JOINT EUROPEAN TORUS

JET–P(90)61

R.S. Hemsworth
and JET Team

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R.S. Hemsworth
and JET Team*

JET-Joint Undertaking, Culham Science Centre, OX14 3DB, Abingdon, UK

* *See Appendix 1*

Preprint of Paper to be submitted for publication in
Nuclear Fusion

SUMMARY OF THE 16TH SYMPOSIUM ON FUSION TECHNOLOGY

R S Hemsworth

JET Joint Undertaking, Abingdon, Oxfordshire OX14 3EA, UK

The 16th Symposium On Fusion Technology (SOFT) was hosted by the JET project and held at the Queen Elizabeth II Conference Centre in London, England between the 3rd and 7th of September, 1990. The majority of the 556 attendance came from European Institutions, but there were also delegates from Japan, the USA, and other countries. This conference has matured considerably over the past decade, and it now attracts numerous industrial sponsors. The conference draws its strength from two themes - invited talks and supporting oral presentations, complemented by many poster presentations. At this symposium, there were 17 invited talks, 15 oral and 321 poster presentations. The following summary groups the various papers by topic.

Fusion Programmes

Europe

Europe, Japan and the USA. all currently have large devices, JET, JT-60, DIIID and TFTR, and all have proposed new machines, either as a precursor to the "Next Step" (CIT), or as domestic versions of the Next Step (NET, FER). Also all have had favourable independent reviews of their fusion programme.

In Europe, JET has made significant steps in $nT\tau$ space, reaching values of ~ 8.10^{20} m^{-3} keV s, less than a factor 2 from Q = 1, the point at which, for a similar condition in a D^{+}-T^{+} plasma, the total fusion power release in the plasma would equal the external power deposited in the plasma. An extension and upgrade of the project is planned, with the auxiliary heating being increased to 50 MW, a pumped divertor being installed, and DT operation planned for 1995-6. The European programme benefits from complementary work on many smaller tokamaks and incorporates the alternative systems, stellarator, heliac, and reversed field pinch. The programme is aimed at providing a physics data base for the Next Step, and, although Europe is actively participating in and supporting ITER, it is continuing with NET, an alternative Next Step design. This differs from ITER in that it does not require current drive with the proposed pulse length (~ 700 s). This has a significant impact on the T handling, and T breeding is unnecessary.

Japan

The present Japanese programme centres around the upgrade of JT-60, following which it is projected to be capable of producing a 100 m^3, 6 MA diverted plasma. The machine, which will be operated in D_2, will have a carbon first wall, using CFC divertor target plates. A significant development will be the new 500 keV, 10 MW negative ion based neutral injector planned for 1994, for current drive studies. Construction has started on a new large helical device, R = 4 m, a ~ 1 m, B_ϕ = 4 Tesla, which will have superconducting coils and long pulse capability. Like Europe, Japan is both participating in ITER and continuing with its own design for a Next Step, FER. FER is envisaged as the smallest machine suitable for an experimental reactor. A pulse length of 400 s is planned with a fusion output of 400 MW.

USA

TFTR and DIIID are the mainstays of the present US fusion programme, whilst studies of the stellarator concept are continuing with the ATF device at ORNL. DIIID has now installed. An advanced divertor, which includes an electrically biased section, and (eventually) a cryo pump. The ECRH power is being increased, eventually to 7 MW at 110 GHz, and ICRH antennae suitable for fast wave current drive are planned. TFTR will continue with high power NBI (32 MW), and they are currently testing SiC composite bumper limiter tiles. The design of CIT, which is designed to be operational after the present generation of large machines, but before the Next Step, is continuing. Following the recent review by the Fusion Policy Advisory Committee, CIT is given the same priority as ITER. Construction is planned to start late 1992, with the first plasma in 1998. The objective is to learn as much as possible about plasmas similar to those expected in ITER, which is the long term US programme. The CIT parameters are: R = 2.57 m, a = 0.8 m, Ip = 11.8 MA, $B\phi$ = 9 Tesla. It will be a graphite machine, with LN_2 cooled coils and a diverted, single null, plasma. The pulse length will be 5 - 20 s, with 50 MW of auxiliary heating, although initially only 20 MW (ICRH) will be available. H_2, D_2, and eventually DT operation is planned.

Safety and Environmental Issues

A total of 21 contributions, including 4 invited talks demonstrates the increasing importance being given to safety and environmental issues by the fusion community.

Safety

In considering safety issues, fusion is considered to provide the baseload electrical power, and a variety of reactor concepts are examined, eg different types of tokamak, differing structural and divertor materials, and, for comparison, reactors using low activation materials. Resourcing is considered in some detail (eg availability of, and risks associated with, obtaining that fuel, such as deaths from mining etc), as is the total radioactive inventory, especially the "toxic content", a measure of the potential to harm adults. Tritium releases, the pathways to man of toxic substances, decommissioning and waste management have also been considered in some detail.

In general the safety issues are found to be favourable for fusion cf other energy sources. For example, the number of deaths resulting from fuel provision is approximately 2 orders of magnitude less than for coal, and the problems of fuel handling is very small compared to conventional dangerous substance, eg chlorine. A major issue is, of course, nuclear waste. The total volume of waste (including decommissioning) is estimated to be similar to that from a fission reactor, but of a very different quality - the vast majority being intermediate or low level waste, much of which will come from the concrete of the shielding building. Considerable progress has been made with safety problems involved with tritium and the modeling of the consequences of the accidental release of T_2 has been considerably improved, making use of data gathered from French and Canadian T_2release experiments. In particular, the conversion of HT and subsequent re-emission from soil is now adequately described by the models.

In a departure from convention, the science correspondent from The Independent (an English daily newspaper) was invited to address the meeting on "Is Fusion Credible?" It was pointed out that the goals of fusion are (probably) well known, but this may be insufficient to convince the public of the need for it, eg safety alone is not enough to justify an expenditure of ~ 50 B$. After considering the history of the UK fission power industry, he concluded that it is important to choose a single reactor type, build to time and budget, keep the public informed of any accidents etc, and to apply strict accounting standards.

Physics and Heating

Boundary layer physics (including divertors), neutral beam heating (NBI) radio frequency heating (RF) and the use of NBI and RF for current drive were the covered in 3 invited papers. NBI was also the subject of 8 posters whilst 27 posters were presented on various aspects of rf heating systems.

Boundary Layer Physics

Understanding and control of the outer region of a confined plasma are recognized as key issues for the future, and JET and DIIID are planning pumped divertors to try to control the plasma wall interaction, hence to control this boundary layer. Electric fields are important in the boundary layer. Defining the plasma as consisting of a core, a core edge and a scrape off layer (SOL), $E_{//}$ (the field parallel to $B\phi$) is important in the sheath and pre sheath of the SOL, whilst $E_{\perp}$ is important in the core edge. Atomic physics (excitation, ionisation etc), surface and material science all play key roles.

It is recognized that α particle build up would lead to a plasma burn of finite duration and it can be shown that the α particle density should be kept below 10 % of the plasma density. Studies at JET have shown the detrimental effects of high Z impurity build up, eg ~ 0.1 % of high Z impurity leads to the need to double the confinement time to obtain the same fusion power output. Pumped divertors or limiters are proposed to control the edge plasma so as to achieve high confinement (H-modes), to exhaust He ash, control the plasma density, to dissipate the core plasma heat flux, and to minimize the influx of impurities. Pumped divertors can exhaust plasma, reduce the heat flux to target plates and limit impurity production when operated in the high recycling mode. Analysis indicates that friction can restrain impurities generated in the divertor from escaping into the core plasma, and a gaseous divertor concept is being suggested that could be applicable to reactors.

Additional Heating

All the large present generation neutral beam injection systems (JET, TFTR, JT-60, DIIID) have delivered > 20 MW to the tokamak plasmas for several seconds (>30 MW on TFTR). Significantly, all the big systems are in routine operation with high reliability and availability, for example delivering > 90 % of the requested power for > 80 % of the shots (JET 1989).

This has been achieved through technical improvements in many areas, eg ion sources with a high monatomic species fraction, > 92 % (JT-60), and extraction grids with integral cooling (JET), which are capable of DC operation. Large scale cryo pumps are in routine operation on all large systems, having demonstrated reliability and robustness.

Neutral beam interfaces with the tokamak have all been solved, eg the beam shinethrough (graphite armour), interlock and control interfaces (careful planning and implementation), control of the stray magnetic field near the ion sources (active field compensation plus passive shielding), and the loading of the walls of the duct to the tokamak (use of good materials plus careful conditioning). Unlike radio frequency techniques, there is no problem coupling the beam power into the plasma.

Neutral beam heating has made a major contribution to the generation of high temperature, reactor relevant plasma, the following values having been obtained.

- Record peak ion temperature, ~ 30 keV in TFTR
- Record β_{tor}, ~ 9.3 %
- Record fusion product $n_i\, T_i\, \tau$, ~ 9 10^{20} m^{-3} keV s in JET
- Generation of high Bootstrap currents, < 0.8 MA
- High driven currents

The TEXTOR injectors have injected 2.7 MW of ^4He and injection of ^3He is planned on JET (< 8 MW per injector). The problem of pumping ^4He and ^3He with condensation cryo pumps has been solved by adsorption onto Ar frozen onto the pumps prior to He introduction. The pumping of ^3He requires operation with the pumping surface at < 4^o K (JET). Plans are in place for the operation of injectors in T_2 (JET, TFTR).

Neutralization efficiency dictates the use of negative ions as the basis of future neutral beam injectors. The proposed systems are much simpler than those of today, yet they offer considerable advantages having low gas flow, simple beam dumping systems, very low beam divergence, and monatomic beams. The problems associated with such injectors are the efficient generation of the negative ions, and the high energies requested (> 1 MeV). These problems are being tackled differently by each of the ITER partners. All have had some success, for example a 10 A H^- beam has been accelerated to 50 keV at JAERI. It is expected that convincing proof of principle experiments will be carried out in the next two years.

ICRH systems are being deployed on many machines, eg on ASDEX and JT-60 the systems are being upgraded to 8 and 10 MW respectively. The largest ICRH system is on JET, with an installed power of 32 MW and 8 antennae. 18 MW have been coupled into the plasma for a few seconds. Significantly, the use of beryllium for the Faraday shield (low self sputtering yield, low Z), and tilting this to make it parallel to the machine magnetic field, has solved the problem of metallic impurity release. This has allowed H-modes to be generated by the use of ICRH alone, and the extension of the density limit, which are major advances. The essential matching between the rf system and the plasma is achieved by adjusting the antenna - plasma distance. On JET the plasma is moved, and an automatic feedback system has been developed which maintains a constant antenna impedance, even during an H-mode transition. ICRH has created electron temperatures of 12 keV in JET. The antennae to be deployed on JET in its pumped diverter configuration and the 2 MW system on DIIID are designed for current drive experiments.

LHCD is widely used, eg 3 MW on ASDEX, and 8 MW is planned for both FTU and Tore Supra. The highest LHCD power coupled to a plasma is 9 MW for a few seconds (JT-60). Currents of 2 MA have been driven with an efficiency close to that required for a reactor, and H-modes have been obtained using LHCD alone (JT-60). Good matching (controlled by the plasma density at the antenna) has been achieved by moving the antenna using large bellows and hydraulic actuators, controlling the position to within the required 1 mm. A feedback control system is in preparation at JET.

1.5 MW for 0.2 s (T10) and 1 MW for several seconds (DIIID) represent the highest ERCH power launched into a plasma, and H-modes have been created on DIIID with ECRH alone. ECRH current drive has been achieved, but with low efficiency (various machines). 3 MW of ECRH is being installed on Tore Supra

RF heating systems have also been used to obtain the following:

- long sawteeth free discharges (ICRH "monsters" - JET, TEXTOR)
- low voltage start-up of a tokamak (ECRH - CLEO, DIII)
- mode control through local heating (ECRH - CLEO, DITE, TFR)

In addition to heating, rf systems for the Next Step may be used to drive current, to assist in plasma start up, for current profile control, and to control plasma disruptions. A prime consideration is the overall efficiency. For heating, the efficiency of ICRH is expected to be acceptable and a large array of up to 60 antennae is proposed for ITER, either integrated into the blanket, or within large ports. LHCD efficiency is close to that required for the Next Step, but current drive is limited to the low density region of the plasma. (> 50 MW of LHCD is requested for ITER). Problems exist with the proximity of launch structures to the plasma, eg water cooled Faraday shields for ICRH, and the complex grill structure of LHCD. For LHCD, the tracking of the plasma edge by the launcher is a formidable technical challenge. ECRH could be used for pre-ionisation, current ramp up assist and mode control. Major problems exist in producing high power sources, and long pulse operation, where cooled windows are necessary. Unfortunately the efficiency at the required frequency (> 140 GHz) is expected to be low. New sources are under active study, eg free electron lasers (Finland, USA).

Materials, Blanket Technology and Tritium Handling

Invited papers were given on developments in superconductors (1), fusion material research (1) and tritium handling (2) in addition to many poster presentations, for example blanket technology was well represented by 48 posters as well as an oral presentation.

Fusion Materials Research

Although the present fusion devices are mainly equipped with copper coil systems, 3 machine, TRIAM 2, Tore Supra, and T 15 are now operating with superconducting coils. A "conductor" in a superconducting coil actually consists of cables, constructed from strands, inside a "case". The strands are the actual superconducting element, and they are typically a bundle of superconducting filaments in a matrix, often copper. Although there exist various superconducting materials, only 3 are available commercially in quantity. These are NbTi, V_3Ga and Nb_3Sn. For fusion applications, the critical temperature (Tc), magnetic field (Bc) and current density (Jc) should all be as high as possible. Jc, is not limited by the intrinsic property of the superconducting material, but the construction of the conductor, of which it is only a few percent. Other parameters such as ductility, homogeneity etc and improvements in manufacturing technology are extremely important. For example the conductors must be produced in long lengths (kms) at acceptable cost. In the last 5 years, a good reliability of manufacturing NbTi has been achieved, and Jc increased from $1.5 * 10^9$ A/m^2 in 1980 to $3 * 10^9$ in 1988. The high fields required for fusion applications impose the use of either NbTi at 1.8 ^oK or $NbSn_3$ at 4.5 ^oK (or below). A breakthrough in the production of $NbSn_3$ was achieved in the early 70's with the "bronze process". This starts with Nb in CuSn bronze rods, where the stabilizing copper is protected from pollution by Sn by a Nb or Ta barrier. The $NbSn_3$ is formed by heat treatment (650 - 700^{o}C). This can be carried out during the coil winding, which also keeps the bronze ductile; this is the so-called "Wind and React" method. Strain can also affect the Jc of the conductor, and this imposes further constraints on the heat treatment and winding process. However, overall the design of superconductors for fusion appears to be progressing satisfactorily.

Blanket Technology

The main areas of work on blanket technology reported were design issues and activation and shielding related issues. There were also presentations on neutronics, T_2 release, corrosion and heat transfer.

A convergence appears to be occurring in the number of designs of blanket. The blanket design proposed for NET is fabricated from 316 stainless steel, with water cooling; there is no breeding, but there is provision for test breeding modules. The ITER blanket includes a breeding facility. The European version for ITER uses $LiAlO_2$ as the breeder material, which is in the form of pellets contained in a steel cladding; Be is used as a neutron multiplier. Studies were also presented on the advantages of Li_4SiO_4 and Pb-17Li as breeder materials.

Materials requirements for fusion systems are varied and demanding, eg the plasma interactive materials must have low Z, low erosion rates, and good thermal shock resistance, whereas radiation shielding could be steel, but with low activation properties. A particular problem at present is the rapid pace of design changes, which can hardly be followed by materials development. Additionally, investigations are complicated by the lack of prototypic testing, eg no intense 14 MeV neutron source exists for this purpose. However, there are plans to build the Energy Selective Neutron Irradiation Test Facility in Japan, capable of a fast neutron flux of $3 * 10^{18}$ n/m^2 over 10^{-5} m^3, and a feasibility study is underway for SORGENTINA, an accelerator based D - T source, irradiating $5 * 10^{-4}$ m^3.

There exist striking differences between fusion and fission reactors, and consequently a new database is needed. For example, in a fusion reactor welds will be exposed to a high neutron flux, and the irradiation behaviour of the weld must be characterized. Data have been gathered for many years on the behaviour of different materials in a fusion environment. To be useful for design purposes, these must be critically evaluated and processed. As a first step, it is planned to use the structure of the High Temperature Materials Databank at Petten/Ispra for collecting data from European laboratories.

Unlike fission, the radioactivity of a fusion reactor is induced, and not intrinsic to the fuel. This presents the opportunity to increase the safety and environmental advantages of fusion by the judicious selection of materials - the surface dose rate 100 years after exposure to 12 MWy/m^2 from "good" elements can be many orders of magnitude (up to 12) less than "bad" elements. Work on "Low Activation Materials" is concentrating on the specification of low activity compositions, and on the elementally modified stainless steels and vanadium alloys.

Tritium Handling

Progress in tritium handling is continuing both in the US and Europe. The JET active Gas Handling System (AGHS) uses both cryogenics and uranium beds for purification of the T_2, and much of that system is considered directly relevant to the Next Step, so valuable information will be gained from its use. This will complement work from other laboratories on purification using hot metallic beds, Pd/Ag membranes and cryosorption. Work is also being carried out with T_2 storage on metallic beds, and uranium remains a candidate material. Zr/Co has a very low hydride pressure at room temperature and it exhibits good T_2 retention. Also, 5 g of T_2 has been stored on LaNiMn for 1 year, with good He retention (99.5 %). A design has been produced for the removal of ^{41}Ar produced by activation of ^{40}Ar used for He pumping in the compound pumps proposed for NET.

The TSTA facility at Los Alamos, which already handles reactor relevant quantities of T_2, has carried out numerous experiments involving tritium, some in collaboration with Japan and Europe, including tests on an electrolysis cell at 10^6 Ci/l, a compound cryo pump, and a palladium alloy diffuser. The facility has a continuous fuel processing loop incorporating a fuel clean up system, isotope separation, T_2 and D_2 storage, tritium waste treatment and a simulation of gas from both the plasma and a neutral beam system. Significantly this loop has operated continuously for 19 days, processing ~ 1 Kg/day, the run being terminated by staff shortages.

The TSTA experience and the developments such as the JET AGHS give confidence that tritium handling will not be a problem.

Next Step Devices

Three invited papers were given on Next Step devices - The Physics of an Ignited Tokamak by Dr F Troyon, The NET and ITER Projects: Critical Design Issues by Dr E Salpietro, and Future Prospects for JET and Next Step Devices by Dr P H Rebut. There were also several oral and many poster presentations directly relevant to the Next Step, such as the materials testing, high heat flux element tests and blanket designs, some of which are covered in preceding sections.

The physics requirements for ignition are reasonably well defined, although they are dependent on the level of impurities in the plasma. Unfortunately, there exists no widely accepted transport model for tokamaks, and we have to rely on extrapolation from empirically determined scaling laws. The ITER-89(P) scaling for L-mode plasmas involves $(P_h)^{1/2}$, where P_h is the auxiliary heating power. This can be re-written to give

$$\langle p\rangle \tau_E = 1.2.10^{-2} A_i (a/R)^{0.3} (\kappa/q)^{1.7} n_e^{0.2} a^{1.7} B_\phi^{2.1}$$

where p is the plasma pressure and the other symbols have their usual meaning. $\langle p\rangle \tau_E$ is a measure of the "confinement capacity". It should be noted that P_h is not present, and that size scaling enters through I, hence through q. Effectively the P_h scaling means that increasing P_h gives horizontal motion on the standard $n_e T_i \tau_E$ versus T_i diagram - this can be considered the discovery of the '80's. From this one can conclude that the best regime is with large κ, low q, high n_e, small a/R (compact devices), large size, and high B .

Improved confinement, ie H-mode plasmas changes the constant in the scaling law to $2.4.10^{-2}$, but as τ_e also increases by a factor 2, a factor 4 overall improvement results.

Using the above scaling law (for H-mode plasmas) as a basis for comparing the various proposed future devices (Ignitor, CIT, ITER etc), one can conclude that a steady burn should be achieved, but that the limit of present technology for magnets and size is a limiting factor. The good news from present devices is that heating to the required T_i is no longer a physics problem, that enhanced confinement is found on all machines - with all heating schemes, and that the pressure limit will not be an obstacle. Real problems remain in maintaining the current for long times, in that current drive will not drive the maximum I, in exhausting the plasma, in refueling, in impurity control, and in controlling disruptions. There is also unknown territory ahead with α particles, transients, and probably other surprises. Concept improvements may be needed, eg using current drive to assist the set-up of the plasma, start-up minimizing P_h, perhaps having $\kappa > 2$ to improve the confinement, obtaining a better confinement regime, new exhaust systems, and perhaps new machine configurations.

Critical Design Issues for NET/ITER

An overview of NET and ITER from the engineering point of view was presented, with the conclusion that in spite of differences in machine size, machine objectives and performance, the technical differences are minor. A great deal of detail of the proposed devices was presented, which can only be appreciated by reference to the papers presented. Overall the design of NET and ITER has reached the stage where testing is needed of the concepts of the main structure, machine assembly, remote handling and maintenance, coil construction, in vessel components etc, indeed this is proceeding in some cases.

A typical example is the divertor assembly, which must handle 60 % of the α power and 60 % of the injected power, and the plasma facing surface must be protected by low Z materials. Tests have already been carried out of carbon brazed onto a cooled molybdenum heat sink and of a variety of heat sink structures. The replacement of a divertor plate is a particularly difficult remote maintenance task, which has to be carried out in the hostile environment of 3.10^{8} rads/hr at 150^{0}C in an inert atmosphere. The proposal is to have a complete circular support rail introduced into the tokamak through the major port system, which encircles the machine. This rail can then carry various devices such as manipulators for armour tile and a telescopic boom/ divertor plate gripper. The latter performs the task of replacing a divertor plate.

Overall it appears that the NET/ITER project engineering is well advanced, and that the R & D programmes to support the designs have been worked out. Some long term R & D has already been carried out, and the results support the design concepts adopted.

Prospects for JET and the Next Step

The requirements for the Next Step can be best determined from the performance of, and the unanswered questions from, today's machines. JET has produced limiter plasmas, and single and double null "X-point" plasmas, and it has exceeded all its design parameters. An importance step was the use of Be, which has enabled plasmas to be produced with D^+ fractions of ~ 80 %, cf only 60 % obtained with a carbon machine. This is particularly important as the fusion power release in a plasma is proportional to $(D^+)^2$. In JET, plasmas with n_e and T_i larger than needed for a reactor have been produced, and electron temperature and energy confinement times comparable to those for a reactor have been achieved. However, the issue of impurity control remains, and this should be tackled before the Next Step is embarked upon. Open questions are:

- What is the volume needed for the energy exhaust of a diverted plasma?
- What is the magnetic field configuration required and what is its impact on the plasma shape?
- What are suitable materials for the wall and divertor plates?

The answers to these questions may determine the size of the next step, and the quest for the answers forms the basis of the extension to JET.

The Next Step must demonstrate; full ignition, high power, semi-continuous operation (~ 1 hr); the viability of a superconducting tokamak; the resistance of sensitive materials to neutron irradiation; and it must test hot blanket modules. Indeed several solutions and concepts may need to be evaluated. As some parameters are probably incompatible, eg current drive and high n_e. It is suggested that a single machine, such as is presently proposed for ITER, cannot accommodate all requirements.

An alternative approach is that the Next Step should consist of three facilities :

M1 - A thermonuclear furnace, capable of generating ~ 2.5 GW for 12 hrs/day for 6 days, with the ability to test blanket modules. The aim is to achieve low cost energy production, ~ 1 ECU/thermal watt.

M2 - An advanced, steady state tokamak. This would be a superconducting device, not operating with T_2. The aim would be continuous operation at high n_e, and the testing of different plasma shapes and advanced divertors.

M3 - A materials test facility, ie a device capable of producing high fluxes of 14 MeV neutrons.

Conclusions

This latest, 16th, Symposium on Fusion Technology has been the largest of the series, with 556 registered attendees. Together with the overall quality of the presentations, invited, oral and poster, this demonstrates the healthy state of fusion research and its continuing forward momentum. On the evidence of this symposium, this is clearly now the major conference concerned with engineering and technical aspects of fusion.

APPENDIX 1.

THE JET TEAM

JET Joint Undertaking, Abingdon, Oxon, OX14 3EA, U.K.

J.M.Adams[1], F.Alladio[4], H.Altmann, R.J.Anderson, G.Appruzzese, W.Bailey, B.Balet, D.V.Bartlett, L.R.Baylor[24], K.Behringer, A.C.Bell, P.Bertoldi, E.Bertolini, V.Bhatnagar, R.J.Bickerton, A.Boileau[3], T.Bonicelli, S.J.Booth, G.Bosia, M.Botman, D.Boyd[31], H.Brelen, H.Brinkschulte, M.Brusati, T.Budd, M.Bures, T.Businaro[4], H.Buttgereit, D.Cacaut, C.Caldwell-Nichols, D.J.Campbell, P.Card, J.Carwardine, G.Celentano, P.Chabert[27], C.D.Challis, A.Cheetham, J.Christiansen, C.Christodoulopoulos, P.Chuilon, R.Claesen, S.Clement[30], J.P.Coad, P.Colestock[6], S.Conroy[13], M.Cooke, S.Cooper, J.G.Cordey, W.Core, S.Corti, A.E.Costley, G.Cottrell, M.Cox[7], P.Cripwell[13], F.Crisanti[4], D.Cross, H.de Blank[16], J.de Haas[16], L.de Kock, E.Deksnis, G.B.Denne, G.Deschamps, G.Devillars, K.J.Dietz, J.Dobbing, S.E.Dorling, P.G.Doyle, D.F.Düchs, H.Duquenoy, A.Edwards, J.Ehrenberg[14], T.Elevant[12], W.Engelhardt, S.K.Erents[7], L.G.Eriksonn[5], M.Evrard[2], H.Falter, D.Flory, M.Forrest[7], C.Froger, K.Fullard, M.Gadeberg[11], A.Galetsas, R.Galvao[8], A.Gibson, R.D.Gill, A.Gondhalekar, C.Gordon, G.Gorini, C.Gormezano, N.A.Gottardi, C.Gowers, B.J.Green, F.S.Griph, M.Gryzinski[26], R.Haange, G.Hammett[6], W.Han[9], C.J.Hancock, P.J.Harbour, N.C.Hawkes[7], P.Haynes[7], T.Hellsten, J.L.Hemmerich, R.Hemsworth, R.F.Herzog, K.Hirsch[14], J.Hoekzema, W.A.Houlberg[24], J.How, M.Huart, A.Hubbard, T.P.Hughes[32], M.Hugon, M.Huguet, J.Jacquinot, O.N.Jarvis, T.C.Jernigan[24], E.Joffrin, E.M.Jones, L.P.D.F.Jones, T.T.C.Jones, J.Källne, A.Kaye, B.E.Keen, M.Keilhacker, G.J.Kelly, A.Khare[15], S.Knowlton, A.Konstantellos, M.Kovanen[21], P.Kupschus, P.Lallia, J.R.Last, L.Lauro-Taroni, M.Laux[33], K.Lawson[7], E.Lazzaro, M.Lennholm, X.Litaudon, P.Lomas, M.Lorentz-Gottardi[2], C.Lowry, G.Magyar, D.Maisonnier, M.Malacarne, V.Marchese, P.Massmann, L.McCarthy[28], G.McCracken[7], P.Mendonca, P.Meriguet, P.Micozzi[4], S.F.Mills, P.Millward, S.L.Milora[24], A.Moissonnier, P.L.Mondino, D.Moreau[17], P.Morgan, H.Morsi[14], G.Murphy, M.F.Nave, M.Newman, L.Nickesson, P.Nielsen, P.Noll, W.Obert, D.O'Brien, J.O'Rourke, M.G.Pacco-Düchs, M.Pain, S.Papastergiou, D.Pasini[20], M.Paume[27], N.Peacock[7], D.Pearson[13], F.Pegoraro, M.Pick, S.Pitcher[7], J.Plancoulaine, J-P.Poffé, F.Porcelli, R.Prentice, T.Raimondi, J.Ramette[17], J.M.Rax[27], C.Raymond, P-H.Rebut, J.Removille, F.Rimini, D.Robinson[7], A.Rolfe, R.T.Ross, L.Rossi, G.Rupprecht[14], R.Rushton, P.Rutter, H.C.Sack, G.Sadler, N.Salmon[13], H.Salzmann[14], A.Santagiustina, D.Schissel[25], P.H.Schild, M.Schmid, G.Schmidt[6], R.L.Shaw, A.Sibley, R.Simonini, J.Sips[16], P.Smeulders, J.Snipes, S.Sommers, L.Sonnerup, K.Sonnenberg, M.Stamp, P.Stangeby[19], D.Start, C.A.Steed, D.Stork, P.E.Stott, T.E.Stringer, D.Stubberfield, T.Sugie[18], D.Summers, H.Summers[20], J.Taboda-Duarte[22], J.Tagle[30], H.Tamnen, A.Tanga, A.Taroni, C.Tebaldi[23], A.Tesini, P.R.Thomas, E.Thompson, K.Thomsen[11], P.Trevalion, M.Tschudin, B.Tubbing, K.Uchino[29], E.Usselmann, H.van der Beken, M.von Hellermann, T.Wade, C.Walker, B.A.Wallander, M.Walravens, K.Walter, D.Ward, M.L.Watkins, J.Wesson, D.H.Wheeler, J.Wilks, U.Willen[12], D.Wilson, T.Winkel, C.Woodward, M.Wykes, I.D.Young, L.Zannelli, M.Zarnstorff[6], D.Zasche[14], J.W.Zwart.

PERMANENT ADDRESS

1. UKAEA, Harwell, Oxon. UK.
2. EUR-EB Association, LPP-ERM/KMS, B-1040 Brussels, Belgium.
3. Institute National des Récherches Scientifique, Quebec, Canada.
4. ENEA-CENTRO Di Frascati, I-00044 Frascati, Roma, Italy.
5. Chalmers University of Technology, Göteborg, Sweden.
6. Princeton Plasma Physics Laboratory, New Jersey, USA.
7. UKAEA Culham Laboratory, Abingdon, Oxon. UK.
8. Plasma Physics Laboratory, Space Research Institute, Sao José dos Campos, Brazil.
9. Institute of Mathematics, University of Oxford, UK.
10. CRPP/EPFL, 21 Avenue des Bains, CH-1007 Lausanne, Switzerland.
11. Risø National Laboratory, DK-4000 Roskilde, Denmark.
12. Swedish Energy Research Commission, S-10072 Stockholm, Sweden.
13. Imperial College of Science and Technology, University of London, UK.
14. Max Planck Institut für Plasmaphysik, D-8046 Garching bei München, FRG.
15. Institute for Plasma Research, Gandhinagar Bhat Gujat, India.
16. FOM Instituut voor Plasmafysica, 3430 Be Nieuwegein, The Netherlands.
17. Commissiariat à L'Energie Atomique, F-92260 Fontenay-aux-Roses, France.
18. JAERI, Tokai Research Establishment, Tokai-Mura, Naka-Gun, Japan.
19. Institute for Aerospace Studies, University of Toronto, Downsview, Ontario, Canada.
20. University of Strathclyde, Glasgow, G4 ONG, U.K.
21. Nuclear Engineering Laboratory, Lapeenranta University, Finland.
22. JNICT, Lisboa, Portugal.
23. Department of Mathematics, Univeristy of Bologna, Italy.
24. Oak Ridge National Laboratory, Oak Ridge, Tenn., USA.
25. G.A. Technologies, San Diego, California, USA.
26. Institute for Nuclear Studies, Swierk, Poland.
27. Commissiariat à l'Energie Atomique, Cadarache, France.
28. School of Physical Sciences, Flinders University of South Australia, South Australia SO42.
29. Kyushi University, Kasagu Fukuoka, Japan.
30. Centro de Investigaciones Energeticas Medioambientales y Techalogicas, Spain.
31. University of Maryland, College Park, Maryland, USA.
32. University of Essex, Colchester, UK.
33. Akademie de Wissenschaften, Berlin, DDR.

CR 88.49.1A