

INSIDE

- * ITER Council Meeting
- * Ways and Means WP Activity
- * ENEA's Fusion Research at Frascati
- * Together for the Future - ITER-Related Activity and the Future at JAERI
- * ITER Major Events

Focus on the future

ITER COUNCIL MEETING

by Paul N. Haubenreich, ITER Council Secretary

The seventh meeting of the ITER Council, in Vienna on 26-27 July, focussed principally on the future. The particular subject of attention was the envisioned Engineering Design Activities (EDA), which would be the logical next step after completion of the current work at the end of 1990.

The Council received reports on information produced thus far by the joint Conceptual Design Activities (CDA) for use by the Parties in considering participation in ITER EDA. The Council's Working Party on Ways and Means reported accomplishment of the task assigned to it, viz., to develop suggestions on supporting elements needed for consideration of and decisions on the possible conduct of ITER EDA. The ITER Management Committee then presented its present estimates of duration and cost of engineering design, supporting research and development, construction and operation of ITER. (It was noted that more refined estimates will be made later this year on the basis of the completed conceptual design.)

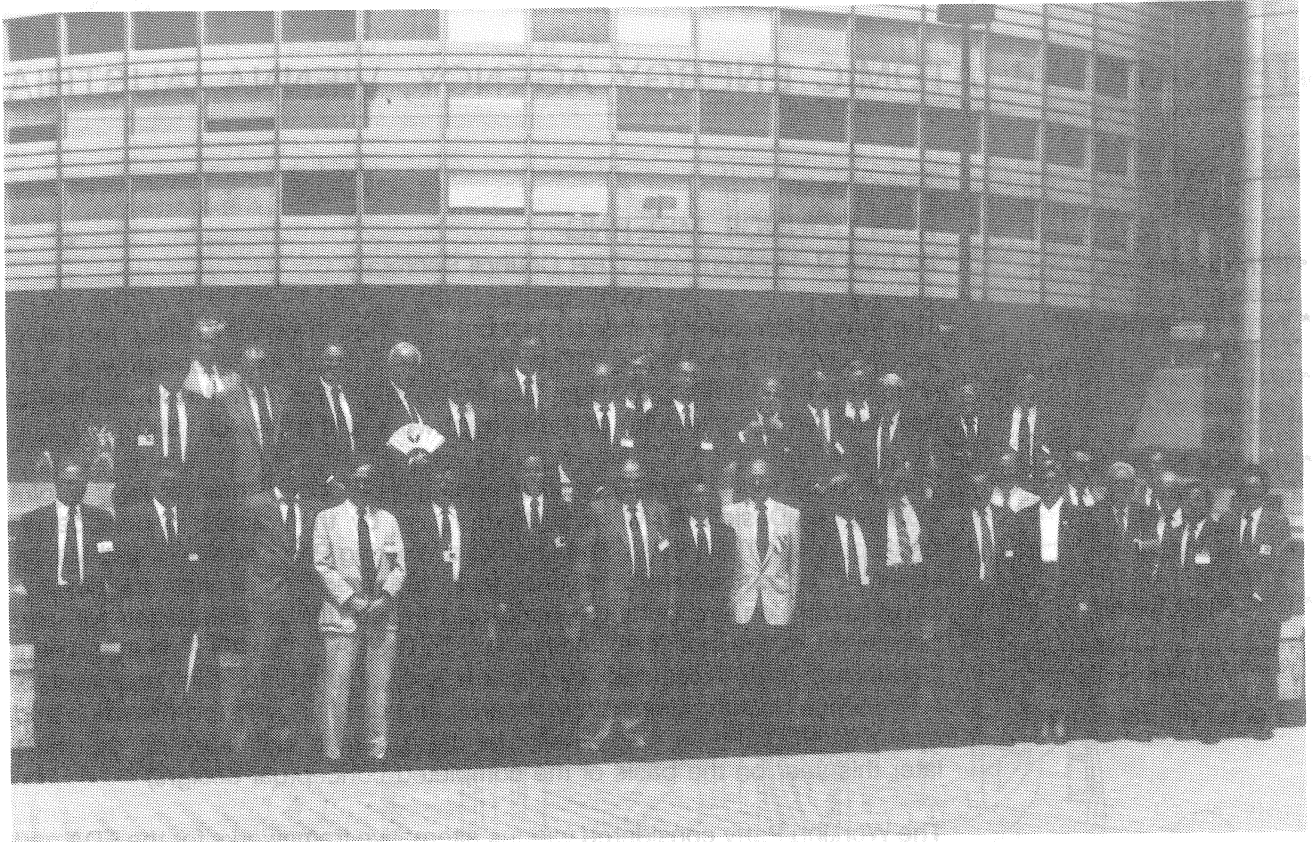
The Working Party considered various alternative frameworks for the EDA and recommended an Agreement at government level, involving IAEA auspices, with successive Protocols defining in detail EDA technical elements and implementation. The Working Party reported broad areas of common views among experts from the four ITER Parties and identified specific items for further consideration in forthcoming discussions. Upon the completion of its assignments, the Working Party was commended by the Council, with expressions of appreciation.

Cost and schedule estimates of the EDA

The IMC estimated that accomplishment of the EDA would require about five years, with technology R&D the most likely critical path. Meanwhile a vigorous effort would need to be applied to the design work, including earliest practicable evaluation of proposed construction sites. The Council reviewed, and after some discussion, approved for publication the IMC's "Preliminary ITER Cost and Schedule Estimates", report (ITER Documentation Series, No. 14). Their report contains the following estimates. The estimated total effort by a Central Team on design and R&D co-ordination and by the four Home Teams on supporting design was 1200 professional man-years. The R&D in support of the design includes physics R&D, basic technology R&D and specific engineering R&D. Costs of the physics R&D are not included in the report since they are integral to the world fusion programme and cannot be separately estimated. The basic technology R&D, providing the data base for the specific engineering and design by studying the properties and behavior of basic component elements, was estimated at \$400M, excluding cost of developing DEMO-relevant blanket modules. The specific engineering R&D, including demonstrations of prototypes and their test facilities, was estimated at \$350M. As part of the joint activities at Garching, comprehensive plans for

ITER-related R&D are being developed under the IMC, with broad participation of specialists from all Parties.

Recognizing the critical importance of organizing technology R&D efforts, the Council appointed an *ad hoc* group of experts to identify those tasks that should be started early in EDA and where, in the Parties' facilities, the tasks could most efficiently be carried out. This information will supplement the other material being supplied for the four-Party discussions.



At the time of the IC Meeting in Vienna, 26-27 July 1990

WAYS AND MEANS WORKING PARTY ACTIVITY

by M. Roberts, Chair, ITER Council's Ways and Means Working Party

Previous activities

The ITER Council established this Working Party at its fourth meeting, July 1989, to assist the Council in fulfilling its function to "explore ways and means to comply with the objective of the co-operation" as stated in section 5.1.2a of the ITER Terms of Reference. In particular, the Working Party was charged with the task of proposing those supporting elements needed for the possible conduct of the Engineering Design of ITER. After hearing an Interim Report from the Working Party at its fifth meeting, November 1989, the Council provided further guidance directing additional exploration into the most complex topics. The Working Party's Report was accepted at the sixth meeting of the ITER Council, April 1990, for publication by the IAEA in the ITER Documentation Series, No. 11 supporting the Council's Status Report, No. 12.

Further tasks assigned by ITER Council

At its sixth meeting, the Council gave the Working Party three further tasks to be completed by the Council's seventh meeting, July 1990. The first was the general charge to continue to facilitate the Parties' exploratory discussions; the second was to finish its exploration of the most complex areas; the third was to work with the IMC to reach "an understanding of the technical basis for the various views" of the Engineering Design Activities.

Common view on a feasible approach to the EDA

Items for further detail exploration identified

Useful contribution to the Parties' discussions and possible negotiations on the EDA

The Working Party met in Tokyo and in Garching to conduct its work on these three tasks. The net result of this work was a common view of all members of the Working Party on a feasible approach to the EDA. The Working Party had considered all the elements necessary to characterize an EDA and concluded there was a sufficient base of information to permit the Parties to prepare any necessary documents by the end of the CDA. At the same time, the Working Party identified a set of specific items where further consideration should be given to illuminate details or understand different assumptions among the Parties' experts. The Working Party was also able to meet with the IMC during one of its busiest times, the initiation of the summer joint work session, to discuss the apparent differences of view noted by the Council. These differences were traced to differences in assumptions and not to differences in any principles.

These findings were reported to the ITER Council at its seventh meeting. They were also provided to the representatives of the Parties in their initial exploratory discussions meeting. These representatives, after hearing the views on an EDA from each Party, and considering the findings of the Working Party, determined that there was sufficient basis for them individually to seek authorization to enter negotiations on an EDA Agreement. The Record of the Council meeting notes in conjunction with this matter that "The Council commended the Working Party for fulfillment of its purpose of providing information that should be useful to the Parties in discussions and, should they so decide, in negotiations of an instrument to allow ITER Engineering Design Activities (EDA)".

Editor's Note

The Newsletter continues the series on the activities of the major fusion research centers of the ITER Parties by presenting Frascati Energy Research Center and by providing more detail information on JAERI's activities.

ENEA'S FUSION RESEARCH AT FRASCATI (ASSOCIATION EURATOM-ENEA)

by R. Andreani and H. Knoepfel

Organization of Italian fusion research

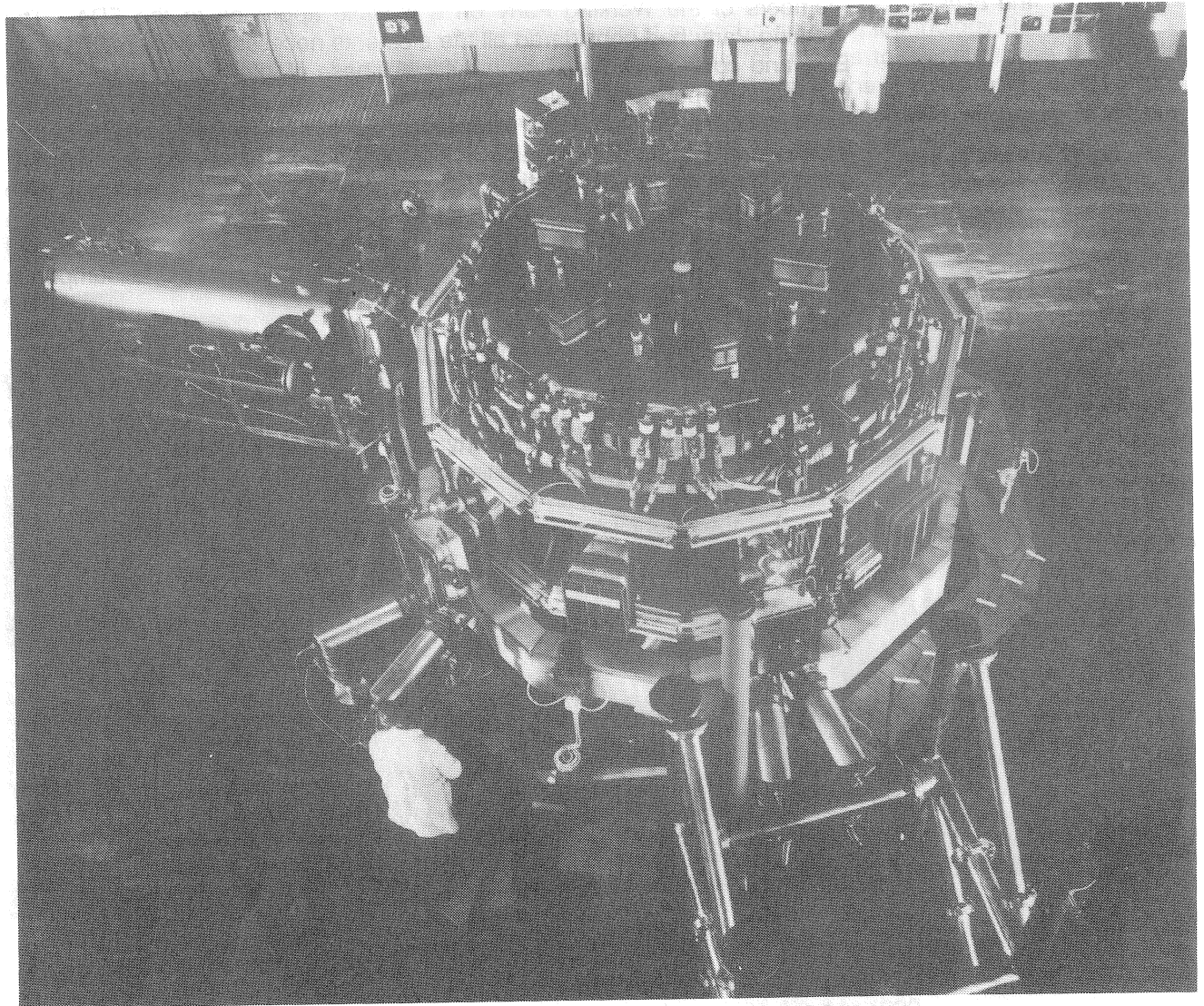
The main nuclear fusion research activities in Italy are carried out at the Frascati Energy Research Center by the Fusion Department of the Italian Commission for Nuclear and Alternative Energy Sources (ENEA) in the framework of the EURATOM-ENEA Association. The Association manages the entire Italian fusion research effort, which also includes activities conducted at the laboratories of the National Research Council (CNR) at Padua (with its substantial reverse field pinch programme) and Milan (theory and electron-cyclotron microwave activities), as well as at many Italian universities. ENEA provides financial and administrative support to these activities, while CNR is responsible for the scientific guidance of its own programmes.

Beginning of fusion research

Fusion related plasma physics research in Italy was promoted by Prof. E. Amaldi and started with a small group led by Prof. B. Brunelli at Rome University in 1957. It was then transferred in 1960 to its own new facilities at the Frascati Center. One year earlier, it had become the second European fusion research programme to be associated to EURATOM (after, and in connection with, the Association EURATOM-CEA related to the French Commissariat à l'Énergie Atomique).

Early research programmes

The research programme in the sixties included some pioneering, ultrafast, high-density experimental approaches to fusion, basically related to inertial confinement concepts: fast theta pinch (CARIDDI), hollow Z-pinch (MIRAPI),



FTU device before being inserted into the cryostat,
and without surrounding diagnostic equipment

high-explosive and capacitor-driven magnetic implosion experiments (MAFIN), plasma focus, and relatively high-power, laser-driven light-plasma interaction and diagnostics (HOT ICE). Starting from the end of the sixties, these small-scale programmes were gradually terminated in favour of the tokamak line and its related technology.

Frascati Tokamak (FT)

The compact, high field tokamak FT came into operation in 1978, attaining a toroidal magnetic field on axis of 8 T and a plasma current of 0.6 MA. (The magnet was tested at the full field of 10 T.)

Although the duty cycle of the FT machine was quite low, due to the time needed to cool back down to liquid nitrogen temperature after each shot, more than 22,000 useful shots were performed. Table I gives the main parameters and performances of the FT plasma.

The main objectives of the physics programme at FT were:

- study in ohmic regime of energy confinement and transport in high density, high temperature conditions,
- study of coupling, propagation, and absorption of RF power in the lower hybrid (LH) frequency range (2.45 GHz and 0.5 MW; 8 GHz and 0.4 MW), and
- study of current and density limits and the related MHD activity.

The FT plasma had very interesting conditions: an average density up to $4 \times 10^{20} \text{ m}^{-3}$ was possible; strongly coupled electron and ion temperatures higher than 1.5 keV, and good plasma purity, i.e. Z_{eff} near unity, at a density higher than 10^{20} m^{-3} . It should be recalled that, in FT, the power flux through the last magnetic surface was of the order of 0.15 MW/m^2 , i.e., the same as in JET with its 25 MW of total input power.

The most important results obtained in FT concern the attainment of high values of the Lawson parameter, $n\tau_E \sim 4 \times 10^{19} \text{ m}^{-3} \text{ s}$, in high density ohmic discharges at 8 T, and the study of heat transport in these conditions; an understanding of the limits of electron-wave interaction with the lower hybrid RF system; the study of parametric decay instability connected with the wave propagation and the characteristics of the plasma edge; the test of the performance of the 8 GHz RF system, which will be applied on the Frascati Tokamak Upgrade (FTU), with respect to the density limit for effective electron heating; and the study of MHD behaviour at the operation limits of the machine.

In mid 1989, the FT programme was terminated in favour of concentrating all available efforts on its successor, FTU.

Frascati Tokamak Upgrade (FTU)

The FTU is a new medium size, high field tokamak, which has come into operation recently in its purely ohmic heating phase (see Table II). After the first month of operation, a plasma current of 1 MA over a pulse width of 1 s at a toroidal magnetic field of 6 T was attained.

The main motivation for building this device was to combine the favourable confinement characteristics of FT-type plasmas (both in confinement time τ and parameter $n\tau$) with a substantial microwave plasma heating capability for which the so-called hybrid mode was chosen.

Lower-hybrid heating

FTU thus includes the necessary access to the plasma for injecting a total of 8 MW at the LH heating frequency of 8 GHz. There are 12 tangential ports ($8 \times 42 \text{ cm}^2$ each) with a total of 0.4 m^2 of access area; 8 of the ports will house the LH launching grills.

The LH heating system is presently under construction. Towards mid 1991, the first MW of LH should be fully operational on the machine, and the complete RF system of 8 MW will then be ordered.

Ion-Bernstein-wave heating

The idea of using ion-Bernstein-wave (IBW) heating on FTU originated from the possibility of combining the characteristics of the low cost and high efficiency of low frequency equipment with the advantages of the waveguide coupling.

In fact, plasma heating by RF power in the ion cyclotron range has the advantage of working at relatively low frequencies for which low cost, efficient equipment is widely available. Usually, in this kind of experiment, the RF power is coupled to the plasma by some antenna-coils placed inside the tokamak vessel; this antenna structure generally gives rise to impurity emission problems. More practical and convenient in this respect are the waveguide systems used at higher frequencies.

FTU is the first device in which IBW heating will be performed using a waveguide coupler. The access port sizes of FTU allow operation with a wave frequency of 433 MHz. This frequency corresponds to the fourth cyclotron-harmonic of hydrogen near the plasma center for a magnetic field of 8 T on axis. The approach selected is to build a system composed of 0.5 MW modular units. It appears that a total 1.5 MW of RF power will suffice to thoroughly explore the potential of IBW in FTU.

Reactor engineering and technology

ENEA's Brasimone Research Center, formerly the site of the PEC fast breeder experimental reactor, was recently assigned to the Fusion Department. The personnel from Brasimone, together with Frascati technical personnel relieved from previous FTU duties, will help to implement a substantial expansion of the activities on reactor engineering and NET/ITER-oriented technology: the blanket and the superconductivity programmes are mentioned below.

Reactor blanket programme

Regarding the blanket programme, the Fusion Department at Frascati is continuing the activities on the design and technological development of the helium-cooled ceramic breeder blanket concept for the DEMO reactor in collaboration with ENEA nuclear departments (Casaccia, Bologna, Brasimone) and with CEA. Experimental investigations on the water-cooled liquid (LiPb) blanket for DEMO were initiated at Brasimone. In collaboration with Ansaldo-Genoa, the construction of a NET first wall mockup, to be tested in 1990-1991 at the European Joint Research Center (JRC) at Ispra, was completed, and the design study of the water cooled ceramic breeder blanket for NET/ITER was started.

Superconductivity programme

The superconductivity programme pursues various activities. The 6-T, 1.3-m bore ENEA-SULTAN superconducting magnet has been modified from a solenoid to a split-coil pair in order to be installed in the updated SULTAN Facility at the Paul Scherrer Institut (PSI) in Villigen, Switzerland. A model magnet made with a Nb₃Sn cable-in-conduit conductor and a wind-and-react technique is under design. The coil is intended as a test of the concept for the ohmic heating solenoid of NET. Collaboration with Kernforschungszentrum in Karlsruhe and NET in Garching has been started in the field of magnet safety assessments to develop and validate two- and three-dimensional computer codes predicting the quench propagation behaviour in a superconducting magnet. Experimental tests are planned for early 1990 at Frascati, using the existing forced-flow facility and the SAFFO magnet. The Frascati 12.5 T test facility has been upgraded to reach 3.5 kA maximum current to measure sub-components of the prototype superconducting cable being developed for the NET ohmic heating coil by the firm EM-LMI (Europa Metalli-La Metalli Industriale) under a NET contract.

Inertial confinement

The activity in inertial confinement research can be traced back to the Frascati experiments of the sixties. At present, the main activity in the Fusion Department concerns the experimental and theoretical studies of laser-driven implosion of thin walled spherical targets.

The rationale of the Frascati programme is to perform experiments of fusion relevance that are realistically scaled down to a laser of the ABC system size, which allow significant studies on hydrodynamic effects and on light absorption, propagation, and refraction.

The experiments are performed by using the ABC two-beam neodymium glass laser system, which delivers an energy of 100 + 100 J in pulses of 2.5 ns at the wavelength of 1.054 μm, with the possibility of converting to 40 + 40 J at 0.527 μm. A large part of the theoretical activity is devoted to code development for the numerical simulation of the laser-target interaction and implosion experiments. At present, these codes are also used to simulate heavy ion beam driven implosions for fusion applications.

Miscellaneous activities

The experience and competence accumulated during all these years of fusion research at Frascati find useful application in numerous other activities. For example, design contributions have been provided for two large new projects: the TJ-II stellarator to be built by the EURATOM-CIEMAT Association in Madrid, and the compact ignition experiment IGNITOR proposed by B. Coppi of MIT, whose design has been developed by an industrial consortium formed by Ansaldo and Fiat under contract to ENEA and under the guidance of B. Coppi.

Collaboration with
ITER, JET and other
laboratories and
industries

In addition to the collaboration mentioned previously, work has also been undertaken with other laboratories and industries: for example, ITER (on LH heating and current drive); JET (on LH current drive and related diagnostics, neutron transport and measurements); the Max Planck Institut fuer Plasmaphysik (IPP) and the Princeton Plasma Physics Laboratory (PPPL) (on the joint LH current drive experiment on ASDEX).

TABLE I. FT MAIN PARAMETERS AND PERFORMANCES

Major plasma radius	0.83 m
Minor plasma radius	0.20 m
Minor liner radius	0.23 m
Max. toroidal field on axis	10 T
Max. plasma current	0.65 MA
Average electron density	$< 4 \times 10^{20} \text{m}^{-3}$
Peak electron density	$< 7 \times 10^{20} \text{m}^{-3}$
Z_{eff} (rel. impurity factor)	1
Energy confinement time	$< 0.05 \text{ s}$
Confinement parameter ($n_{e0} \tau_E$)	$< 4 \times 10^{19} \text{ m}^{-3} \text{ s}$

Main parameters of
tokamaks FT and FTU

TABLE II. FTU MAIN PARAMETERS

Major plasma radius	0.93 m
Minor plasma radius	0.30 m
Max. toroidal field on axis	8 T
Flat-top of field pulse	1.6 s
Max. plasma current	1.6 MA
Max. LH microwave heating power at 8 GHz during 1 s	8 MW
Total power of ion-Bernstein-wave system at 433 MHz during 1 s	1.5 MW

TOGETHER FOR THE FUTURE - ITER-RELATED ACTIVITY AND THE FUTURE AT JAERI

by T. Iijima, Director General, Naka Fusion
Research Establishment, JAERI

National activities on ITER Conceptual Design in Japan are conducted by the Fusion Experimental Reactor (FER) Team at Naka Fusion Research Establishment. The activities are supported by a broad research programme at JAERI (Japan Atomic Energy Research Institute), especially by the dedicated fusion research at Naka Fusion Research Establishment, and by other research centers in Tokai, Oharai and Takasaki whose experience and efforts in fission technology are incorporated in fusion research. This article is to present integrative research activities at JAERI related to the ITER design. The overview of the basic fusion research at JAERI was made earlier by M. Tanaka (see ITER Newsletter Vol. 3, No. 5).

Emphasis in recent JT-60 experiments was placed on 1) improvements in plasma confinement with profile control and 2) steady-state operation study, both of which are critical issues in physics R&D for the ITER design. Improvements in confinement were demonstrated with pellet injection, LH current drive, high-T_i mode operation or ICRF heating. Peaked density profiles of $n_e(0)/\langle n_e \rangle$ up to 4.5 were produced by successive injection of three or four pellets with a velocity of 2.3 km/s. The achieved $n_e(0)\tau_{E,T_i}(0)$ is $1.2 \times 10^{20} \text{ m}^{-3} \text{ sKeV}$ for $I_p = 3.1 \text{ MA}$, which is twice that of gas fuelled discharges.

Recent JT-60 results

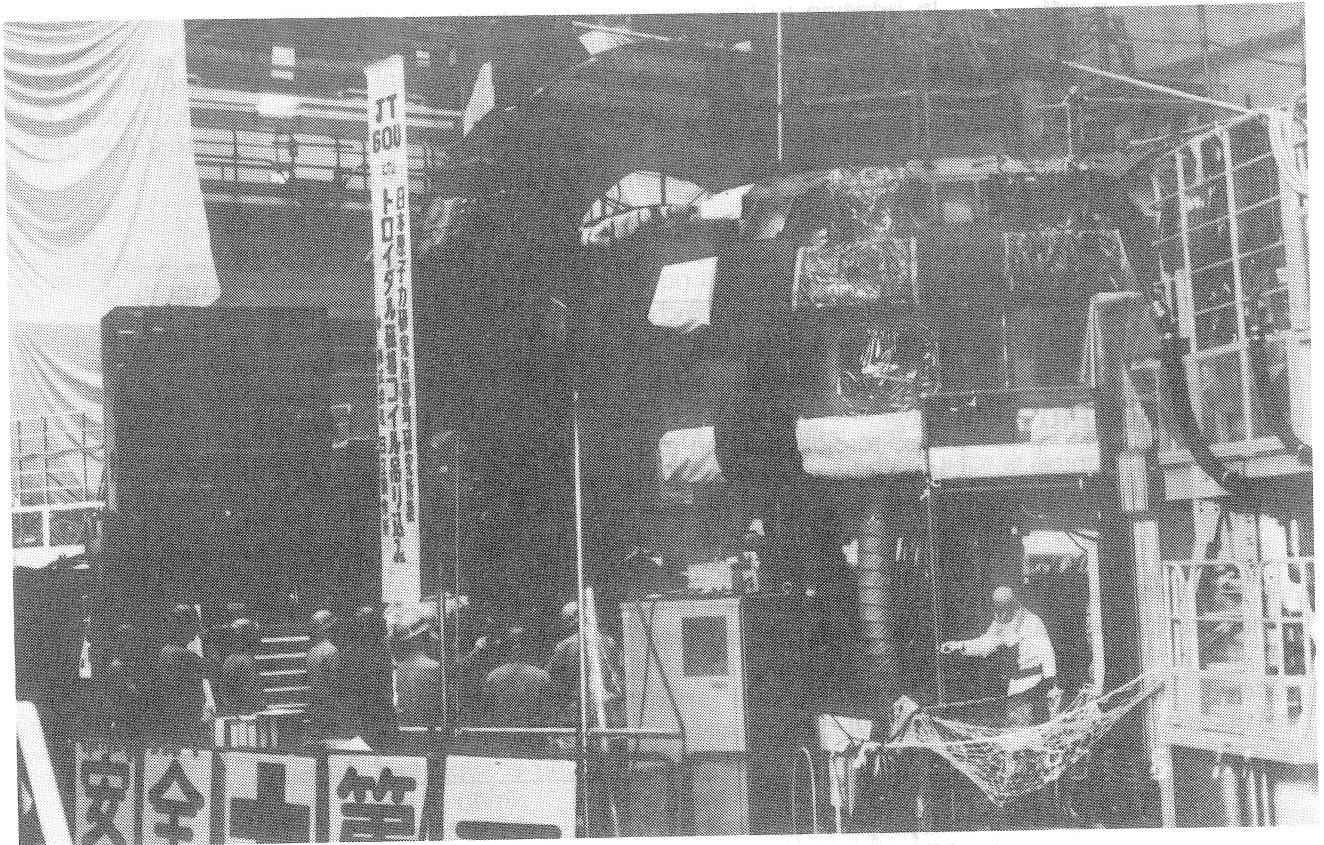


Fig. 1. JT-60U under construction. The first toroidal coil was attached on June 11, 1990.

Current profile controllability with energetic electrons was improved by the new LH launcher. Low- $N_{||}$ LH current drive discharges with high power NB heating showed centrally peaked hard X-ray and ion temperature profiles during the sawtooth-free period of up to 1.8 s. The improvement of the confinement capability in a peaked density and temperature resulted in a promising data base for ITER. The H-mode was obtained in limiter discharges with LH current drive. The threshold LH power was as low as ohmic heating power with hydrogen plasmas. Nearly steady-state ELM-free H-mode with duration up to 3.3 s was achieved without significant impurity accumulation. LH current drive experiments were carried out for a wide range of $N_{||}$ (1.0-3.4). The maximum current drive efficiency of $3.4 \times 10^{19} \text{ m}^2 \text{ A/W}$ was achieved with LH injection of 2-4.5 MW. The achieved value is approaching $5 \times 10^{19} \text{ m}^2 \text{ A/W}$, which is the goal value for the ITER steady-state operation. The bootstrap current was confirmed in the wide range of poloidal beta values, β_p . The ratio of the bootstrap current to the total current increased proportionally to β_p and reached 80% for $\beta_p \sim 3.2$. JT-60 has been contributing to the ITER design through joint works and the short-term R&D on many other issues such as divertor characteristics, disruptions, volt-second saving experiments by LH and so on.

JT-60 future plan

The JT-60 tokamak was shut down in November 1989 for the modification to JT-60U, which will have 6-MA plasma current, 4.2-T toroidal field, aspect ratio of 3.4-3.8 with lower X-point divertor configuration and a plasma volume up to 100 m^3 (Fig. 1). Deuterium will be used as the working gas. Experimental emphasis will be put on improvement in confinement characteristics and the research of steady-state operation. The JT-60U experiments will be started early in 1991. The results will contribute to the long-range physics R&D of ITER design.

Tokamak JFT-2M

The JFT-2M tokamak has shown various improvement modes of confinements: H-mode, H-mode with pellet injection (Fig. 2), Improved-L(IL)-mode and steady H-mode by using an ergodic magnetic limiter. The specific

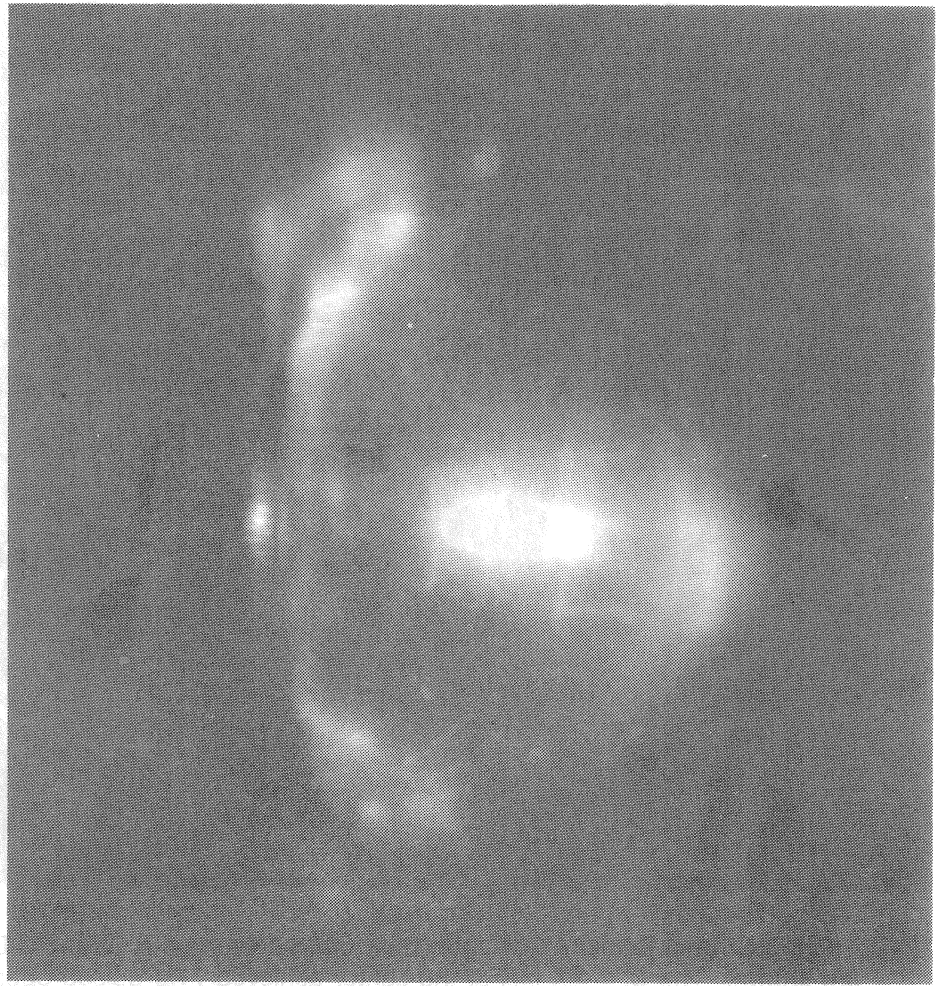


Fig. 2. Pellet-injected H-mode in JFT-2M. The pellet is injected from the outside of the torus (at the left hand side) and ablated at the plasma center.

H-mode data were assessed in details and compared with the data of JET, DIII-D, ASDEX and PDX/PBX at the ITER-related H-mode workshops to establish H-mode confinement scaling law for ITER. The data of divertor performance, density limit for Ohmic, L- and H-modes etc. were also presented to the ITER joint works and the short-term R&D. Diagnostics are being installed to measure the time evolution and profiles of plasma parameters such as ion and electron temperature, density and plasma rotation for the study of confinement mechanisms. The biased divertor experiment is planned to study and improve the divertor performance. A FW current drive experiment is also being made in combination with ECH. The results will contribute to the long-range R&D.

Superconducting Magnet Laboratory

The major activities in JAERI's Superconducting Magnet Laboratory at Naka are the R&D work of toroidal and poloidal coils, cryogenic system, power supply and protection systems. The activities on toroidal coils are the design, fabrication, and experimental evaluation of candidate conductors and testing magnets for the Proto Toroidal Coil Programme. The Demo Poloidal Coil (DPC) project is also in progress. In the DPC project, a 1-m bore Nb₃Sn coil (DPC-EX) has recently achieved the 7 T/s pulsed operation where the coil was charged up to 17 kA and 7 T in one second with the background field of two Nb-Ti pulsed coils (DPC-U1 and U2). The pulsed power supplies of the JT-60 are available for the operation of the coil system. The inner view of the DPC coil system is shown in Fig. 3. An international collaboration between JAERI and the U.S. Department of Energy is underway to use this facility more widely and efficiently. In this collaboration, Massachusetts Institute of Technology will provide one test coil (US-DPC). The fabrication of US-DPC was completed in May 1990. The coil is scheduled to be tested

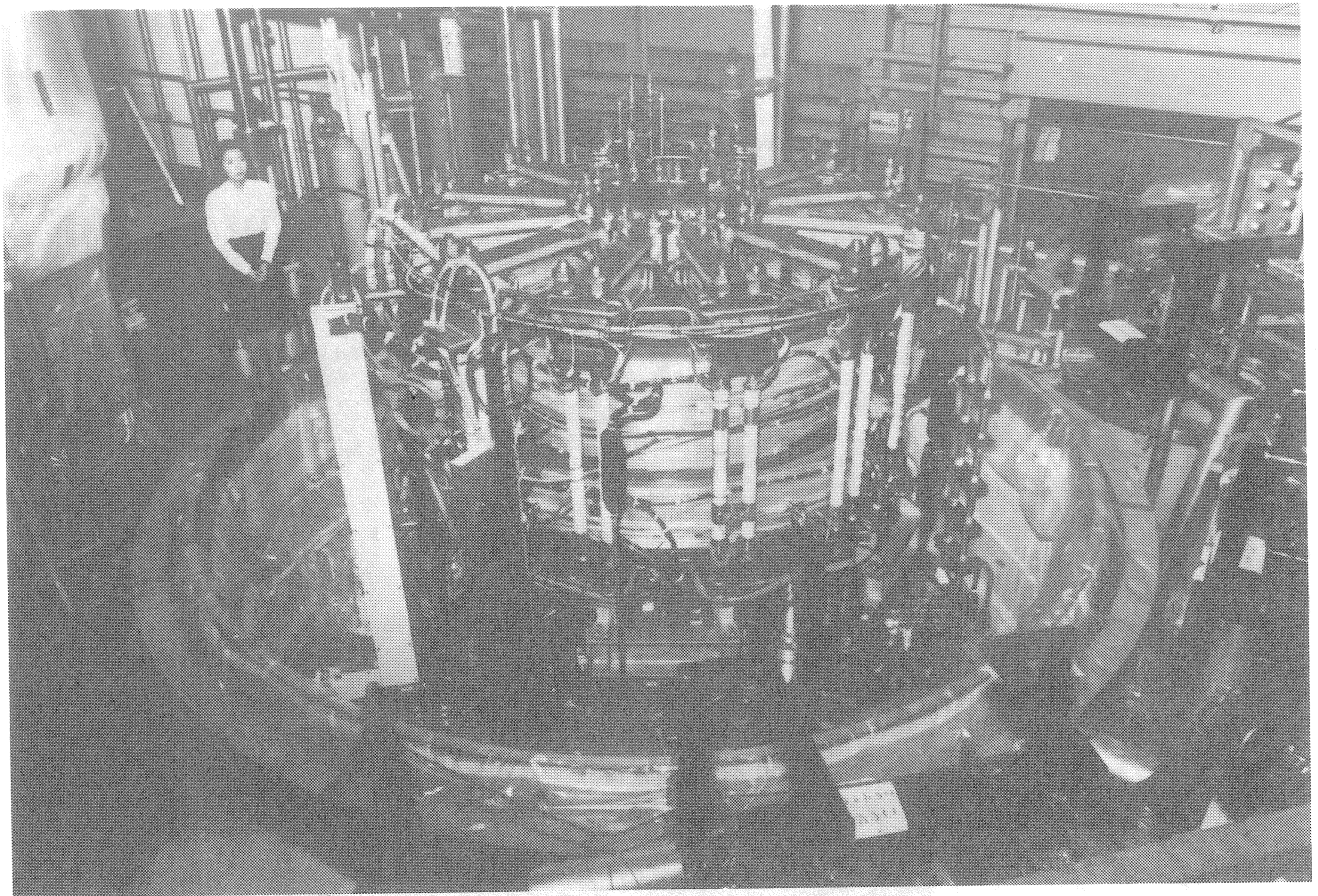


Fig. 3. Demo Poloidal Coil system. From top to bottom, DPC-U1, DPC-EX and DPC-U2 are shown. The DPC-EX coil at the center achieved 7T/s pulsed operation with background field of DPC-U1 and U2.

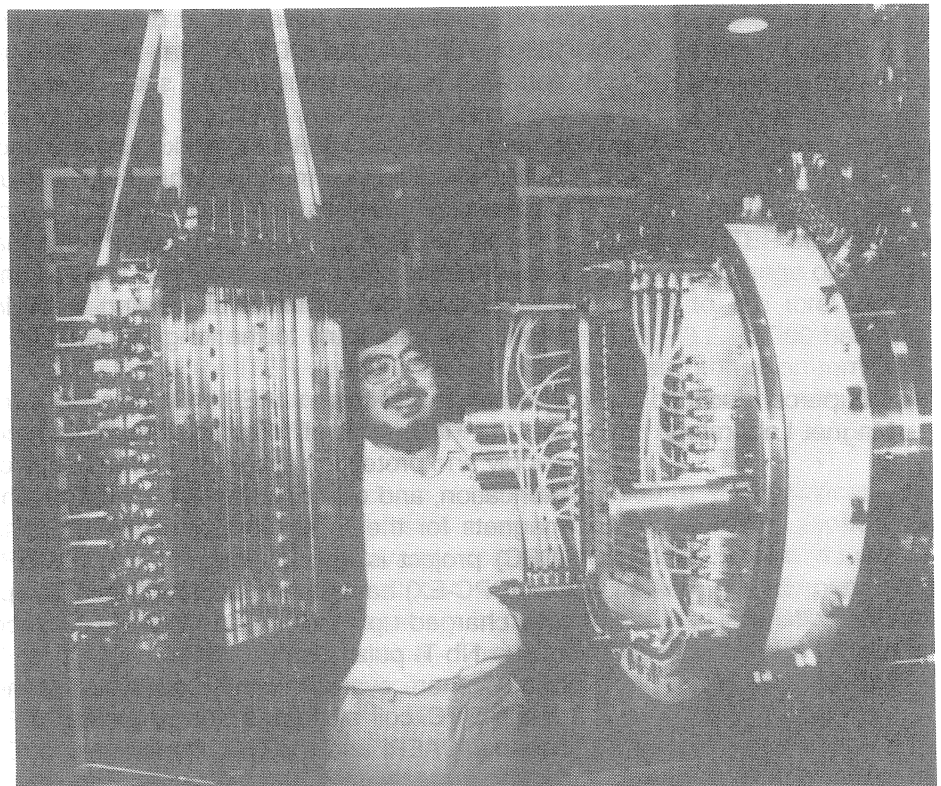


Fig. 4. Negative ion source, which reached 10A with 50keV for 0.1 seconds.

in a test stand (DPCF) from this autumn. Based on the long experience in the development and testing of superconducting magnets, JAERI has contributed to the ITER design and will develop the coming R&D works.

**Negative-ion
system**

Negative-ion-based neutral beam injectors have been developed since 1984 at JAERI. The first ion source produced only 7.5 mA in 1984 and it has been improved to reach 3.4 A in 1989. The ion source is shown in Fig. 4. Recently the current was tripled by introducing cesium in the source to get 10.1 A at 50 keV for 0.1 seconds. The experiment with higher energy beam with 350 keV and 0.1 A has started aiming at a negative-ion-based NB injectors with 0.5 MeV and 10 MW deuterium beams for the JT-60U by 1994. The Advanced Injector Test Stand (AITS) is being planned aiming at 1.3 MeV and 5 A for ITER.

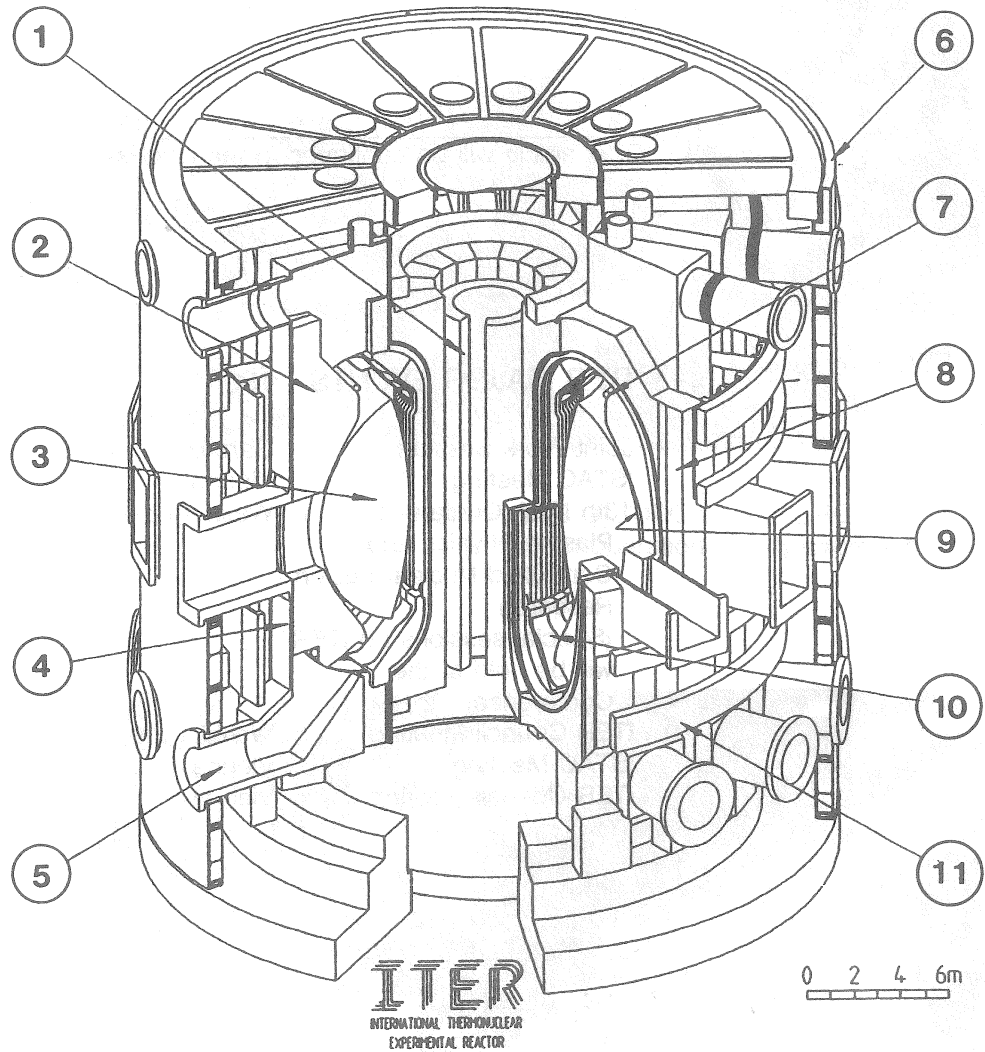
(This article will be continued in the next issue of the Newsletter)

ITER MAJOR EVENTS - 1990

Joint Work Session	Garching	2 July - 16 Nov
ISTAC Meeting	Vienna	12 - 14 Sep
13th IAEA Conference on Plasma Physics and Controlled Nuclear Fusion Research (Special sessions on ITER will be held at the Conference)	Washington	1 - 6 Oct
ITER Council Meeting	Washington	8 - 9 Oct
ISTAC Meeting	Vienna	28 - 30 Nov
ITER Council Meeting	Vienna	11 - 12 Dec

ITER REFERENCE PARAMETERS

Plasma major radius, R (m)	6.0
Plasma half-width at midplane, a (m)	2.15
Elongation, 95% flux surface	1.98
Toroidal field on axis, B_0 (T)	4.85
Nominal maximum plasma current, I_p (MA)	22
Nominal fusion power, P_f (MW)	1000



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|-------------------------|-------------------------|--------------------------|
| 1- CENTRAL SOLENOID | 5- PLASMA EXHAUST | 9- FIRST WALL |
| 2- SHIELD/BLANKET | 6- CRYOSTAT | 10- DIVERTOR PLATES |
| 3- PLASMA | 7- ACTIVE CONTROL COILS | 11- POLOIDAL FIELD COILS |
| 4- VACUUM VESSEL-SHIELD | 8- TOROIDAL FIELD COILS | |

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