Figure 1.
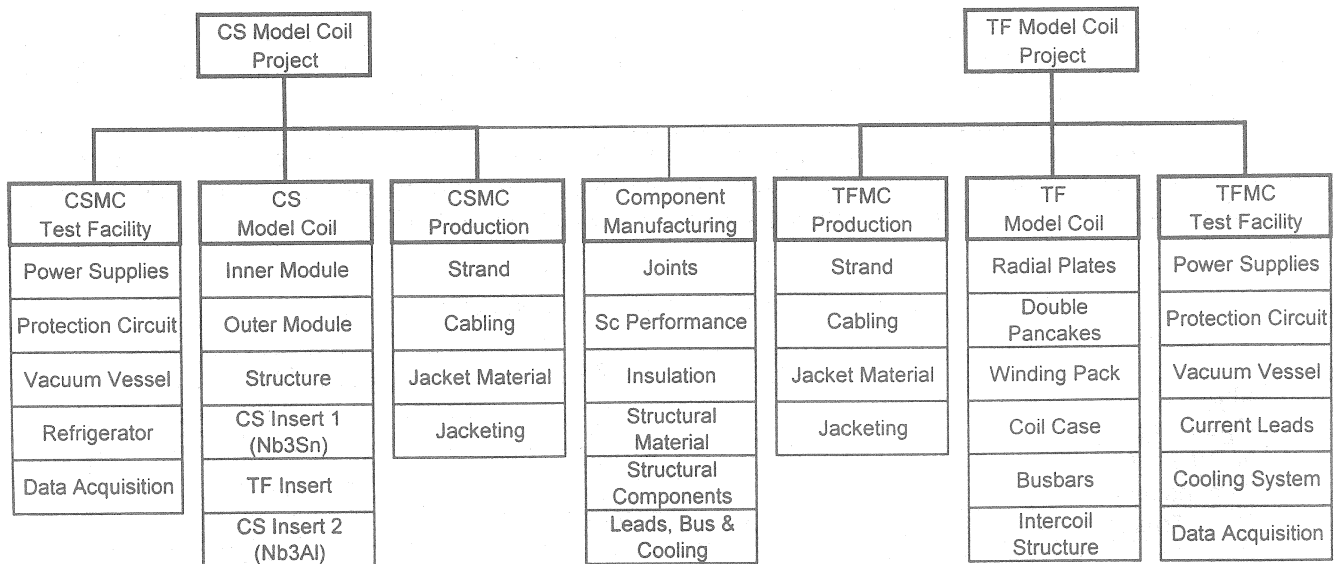
CSMC Test Facility	CS Model Coil	CSMC Production	Component Manufacturing	TFMC Production	TF Model Coil	TFMC Test Facility
Power Supplies	Inner Module	Strand	Joints	Strand	Radial Plates	Power Supplies
Protection Circuit	Outer Module	Cabling	Sc Performance	Cabling	Double Pancakes	Protection Circuit
Vacuum Vessel	Structure	Jacket Material	Insulation	Jacket Material	Winding Pack	Vacuum Vessel
Refrigerator	CS Insert 1 (Nb3Sn)	Jacketing	Structural Material	Jacketing	Coil Case	Current Leads
Data Acquisition	TF Insert		Structural Components		Busbars	Cooling System
	CS Insert 2 (Nb3Al)		Leads, Bus & Cooling		Intercoil Structure	Data Acquisition

Figure 1. Outline of Magnet R&D

The facility at the Japan Atomic Energy Research Institute (JAERI) in Naka to carry out the test of the CS model coil (CSMC) is ready. The modifications of facilities at the Forschungszentrum Karlsruhe (FzK) for the test of the TFMC are almost complete.

Objectives

The technology required to build the TF coils represents a significant advance on that existing today. Consequently, there is a TFMC programme to demonstrate the industrial feasibility of the ITER TF coils manufacturing procedure and to assess the operating margins, namely;

a) to develop and verify the full scale TF coil manufacturing techniques,
b) to establish realistic manufacturing tolerances,
c) to upgrade the production capacity for conductor,
d) to benchmark methods for the ITER TF coil acceptance test,
e) to gain information on the coil's mechanical behaviour and operating margins by testing.

The experience with all coil manufacturing processes including quality assurance (QA) methods and equipment will allow the full scale design to be modified to ease manufacture and improve the quality and reliability of the full scale coils.

It should be noted that high field testing of the ITER TF coil conductor under both steady and pulsed conditions will be a part of the high field CSMC test, with a single layer TF conductor insert being designed and built by the RFHT. The CSMC test facility gives a much more convenient configuration for conductor testing, with long lengths (up to 100 m) at fields up to 13 T.

The TFMC programme objectives are focussed on the manufacturing programme and the final testing is not the primary goal of the work. The TFMC programme concentrates on the wind, react and transfer process by performing development on:

- techniques for winding the conductor and clamping it during the Nb3Sn reaction heat treatment so that the 0.5 mm tolerance fit to the grooves can be achieved;.
- closure of the plate grooves with welded caps without distorting the plate or damaging the insulation;
- application of insulation to the conductor during transfer and epoxy penetration into the groove during impregnation;
- design of joint region to avoid local stress concentrations;
- flatness achievable on radial plates after completion of plate insulation layer;
- case manufacture (both welding assembly from plates and final tolerances);
- winding pack assembly into case (gap filling and case closure welding).

The manufacturing feasibility of the TF coil cases and radial plates will be demonstrated. The models manufactured include the key design features of the final full-size TF coil, and use similar fabrication techniques. Selection and qualification of the welded joints design, weld processes to minimise the distortion, and the QA methods are the main objectives. A major effort will be devoted to the final closure-weld joint of the case, and of the radial plate conductor cover which can be accessed from the outer side only. The welding must not damage the electrical turn insulation of the superconductors integrated into the radial plate steel structures, and electrical ground insulation of the winding pack integrated into the stainless steel case.

In several cases, development of cost effective techniques is required as well as the resolution of technical problems. Development of QA procedures that will be applicable to the full-size coils is an important consideration at all steps.

TFMC Design

The TFMC is a "racetrack" shaped coil about 3.8 m high and 2.7 m wide, which includes some of the technical features and manufacturing approaches foreseen for the full-size TF coils. The main operating conditions for the TFMC are summarized in Table 1 where they are compared with those of the full scale coils.

The TFMC conductor is shown in Figure 2 and is composed of a Nb3Sn superconducting cable, inserted in a thin jacket and compacted. It is similar to the full size conductor of the ITER TF Coils, but uses a stainless steel jacket instead of Incoloy 908. The winding pack has a similar layout to an ITER TF coil.
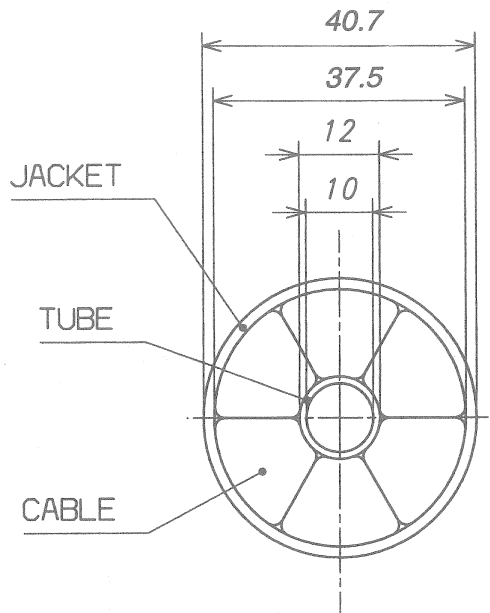
JACKET

TUBE

CABLE

40.7
37.5
12
10

Figure 2. TFMC Conductor

The TFMC consists of 5 double pancakes (D.P.'s) and 5 radial plates enclosed in a rectangular coil case. Figure 3 shows the coil cross-section in the straight leg, compared to the full-size coil cross-section. Each D.P. consists of a radial plate with machined grooves on both sides, in which the insulated round conductor is placed and then held in place by covers which are welded at the corners of the groove. The two single pancakes of a D.P. are connected by a joint at the inner circumference. The 5 D.P.'s are each wrapped with insulation impregnated with epoxy resin which fills the space around the conductor and the outer insulating layer. The D.P.'s are then stacked together, separated by an additional glass sheet, wrapped with ground insulation, impregnated with epoxy resin and connected by joints at the outer circumference. The main parameters of the TFMC are summarised in Table 2.

Table 2: Main Parameters of the TFMC

Length of conductor (m)	809 (net length in coil)
Weight of conductor (t)	6.27
Weight of Nb$_3$Sn strand produced (t)	3.9
Weight of winding pack (t)	17
Weight of coil case (t)	16

Testing of the TFMC

The TFMC will be tested in the TOSKA Test Facility at FzK. It will be coupled by means of a supporting steel structure to an existing large superconducting coil (the EURATOM LCT coil) that provides the background field to generate additional loads on the TFMC (see Figure 5).
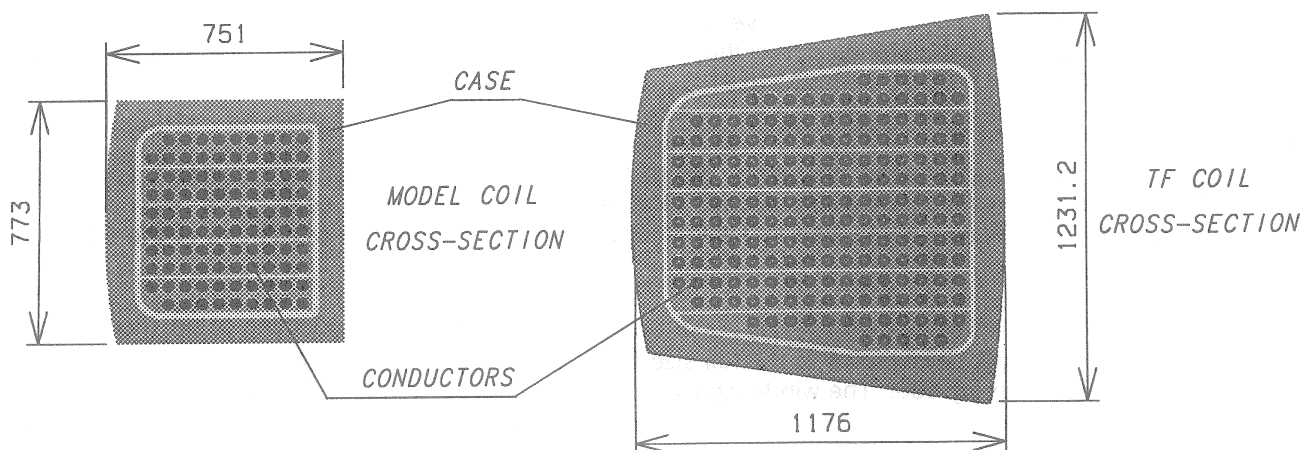

CASE

MODEL COIL
CROSS-SECTION

CONDUCTORS

TF COIL
CROSS-SECTION

751
773
1231.2
1176

Figure 3. TFMC Cross-Section Compared to the Full Size Coil
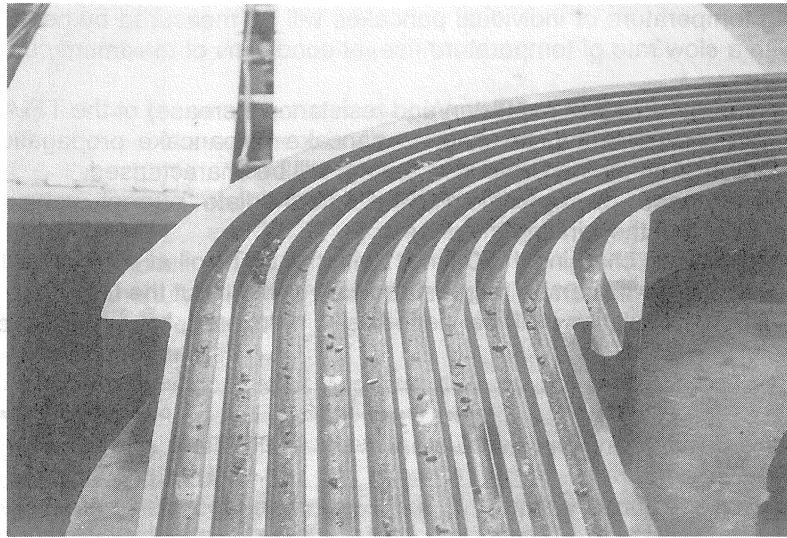
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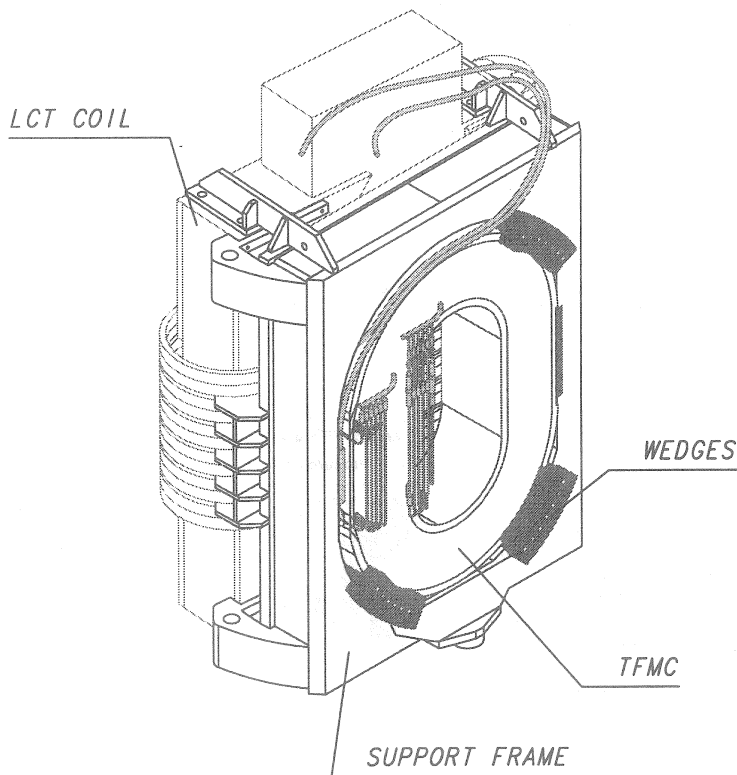*Figure 4. Machined Radial Plate (Photo: Preussag Noell, Wuerzburg, Germany)*

*Figure 5. TFMC assembled with intercoil structure
and the LCT coil*

The TFMC will interface with the LCT background coil such that the TFMC is located at the side of the LCT coil, with a small angle between the two coils (roughly the same configuration of two adjacent TF coils in the ITER tokamak). This arrangement allows for testing of the coil under independently varying field and current and permits the application of both TF coil in-plane and out-of-plane loading.

The TFMC rests inside a supporting framework and is connected to it through four U-shaped wedges at the corners that allow the TFMC to develop both in- and out-of-plane bending without transmitting extra stresses to the LCT coil. The supporting framework extends between the coils to react the forces between them.

The TOSKA test programme will contain the following major items:

- Pressure drop will be measured on every cooling circuit of the TFMC for mass flow rates from 0 to 18 g/s per channel.
- The TFMC and LCT coil will be separately and jointly charged in steps toward the maximum current of the TFMC. Changes in outlet temperature, flow rate, stress levels and displacement of the TFMC will be measured.

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- The current sharing temperature of individual pancakes will be measured by heating the inlet helium at constant current with a slow rate of temperature rise, at conditions of maximum current in the TFMC and the LCT coil.
- Quench performance (rate of helium expulsion and resistance increase) of the TFMC, with and without a delayed discharge, will be measured to examine pancake to pancake propagation and heat transfer effects from radial plates. The quench detection system will be characterised.
- The TFMC will be ramped down from high to intermediate current values and the transient electromagnetic behaviour of the winding measured.
- The TFMC will undergo cyclic charging 1000 times, with the LCT coil at a constant level of current. The DC performance parameters will then be remeasured with and without the LCT coil.
- The state of the TFMC insulation will be periodically monitored by a partial discharge diagnostic procedure.
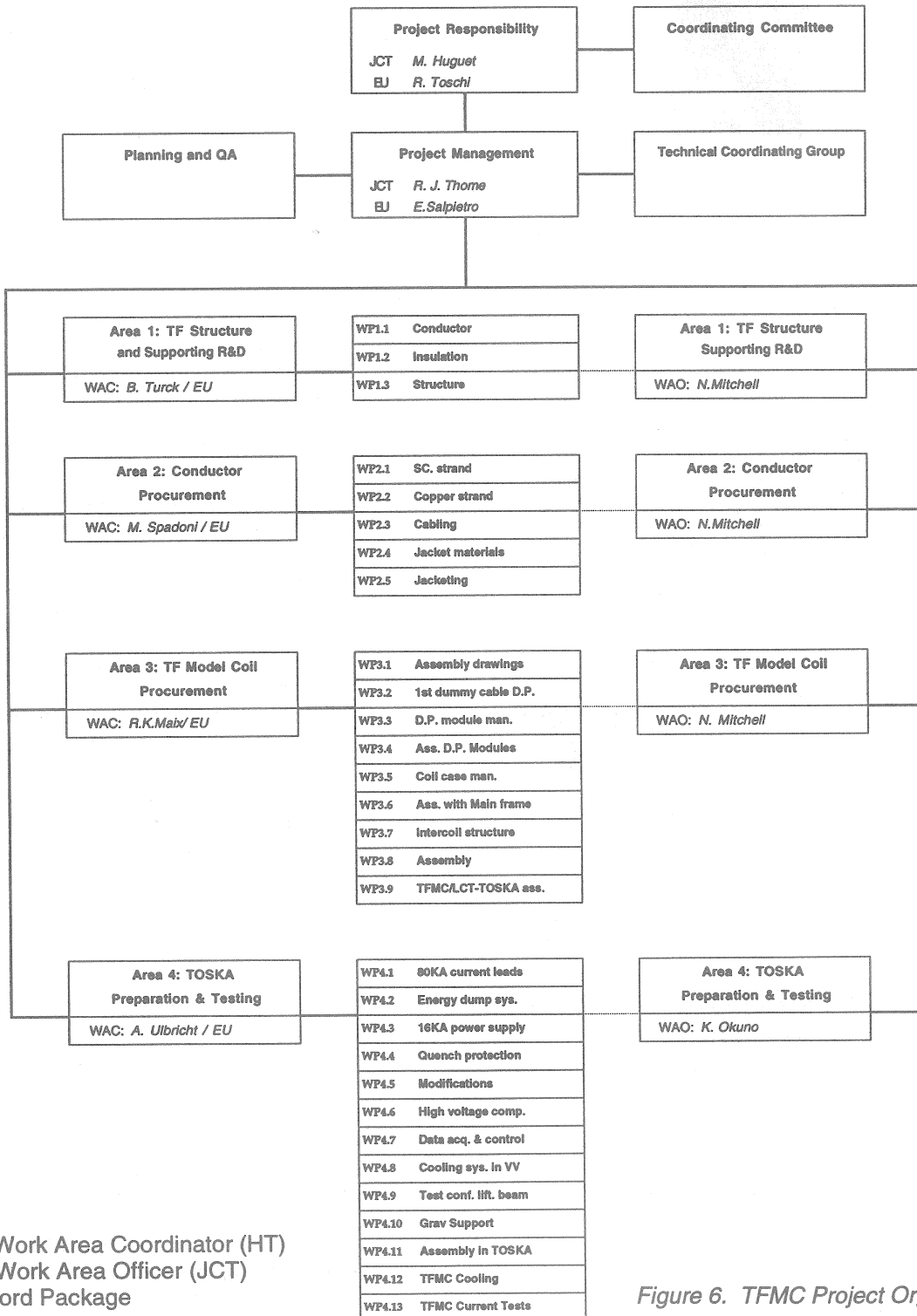
WAC = Work Area Coordinator (HT)
WAO = Work Area Officer (JCT)
WP = Word Package

Figure 6. TFMC Project Organization

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Project Management
The organization and management structure is summarized in Figure 6.

The Project Responsibility lies with the Naka Joint Work Site (JWS) Deputy Director and the EU Home Team Leader. The Project Management is also a joint responsibility of the Joint Central Team (JCT) and the EU Home Team (HT). The JCT is primarily responsible for defining the deliverables of the project, and its objectives, and the HT is primarily responsible for the implementation.

Work Status and Conclusions
- Both the detailed manufacturing drawings of the TFMC and the stress analysis of the test configurations are nearly complete.
- 85% of strand manufactured.
- About 500 m of dummy conductor and first 190 m unit length of superconductor have been finished.
- The forgings for all the radial plates have been done and the final machining of the plates and covers is underway (Figure 4).
- The winding line is operational and the part of the dummy conductor has been used to establish the winding parameters for the pancake winding (Figure 7).
- An R&D program with Incoloy jacketed TF conductor has been agreed upon in order to demonstrate the coil manufacturing process also with this material.
- A 1000 m jacketing line has been set up for an Incoloy conductor by the RFHT. A 1000 m dummy cable is available from the JAHT for a 1000 m pull-through and a compaction demonstration for the full-scale-conductor is scheduled for mid 1997.
- The TOSCA test facility modifications for the TFMC are underway at the Forschungszentrum Karlsruhe and will be ready by mid-1997. The EURATOM LCT coil has already been successfully tested at 1.8 K and over 11 T.

In addition, there has been extensive progress in supporting R&D activities on components and material properties. Effective collaboration has resulted in significant strides in developing the ITER design and establishing the R&D program that is essential in verifying design parameters.
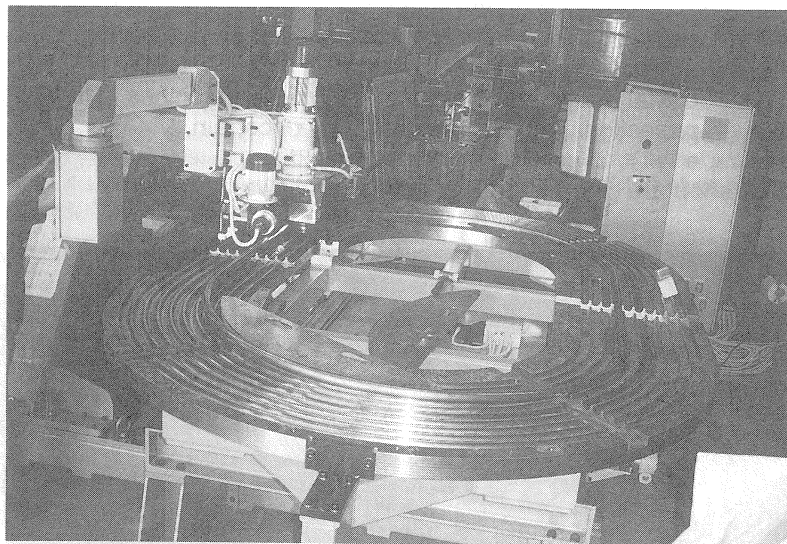

Figure 7. Winding line with reaction mould (Photo: Ansaldo, Genoa, Italy)

COMBINED WORKSHOP OF THE CONFINEMENT MODELING AND DATABASE AND CONFINEMENT AND TRANSPORT EXPERT GROUPS

by Dr. D. Boucher, ITER JCT, Dr. J.G. Cordey, JET Joint Undertaking, Dr. V. Mukhovatov, ITER JCT, and Prof. M. Wakatani, Kyoto University

The Combined Workshop of the Confinement and Transport and Confinement Modeling and Database Expert Groups took place April 14th to 18th at the San Diego ITER Joint Work Site. The Workshop lasted for 5 days. Four days began with 1-2 hour Joint Sessions with topics of interest to both Expert Groups. The Joint Sessions were followed by two or three parallel sessions. The fifth day was devoted to a Joint Summary Session.

Progress and status on implementing the ITER Confinement R & D Needs as specified at the last Workshops of the Expert Groups in Montreal (October 1996) were reported. The main results are as follows:

- The H-mode threshold database has been extended and $\beta_N^{tot} \approx 3.2$ improved. Preliminary analysis of the new version of database yields a threshold power prediction for ITER which varies between 50 and 170 MW (cf. with P_{thr} = 50-200 MW reported at the last Workshop). Results of further analysis will be reported at the 24th EPS Conference in June 1997.
- Recently discharges with the same $\nu*$ as ITER and   been obtained in JET with a good confinenement factor H_H=1.3.
- A new version of the global H-mode confinement database (DB3) was released just before the Workshop. Preliminary analysis shows that the β degradation is still strong in the τ_E scalings for ELM-free and ELMy H-mode. The confidence interval for τ_E seems to be close to that for DB2, i.e. 3.5 - 9 sec. Analysis of DB3 is in progress, and further results will be presented at the 24th EPS Conference.
- Benchmarking of local transport codes is well under way and should allow the standardized procedure to test the transport models and study their projections for ITER to be completed as planned by the next meeting in September.
- Critical R&D issues associated with ITER operation in RI-mode and in regimes with internal transport barriers have been identified. Those for the RI-mode are expected to be addressed in the next few months.
- Preliminary results from gyro-kinetic codes suggest a lower transport compared to the IFS/PPPL model.
- Theoretically, a low frequency turbulence in some conditions can provide a moderately peaked density profile that should be beneficial to ITER performance.
- Scalings for edge pedestal width and height in individual machines (DIII-D and JT-60U) have been suggested; however, projections to ITER remains fairly uncertain.

The results of the Workshop are given in greater detail as follows.

Participants in the Meeting

URGENT RESEARCH NEED 3.1: H-MODE THRESHOLD PHYSICS

Several talks were presented on how to relate the existing data to theoretical models of the LH transition. It is suggested that the most effective test of theoretical models will involve comparison of local non-dimensional edge parameters with the model predictions. All facilities are now devoting considerable effort to measurement of local edge parameters corresponding to the L-H transition. Results from ASDEX Upgrade, including scaling relations, were presented. A novel physical picture explaining the observed n and B dependences of the threshold power (P_{thr}) by cross-field neoclassical fluxes driven by poloidal temperature gradients at the edge has been suggested by T. Carlstrom. This model predicts very low $P_{thr} \sim 12$ MW for ITER. The role of neutrals in experiments in DIII-D and JT-60U was discussed and shown to be important. Reduction of n_0 at $r_{95\%}$ was shown to correlate with reduced threshold powers in both devices. D. Post presented calculations demonstrating that neutral effects in ITER should be strongly reduced relative to present devices. Essentially all proposed new experiments are still intended to be carried out in 1997 campaigns (most of which have not yet begun or are just underway).

URGENT RESEARCH NEED 3.2: ITER DEMONSTRATION DISCHARGES

Experimental work has continued in all of the X-point devices, on the setting up of ITER Demonstration discharges having the same dimensionless geometric parameters as ITER. Several devices have recently reported gyro-Bohm scaling of the H-mode: ASDEX Upgrade, Alcator C-Mod, JT-60U, confirming the earlier findings of DIII-D and JET. The scaling with dimensionless parameters β and ν^* (collisionality) has been investigated by DIII-D and JET. The scaling of $B\tau_E$ with ν^* is similar to that of the ITERH-93P scaling expression ($B\tau_E \propto \nu^{*-0.28}$); however the scaling with β is found to be very weak in contrast to that of ITERH-93P ($B\tau_E \propto \beta^{-1.2}$). The reason for the difference in the β scaling is being actively studied since it has important consequences for ITER. Initial scoping studies on dimensionless identity discharges between Alcator C-Mod and DIII-D, Alcator C-Mod and JET, and COMPASS and DIII-D/ASDEX Upgrade have been carried out. It is expected that these experiments will be completed this year along with further studies on the scaling of τ_E with q, isotope mass and Mach number.

URGENT NEED 4.1: H-MODE GLOBAL DATABASE

The new version of the global H-mode confinement database (DB3) now includes data from 12 Tokamaks (ASDEX, ASDEX Upgrade, Alcator C-Mod, COMPASS-D, DIII-D, JET, JFT-2M, JT-60U, PBX-M, PDX, TCV, TEXTOR{RI-mode}). Extensive analysis of DB3 has been initiated including an offset non-linear analysis by T. Takizuka. The results of the first analysis show that the condition of the DB3 standard dataset is slightly better than that of DB2. The ELMy data now also satisfies the high β constraint. The beta degradation has been reduced from -1.2 to -0.8 in the ELM-free H-mode scaling. For the ELMy H-mode scaling it is close to -0.8 for DB2 as well as for DB3. Including the limited number of Ohmic H-mode data from COMPASS-D and TCV did not alter the scalings. The 95% confidence interval of the ITER confinement time prediction based on the DB3 scalings will be determined as the updated database is more fully analyzed in a variety of fashions. Since the scalings changed only marginally with the updated database, it was sensed that this confidence interval will be close to the 3.5 to 9 sec range determined from DB2. G. Hammett presented results of a modified jacknife study suggesting that the uncertainty range, as determined from DB2, was slightly larger. M. Zarnstorff disputed the use of any statistical uncertainty because of the lack of full knowledge of the confinement behavior, in particular the possible impact of hidden variables (e.g. wall conditioning and rotation) not included in the regression analysis. As was presented by O. Kardaun in previous meetings, the interval estimate is a result of a number of considerations, including projections based on different functional forms of equal significance as opposed to the uncertainty associated with one particular functional form. S. Scott, Y. Muira and J.G. Cordey presented data on the effect of rotation on energy confinement in TFTR, JFT-2M and JET but with different conclusions. Further analysis of the data is required and will be pursued. The status of the database and results of further analysis will be reported at the EPS Conference in June 97. Until the next meeting, variables concerning rotation and pedestal values will be added as well as new data from ASDEX Upgrade, Alcator C-Mod, DIII-D, JET and TCV.

URGENT RESEARCH NEED 4.2: H-MODE POWER THRESHOLD DATABASE

The following topics were addressed: status of the database (DB2) and new data, origin of the scatter of the data, and analyses of the new version of the database. Almost all the tasks decided at the previous Workshop (Montreal, October 96) have been completed. The scattering of the data has been re-analyzed in several devices. The scattering in ASDEX Upgrade is rather small. In JFT-2M the scattering remaining after applying the selection criteria is attributed to wall conditioning. Wall conditioning is also playing a role in JT-60U. The scattering in Alcator C-Mod cannot be attributed to any measured parameter such as plasma-wall distance, radiation, neutral pressure and apparently also depends on wall conditions. The scattering in DIII-D is large, only a part of it can be attributed to parameters such as X-point height. Discriminant analysis made in TCV on the LH transition provides a parameter which discriminates with precision between L and H mode. Further work will be done along this line with the complete database with the aim of predicting the power threshold. For this purpose more L-mode points are necessary. Analyses of the complete database with global parameters were performed with 3 different data selections: either only the ITER appropriate points, or only the "points with the lowest threshold power", or only the minimum of the threshold at each magnetic value. These different approaches yield threshold predictions for ITER which vary between 50 and 170 MW. The R dependence is the main effect leading to such variations of the prediction. The availability of edge data in the database has been significantly improved and preliminary analyses of the new version of DB were made. Further work must be done until reliable results can be given.

URGENT RESEARCH NEED 4.4: PROFILE DATABASE AND TESTING OF TRANSPORT MODELS

New contributions into the Profile Database have been made from Alcator C-Mod, TFTR, JET and DIII-D. The present status of discharges in the database has been reviewed in order to define a standard dataset for the model testing. Development of standardized procedures to understand and reduce the variability between the predictions of different transport codes when using ostensibly the same model had been proposed by D. Boucher in preparation to the meeting. Preliminary implemention of these procedures by D. Mikkelsen, J. Kinsey and D. Boucher showed that convergence can indeed be achieved between their three independent transport

codes. W. Houlberg made some comments on the use of metric coefficients in extending models to general geometry. D. Mikkelsen gave a description of his analytic benchmark for checking whether codes could recover a known result. Examples of this analytic benchmark testing were performed by Y. Ogawa et al. and presented by T. Takizuka. D. Mikkelsen also explored the sensitivity of predictions to experimental uncertainties in the data and the use of differing algorithms. R. Waltz described plans to run localized, pulsed ECH on DIII-D to distinguish between models according to their "stiffness". J. Kinsey described work with the MLT code and four models (GLF23, IFS/PPPL, Multimode, CDBM) using the ITER Profile Database and additional DIII-D shots. The Multimode model performed best; inclusion of $\underline{ExB}$ shear improved GLF23, but not IFS/PPPL, and all models recovered τ_E scalings observed in DIII-D scans; the methodology was unable to significantly distinguish models with different physics. Yu. Dnestrovskii showed that the Canonical Profile Transport model was also able to recover L and H-mode scalings in JET and DIII-D, including a weak β-scaling which contrasts with the global ITER scaling laws for τ_E.

New simulations have been submitted following the procedure agreed upon at the last Workshop in Montreal (October 1997) for standard ITER predictions. Theoretically derived models based on ITG turbulence were the most pessimistic (IFS/PPPL and GLF23 models) and would require driven operation. Other models such as the CDBM model or RLW and RLWB models were more optimistic and predicted ignition. A. Kritz showed that the Multimode model - although also based on ITG turbulence - would predict ignition in ITER even under conditions close to the L-mode. This model differs from other ITG models by the κ^{-4} elongation dependence introduced in the Multimode model based on global scaling considerations as well as direct comparison with experimental profiles. If this term is removed from the model, it was found that the simulated temperatures would fall below the experimental ones and that ITER predictions would no longer ignite even for an edge temperature of 2 keV. J. Weiland showed a new improved ITG model and model predictions using the density profile and input power from two target ITER reference cases. The predicted temperatures were somewhat above the target ones and ignited. O. Mitarai showed ITER performance projection from 0-D analysis using two global scaling laws: ITERL-89P and ITERH-93P. Yu. Dnestrovskii presented the ITER predictions using the Canonical Profile Transport model.

URGENT RESEARCH NEED 4.9: EDGE TRANSPORT BARRIER / PEDESTAL

D. Post discussed the edge operational space for ITER on the basis of results from ASDEX-Upgrade, DIII-D, Alcator C-Mod, JET and JT-60U. T. Osborne presented data from DIII-D and Alcator C-Mod showing a correlation between the global energy confinement time and edge pedestal parameters. The $H_{ITER93H}$ confinement factor in DIII-D was found to be proportional to $(\beta_T^{ped})^{0.5}$, where β_T^{ped} is beta value at the edge pedestal. The pedestal width in discharges with Type I ELMs exhibits a good correlation with $(P_e^{ped})^{0.51}/B_\rho^{0.92}$ and $(T_e^{ped})^{0.51}/B_p^{0.51}$. It was observed that the total plasma pressure can continue to increase when the edge pressure saturates which would not be expected for very 'stiff' profiles. J.G. Cordey compared pressure profiles of JET plasmas obtained at low gas fueling, when Type I ELMs were observed, with those obtained at Ne/N$_2$ injection and strong D$_2$ gas puff, when Type III ELMs were presented. No strong profile consistency was deduced from these experiments, i.e., effective thermal diffusivity in the plasma core in both cases was approximately the same whilst the edge electron temperature changed significantly. M. Mori discussed ELM behavior in JT-60U. The $H_{ITER89p}$ factor was shown to decrease with increasing frequency of Type I ELMs. The ELM frequency increased almost linearly with heating power and tended to increase with rising density. The edge pedestal width in the ELMy phase was found to be larger (by 2-3 times) than that in ELM-free phase, the latter obeying a scaling relation $\Delta=(3.3\pm0.5)\varepsilon^{0.5}\rho_p$ which differs from DIII-D scalings (ρ_p is the ion poloidal gyroradius). T. Osborne also presented data showing that $H_{ITER93H}$ factors were relatively low in low-density H-mode discharges with Type III ELMs, whilst in high-density discharges with Type III ELMs there seemed to be no strong reduction in confinement.

HIGH-PRIORITY NEED 3.5: ALTERNATIVE SCENARIOS FOR ITER

For a device of ITER's scope and mission, it is essential to develop multiple high-confinement operational scenarios. These confinement scenarios can be classified as follows:
- Alternate scenarios. Examples of alternate scenarios are the RI-mode, if confirmed in large divertor devices, and operational modes with peaked density profiles. They can broaden ITER operational capabilities and be a back-up for the standard scenario. They are characterized by having similar performance to the ELMy H-mode and their basic properties can be documented before the FDR. There are a few critical R&D issues related to these scenarios that must be addressed in the next few months.
- Advanced scenarios. Examples of advanced scenarios are the large variety of operational modes with internal confinement barriers. They offer the promise of very high performance but their applicability to ITER is still uncertain. Physics R&D must actively explore these options during the next ITER phase.

M. Zarnstorff gave a general talk raising critical issues of advanced confinement regimes with an emphasis on the importance of peaked density profiles. J. Ongena discussed in detail the new operational regime, the RI-mode, established in TEXTOR that combines many features that are attractive for reactor operation. X. Garbet discussed issues regarding applicability of the RI-mode to ITER operation. They include allowable impurity content and required momentum input. P. Gohil presented a comprehensive review of present results in advanced confinement modes based on weak and reversed magnetic shear configurations. Recent results from JT-60 reversed shear experiments were reported by M. Mori and Y. Koide. A list of planned experiments on major devices addressing alternative scenarios has been presented and issues to be resolved in order to extrapolate these regimes to ITER were proposed. Progress in this area will be reported at the next Expert Group Workshop in September 1997.

HIGH-PRIORITY NEED 3.6: TRANSPORT MODEL DEVELOPMENT

F. Perkins set the scene for this session, suggesting experimental tests of global scaling laws and plasma profiles to address issues that had arisen from theoretical transport modeling (e.g., the IFS/PPPL ion temperature gradient model), such as stiffness (or resilience) of profiles and the role of the edge pedestal. At a theoretical level, tests of the approximations underlying the IFS/PPPL gyro-fluid model are underway both using gyro-kinetic codes, as reported by R. Sydora, and examining the poloidal flow damping which controls the turbulence saturation in the gyro-fluid simulations, discussed by M. Rosenbluth. Early results from the gyro-kinetic codes suggest lower transport than in the IFS/PPPL model. In another analysis of basic physics, J. Weiland discussed the role of kinetic

effects in the context of his fluid drift-wave model. A less stiff model than the IFS/PPPL model is the Current Diffusive Ballooning Mode (CDBM) model and M. Wakatani described its theoretical basis and how it could describe a variety of confinement modes; with careful current drive, internal transport barriers can be sustained. W. Houlberg described a comprehensive neo-classical and radial force balance code for help in analyzing bootstrap currents, radial electric fields and rotation in the presence of multiple ion species. A peaked density profile can be beneficial to ITER performance and M. Rosenbluth showed how low-frequency turbulence can predict the ratio between the axis and the edge. Finally W. Houlberg demonstrated how difficult it was to deduce the density scaling of local transport from the global scaling as it could be obscured by the influence of sources.

HIGH-PRIORITY NEED 4.8: MODELING OF SAWTOOTH ACTIVITY

D. Boucher presented transport code calculations of ITER sawtooth using the PRETOR code that implements a model developed by F. Porcelli. A sawtooth period of about 100 seconds was found. J. Connor described potential refinements to this model that could produce a shorter period. Both short ($\sim \tau_E$) and long (>100s) sawtooth periods are therefore still being used as design assumptions for ITER.

LONG-TERM NEED 3.7: EFFECT OF WALL CONDITIONING ON PLASMA CONFINEMENT

Presentations were made by ASDEX-Upgrade (F. Ryter), Alcator C-Mod (M. Greenwald), DIII-D (G. Jackson), JET (J.G. Cordey), TEXTOR (J. Ongena), and TFTR (S. Scott). There is strong evidence that proper limiter and wall conditioning can improve global energy confinement. In Alcator C-Mod, wall conditioning (boronization) has decreased the impurity influx, which has led to higher edge temperatures and higher τ_E. Similarly, DIII-D reports a correlation of energy confinement and recycling in H-mode plasmas. There is contradictory evidence regarding the effect of conditioning on the heating power required to obtain an H-mode transition, or transitions to other favorable confinement regimes. The high-performance VH mode was discovered in DIII-D only after boronization. In TFTR, reduced recycling rates for hydrogen seem to reduce the power threshold for enhanced reverse-shear plasmas and limiter H-modes. However, these trends are not universal. High performance 'optimized shear' discharges in JET were obtained within the first week of operation, during which the wall conditioning was still improving, while much more extensive conditioning was required to obtain good hot-ion H-modes. DIII-D observed only a modest effect of boronization on τ_E for ELMy H-mode discharges.

The overall ITER design has incorporated a number of features which should reduce the influx of hydrogen and impurities from the plasma-facing surfaces. It is recommended that a limited experimental campaign be undertaken to assess whether confinement degrades in present devices over a series of consecutive discharges, whose integrated duration is ~100 seconds, without any intervening wall conditioning.

At the session devoted to preparation of the ITER Physics Basis document, the outline of the Chapter II "Confinement and Plasma Performance" was modified taking account of progress in confinement studies and a general guidance from the ITER Director. A writing schedule for the Chapter II that fits schedule for the whole document was agreed.

The last day of the meeting was devoted to a joint session were all the session Chairmen presented summaries of presentations and discussions that took place during their sessions.

The next Workshop will also be joint between the two Confinement Expert Groups and planned from September 25th to 30th, 1997 in Garching, Germany.

LIST OF PARTICIPANTS

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97-02111