Conf-920913--17

INTERNATIONAL ATOMIC ENERGY AGENCY

FOURTEENTH INTERNATIONAL CONFERENCE ON PLASMA PHYSICS AND CONTROLLED NUCLEAR FUSION RESEARCH

Würzburg, Germany, 30 September – 7 October 1992

IAEA-CN-56/E-2-5

PPPL-CFP--2756

DE93 002102

NOVEL CURRENT DRIVE EXPERIMENTS ON THE CDX-U, HIT, AND DIII-D TOKAMAKS

M. ONO, C.B. FOREST, Y.-S. HWANG, R.J. ARMSTRONG, W. CHOE,
D.S. DARROW, G. GREENE, and T. JONES
Plasma Physics Laboratory, Princeton University, Princeton, New Jersey 08543, USA,

T.R. JARBOE, A. MARTIN, B.A. NELSON, D. ORVIS, C. PAINTER, L. ZHOU,
and J.A. ROGERS
University of Washington, Seattle, Washington, USA,

M.J. SCHAFFER, A.W. HYATT, R.I.PINSKER, G.M. STAEBLER, R.D. STAMBAUGH,
E.J. STRAIT, K.L. GREENE, J.A. LEUER, and J.M. LOHR
General Atomic, San Diego, Calif. 92186-9784, USA.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

Received OSTI
OCT 29 1992

MASTER

This is a preprint of a paper intended for presentation at a scientific meeting. Because of the provisional nature of its content and since changes of substance or detail may have to be made before publication, the preprint is made available on the understanding that it will not be cited in the literature or in any way be reproduced in its present form. The views expressed and the statements made remain the responsibility of the named author(s); the views do not necessarily reflect those of the government of the designating Member State(s) or of the designating organization(s). *In particular, neither the IAEA nor any other organization or body sponsoring this meeting can be held responsible for any material reproduced in this preprint.*

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

NOVEL CURRENT DRIVE EXPERIMENTS ON THE CDX-U, HIT, AND DIII-D TOKAMAKS

ABSTRACT - Two types of novel, non-inductive current drive concepts for starting-up and maintaining tokamak discharges have been developed on the CDX-U, HIT, and DIII-D Tokamaks. On CDX-U, a new, non-inductive current drive technique utilizing fully internally generated pressure driven currents has been demonstrated. The measured current density profile shows a non-hollow profile which agrees with a modeling calculation including helicity conserving non-classical current transport providing the "seed current". Another current drive concept, dc-helicity injection, has been investigated on, CDX-U, HIT and DIII-D. This method utilizes injection of magnetic helicity via low energy electron currents, maintaining the plasma current through helicity conserving relaxation. In these experiments, non-ohmic tokamak plasmas were formed and maintanined in the tens of kA range.

I. Introduction

To find acceptable methods for driving steady-state current is one of the important topics for improving tokomak reactor. Two novel methods for non-inductive tokamak start-up and current drive are presented here. One method utilizes internally generated pressure driven currents and the other employs an external injection of magnetic helicity. The current drive efficiencies for the methods investigated here are, in principle, independent of plasma density and, therefore, may be used to start-up and/or drive currents in high density high β_{pol} reactor plasmas. Experimental results presented here were conducted without ohmic heating in the Current Drive Experiment-Upgrade Tokamak (CDX-U), Helicity Injected Tokamak (HIT) and DIII-D Tokamak. The CDX-U experiment (R_0 = 32 cm, a = 23cm, elongation =1.5, and steady-state toroidal field current NI = 300 kA), and the HIT experiment (R_0 = 36cm, a = 25cm, elongation = 2, and toroidal field current NI = 500 kA) are low-aspect-ratio tokamak research facilities, and DIIID (R_0 = 170cm, a = 67 cm, elongation = 2, and toroidal field ≈ 2.1T) is a conventional tokamak with divertor cold electrode.

II. ATTAINMENT OF 100% PRESSURE DRIVEN TOKAMAK IN CDX-U

A new, non-inductive current drive concept, utilizing internally generated pressure driven currents, has been demonstrated on CDX-U[1]. Electron cyclotron heating (ECH) ($P_{rf} \leq 8$ kW, f = 2.45 GHz) is used to provide the heating power necessary to create and maintain a low collisionality ($\nu_e^* < 1$), high-β_{pol} ($\varepsilon\beta_{pol} \approx 1$ where $\varepsilon = a/R$) plasma, the regime in which pressure driven currents are important. Taking advantage of the low-aspect-ratio geometry, a toroidal mirror configuration [Fig. 1(a), left] has been utilized to provide initial confinement for a hot, ECH produced, trapped electron (banana) population. With application of ECH through a simple non-phased waveguide, the plasma is initiated and the plasma current is generated from zero current. The generated current scales inversely with the background neutral particle density, showing the importance of reducing electron-neutral collisionality. The generated current direction depends only on the applied poloidal field direction, not on the toroidal field direction, as expected from the theoretical model below. The currents flowing into segmented limiters are very small (<< 1A), confirming that the currents are internally generated. With application of ≈ 8 kW of ECH power, a toroidal plasma current of up to Ip ≈ 1.4 kA has been generated with the CDX-U aluminum chamber acting as a passive stabilizer. This level was increased to $Ip \approx 2.4$ kA by an addition of simple vertical field programming. For $Ip \geq 600$ A, the poloidal fields from the plasma currents are sufficiently large to form a low-aspect-ratio tokamak plasma. A two-dimensional reconstruction of the magnetic flux contours is shown in Fig. 1(a), right. The mid-plane current profile is shown as a solid curve in Fig. 1(b). The β_{pol} in this experiment is high with $\varepsilon\beta_{pol} \approx 1$.

To understand the nature of the generated currents, two dimensional modeling has been performed.[2] Two pressure-driven toroidal currents are important here: the banana precessional drift current and the neoclassical bootstrap current. In the open field line (toroidal mirror) configuration where all particles are trapped ($\varepsilon \approx 1$), the generated net toroidal current is mainly due to the banana precessional current. The precessional current can be estimated as $J_{prec} \approx \varepsilon^{1/2} [P/(RB_{pol})]$ where P is the electron pressure. The diamagnetic current, which itself generates no net current, plays an important role of enhancing the precessional current by reducing B_{pol} in the high pressure central region. When the flux surfaces close ($Ip \geq 600$ A), the bootstrap current, $J_{boot} \approx 2\,\varepsilon^{1/2} [(\partial P/\partial r)/B_{pol}]$, becomes dominant. From the measured plasma pressure profiles obtained from the two-dimensional microwave interferometer system for $n_e(x,y)$ and Langmuir probe measurement for $T_e(r)$ ($n_{e0} \approx 2 \times 10^{17}$ m^{-3}, $T_{e0} \approx 30$ - 40 eV, and $\nu^* \leq 1$), the current profile has been calculated using the neoclassical model. Although the calculated total current level is within a factor of 2 of the experiment, the calculated current profile is hollow (since the neoclassical currents vanish at the plasma center) while the measured profile is non-hollow as shown in Fig. 1(b). Furthermore, the neoclassical hollow current profile cannot exist self-consistently without the current on axis (the "seed" current). So resolve this question, we added the helicity conserving non-classical current transport term [3] in the modeling calculation which current profile has indeed led to a reasonable agreement with the experimental measurement as shown in Fig. 1(b). We note that non-classical current transport has been observed previously in the dc-helicity injection current drive experiment [4] and the possibility of a fully internally driven tokamak, which includes the effect of tearing modes and bootstrap current, has been pointed out by Boozer [3]. As an extension of the present result, one could envision starting-up a tokamak plasma by an auxiliary heating method and then maintaining the burning plasma with heat from fusion alphas.

III DC-HELICITY INJECTION CURRENT DRIVE IN CDX-U, HIT AND DIII-D

DC-helicity injection current drive (DCHI) is achieved by building up and sustaining the magnetic helicity, allowing plasma relaxation to maintain the equilibrium. This method is theoretically more efficient than RF or NB based systems and allows the use of relatively low voltage, low cost dc power supplies. Preliminary experimental results from CDX-U, HIT and DIII-D are encouraging, demonstrating non-inductive tokamak startup currents in the tens of kA range.

CDX-U - In the CDX-U experiment, helicity is injected from a toroidally localized emissive cathode and the driven current has reached 10 kA level, maintaining a tokamak plasma with $q(a) \approx$ 6-10. The plasma equilibrium was mostly provided by the 3 cm thick aluminum vacuum vessel wall acting as a passive stabilizer with an addition of some vertical field programming for the higher current range. The magnetic reconstruction of flux contours has shown a formation of a tokamak plasma in a single-null divertor configuration. The current multiplication factor, a ratio of driven to injected current, is quite high ≈ 30, with an applied cathode voltage of ≈ 400 V.

HIT - The HIT experiment is a low aspect ratio tokamak designed for coaxial helicity injection current drive with vertical field provided by a passive conducting boundary as in CTX [5] without an ohmic heating transformer. After conducting successful helicity source characterization experiments, initial measurements of current drive were made in the following non-optimized configuration. The conducting boundaries are a 2 mm thick copper sheet placed in side the 1.2 m diameter stainless steel vacuum wall, 30 mm thick aluminum end plates, and a 220 mm diameter 12 mm thick aluminum pipe as the inner wall. Non-inductive tokamak currents of over 30 kA have been driven with ≈ 650 V source voltage. The toroidal field at R_0=0.36 m is 0.3 - 0.4 T. The helicity injection rate is 2 $V_g\psi_g$ where V_g is the voltage between the electrodes of the injector and ψ_g is the flux that penetrates the electrodes. For ohmic heating transformer the helicity injection rate is 2 $\Phi_t V_l$ where V_l is the toroidal loop voltage and Φ_t is the toroidal flux. Thus, assuming helicity conservation, an effective loop voltage from electrostatic helicity injection can be defined as $V_{eff} = V_g\psi_g / \Phi_t$ [6]. For the results discussed here ψ_g = 4 mWb and Φ_t = 170 mWb. The plasma density is estimated from the Alfven transit time and the electron temperature is estimated from the spectroscopic lines of C, N, and O.

Two conditions in the HIT experiment support the helicity model [6]. The plasma current evolution traces for the two cases are shown in Fig. 2(a). In the first case, internal probes are used without discharge cleaning or Ti gettering. The observed plasma parameters are as follows: $I_p \approx$ 10 kA, $T_e \approx$ 6 eV, $V_g \approx$ 540 V. The V_{eff} is 13V with the observed plasma current of 10 kA. The density is 8 x 10^{19} m^{-3} at the beginning of discharge which drops to 1-3 x 10^{19} m^{-3} during the discharge. The total electrode current is about 5 kA. The measured poloidal flux profile at the mid-plane is shown in Fig. 2(b). In the second case, with discharge cleaning and Ti gettering and no internal probe, the estimated T_e ≈ 12 eV and the density is around 0.5-1.5 x10^{19} m^{-3} with no high beginning. The V_{eff} is 15V and the current increases as expected, ≈ V_{eff} $T^{3/2}$, to 30 kA. The injected current was 7kA giving an overall power efficiency (defined as $I_p\psi_g / \Phi_t I_g$) of 10%. Preliminary data with Ti gettering indicate closed poloidal flux, demonstrating tokamak plasma formation.

COMBINED ELECTRODE AND ECH STARTUP IN DIII-D - A tokamak plasma was formed and sustained in DIII-D by a combination of DCHI using the cold axisymmetric divertor ring electrode [7] and ECH heating assist with the 60 GHz, 0.7 MW, extraordinary mode, high field launch ECH system. This experiment represents a considerable scale up test of DCHI from the CDX-U and HIT experiments. A steady-state equilibrium field, B_{eq}, as shown in Fig. 3a, was designed to connect the ECH resonant absorption point (R = 1.59m, midplane, B_T = 2.14 T) to the divertor electrode, to magnetize the electrode, and to have favorable curvature for plasma stability. The vessel was filled to 0.04 - 0.08 mbar of D_2 just prior to discharge initiation. The electrode was driven to + 700 V with respect to the wall.

ECH alone broke down the gas but drove only ≈ 1 kA of toroidal current. The electrode alone could not break down the gas. Simultaneous application of electrode and ECH formed plasmas carrying 10 to 20 kA of toroidal current. An example is shown in Fig. 3(b). After a 20-30 msec transient, a steady-state discharge was sustained for the duration of the ECH pulse (300 ms). The current multiplication factor was ≈ 100 which is also approximately the number of times a magnetic line circled the torus between the electrode and the top of the vessel.

Reconstruction of the magnetic configuration, by a modified version of the MFIT code [8], in Fig. 3(c) shows a closed separatrix boundary with R/a ≈ 1.75m / 0.4m. The closed configuration was established in less than 10 ms, and it continued to increase in volume and internal poloidal flux until reaching steady-state. The necessary condition for the closed flux formation appears to be a toroidal current in the plasma shell large enough to null B_{eq} in the inner radius region. The plasma was at the toroidal equilibrium limit, $\varepsilon\beta_{pol} \approx 1$, which may result in generation of significant bootstrap current. The monitoring of electric current flowing out of an instrumented protective tile covering 0.14m radially inward from the electrode [9], suggests a possible helicity transport inward from the flux surfaces carrying directly driven electrode current.

* Work supported by DoE contract Nos. DE-AC02-76-CHO-3073, DE-AC03-89ER51114, DE-FG06-90ER54095.

REFERENCES

[1] FOREST,C.B., HWANG, Y.S., ONO, M., Phys. Rev. Lett. 68 (1992) 3559.
[2] HWANG, Y.S., PhD Thesis, Princeton University, (1992).
[3] BOOZER, A.H., Phys. Fluids 29 (1986) 4123.
[4] ONO, M., et al., Phys. Rev. Lett. 59 (1987) 2165.
[5] BARNES, C.W., et al., Phys. Fluids 29 (Oct. 1986).
[6] JARBOE, T., Fusion Tech. 15 (1989) 7.
[7] ANDERSON, P.M., et al., Fusion Technology 1990 (Proc. 16th Symp. on Fusion Technol., London, UK, 1990) North Holland (1991), Vol. I, p. 336.
[8] LAO, L.L., ST. JOHN, H., STAMBAUGH, R.D., PFEIFER, WM., Nucl. Fusion 25 (1985) 1421.
[9] SCHAFFER, M.J., LEIKIND, B.J., Nucl. Fusion 31 (1991) 1750.

FIGURE CAPTIONS

Fig. 1 100% pressure driven tokamak formation in CDX-U. (a) Vacuum flux contours for low-aspect-ratio toroidal mirror configuration before application of ECH.(left) Reconstructed flux contours for the 1 kA generated current case showing formation of a low-aspect-ratio tokamak topology.(right) (b) Comparison of experimental current density profile with model calculations.

Fig. 2. Helicity Injection Experiment in HIT. (a) Ip for two conditions. Upper curve is with Ti gettering. (b) Poloidal flux versus radius at the mid-plane.

Fig. 3. Combined Electrode and ECH Startup in DIII-D. (a) Predischarge conditions including equilibrium flux surfaces, divertor electrode, approximate ECH antenna pattern and resonant surface, and B_T direction. (b) Time traces of various parameters. (c) Magnetic reconstruction of plasma at t = 150 ms.

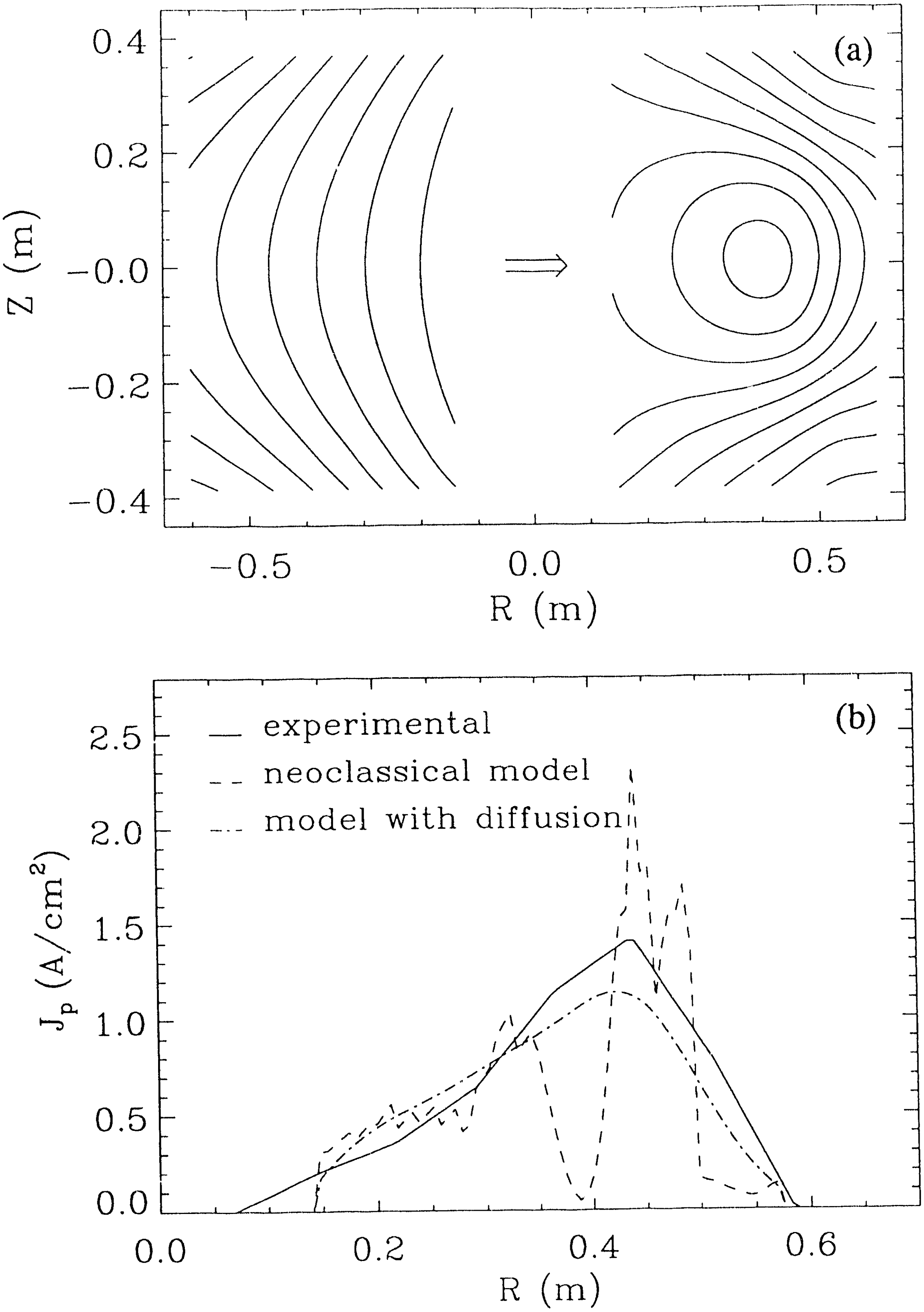
(a)
Z (m)
0.4
0.2
−0.0
−0.2
−0.4
−0.5
0.0
0.5
R (m)
(b)
experimental
neoclassical model
model with diffusion
J_p (A/cm^2)
2.5
2.0
1.5
1.0
0.5
0.0
0.0
0.2
0.4
0.6
R (m)

Fig. 1

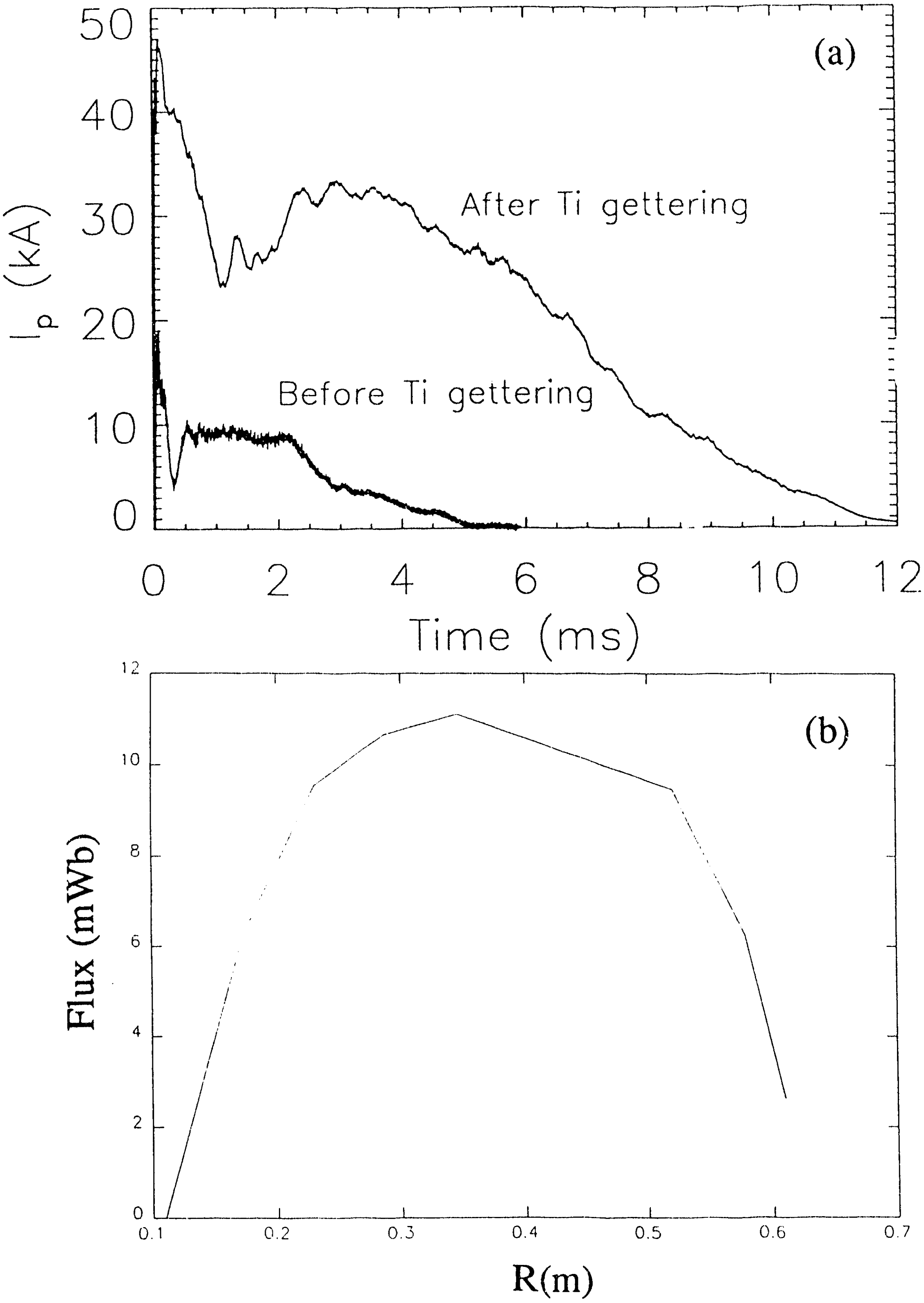
(a)
50
40
30
20
10
0
I_p (kA)
After Ti gettering
Before Ti gettering
0
2
4
6
8
10
12
Time (ms)
(b)
12
10
8
6
4
2
0
Flux (mWb)
0.1
0.2
0.3
0.4
0.5
0.6
0.7
R(m)

Fig. 2

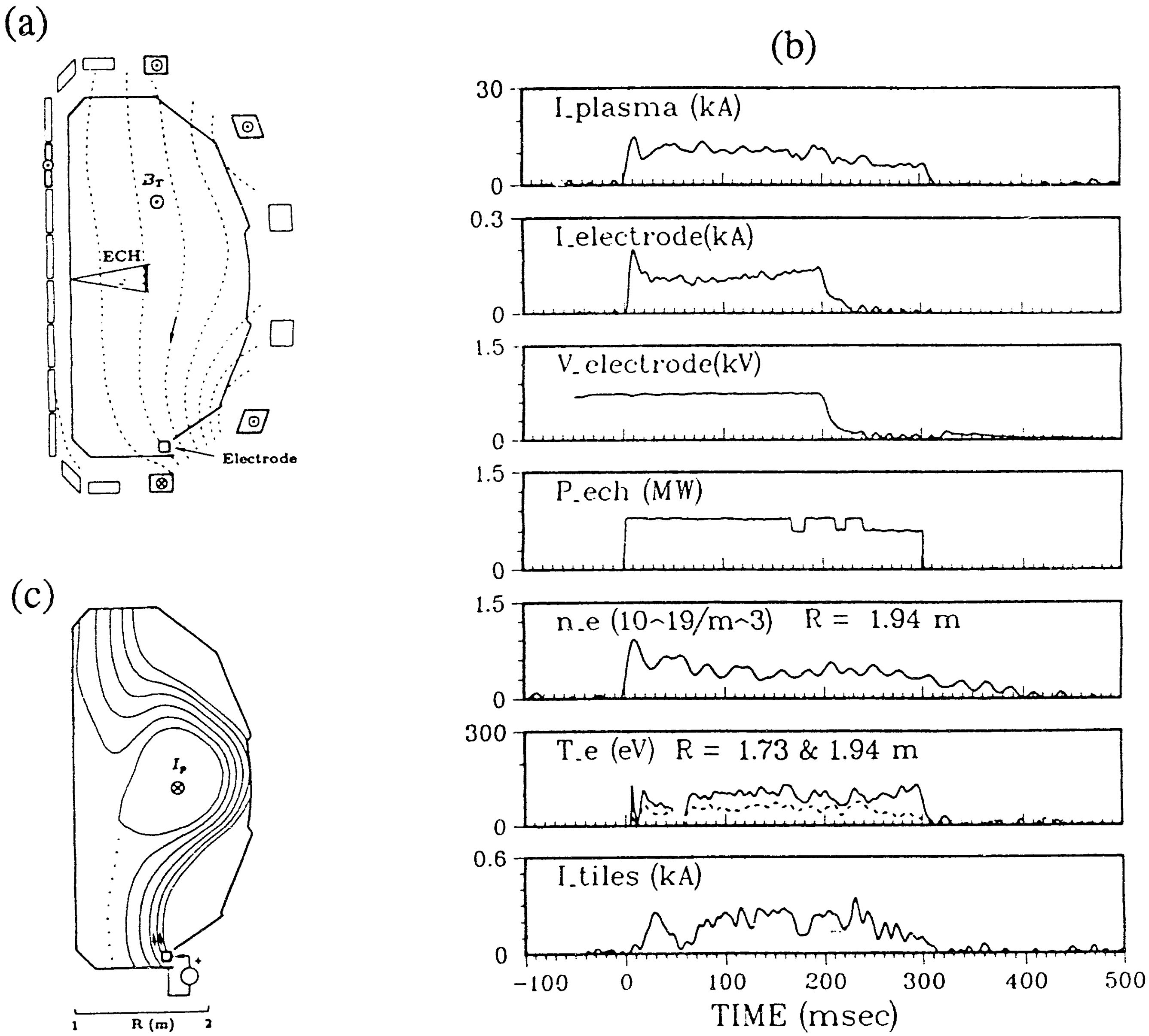
(a)
B_T
ECH
Electrode
(b)
I_plasma (kA)
I_electrode(kA)
V_electrode(kV)
P_ech (MW)
n_e (10~19/m~3) R = 1.94 m
T_e (eV) R = 1.73 & 1.94 m
I_tiles (kA)
30
0
0.3
0
1.5
0
1.5
0
1.5
0
300
0
0.6
0
-100
0
100
200
300
400
500
TIME (msec)
(c)
I_p
1
R (m)
2

Fig. 3

END

DATE
FILMED

12/28/92