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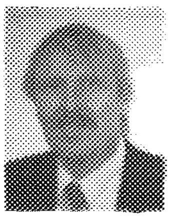
DIVERTOR DEVELOPMENT PROJECT *

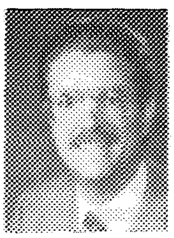
by Drs. R. Tivey and M. Ulrickson



Richard Tivey joined the Divertor & Plasma Interface Division of the Garching JCT in March 1994 and has been Group Leader of the Divertor Engineering Group since August 1996.

Prior to joining ITER he spent 12 years at JET initially in the Neutral Injection Group designing high heat flux components for the beamlines and later in the First Wall Division designing divertors.

He holds a B.Sc. degree in mechanical engineering from the University of Salford, England.



Michael Ulrickson joined the US ITER Home Team in August 1994 as manager of the divertor development effort.

Prior to joining the ITER team, he spent 18 years at Princeton Plasma Physics Laboratory developing plasma facing materials and components for TFTR and several other machines. He won the David Rose award for excellence in fusion engineering in 1988.

He holds a Ph.D. degree in nuclear physics from Rutgers University.

The design of the ITER Divertor, which must sustain high heat flux and be capable of regular replacement, is one of the most demanding problems faced by the ITER team. Two large projects have been put in place: one to cover the remote replacement of the divertor (reported in Newsletter Vol.5, Nr.8); and one to develop the technology needed to reliably construct full-scale divertor components; this is described below.

The ITER Divertor, with an X-point inside the vacuum vessel, channels particles and energy along the open magnetic field lines just outside the separatrix towards a target and pumping region. To minimize heat loads and erosion of the divertor target plates, the ITER Divertor is based on a concept where much of the power is exhausted perpendicular to the field lines through atomic and molecular interaction with a neutral gas target in front of the plates. It is predicted that most of the energy will be lost by radiation and charge exchange, and with the exhaust area increased to in excess of 250 m^2 the average power density on the plates should be less than 1 MW/m^2. However, the design anticipates that the plasma might partially reach the target plates, and hence a steady-state flux to the targets of up to 10 MW/m^2 at the strike point is catered for. In addition, occasional so-called slow transients are expected to deposit up to 20 MW/m^2 for as long as 10 s on to the target plates and the design assumes 10% of the pulses will end in disruption, depositing 100 MJ/m^2 in 1-10 ms on the plasma facing components (PFCs).

The Divertor installed inside the Vacuum Vessel is a 1500 ton assembly of 60 separate modules or "cassettes" and the segmentation of the Divertor in this way facilitates the rapid exchange of a divertor assembly in the Vacuum Vessel. A cassette is based on a stainless steel body which performs several functions: it contributes to the neutron shielding of the magnets, through pumping ducts incorporated in the body it allows helium ash and other impurities to be pumped away, and it both mechanically supports and acts as a coolant routing for the PFCs. The PFCs comprise a dome to protect the leading edges of the wings which form the transparent wall of the gas box, a power exhaust region, where the gas box louvered

* This article is a continuation of the publications relating to the ITER 7 Large Projects. For the previous articles in this series, see Newsletter Vol. 5, No. 8 (Divertor Module Remote Handling Project) and Vol. 5, No. 9 (Vacuum Vessel Sector Project Overview).

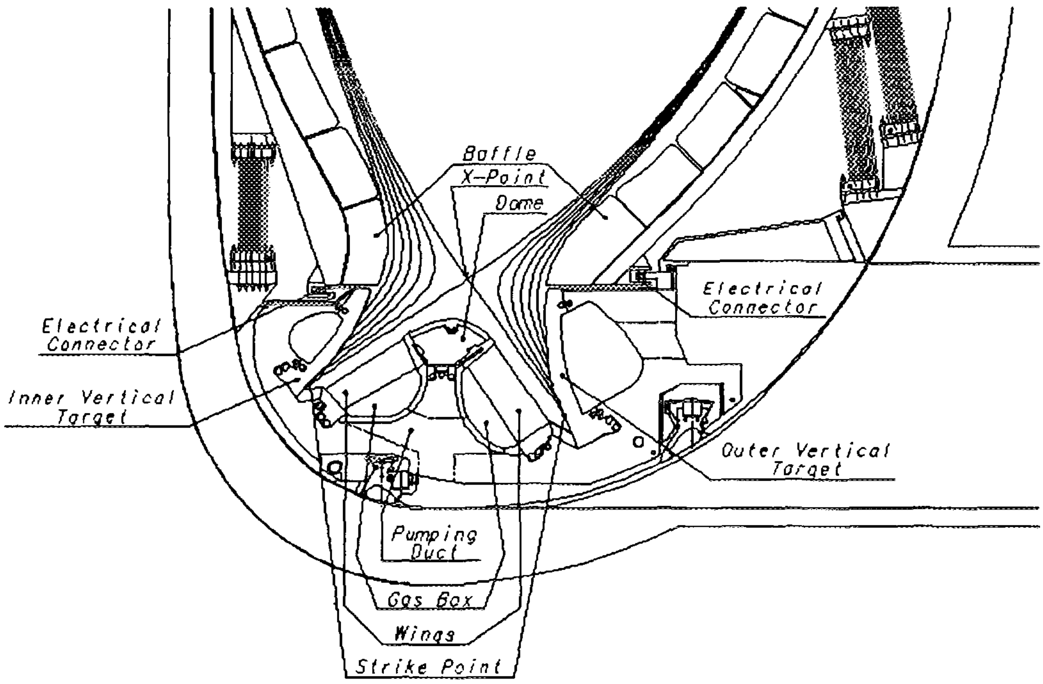


Fig. 1. Cross-Section of the ITER Divertor

structures intercept the hot neutrals from the divertor plasma while allowing the cold neutrals to circulate freely, and a vertical target which intercepts the remaining steady-state power load at the strike point (see Fig. 1). The entire cassette assembly must be capable of withstanding the thermal loads from the plasma and from the neutron flux, and the loads generated by electromagnetic forces. Since the flux strikes the target plates at a glancing angle, cassettes must be accurately aligned in order to avoid steps which would produce high-heat flux zones, such as occur when leading edges are exposed to the plasma.

The design is required to sustain 10^4 pulses of 1000 s duration during the Basic Performance Phase of ITER (the first 10 years), with up to 3 exchanges of the PFCs. These are very demanding requirements on the three candidate armours considered (beryllium, tungsten and carbon-fibre-composite). CFC capable of sustaining the slow transient is selected for the region intercepting the strike point, W with a low sputter rate appears suitable for the upper target/baffle, and both W and Be are considered as options for the dome located immediately beneath the X-point.

There existed only limited experience of constructing water-cooled high-heat flux armoured components for tokamaks, and the added complication incurred by ITER's demands for a silver-free joining technique to be employed (since silver would transmute to cadmium, an element incompatible with ultra-high vacuum) meant a whole new joining development programme was needed. Bearing in mind that the divertor requires high reliability in the performance of more than 250 m^2 of PFCs, and 100% reliability in the 100 m^2 region of the strike point, a single R&D project was conceived to focus all the R&D efforts being performed world-wide into the design, development, manufacture and testing of full-scale PFCs suitable for ITER. The task aims at addressing all the interconnected issues facing ITER Divertor design, such as providing adequate armour erosion lifetime, meeting the required armour-heat sink joint lifetime and heat sink fatigue life, sustaining thermal-hydraulic and electro-mechanical loads, and all this while seeking to identify the most cost-effective manufacturing options. The project aims are formalized in the list of objectives overleaf.

With these objectives in mind,the JCT and the 4 Home Teams will design, develop and fabricate full-scale components for integration onto a toroidal half of a cassette body. In addition, near full-size, armoured PFCs will be built and tested under high-heat flux in existing test facilities. The US, through McDonnel Douglas in St. Louis, have the responsibility for the overall integration of the task and the maintenance of the interface controls. They will fabricate a stainless steel toroidal half of the cassette body. In addition, McDonnel Douglas are responsible for the development of an armoured dome PFC. Japan is developing CFC and tungsten armoured PFCs for the inner plasma channel of the Divertor, while both the EU and the US are developing PFCs for the outer channel: the EU with tungsten and CFC armours, and the US with beryllium or tungsten armour. Finally, the RF is charged with the development of beryllium and tungsten liners to protect the remainder of the cassette body. Figure 2 illustrates the extent of the involvement of the Parties in the task.

PROJECT OBJECTIVES

- Demonstrate reliable performance of PFCs under maximum predicted steady-state load conditions and under maximum transient load conditions.
- Demonstrate feasibility of achieving desired manufacturing tolerances.
- Select manufacturing technologies for Divertor Cassette Body and PFCs.
- Demonstrate reliability of manufacture of large armoured PFCs.
- Show satisfactory armour lifetime due to sputtering and disruptions.
- Establish T retention and permeation.
- Build full-size components and identify design modifications to improve performance and manufacturability.
- Provide input necessary for producing divertor component procurement specifications.
- Develop non-destructive examination (NDE) techniques suitable for full-size components. Assess feasibility of NDE in place of thermal load testing of all PFCs.
- Refine divertor cost estimates and identify more cost-effective manufacturing methods.
- Assess manufacturing time scales.
- Demonstrate co-operation of the four Parties and the JCT in component production and testing on an engineering scale relevant to ITER and within an acceptable schedule.

The study and evaluation of plasma materials interactions is supported by all four Parties giving an input on armour erosion rates and the level of tritium expected to be retained by the armour materials.

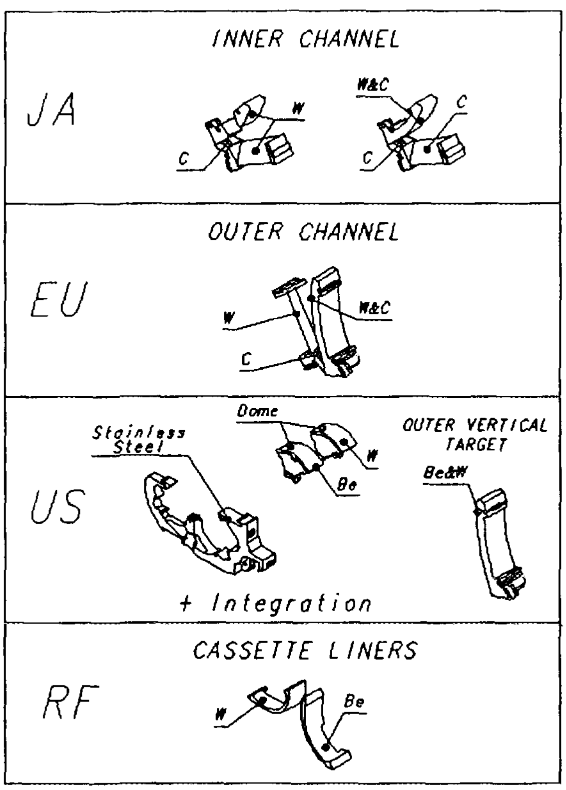


Fig. 2. Task-Sharing of the Full-Scale Divertor Components Between the Four Parties

Thermal hydraulic performance of partial models of the heat sinks at ITER-relevant coolant parameters has demonstrated that the peak flux can be handled using turbulence promoters such as twisted tapes.

Good progress has been made in developing armour joints for the strike point region of the target (see Figure 3 for summary results). In the EU, small-scale CFC monoblocks[1)] built by Plansee Metals, employing an intermediate pure copper layer cast into the CFC monoblock, which is then brazed to a Cu alloy tube using a titanium brace, have sustained 1000 cycles at 30 MW/m^2. In Japan CFC saddleblocks[2)], albeit joined to a pure Cu tube using a silver-based braze, have survived 10^4 cycles at 5 MW/m^2 and a similar design 900 cycles at 30 MW/m^2.

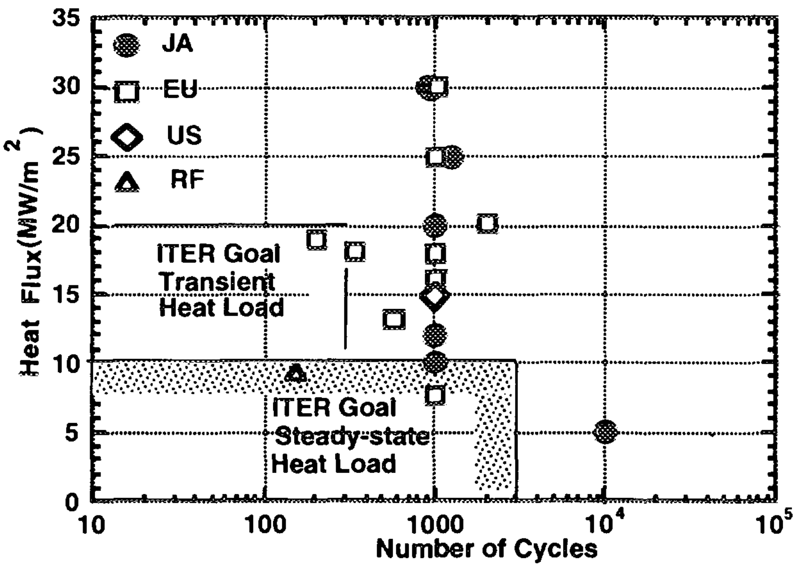


Fig. 3. Results of High-Heat Flux Tests of CFC Armour to Cu Heat Sink Joints

In the EU, a medium scale tungsten "macro-brush" mock-up, with 4 mm square pins set in a cast pure copper matrix to form tiles which are subsequently electron-beam welded to a Cu alloy heat sink, has withstood 1000 cycles at 9 MW/m^2 and a further 1000 cycles at 16 MW/m^2, where only one pin detached. This is a very good result considering that the maximum design load for tungsten armour is 5 MW/m^2. For beryllium armour, flat tiles attached to a Cu alloy heat sink with CuMnSnCe brace show no damage after 1000 cycles at 8 MW/m^2.

These are encouraging results and the next phase will test the performance of irradiated joints, and investigate the Problems of scaling up with the near full-size mock-ups.

Work by the US on the design of the cassette body support structure is well advanced in the optimization of the coolant routing to meet the hydraulic requirements of the PFC circuit and for the cooling of the bulk of the stainless steel from the effects of neutron heating. Progress is being made in the qualification of manufacturing procedures involving construction of the cassette body from near net shaped pieces obtained from metal powder or castings and densified using a hot Isostatic Pressing technique (HIP). Activities involved in the fabrication of the cassette body are on schedule and should lead to a reliable and cost effective manufacturing route being demonstrated within the EDA.

The interfaces between the components produced by the four Home Teams are essentially agreed upon and the designs of the full-scale pieces due for integration on the cassette body are well advanced with the progress in each Party beyond conceptual design and into the stage where industrial partners are performing the detailed manufacturing design. Delivery of the full-scale PFCs from the Home Teams is promised before April 1998 and their integration onto the cassette body is on schedule for completion by July 1988. However, completion of the testing of this assembly will continue beyond the EDA, as will the high-heat flux cycling of the near full-scale armoured PFCs. The project, with all four Parties contributing, is a microcosmos of the management challenges to be found in ITER as a whole, and is a demonstration that physicists and engineers from the four Home Teams can come together to work efficiently as one team striving towards a common goal.

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1) A solid rectangular block of armour with a single drilled hole is slid on and joined to a cooling tube.

2) Armour has a saddle shaped interface with the cooling tube.

16th Energy IAEA Fusion Conference

(ITER and Other Tokamak Issues)

by Drs. T. Dolan and U. Schneider, Physics Section, IAEA

The 16th IAEA Fusion Energy Conference held 7-11 October 1996 in Montreal, Canada, was attended by 547 participants and 34 observers from 36 countries. 329 papers were submitted to the Conference, from which 287 were selected by the Programme Committee for presentation (109 oral papers including 13 rapporteured talks, 178 posters including 3 post deadline papers). The number of papers submitted has increased from 270 (1992) to 309 (1994) to 329 (1996).

In his welcoming remarks Mr. Machi, Deputy Director General of the International Atomic Energy Agency, acknowledged the important progress being made on the ITER Engineering Design Activities, the inertial fusion experiments that are planned in the USA and France, the stellarators under construction in Germany and Japan, fusion research progress in some non-ITER countries, and the spin-offs from fusion technology. Referring to ITER and the other tokamaks which continue to be the main line of fusion research. He said that the ITER Engineering Design Activities are addressing the key physics and technology issues, which include H-Mode physics, divertor and edge physics modeling, tritium handling, rapid shutdown, disruption control, coil design, remote handling, and tritium inventory. He said, that.fusion research is prospering in countries, including some non-ITER countries (China, India, Korea), where the leaders are farsighted. The IAEA is working to facilitate cooperation in nuclear fusion research and therefore continues to support many activities world-wide. Mr. Machi emphasized that the IAEA is proud to be the organizer of this meeting in collaboration with the Canadian National Fusion Programme.

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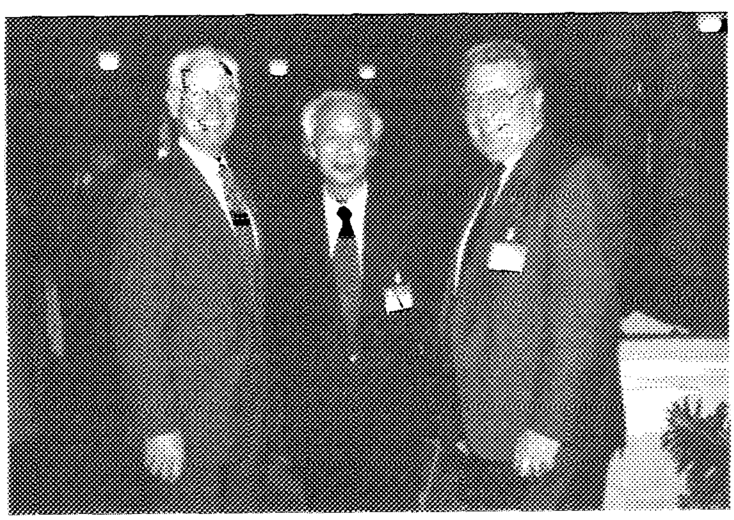
The first day of the Conference was devoted to overview talks on recent results. The progress of the ITER-EDA was the focus of attention, as in other international meetings on fusion. In the first scientific talk at the meeting R. Aymar, ITER Director, presented an overview of progress on the ITER EDA. He reported on the design of the machine and detailed plans for the design of the needed infrastructure. Mr. Aymar emphasized that it is highly desirable that the ITER device be able to operate reliably in the H-Mode (or ELMy H-Mode), the confinement time being about 2-3 times longer than in the L-Mode. He reported on the current research aimed at defining the heating power threshold, above which H-Mode operation occurs, and the precise mechanism of the H-Mode. As reported to the audience a comprehensive database of tokamak disruptions has been assembled for correlation with theory, in order to provide reliable predictions of the operating limits for ITER to avoid disruptions. Six tokamaks have conducted experiments involving the injection of "killer pellets" of impurities into tokamak plasmas to attenuate the plasma current rapidly, mitigating the effects of oncoming disruptions. As Mr. Aymar said, there is still a problem of runaway electrons produced during disruptions with killer pellets. In addition to Mr. Aymar's talk, there were five more specialized talks and 28 posters presented in sessions devoted to ITER (under the main title ITER-EDA), and many of the other conference papers dealt with ITER issues.



At the Conference (from left to right): Dr. R. Bolton, Executive Director, Centre Canadien de Fusion Magnétique, Conference Chairman; Dr. S. Machi, Deputy Director General, IAEA; Dr. R. Aymar, ITER Director

At its 11th Meeting (December 1996), the ITER Council commended the ITER Joint Central Team and the Home Teams for the quality of the ITER presentations to the IAEA Fusion Energy Conference and noted that the new experimental results have, in general terms, confirmed and strengthened the physics basis for the ITER design.

Major Scientific Results Achieved on Other Major Tokamaks

K. F. McGuire from the Princeton Plasma Physics Laboratory (PPPL) reported on recent results of the **Tokamak Fusion Test Reactor (TFTR)**, the largest tokamak in USA, working with deuterium-tritium (DT) plasmas, the fuel that will be used in ITER. He reported on self-heating effects in plasma discharges by fusion-product α-particles; the isotope effect (between hydrogen, deuterium and tritium) on the scaling of physical parameters; and investigations of the **"reversed-shear mode"** of operation. It was found that the shear reversal is accompanied by reduced plasma turbulence and lower thermal conductivity (a "transport barrier"), resulting in better energy confinement. New theories are evolving about how the variation of the ExB drift velocity in the reverse shear region suppresses the plasma turbulence. Similar reverse shear experiments are underway in other tokamaks.

K. Ushigusa, JAERI, Japan, described the successful implementation of a negative-ion based neutral beam injection for heating the **JT-60 Upgrade** plasma (Naka, Japan). With a negative ion source, neutral beams with high energies of 0.5 MeV and a high injection power of 10 MW can be efficiently neutralized permitting very deep penetration into the plasma, which will be important for the performance of the ITER experiment. The JT-60U plasma current was sustained for long periods without inductive current drive by using 4 MW of neutral beams injected parallel to the plasma current and 21 MW injected perpendicular to the current for plasma heating, with a plasma triangularity of 0.33. The JT-60U tokamak has achieved a record product of density, confinement time, and temperature, $n_o\tau_\varepsilon T_o$ = $1.5\times10^{21}m^{-3}s$ keV. Later experiments on JT-60U achieved plasma conditions that would be equivalent (if DT fuel had been used) to a ratio Q_{DT} = (fusion power)/(input power) greater than one, the **"break-even condition"** by optimizing the reversed-shear mode. The Japanese are designing a new "super-upgrade" of the present JT-60U tokamak, which will be much larger and more powerful, capable of sustaining 5-6 MA current for over 1000 s, using 60 MW of 0.75 MeV neutral beam injection plus 120 GHz electron cyclotron heating. Its parameters will be intermediate between the largest present tokamaks (JT-60U and JET) and the ITER device.

J. Jacquinot, JET Joint Undertaking, EC, gave an overview of the **Joint European Torus (JET)** research at Culham, UK, which is focusing on divertor operations. With the old Mark I divertor configuration the plasma current was increased to 6 MA, the total heating power to 32 MW and the plasma stored energy to 13.5 MJ. The new Mark II configuration, with various target plates mounted on a water-cooled structure, will demonstrate divertor performance to maintain the plasma purity under ITER-relevant plasma conditions and high heat loads. A new scaling law correlates the increase of the central plasma impurity content to the use of impurity injection for edge plasma cooling. Divertor effectiveness optimization will be needed to maintain low impurity content in ITER plasmas. JET operations in 1997 will use DT fuel and seek to demonstrate high values of Q_{DT}.

M. Kaufmann, Max-Planck-Institut für Plasmaphysik, Germany, presented recent results from the **ASDEX Upgrade** tokamak, Garching, Germany, which investigated tokamak operational limits. The empirical plasma density limit (the "Greenwald limit") was exceeded on ASDEX Upgrade by means of pellet fueling, which is of interest for the ITER experiment. Other studies included plasma edge cooling and operation of the divertor plates coated with high-Z material (0.5 mm tungsten on carbon). The use of tungsten as a plasma-facing material is considered for some components of the ITER experiment, because of the low sputtering rate of tungsten. The ASDEX-U completely-detached high-confinement (CDH) mode, which combines small, high-frequency edge-localized modes (ELMs), complete divertor detachment (separation of the hot plasma from the divertor target plates) and H-mode confinement, may also be promising for ITER operations.

V. S. Chan, General Atomics, USA, reported that the strong plasma-shaping capabilities of the **DIII-D** tokamak (San Diego, USA) have been used to achieve high-β (high plasma pressure) operation and to facilitate plasma stability and transport studies, especially for reversed-shear mode operations. Experimental evidence indicated that turbulence was suppressed in the reversed-shear region. The plasma current density profile was controlled using fast wave current drive (FWCD). The total non-ohmic plasma current driven in DIII-D exceeded 250 kA, and the current drive figure of merit is consistent with theory, in particular with the ITER scaling for FWCD.

The investigations on the **Tokamak de Varennes** (TdeV), reported by R. Décoste, Centre Canadien de Fusion Magnétique, Canada, focus on the divertor performance with regard to the power flow and helium exhaust. The conditions under which the hot plasma was "attached" to the wall or "detached" from the wall were investigated. The power input to the plasma has been increased from ohmic heating conditions (about 200 kW) by the addition of up to 1 MW of lower hybrid (LH) power. The plasma detaches from the divertor plates when the line-averaged density exceeds approximately 5×10^{19} m^{-3}. This is a well-controlled process, progressing continuously from the attached to the detached state. Divertor plate biasing has also been applied to vary the detachment characteristics and the plasma compression ratio. Researchers at TdeV are also studying central plasma fueling by compact toroid injection, a method which has the potential of facilitating operation at densities above the Greenwald limit.

The operating control system of **Tore Supra** tokamak was upgraded to facilitate real-time feedback control of global plasma parameters (internal inductance l_i, edge safety factor, plasma surface flux, etc.), as reported by B. Saoutic, Association Euratom-CEA Cadarache, France. Plasma currents of 1.7 MA were sustained for 35 seconds, and 0.65 MA pulses were sustained non-inductively (zero loop voltage) for 70 seconds. Two-minute discharges were produced in the lower-hybrid enhanced-performance (LHEP) regime, in which the electron thermal energy content exceeds the Rebut-Lallia-Watkins (RLW) scaling predictions by a factor of 1.5. Substantial progress was achieved in particle and heat control as well as in heat exhaust capabilities. Tore Supra is equipped with special modular limiter tiles that (in separate tests) sustained heat fluxes of 6 MW/m^2 under steady-state conditions at 14.7 MW/m^2 during 1000 thermal cycles without any damage. Such tiles could satisfy the thermal constraints for the ITER baffles.

These tokamak results were complemented by additional overview talks on **Alcator C-Mod**, the high-field compact-divertor tokamak (MIT, USA), on the **JFT-2M** tokamak (Japan), and on the **TEXTOR-94** tokamak (Germany), and by numerous other tokamak talks and posters under the main title "Magnetic Confinement Experiments I". Progress reports and recent results on other tokamak and non-tokamak fusion research were presented in talks and posters during sessions under the following titles: Magnetic Confinement Experiments II (helical and alternative concepts), Inertial Confinement Fusion, Magnetic Confinement Theory, Plasma Heating and Current Drive, and Fusion Technology and Reactor Concepts.

A Workshop on "Remote Participation Through Telecommunications" organized by the International Energy Agency (IEA) was held on 9 October 1996 in conjunction with the Conference. The speakers described the increased telecommunication transmission capacity and improved computer technology (hardware and software), which are rapidly expanding the potential for enhancing the effectiveness of collaboration among remote partners, especially in fusion research given its international character. The potential of remote telecommunications for ITER, the requirements, the current experience, and the critical issues were discussed.

The proceedings of the 16th IAEA Fusion Energy Conference will be published in the course of 1997 in three volumes. The 17th IAEA Fusion Energy Conference will be held 18-24 October 1998 at the Pacific Convention Plaza in Yokohama, Japan.

Superconducting toroidal field coils (20 coils) Superconductor Structure	Nb_3Sn in circular Incoloy jacket in grooved radial plates Pancake wound, steel encased wind, react and transfer technology
Superconducting poloidal field coils (CS, PF2 - PF8) Superconductor Structure	CS, PF2&7:Nb_3Sn, PF3 - PF6, PF8:NbTi Square Incoloy Jacket, layer wound for CS, pancake wound for all others CS, PF 2&7 : "wind react transfer" technology
Vacuum Vessel Structure Material	Double-wall welded ribbed shell Stainless Steel 316 LN
1st Wall/Blanket (Basic Performance Phase) Structure Materials	armour-faced modules on toroidal backplate Be armour Copper alloy heat sink Stainless Steel 316 LN structure
Divertor Configuration Materials	single null 60 solid replaceable cassettes W alloy and C plasma facing components Copper alloy heat sink Stainless Steel 316 LN structure
Cryostat Structure Maximum inner dimensions Material	Double-wall welded ribbed cylinder with flat ends 36 m diameter, 30 m height Stainless Steel 304L
Heat Transfer Systems (water-cooled) Heat released in the Tokamak during nominal pulsed operation	2200 MW at ~4 MPa water pressure, 150°C
Cryoplant Nominal average He refrigeration / liquefaction rate for magnets and Divertor cryopumps (4.5K) Nominal cooling capacity at 80 K	100 kW / 0.35kg/s 225 kW
Additional Heating and Current Drive Total injected power Candidate Additional Heating and Current Drive (H&CD) systems	100 MW Electron Cyclotron, Ion Cyclotron, Lower Hybrid , Neutral Beam from 1 MeV negative ions
Electrical Power Supply Pulsed Power supply from grid Total active/reactive power demand Steady-State Power Supply from grid Total active/reactive power demand	 650 MW / 500 Mvar 230 MW/160 Mvar

MAIN FEATURES OF THE ITER SYSTEMS AS CONTAINED IN THE ITER DETAILED DESIGN REPORT, COST REVIEW AND SAFETY ANALYSIS (DDR)

At its 11th Meeting (December 1996), the ITER Council accepted the DDR for consideration by the Parties and agreed that the technical work of the ITER EDA should continue on its basis.

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