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INSIDE

ITER PROGRESS NOTED AT US-USSR SUMMIT

Quest for fusion power being advanced by ITER

Presidents Bush and Gorbachev, at the conclusion of their meeting in Washington at the beginning of June, issued the following joint statement.

"At their meeting in Geneva in 1985, the leaders of the United States and the Soviet Union emphasized the importance of the work aimed at utilizing controlled thermonuclear fusion for peaceful purposes, and advocated the widest practical development of international co-operation in obtaining this essentially inexhaustible source of energy for the benefit of mankind.

"The International Thermonuclear Experimental Reactor (ITER) project, involving joint efforts by the USSR, the United States, Japan and the European Community, under the aegis of the International Atomic Energy Agency, is making significant progress towards this end. A conceptual design will soon be completed.

"Noting with satisfaction the results being attained under this project, the United States and the Soviet Union look forward to continued international efforts aimed at promoting further progress in developing controlled thermonuclear fusion for peaceful purposes."

REVIEWS BY ITER PARTIES OF THEIR FUSION PROGRAMMES

by John F. Clarke, ITER Council Chairman

Concurrent reviews of fusion programmes of ITER Parties under way

Fundamental reviews of the ITER Parties' fusion programmes are presently being conducted. The EC programme is being reviewed by the European Communities Fusion Review Board, chaired by Professor U. Colombo. The US programme is being reviewed by the Department of Energy's Fusion Policy Advisory Committee, chaired by Dr. H. Guyford Stever. Following a recent technical review led by Professor T. Sekiguchi, the Japanese programme will undergo a policy review by a senior panel to be set up under the Atomic Energy Commission and its Nuclear Fusion Council. In the Soviet Union, based on a broad review of vitally important science and technology topics, the fusion programme has recently been included in the list of high-priority state programmes.

An important feature of these domestic program policy reviews is the willingness and interest on the part of the review boards to seek and listen to the views of leaders of the other fusion programmes, especially with regard to international collaboration, including ITER.

Each Party's review is hearing leaders of other fusion programmes support collaboration

ITER Council members Evgenij P. Velikhov, James F. Decker and Katsuhisa Ida have each recently spoken to the EC's Colombo Board. In the USA, ITER Council members Velikhov and Charles Maisonnier and JAERI Executive Director M. Yoshikawa have spoken to the Stever Committee. Similar appearances are expected at the Japanese review. A theme running throughout the leaders' presentations has been the value of international collaboration. In particular, they have spoken of the importance of the current experiences on ITER and the expected value of continuation of ITER activities into engineering design.

IAEA DG Blix is looking ahead to possible continuation of ITER

In the same spirit of valuing ITER work thus far, IAEA Director General Hans Blix has recently written to each of the Parties expressing his view of the productive experience. In his letters, he notes the approaching conclusion of the ITER Conceptual Design Activities, the willingness of the IAEA to continue its support of ITER if desired by the Parties, and the timely nature of the programme reviews under way. He closes by stating his hope that during these reviews "due consideration will be give to the technical and economic benefits that might accrue from a continuation of the joint effort on ITER by the world's four major fusion programmes."

OVERVIEW OF SPECIALISTS' MEETINGS

Editor's Note

Several ITER Specialists' meetings on various technical aspects of ITER design were held at Garching in February and March 1990, during the winter joint work session. The Newsletter readers have been partially informed of these events. In this issue, we present summaries of three more meetings. This time, the summaries contain a number of scientific details which, we hope, will be of interest for the fusion specialists.

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1. SHIELDING EXPERIMENTS & ANALYSIS

by W. Daenner, Nuclear Engineering Project Unit

Experimental results analyzed to provide better support of ITER design

Shielding analysts working on ITER design usually apply safety factors to their calculated results in order to account for nuclear data uncertainties, imprecise geometrical modelling, and unforeseen local shielding deficiencies coming along with the engineering design. These safety factors are in the range of 1.3 to 2.0. They are the today's best estimates, but need to be verified by uncertainty analyses and by experiment. The purpose of the specialists' meeting on Shielding Experiment and Analysis, held at Garching from 12 - 14 February 1990, was to evaluate the applicability of previous experimental results and to discuss the needs for additional experiments in support of the ITER engineering design.

Towards the end of 1990, the first version of the Fusion Evaluated Nuclear Data Library (FENDL), prepared by the IAEA, will become available. It will be immediately processed into working libraries for the most important transport codes presently in use. It combines the best data sets for 25 elements and will be distributed without any restrictions. A few additional elements were identified which should be included in the next version as well as others which require further refinement. It was also suggested to add, at a later stage, covariance data as a basis for uncertainty analyses.

The tools required for uncertainty analysis also need further development in order to evaluate the impact of uncertainties in secondary energy and angular distribution.

Previous experiments performed by ITER Parties

Previous shielding experiments were performed by all four ITER Parties, most of them in Japan. They covered bulk shielding, transmission, streaming and skyshine problems. Many of the test assemblies included materials and/or configurations which are relevant for ITER. Most of the experiments were also numerically analyzed. The deviations between calculated and measured responses are in the range of 0.4 to 2.0. It remains open whether a recalculation using the most recent database would lead to a better agreement. In this context, it should be recognized that the accuracy requested by the design engineers is demanding. The radiation loads on the toroidal field coils, for instance, should be predicted with approximately 10 to 30% accuracy.

More systematic experimental programme recommended

Because the information provided by the past experiments does not seem to be sufficient, a more systematic experimental programme is recommended for supporting the ITER engineering design work. Bulk shield experiments in a homogeneous block of 316 stainless steel, and in layered configurations of stainless steel and water, both with and without special materials to augment the shielding effectiveness, and with and without internal voids and discontinuities reducing the shielding effectiveness are proposed. Also, streaming experiments with slots, ducts and larger penetrations which are representative for the ITER design are recommended. This programme requires close international collaboration and should involve all existing and upcoming intense 14-MeV neutron sources around the world. At this time, seven suitable sources with intensities up to 5×10^{12} neutrons per second are available: two in Japan, three in the USSR, one in the U.S. and one in the German Democratic Republic; another one located in the EC will become operational next year.

2. PHYSICS AND MODELLING OF LHW-ASSISTED RAMP-UP

by J.-G. Wegrowe, Physics Project Unit and CD&H Design Unit

Lower Hybrid Waves (LHW) will be used on ITER for partial current drive and current profile control during the burn phases in the outer plasma regions. Non-inductive current ramp-up assist by LHW, for which a broad experimental data base is available, is attractive for ITER as a means to extend the burn duration or to reach higher plasma current by saving transformer flux.

A meeting was held at Garching from 26 February to 1 March 1990 with the objective to estimate the achievable performance, to identify workable operation scenarios and parameter ranges, and to define the technical requirements for this application. Modelling of LHW-assisted current ramp-up was reported; recent experimental results and the constraints imposed by the power exhaust system and the poloidal field system of ITER were presented and dicussed.

Modelling: fast and slow ramp-up scenarios

Two distinct ramp-up scenarios have been studied. The first scenario aims at assessing the achievable volt-seconds (V·s) saving in LHW assisted ramp-up done in a time comparable to the inductive ramp-up. This scenario is interesting to ensure a certain reserve in burn time and, in the physics phase, for achieving the highest current in ITER. In the second scenario, the objective is to explore slow ramp-up with large V·s saving in order to minimize the number of shots for a given fluence.

The results summarized below have still to be considered as preliminary (in particular, the plasma geometry was kept fixed during ramp-up, corresponding to a fully elongated plasma; further, the fast ramp-up studies were conducted using a confinement time reduced by a factor 1.5 to 3 with respect to the ITER scaling).

Fast ramp-up

Using a modest amount of 20 MW of LHW current drive power in a plasma of volume averaged density 1.5 to $2 \times 10^{19}\ m^{-3}$ results in a saving of more than 20 to 30 V·s with a ramp-up rate comparable to the ohmic one. This would provide an extension of the burn duration (according to ITER guidelines) by 120 to 180 s. The safety factor profiles remain essentially monotonic in these conditions. The V·s savings are found to scale approximately like the absorbed LH power during the ramp-up time; the q-profiles become non-monotonic at power levels of the order or 30 to 40 MW. The order of magnitude of flux saving (20 - 40 V·s) is compatible with the poloidal field system requirements.

Though not optimized as yet, ramp-up assist at somewhat higher plasma density, injecting a correspondingly higher LHW power (e.g. around $4 \times 10^{19} m^{-3}$, using up to 40 MW of LHW power) also offers good prospects.

One presentation included a simulation of JT-60 LHW-assisted ramp-up (using Kaye-Goldston scaling), which displays a fair agreement with the available experimental data.

Slow ramp-up at zero surface voltage

This scenario implies long ramp-up time (5 to 50 times the inductive duration) but would permit saving of the order of 100-130 V·s, if compatible with the requirements of the poloidal field system. It would offer the possibility to decrease by a factor of up to 6 the number of pulses in the technology phase, using 20 to 40 MW of LHW power at low plasma density ($n_e \simeq 1$ to $2 \times 10^{19} m^{-3}$). In this study, the ITER confinement law has been used and a limitation has been imposed on the injected power by requiring moderate non-monotony of the q-profiles through the condition $q(a)-q(0) \geq 1$.

Specific ramp-up scenario proposed

It is proposed to study an LHW-assisted ramp-up scenario in which the plasma configuration evolves in the following way: the plasma is initiated outboard (close to the LHW launchers); LHW power is then injected in a circular plasma of growing size, till elongation begins, and finally the divertor configuration is installed. An adequate path in the q-ℓ_i plane to optimize plasma stability during the ramp-up will be aimed at. This scenario with outboard initiation is similar to the one used for fast ohmic ramp-up in JET with low Vs consumption.

Technical requirements

The wave spectrum to be launched during the initial ramp-up phase requires higher $N_{\parallel}$ components than needed during the burn phase (and implemented in the present system design). One could rely upon the 'natural' broadening of low-$N_{\parallel}$ spectra observed in today's experiments, which operate in the same range of plasma parameters as envisaged for the ramp-up in ITER, to provide the necessary upshift.

Further, a flexible way to tailor the driven current profile is desirable. A means to achieve this, as demonstrated by experiments in ASDEX reported at the meeting, is to use 'composite' wave spectra, in which a small fraction of the power is launched at larger $N_{\parallel}$ than the bulk. Such a device further provides a back-up in case the 'natural' upshift would not be sufficient. Using such a scheme appears compatible with the present design of the launching system, but this point needs further analysis.

The combinations of power and densities envisaged above comply with the power exhaust system requirements in the divertor configuration. However, in the early phase of the proposed ramp-up scenario, the plasma will be circular and leaning on an outboard limiter, the implications of which have to be analyzed.

New experimental results

Results of LHW experiments on ASDEX and JT-60 were presented and discussed; they included reports on the experiments on V·s saving in JT-60 and their evaluation (full saving of the resistive V·s and 10% saving of the inductive V·s), profile control by means of composite wave spectra in ASDEX and comparison of experimental and theoretical current drive efficiency, both with and without electric field in ASDEX. These informations, together with the broad data base already existing confirm the validity of the basic features of the models and guided the definition of the technical requirements.

Recent experimental and theoretical results in LH current drive have also been presented. Among them, an H-mode obtained in limiter discharges in JT-60 and the stabilization of sawteeth and of the m=1 mode, accompanied by strong peaking of the electron temperature in ASDEX, seem worth to be mentioned here because of their potentials for LHW applications in ITER.

In conclusion, these preliminary results confirm the attractiveness for ITER of both slow and fast LHW-assisted ramp-up scenarios, though the modelling results have been obtained using various assumptions and simplifications, the impact of which has still to be assessed. Further work is needed and will be subject of continuing research activities, in particular to account for the changes in the plasma configuration during ramp-up, to ensure an adequate path in the q-ℓ_i diagramme and to explore a wider range of parameters.

3. BETA LIMITS AND PROFILES

by T. Tsunematsu and J. Hogan, Physics Project Unit

Beta limit and ITER operational space

The beta limit is one of the fundamental constraints to ITER's operational space, especially for non-inductive operation. The dependence of this limit on plasma parameters plays a major role in determining the operating parameters. The ITER physics guidelines refer to four main issues: (1) the overall beta limit, and its shape dependence, (2) the q limit and the possibility of operation with $q_\psi < 3$, (3) the profile requirement, i.e., the need to provide optimal current density profiles for the range of peaked to flat pressure profiles which may be generated in ITER, and (4) the m=1 modes, which are related to sawtooth, fishbone and internal kink effects.

Wide representation at the meeting

To confirm and improve these guidelines, an MHD specialists' meeting on Beta Limits and Profiles for ITER was held in Garching, on 5 - 7 March, 1990. There were 24 participants in the meeting: 6 from EC, 2 from Japan, 2 from the Soviet Union, 6 from the U.S., and 8 from the ITER team. A total of 18 presentations were made. In addition to theoretical reports, data from JET, DIII-D, TFTR, JT-60, PBX-M and ASDEX were presented. Twelve written summary papers were contributed. The MHD specialsts at the meeting were asked to contribute to the discussion of the impact of shape on the beta limit, the impact of profiles on the beta limit, the present status of operating regimes and margins of safety for ITER operation, and, in all these areas, to discuss the relation between theory and experiment.

From the data and discussions on these issues at the specialists' meeting, the following conclusions can be reached:

Discussions and conclusions

Concerning plasma shaping, no strong enhancement of the beta limit has yet been observed for k>2.2. The experimental results from DIII-D and PBX-M show g constant for elongations up to k=2. The operation region and the consistency of current density profiles with pressure profiles were the major areas discussed at the meeting. For $q_\psi \sim 3$, the safe value of the Troyon coefficient, g_s (recommended for ITER operation) is found experimentally to occur at low internal inductance ($\ell_i(3)\sim 0.65$). The operating space (g-ℓ_i) in the high g (≥ 3), $q_\psi \sim 3$ region is considerably narrower than for higher q_ψ or for lower g. Values of $g^{max} \simeq 4$-5 have been observed for $q_\psi \sim 5$-6, and a broad region in ℓ_i with $g>3$ is observed (g^{max} is the largest value of the Troyon coefficient observed, even on a transient basis). The operational space for $q_\psi < 3$ is narrower than for $q_\psi \geq 3$, and the disruption frequency (which has a value 0-10% for $g \leq g_s$) increases for $g/g_s > 1$.

A theoretical model based on neoclassical MHD equilibrium shows that strong shear stability due to $q(0) \lessapprox 1$ is found to stabilize Mercier and ballooning modes. When $q(0)<1$ the sawtooth (reconnection) or resistive kink modes become unstable. It is important to include these effects in the model. Furthermore, self-consistent approaches (e.g., using the neoclassical MHD stability model) are necessary for further work. Non-MHD effects were also discussed. Tearing modes are found to be stable for ITER parameters with $k>1.5$, except at low beta. High-n modes are stabilized by compressional effects of alpha particles in the low frequency regime, and destabilized by resonance with trapped particle precession.

Based on discussions and the conclusions of the meeting, the Physics Project Unit recommends:

Present ITER guidelines generally confirmed; recommendations for further study provided

1. The present ITER guidelines are confirmed in general. There is some possibility for an increase in g_s (the safe Troyon factor) for operation with higher q_ψ.

2. Although high beta values are obtained with $q_\psi < 3$, there is no strong recommendation for operation at lower q because of the observed increase in disruption and/or beta collapse in this region. Important new data on reliability against disruption and beta collapse as a function of g/g^{max} has been presented which shows that the disruption frequency is $\sim$0-10% for $g \leq g_s$.

3. Further detailed study of $m=1$ processes is required, both as to the effect on ideal limits, and with regard to the interaction with kinetic processes.

Editor's Note

This article is continued from the previous issue of the Newsletter, where the background information on the ORNL and its missions was presented and the Laboratory activities in ITER-related engineering design and analysis, and fusion materials studies were described.

THE OAK RIDGE NATIONAL LABORATORY (ORNL) *

by C.C. Baker, E.E. Bloom, M.J. Gouge, J.T. Hogan, R.C. Isler, D.C. Lousteau, N.A. Uckan, D.W. Swain

Pellet fuelling

The ITER design uses an advanced, high-velocity pellet injection system to achieve and maintain ignited plasmas. ORNL staff have participated in a series of fuel cycle expert meetings at Garching over 1988-1990 and have developed a conceptual design description for a flexible plasma fuelling system.

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* Operated by Martin Marietta Energy Systems, Inc., under contract DE-AC05-84OR21400 with the U.S. Department of Energy.

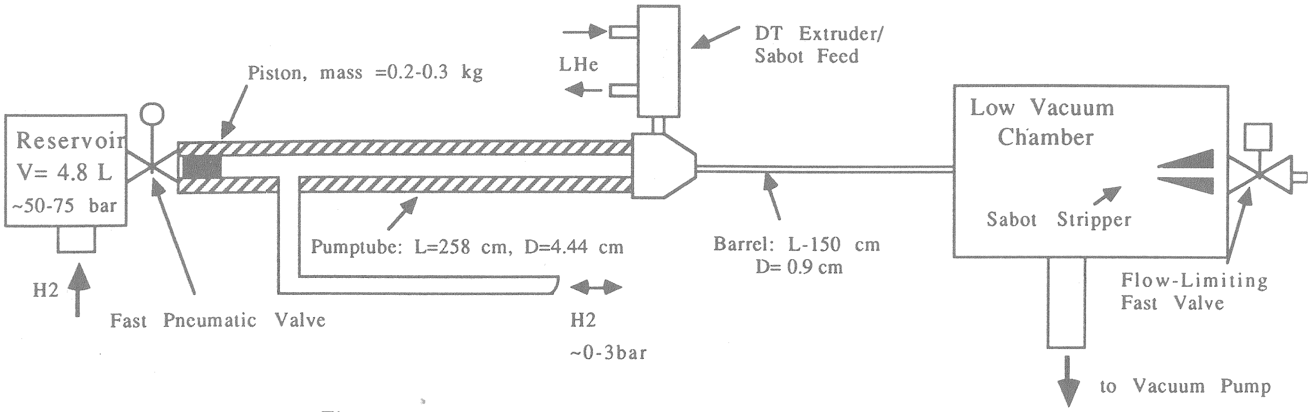


Fig. 1. Two-stage pneumatic pellet injector for ITER.

Three pellet injectors are proposed in addition to the traditional gas-puffing system. For ramp-up to ignition, a highly reliable, moderate velocity (1-1.5 km/s) single-stage pneumatic injector and a high-velocity (4-5 km/s) two-stage pneumatic pellet injector (Fig. 1) using frozen hydrogenic pellets encased in sabots will be used. For the steady-state burn phase, a continuous, single-stage pneumatic injector is proposed which will provide a flexible fuelling source well beyond the edge region to aid in decoupling the edge region (constrained by divertor requirements) from the high-temperature burning plasma. All three pellet injectors are designed for operation with tritium and D-T feed gas. Issues such as performance, neutron activation of injector components, maintenance, design of pellet injection vacuum line, gas loads to the reprocessing system and equipment layout have been addressed.

ORNL is also conducting research and development in support of ITER fuelling requirements. Tritium single-pellet fabrication and acceleration experiments were conducted at the Tritium Systems Test Assembly at the Los Alamos National Laboratory in 1988-1989, and a programme has been initiated to develop tritium-compatible extruders for long-pulse fuelling. High-velocity pellet accelerators are under development, including a repetitive two-stage pneumatic injector (4-5 km/s) and, in the longer term, an electron-beam-heated, mass-ablation system. ORNL is also involved in operating and maintaining state-of-the-art pellet fuelling systems on TFTR, Tore Supra, and JET, which will provide an operations and reliability data base for the ITER extrapolation, in addition to their contribution to the physics of pellet-fuelled high-temperature plasmas.

Fast-wave current drive

The ORNL theory and RF development groups have been working for the last year and a half on the design of a current drive system for ITER that uses waves in the ion-cyclotron range of frequencies (ICRF) to drive a steady-state current. For ITER, fast-wave current drive (FWCD) has been calculated to be competitive with using neutral beams as current drive sources. Although the efficiency of FWCD is less, the absence of a high energy beam beta component allows a comparable amount of current to be driven for equal injection powers.

In conjunction with other ITER collaborators, a FWCD system has been designed which will operate from 20-60 MHz. This system can be operated at 20 MHz to deliver all of the RF power to the electrons for current drive, or it can be tuned to ion resonance frequencies for direct ion heating to ignition. The FWCD system will use an array of up to 60 contiguous antennas that will fit in the space above the ITER main horizontal ports, as shown in Fig. 2. Each antenna is driven by a separate RF transmission and matching line, which is connected to a 3-MW RF power system. Up to 150 MW of RF power can be delivered to the plasma using this system.

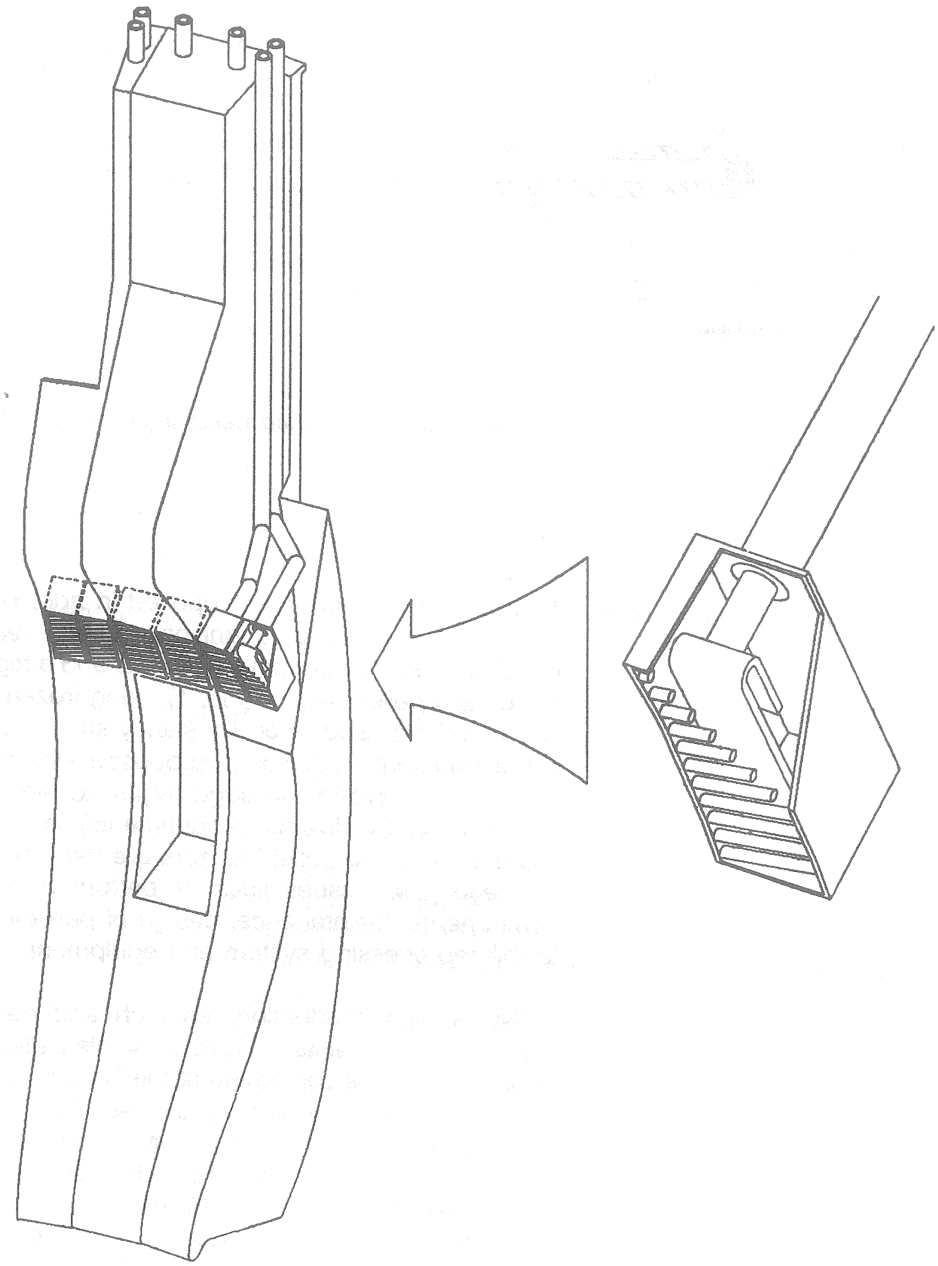


Fig. 2. ITER fast-wave current drive antenna located in outboard blanket.

Improvements in antenna technology to allow higher power per antenna (with correspondingly fewer antennas) will result in a more compact system that would take less blanket space. A conceptual design for a high-power ITER antenna with Faraday shields that can withstand the steady-state radiation environment of ITER, and be remotely maintenable, is being done. The RF development group also is working with the ITER team to clarify the interfaces between the antenna and transmission line system with the rest of the tokamak, most notably the blanket/shield assembly, and the routing of the transmission lines in the torus room.

ORNL is also working on a collaborative FWCD programme on the DIII-D tokamak at GAT. A FWCD proof-of-principle experiment, using a four-element antenna array designed and built by ORNL, should provide initial experimental results by autumn of 1990.

Plasma theory

The ORNL theory programme is involved in several aspects of ITER Physics Group activities. Members of the group have served on the U.S. team at IPP

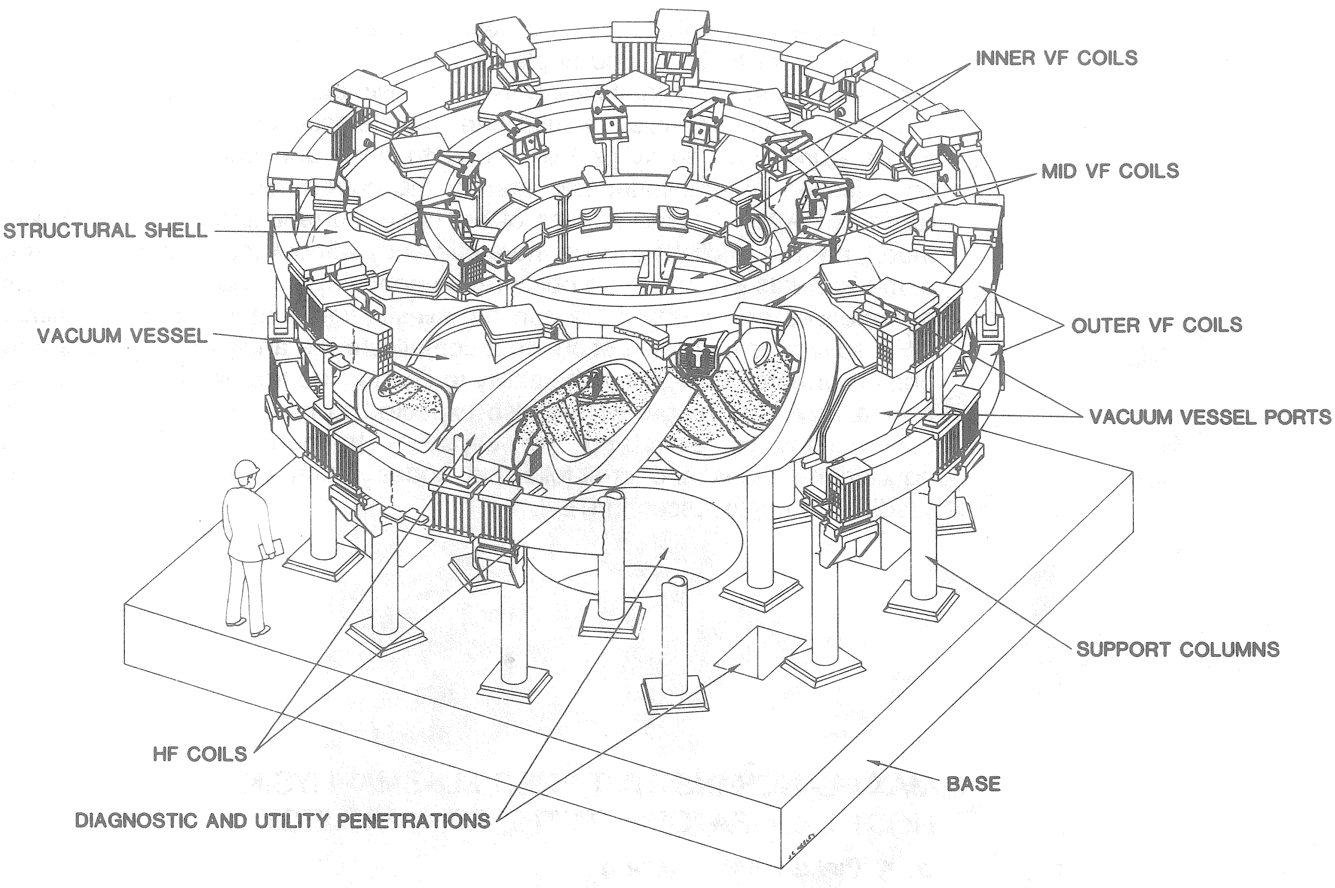


Fig. 3. Schematic diagramme of the Advanced Toroidal Facility showing the two sets of helical and three sets of vertical field coils.

during the 1988-1990 Joint Work sessions. Contributions have been made in preparing physics specifications, describing and evaluating confinement scaling predictions for ITER performance, estimating limitations to the operational space due to MHD instability (ideal and resistive, linear and non-linear, fluid and kinetic), assessing volt-seconds requirements for current ramp-up by analyzing DIII-D and TFTR data bases, simulating ITER operational scenarios, calculating prompt ripple losses, assessing fuelling requirements and in helping to co-ordinate the many ITER physics R&D contributions which have been received from experimental groups. Work at present is focussed on updating the physics guidelines in all areas, assessing the impact of current-driven operation on stability, analyzing MHD aspects of single null operation, looking at ramp-rate scaling of volt-seconds consumption (with the TFTR group), improving the guidelines for helium ash concentration (with the TEXTOR group at KFA/Julich) and preparing for this year's R&D submissions.

Advanced Toroidal Facility

Experimental plasma confinement studies center on the ATF, which began operating in January 1988. ATF is an $\ell = 2$ torsatron having 12 field periods. The major radius is 210 cm and the average minor radius is 27 cm. Plasmas are formed using up to 400 kW of 53 GHz ECH power; 2 MW of neutral beam power is available for additional heating. A versatile set of vertical field coils (Fig. 3) permits a great deal of flexibility for studying different magnetic configurations, for changing the shear of the field, and for producing either a magnetic well or a magnetic hill. Since currents are not necessary to generate a confining configuration, the device can, in principle, be operated in a steady state. ATF was designed specifically to have direct access to the second stability regime, and in the first year of operation this feature was verified experimentally [J.H. Harris et al., *Phys. Rev. Lett* 63:1249(1989)].

ATF has been contributing to the near-term ITER R&D effort in the areas of transport, density limits, and ripple loss studies. Because it is a currentless device, it has some unique capabilities for basic plasma physics studies. For example, it is possible to examine the effects of changing electron temperature and density profiles on fluctuations and transport without the complication of simultaneously altering the current profile as occurs in a tokamak. Also, detailed studies of bootstrap current and comparison with theory are accomplished much more efficiently without the presence of large ohmic currents which tend to obscure the bootstrap contribution. The effects of impurities on density limits can be investigated without the concomitant MHD activity or hard disruptions that complicate the analysis in devices with ohmic currents. Because of the ability to change the trapped particle fraction over a very wide range, ATF is also expected to make significant research contributions by exercising this capability to benchmark nonzero-beta equilibrium codes and orbit-following codes which can then be used to predict the ripple losses in ITER.

MAX-PLANCK-INSTITUT FUER PLASMAPHYSIK - HOST LABORATORY TO THE ITER PROJECT

by K. Pinkau, IPP Garching

IPP history

The "Institut fuer Plasmaphysik" (IPP) was established in 1960 as a joint stock company at Garching bei Muenchen, Federal Republic of Germany. The company associates were the Max Planck Society and Professor Werner Heisenberg. Since 1961, IPP and the European Atomic Energy Community (EURATOM) have been party to an agreement on association in the field of plasma physics and controlled thermonuclear fusion. In 1971, IPP was incorporated in the Max Planck Society. It is located at Garching research site, one of the largest research centres in Europe, which also accommodates a number of other scientific institutes of the Max Planck Society, of Munich's two universities, and of the Bavarian Academy of Sciences.

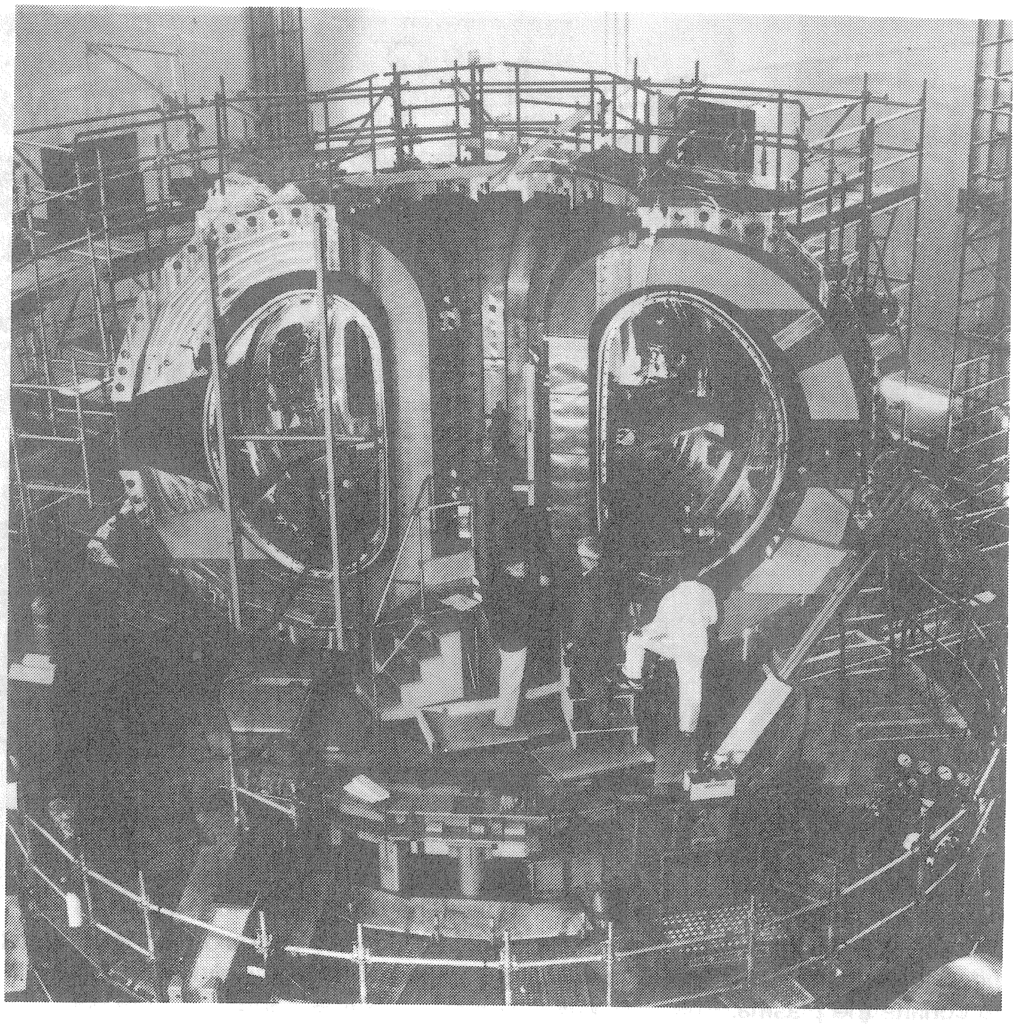
Main research directions

IPP has some 1,000 members of staff, a quarter of these being scientists. They are conducting research in three fusion experiments: the ASDEX tokamak and its successor, ASDEX Upgrade - construction of which is nearing completion - and the WENDELSTEIN VII-AS stellarator experiment. IPP is the only fusion centre in Europe that is active in both the tokamak and stellarator lines. In addition, IPP is treating questions of plasma physics in the divisions for theory, surface physics, technology and computer science.

Tokamak experiments

The main objective of the ASDEX tokamak experiment was to produce a plasma with the degree of purity required in a future fusion reactor. For this purpose special magnetic field coils generate an auxiliary field which cleans the plasma by removing its boundary layer. This divertor system has made it possible to reduce impurities in the plasma substantially in relation to comparable devices without divertor. In 1982, the divertor on ASDEX allowed the extremely important discovery of a plasma state characterized by good thermal insulation and high density, the so-called H-regime. The good results thus obtained on ASDEX have led to inclusion of the divertor in the designs being prepared throughout the world for the next generation of fusion devices. In anticipation, the ASDEX Upgrade follow-up experiment will pursue two main aims: to find a divertor configuration suitable for a fusion reactor and to study the interaction between the plasma and vessel wall under reactor-like plasma boundary layer conditions. ASDEX Upgrade will commence operation in autumn 1990.

Discovery of H-mode



View of the interior of the ASDEX Upgrade fusion experiment during assembly. This first half of the torus, comprising half of the ring-shaped plasma vessel enclosed by magnet coils and their support structure, was completed in January 1989.

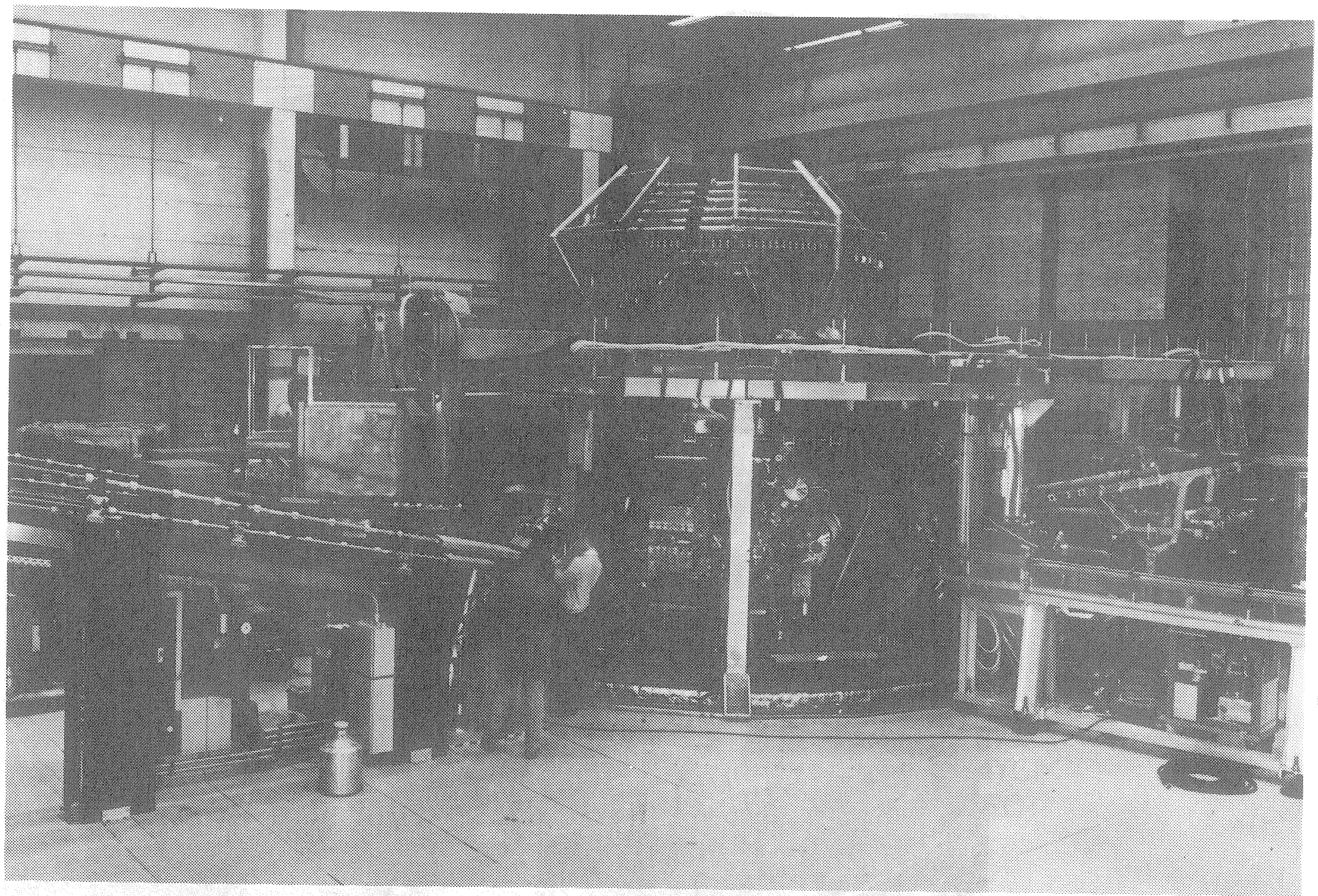
Main characteristics of ASDEX Upgrade

Plasma major radius	1.65 m
Plasma minor radius	1.00 m
Plasma vertical elongation	1.6
Toroidal magnetic field	3.9 T
Plasma current (max)	2 MA
Discharge duration	10 s
Heating power	15 MW

Stellarator programme

Besides the tokamak, the alternative confinement concept being investigated by IPP is the stellarator. The stellarator principle, unlike the tokamak, allows the plasma to be confined without enlisting the magnetic field of the plasma current, i.e. by applying a field generated solely by external magnet coils. This dispenses with the need for a transformer, thus affording the possibility of steady-state operation. Plasma confinement in an externally generated magnetic field also promises improved control of plasma instabilities, which in tokamaks are induced by the plasma current.

In 1980, it was demonstrated on the WENDELSTEIN VII-A stellarator for the first time that it is possible to confine a hot plasma solely with external magnetic fields. Since 1988, IPP have been operating WENDELSTEIN VII-AS as successor to WENDELSTEIN VII-A. This device is the first of a new



The WENDELSTEIN VII-AS stellarator experiment, put into operation in 1988: a stellarator dispenses with the need for the magnetic field of the plasma current and relies solely on an externally generated magnetic field to confine the plasma.

generation of advanced stellarators in which novel improved strategies developed at IPP are being subjected to first experimental tests. With WENDELSTEIN VII-AS, the aim is to investigate plasma confinement without internal current for lengthy periods and with improved, "optimized" magnetic fields. For this purpose the plasma, four times as large as that of WENDELSTEIN VII-A, is confined by 45 modular magnetic field coils of a new type.

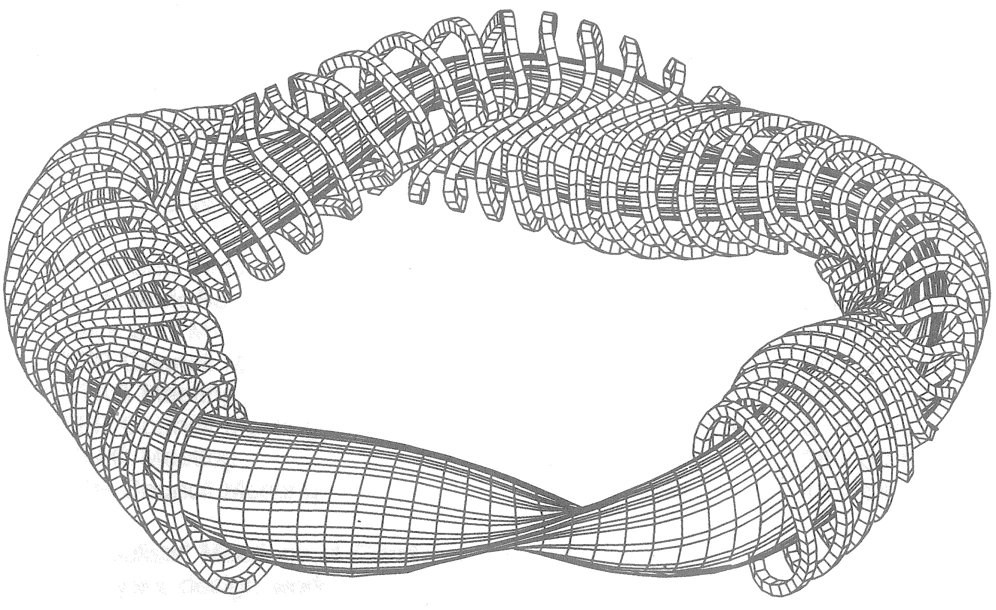
Main characteristic of WENDELSTEIN VII-AS

Plasma major radius	2 m
Plasma minor radius (mean)	0.2 m
Number of nonplanar coils	45
Toroidal magnetic field	2.5 - 3 T
Rotational transform	0.25 - 0.6
Plasma current	0
Discharge duration	3 s
Heating power	2.6 MW

While WENDELSTEIN VII-AS was being built, the numerical and theoretical stellerator studies were intensified. Thus, there are grounds for hoping that a further optimized stellarator experiment, WENDELSTEIN VII-X, now being planned, will be capable of demonstrating that the new stellarator concept is suitable for a fusion reactor.

International collaboration

The work being done at IPP is subject to international co-ordination. European co-operation is regulated by an agreement on association with the



Computer graphic of the magnet coils and plasma of the WENDELSTEIN VII-X stellarator experiment, now being planned at IPP.

European Atomic Energy Community (EURATOM), who also share the funding of IPP with the Federal Government and the State of Bavaria. This collaboration is manifested in the active participation of IPP in the Joint European Torus (JET) experiment. IPP supports this experiment not only by delegating scientific and technical personnel but also by developing, constructing and operating various plasma facilities. These are based on the experience gained at IPP and are primarily concerned with the problems of plasma contamination and plasma position control.

All these investigations serve as the basis for IPP's contribution to planning the next generation of large fusion experiments in the European context with Next European Torus (NET), or - as an alternative - on an international scale with the International Thermonuclear Experimental Reactor (ITER). In 1983, an international study group was installed at IPP to handle the design phase of NET. The NET team comprises about 60 scientists from every country of the European Community and from Sweden and Switzerland. The ITER team has been hosted by the EC Party through IPP since 1988, when the joint work for the ITER Conceptual Design Activities began.

Hosting the ITER team

ITER EVENTS CALENDAR - 1990

Major Events:

Ways and Means Working Party Meeting	Tokyo	6 - 8 June
Joint Work Session	Garching	2 July - 16 Nov
Ways and Means Working Party Meeting	Garching	10 -13 July
ITER Council Meeting	Vienna	26 - 27 July
ISTAC Meeting	Vienna	12 - 14 Sep
ITER Council Meeting	Washington	8 - 9 Oct
ISTAC Meeting	Vienna	28 - 30 Nov
ITER Council Meeting	Vienna	13 - 14 Dec

Specialists' Meetings at Garching in support of joint design work:

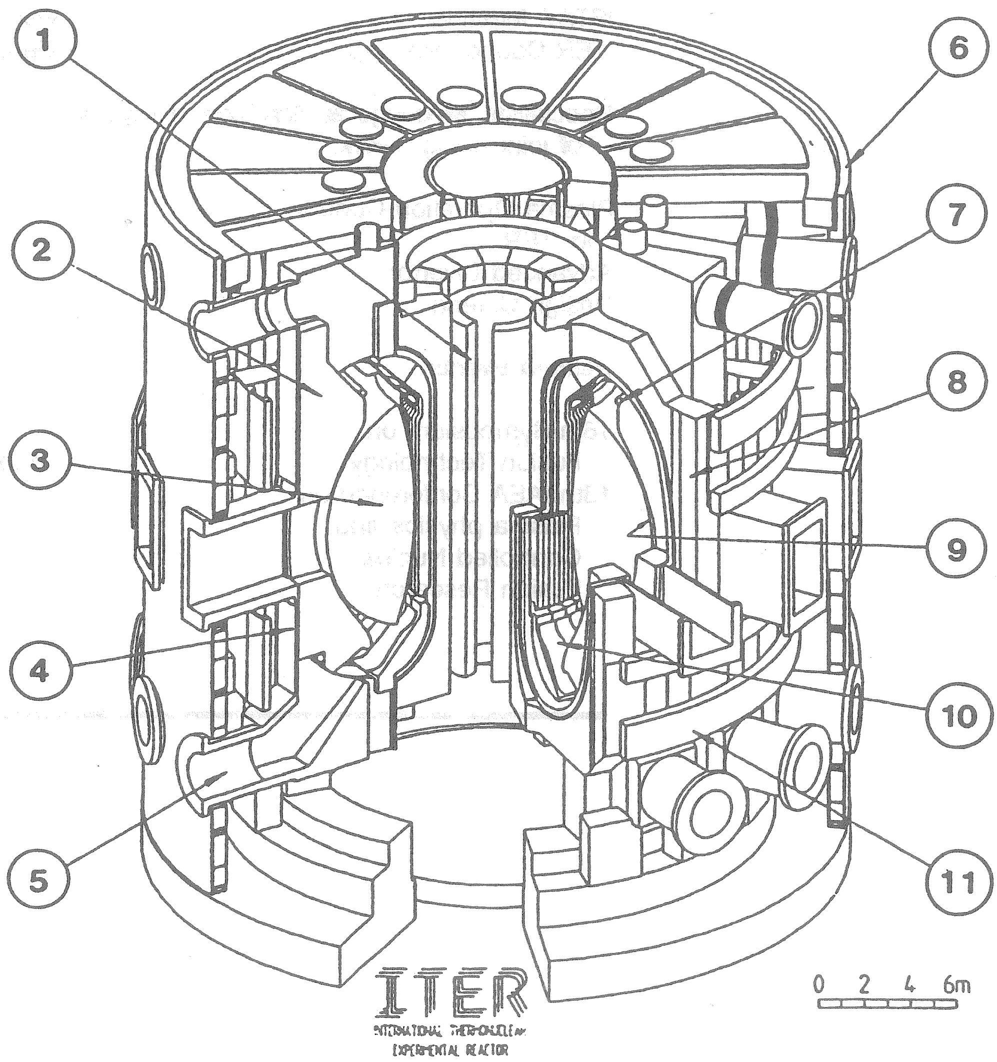
Plasma Operation Control in ITER		23 - 27 July
Advanced Divertor		20 - 24 Aug
Design Criteria		10 - 14 Sep

Related Events:

16th Symposium on Fusion Technology	London	3 - 7 Sep
13th IAEA Conference on Plasma physics and Controlled Nuclear Fusion Research	Washington	1 - 6 Oct

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_o (T)	4.85
Nominal maximum plasma current, I_p (MA)	22
Nominal fusion power, P_f (MW)	1000



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

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