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Ion Cyclotron Resonance Heating of D-T Divertor Plasmas in JET

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1. INTRODUCTION.

Results of ion cyclotron resonance heating (ICRH) of deuterium-tritium (D-T) divertor plasmas of JET in ITER-relevant scenarios and in ITER-like configuration are presented. H-mode experiments have been performed in D-T plasmas with up to 95% of tritium in scenarios such as the second harmonic heating of tritium ($2\omega_{CT}$), (D)-minority heating in T (ω_{CD}) and He^3-minority in DT mixtures. Relevant to the initial non-active campaign of ITER, 'inverted' minority schemes (Z/M of the minority species being higher than that of the majority) such as He^3 and D-minority in H plasmas including T-minority in D plasmas have also been studied in L-mode. In the latter schemes requiring operation at lower frequencies, the coupled power capability is lower (see below) and the H-mode threshold power is higher.

Operation in such a variety of scenarios has been possible in JET due to the existence of a very wide frequency range (23-56 MHz) of the ICRH plant. For an operation of the JET tokamak with a toroidal field on-axis of 3.4-3.7 T, the above scenarios, for example, required operation at 23, 28, 34, 42, 51 and 56 MHz. The JET ICRH system couples power via four antennas distributed around the torus. Antennas are equipped with Faraday shields made out of beryllium. Each antenna has four straps which can be phased independently. Up to 9 MW of ICRH power has been coupled in single-null DT divertor H-mode plasmas.

2. EXPERIMENTAL RESULTS.

2.1 Antenna-Plasma Coupling Resistance (R_c).

The R_c as measured experimentally in JET can be described as the value of antenna resistance seen at a current antinode of the feeder line. By transmission line relations, this can be expressed as $R_c = 2\, Z_{0L}^{\,2}\, P_c / V_{max}^{\,2}$ where Z_{0L}=30 Ω and V_{max}= 25-32 kV depending on the conditioning. Experimentally measured values of R_c (averaged over the four straps of an antenna and subtracting line losses) are plotted as a function of frequency in Fig. 1 for L-mode single-null divertor plasmas for a range of antenna strap to separatrix distances indicated in the figure. The variation of R_c as a function of frequency is a reflection of the antenna resonance characteristics and the fact that the radiation resistance for fast waves increases with frequency. For H-mode density profiles, R_c is lower by factor of ~0.6-0.8.

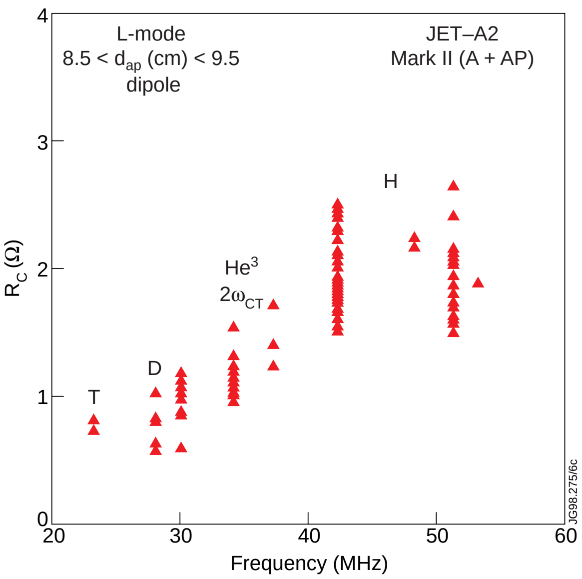

FIG. 1. Coupling resistance vs frequency. Symbols represent minority ion used.

2.2. $2\omega_{CT}$-heating scheme and He^3-minority heating in DT-plasmas.

In JET, experiments at $2\omega_{CT}$ were conducted at a power level of 8MW [1]. The second harmonic heating scheme utilizes the finite Larmor radius effect and the heated population absorbs the power predominantly, producing a large tail. This was decreased by operating at high plasma density ($5x10^{19}m^{-3}$) but it was not sufficient to achieve dominant ion heating. However, this should not be a limitation in ITER for its high density operation. Single-pass absorption in this scheme is weaker in JET. Puffing a small amount (1-2%) of He^3 (in addition to the He^3 present due to radio-active decay) improves significantly the wave absorption and the energy confinement [2]. In such a case, minority (He^3) ion absorption dominates and a significant He^3 tail is produced. By adding He^3 to a level of 5-10 %, He^3-tail can be sufficiently lowered to produce strong ion heating. Time traces of such a discharge are shown in Fig. 2 where T_{i0}≈13 keV is achieved by ICRH alone. Other parameters are B_φ=3.7 T, I_P=3 MA, f=37 MHz and $n_T/(n_D + n_T)$=0.5. He^3-minority heats ions efficiently as He^3-tail is smaller and its critical energy (P_e=P_i) is 3 times higher than that of hydrogen.

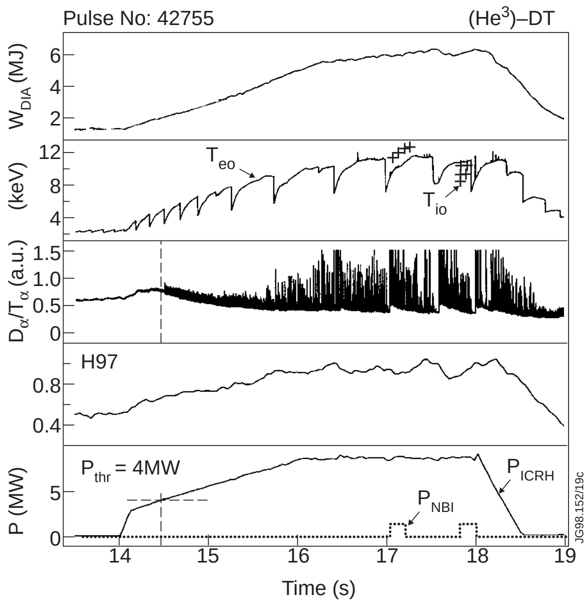

FIG. 2. He^3-minority (~10%) in a 50:50 DT plasma with strong ion heating (T_{i0}=13 keV)

2.3. D-minority heating in T-plasmas.

Time traces of an illustrative discharge in the (D)-T scheme are shown in Fig. 3 where the ICRH power level is about 6 MW. Other parameters are B_φ=3.7 T, I_P=3 MA, f=28 MHz and $n_T/(n_D+n_T)$=0.91. D-minority ion tail temperature could be sufficiently lowered to about 100 keV (close to the peak of DT fusion cross section), by operating at higher plasma density and with D-minority concentration of about 10%. In this discharge, the neutrons thus produced are largely of non-thermal origin. A fusion power of about 1.6 MW was obtained leading to a steady state Q≈0.22 whereas the bulk T_{i0}≈6 keV was low. The neutron yield could be

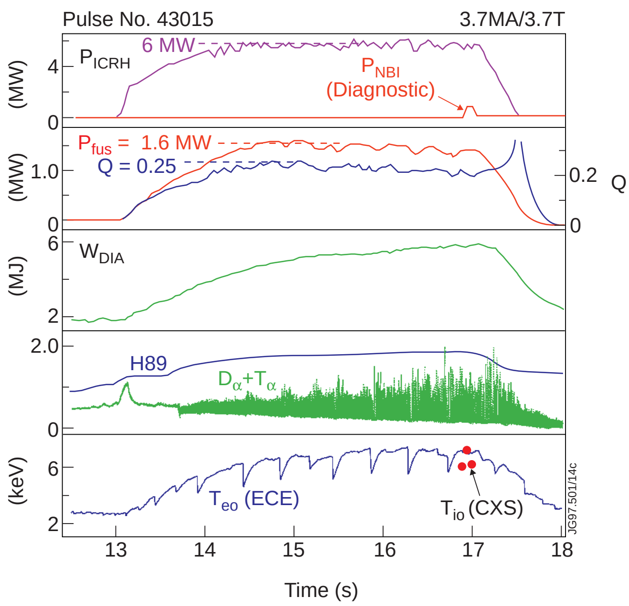

FIG. 3. D-minority heating in T-plasma. Non thermal DT neutrons with Q≈0.25 for ~ 2s.

well reproduced by the PION code[3] and confirms the non-thermal origin of these neutrons. A comparison [1] of observed neutron yield as a function of the expected thermal neutron yield shows that in the (D)-T scheme, non-thermal yield can be as large as 10 times the thermal value whereas in the (He^3)-DT and $2\omega_{CT}$ schemes, it is largely thermal. A comparison of these (D)-T results and the 1991 JET Preliminary Tritium Experiments (PTE) NBI results [4] shows (see Table 1) that for the same n_D/n_T or n_T/n_D (~0.1), fusion power amplification factor (Q) values obtained with ICRH are steady state whereas with NBI they are transient.

Table 1. A comparison of ICRH (D)T and NBI (PTE) Results.

Parameter	**ICRH (1997) (D)-T**	**PTE (NBI) (1991) Hot-ion H-mode**
n_D:n_T	9:91	88:12
B_T (T)	3.7	2.8
n_{e0} ($10^{19}m^{-3}$)	5.4	3.6
P_{FUS} (MW)	1.7	1.8
P_{in} (MW)	6.9	15.3
E_{FUS} (MJ)	5.0	2.2
Q	0.22	0.19
	(dW/dt=0)	(dW/dt=6MW)

2.4 Inverted Minority Schemes:

2.3.1 Dispersion Diagrams.

In Fig. 4, we illustrate and compare the dispersion diagrams for the fast-wave (FW) and ion-Bernstein wave for 'normal' (H)-D and 'inverted' (D)-H minority heating schemes. Minority cyclotron layer position is kept fixed by choosing appropriate frequencies (51 and 25.5 MHz respectively). For the inverted scheme, the minority cyclotron layer lies on the high-field side of the central FW cut-off. Therefore, for ion heating, it is important to avoid this cut-off by using low minority concentration. When minority concentration is increased, the mode conversion-cut-off layers move away to the low-field side as opposed to the normal scheme where they move to the high-field side. The difference in the evanescent region at the edge is largely due to the different frequencies used. In the inverted scheme, at low density at the edge, an additional Shear Alfven wave (SAW) cut-off-mode conversion pair appears which is illustrated in Fig. 5 by choosing an artificially slow variation in the density profile. In the usual tokamak profiles, the mode conversion occurs in a very narrow region and little (~1%) [5] power is mode converted. Therefore, its occurrence is of little concern.

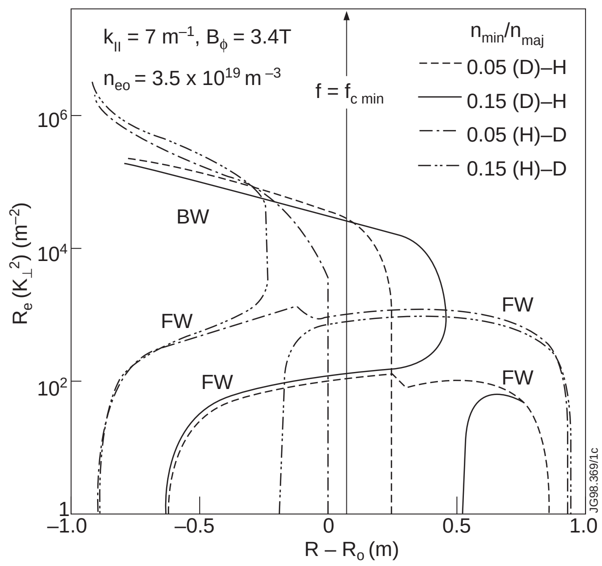

FIG. 4. Dispersion diagram for the inverted and normal minority scenarios in a tokamak.

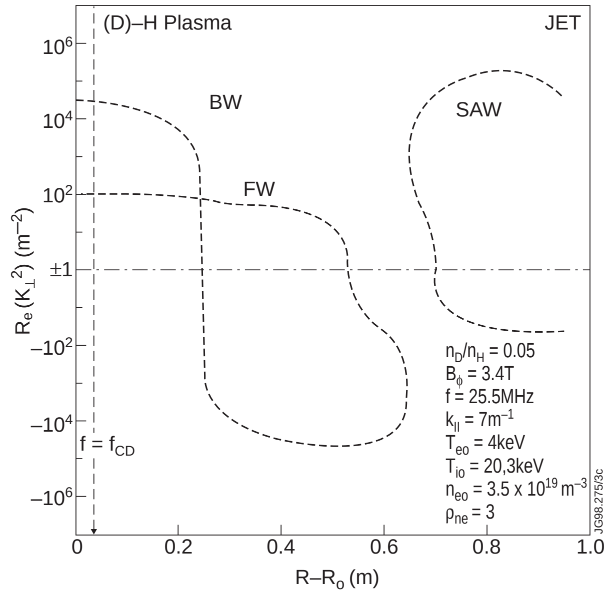

FIG. 5. Low density dispersion diagram in the inverted scheme. ρ_{ne} : parabolic exponent.

2.4.2 An Experimental Comparison of Inverted and Normal Schemes.

In Fig. 6, we compare sawteeth in electron temerature (T_e) and confinement factor (ITERH-89P) with respect to L-mode for normal, (H)D and (He^3)D, and inverted schemes (He^3)H and (D)-H at $P_{ICRH} \approx 3$ MW. Both the larger sawteeth and higher H-89P factor for normal schemes indicate higher fast-ion energy in these discharges. In the inverted schemes, power in the lower $k_{//}$-spectrum is mode converted (electron heating) before reaching the cyclotron layer. Operation at higher power may boot-strap the production of fast-ions. Experiments of T-minority heating in a D-plasma show that significant, DT neurons are produced as the tritium tail develops [6]. In

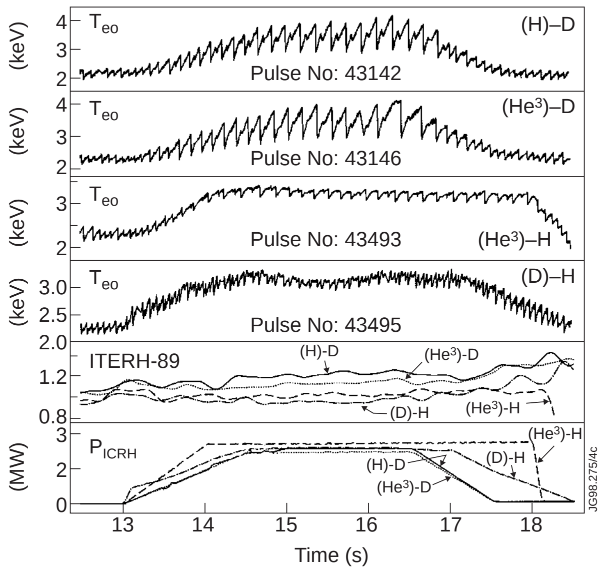

FIG.6 T_e and confinement factor for normal and inverted minority heating scenarios.

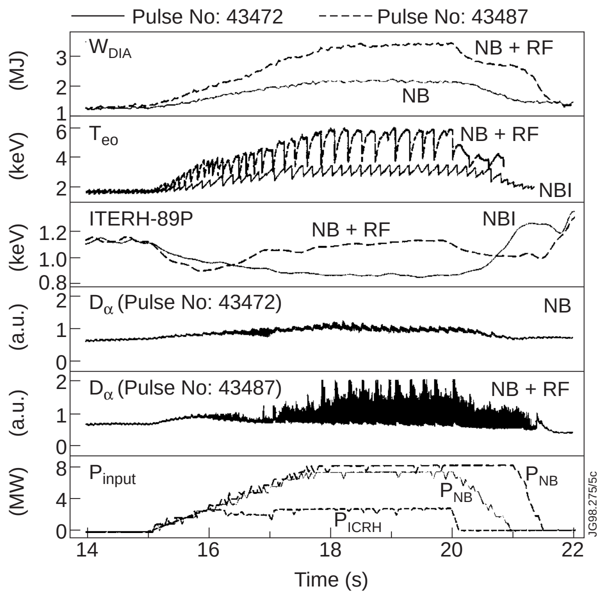

FIG. 7. Compare NBI only and NBI+ICRH (inverted (He3)-H discharges.

Fig.7, we show the effect of additional ICRH power in the inverted (He^3)-H scheme, to make the transition to an H-mode (compare the ELMs in D_α-signal) in the combined H°-neutral beam (NB) + ICRH discharge. A selected similar NB heated discharge without ICRH remained in L-mode.

3. CONCLUSIONS.

By optimising the D-tail to about 100keV in the (D)-T scheme, significant D-T fusion power was obtained permitting the achievement of Q≈0.22 for a record duration of more than 2s limited primarily by neutron economy. Strong ion heating (T_{i0}=13 keV) has been demonstrated with He^3-minority in DT plasmas as He^3 features shorter tails and E_{crit} ($P_e=P_i$) 3 times larger than H. Inverted minority schemes, (He^3)-H and (D)-H, show good absorption of power. But, at low but comparable powers, fast-ion tails were not as well developed as in (H)-D and (He^3)-D. For good minority ion heating, it is important to avoid the central cut-off by using low minority concentration. Mode conversion to the shear Alfven wave at the low density edge is expected to be small. Finally, the experimental results obtained at JET constitute a firm experimental basis for the application of ICRH on ITER.

ACKNOWLEDGEMENTS.

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REFERENCES

[1] START, D.F.H. et al, submitted to Phys. Rev. Lett. (1998)

[2] JACQUINOT, J. and the JET Team, 24th EPS Conf. on Contr. Fus. Plasma Phys., Berchtesgaden 1997, Europhysics Conf. abs, vol. 21A, part IV, p. 1865.

[3] ERIKSSON, L.G. et al., submitted to Nucl. Fusion.

[4] JET TEAM, Nucl. Fusion 32 (1992) 187.

[5] START, D.F.H. et al, Proc. RF Heating and Current Drive of Fusion Device, Brussels, 1998, Invited paper to be published in Plasma Physics and Controlled Fusion.

[6] Van Eester, D., Private Communication.