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EFFECT OF BERYLLIUM ON PLASMA PERFORMANCE IN JET

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ABSTRACT

JET is investigating beryllium as material for walls and limiters. Studies were carried out with initially a thin layer of beryllium evaporated onto the walls of the machine, later in addition the graphite material of the belt limiter was exchanged against beryllium. The use of this material was generally beneficial for the plasma behaviour. Combined with a reduction in the oxygen content, strong pumping of hydrogen isotopes was found which allowed JET to widen considerably the operational space with respect to ion temperatures, densities and plasma purity. In this paper a comparison of the plasma performance with graphite and beryllium will be presented. We will discuss especially the impurity behaviour with respect to fluxes, concentrations, effective charge and dilution, we will report on density limits, disruption behaviour, wall pumping, hydrogen isotope retention, and the power handling capability of the beryllium limiter in the present design. Examples of improved plasma performance will be given.

KEYWORDS

Beryllium; limiter; wall; impurities; wall pumping; hydrogen retention; density limit; disruptions.

INTRODUCTION

The use of beryllium in JET was proposed already during the design phase (Rebut, 1975). From the start of machine operation in 1983 studies were initiated by JET to assess beryllium as an alternative to graphite, the limiter material selected initially. These studies comprised of the investigation of beryllium limiters for the tokamaks ISX-B (Mioduszewski *et. al.*, 1986) and UNITOR (Hackmann and Uhlenbusch, 1984; Bessenrodt-Weberpals et. al., 1989), as well as the evaluation of single processes, i.e. measurements of sputtering yields (Bodhansky *et. al.*, 1985) and hydrogen retention (Möller *et. al.*, 1986; Causey *et. al.*, 1990).

In tokamak experiments it was found that beryllium is a well suited material for limiters which facilitates the conditioning of the respective machine, reduces the oxygen content of the plasma by gettering and allows for increased density limits. Thermal overloading of the beryllium limiters leads to surface melting which results in increased beryllium content of the plasma even then when the power to the limiters is reduced to values below the threshold for melting. This is due to hot spots at protrusions on the surface which were generated during the melting phase.

The investigation of single processes showed that beryllium has a sputtering yield comparable to that of graphite at ambient temperature, but which in contrast to graphite, is only moderately temperature dependent until evaporation becomes the main release process. At 600 K its long term hydrogen retention is by about a factor of three lower than the one for graphite. The mechanism for hydrogen pumping is comparable to that of a metal which dissolves hydrogen and does not form hydrides.

These results were analysed (Rebut *et. al.*, 1985; Hugon *et. al.*, 1989) and it was expected that due to the lower nuclear charge of beryllium compared to graphite the plasma should contain more deuterons for a given impurity concentration. Unless hot spots become dominant, the beryllium concentration should be less than that for carbon because in contrast to graphite, beryllium does not show chemical erosion (hydride formation), radiation induced sputtering or for perpendicular incidence self-sputtering yields above unity. Beryllium is a getter for oxygen and therefore the oxygen content of the plasma and the sputtering by oxygen will be decreased. This will lead to reduced impurity contents and, based on the model which relates the maximum obtainable density to the radiation outside the $q = 2$ surface, to increased density limits.

Areas of concern remain the angular dependence of the self-sputtering rate for beryllium and the risk of thermally overloading the limiter surface. For the same surface temperature beryllium can take about 1.7 times the load of graphite but overloading, which never can be avoided, will inevitably result in surface melting whereas graphite only experiences increased erosion. On the other hand the maximum permitted surface temperatures for graphite and beryllium are similar. It was shown by theoretical analysis (Hugon *et. al.*, 1989) that 1300 K should not be exceeded for beryllium (evaporation limit) and for graphite the surface temperature should remain below 1500 K to avoid the carbon bloom by enhanced self-sputtering. That was confirmed for graphite surfaces during discharges in JET (Summers *et. al.*, 1990). The apparent sputtering yield of graphite rises from 5% for temperatures of 1300 K to unity for 1700 K. This can be understood in terms of self-sputtering and/or radiation enhanced sublimation. This result excludes graphite as a material for high heat flux components in case surface temperatures would exceed 1700 K.

EVALUATION OF BERYLLIUM AS WALL AND LIMITER MATERIAL

Preparation

In parallel to the experimental investigations into the properties of beryllium with respect to its behaviour in the presence of tokamak plasmas, designs were carried through to introduce into JET a beryllium limiter and at the same time to cover the internal surfaces by a thin evaporated beryllium layer.

The switch of the limiter material from graphite to beryllium was planned to be implemented in connection with the belt limiter which was designed to allow for an easy exchange of materials (Celentano *et. al.*, 1986). The limiter material is inserted in form of tiles between cooling fins which are welded to the water-cooled support structure. The tiles are 380 mm long and 75 mm deep, they have a width of 20 mm (beryllium) and 26 mm (graphite). They face the plasma with their narrow side. Any two plates are assembled into one unit. Each unit is held by disk springs which are located in slots in the fins. The tile pairs are aligned to each other to an accuracy better than 0.3 mm before more serious misalignment may have occurred

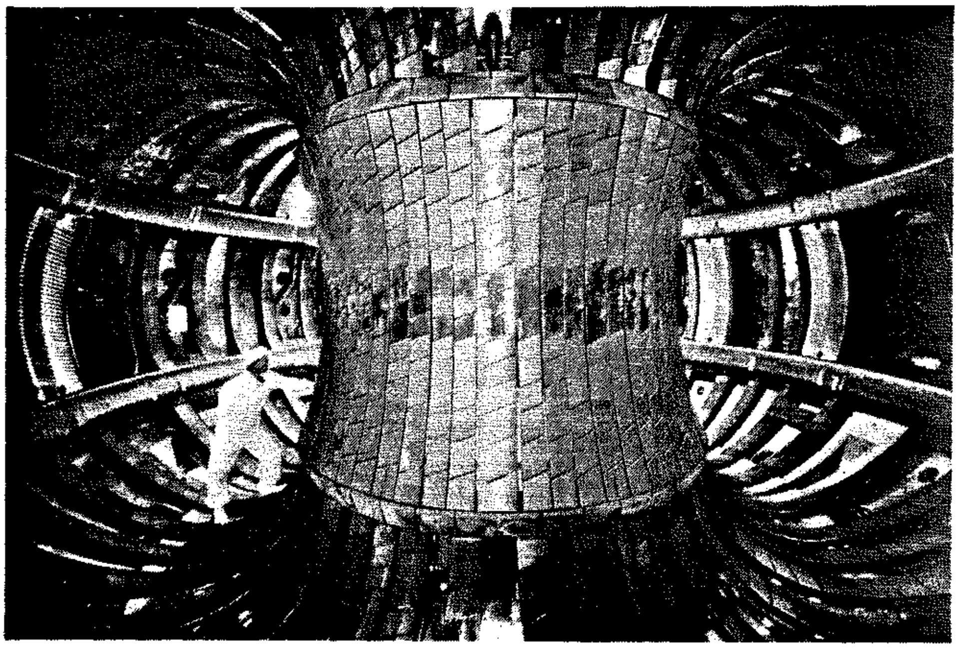

Fig. 1. The inside of the vacuum vessel

due to movements during pumping down of the vacuum vessel, during baking to the operation temperature of 300 C or during plasma operation. The plasma facing surface of the beryllium tiles is slotted every 20 mm to minimize the effect of thermal stresses.

The beryllium evaporators (Sonnenberg *et. al.*, 1986) consist of a hollow beryllium cylinder (mass 3 kg) which is closed at one end and supported by a carbon fibre tube. Inside there is a spirally wound carbon-fibre heater which can be electrically heated to 2400 C. Four such evaporators can be inserted into the vessel by 300 mm for evaporation and retracted again for pulse operation.

Operation with beryllium

Dilution of the JET plasma by low-Z impurities and the absence of proper density control were the main limitations to the performance during the use of graphite as limiter and wall material. The massive graphite elements in the vessel as shown in Fig. 1 were the tiles for the belt limiter, the inner wall protection, the X-point and the RF-antennae side protection. The rest of the vessel was carbonized including the screens for the RF-antennae.

Three phases were foreseen for the evaluation of beryllium. The carbon phase saw the operation of JET as a graphite machine in the configuration described above to establish reference discharges. The carbon/beryllium phase was characterized by beryllium evaporation. Twenty-six evaporations were made and 240 g of beryllium were deposited. Discharges on the inner wall, the X-point protection tiles and the belt limiter were carried out during this phase. The area in contact with the plasma was graphite covered with only a thin beryllium layer of about 100-300 Å thickness. The beryllium phase started after the exchange of the belt limiter material against beryllium. In addition evaporation was still carried out. Only the discharges run on the belt limiter can be considered to be made in a beryllium environment, the X-point discharges during the beryllium phase were still run on beryllium covered graphite.

Impurity behaviour

The impurity behaviour is discussed in detail by Hawkes *et. al.*, (1989) and by Thomas (1990). With carbon limiters and walls, after prolonged operation and glow discharge conditioning in helium, Zeff values between 2 and 3 can be obtained for

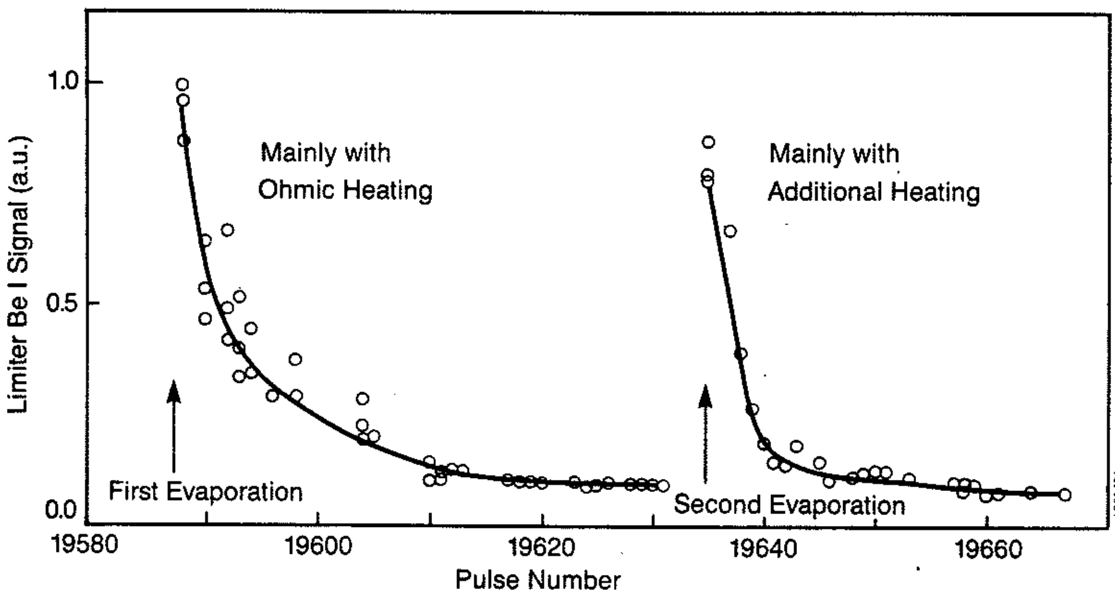

Fig. 2. Erosion of evaporated beryllium layers with pulse number

moderate plasma currents and low densities (3 MA, $1 - 2 \times 10^{19} m^{-3}$). After tokamak discharge conditioning in helium slightly lower values can be achieved. The main impurities are carbon with typical concentrations of 5%, oxygen with 0.5 - 1%, and nickel with 0.01 - 0.1%.
With the start of the beryllium/carbon phase Z_{eff} decreased to 1.5 - 2, mainly due

to the reduction of carbon in the plasma to about 2%. The oxygen content is reduced by a factor of about ten and its contribution to Z_{eff} becomes negligible. This behaviour can be explained by oxygen gettering and the resulting decrease of carbon sputtering by oxygen. Nevertheless carbon remains the dominant impurity. Beryllium concentrations reach ~ 3% immediately after evaporation but fall rapidly to about 0.5% after a few discharges.

The apparent lifetime of the evaporated beryllium layer, derived from the decrease of the beryllium flux from the limiter, was short as shown in Fig. 2. This decrease is especially fast for discharges with additional heating were the beryllium flux is halved after about four discharges. The reduction in Z_{eff} together with the reduced content of oxygen and carbon do not depend on the surface coverage with beryllium of the components in contact with the plasma and remain unchanged over tens of discharges.

It was found that a small deuterium puff (~50 mbarl) during high power heating could reduce substantially the plasma contamination. Consequently Z_{eff} depends now to a large degree on the method of setting up the discharge and can be strongly influenced by gas influx. Figure 3 shows a comparison of discharges with and without that additional gas influx.

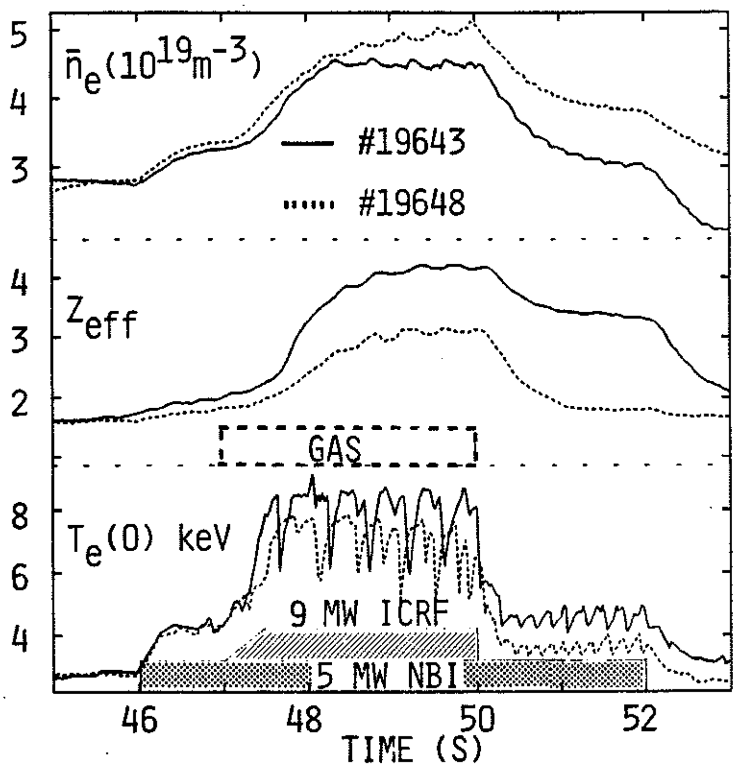

Fig. 3. Evolution of average electron density, effective charge and central electron temperature with and without additional gas injection

The impurity reduction could be achieved with affecting only marginally the central electron temperature or density and in addition the total neutron yield increases during the discharge with the additional gas feed. This behaviour is discussed in detail by Gondhalekar *et. al.*, (1990). It is observed that the additional gasflow into the edge increases the deuterium flux from the limiter into the plasma whereas the beryllium flux from the limiter remains unchanged. The concentration and accordingly the impurity content (mainly carbon) in the plasma is however reduced. This behaviour cannot be explained by the impurity screening model (G. M. McCracken *et.al.*, 1985). As an alternative it has to be assumed that the global particle confinement time has decreased. The central electron temperature did only change marginally with the gas puff indicating that the energy confinement time did not change substantially in contrast to the particle confinement time which is reduced by about 50%.

During the operation with the beryllium limiter beryllium became the dominant impurity, with carbon and oxygen both contributing negligibly to Z_{eff} or radiation. For low level additional heating (2 MW neutral beam, 2 MW RF) the radiated power was below 20% with nickel from the RF-antennae screens radiating about half of this value. Discharges with more than 10 MW input power lead to hot spots on the limiter and to a severe increase of beryllium influxes. It was found, similar to the operation with beryllium gettering, that a gas puff of a few 100 mbarl during the additional heating period reduces Z_{eff} to values of about 1.5 and this even with heating powers of up to 30 MW for several seconds.

The Z_{eff} values of 1.5 achieved with an additional gas pulse at heating powers of up to 30 MW are comparable to the best ones ever obtained for a well conditioned graphite limiter for ohmic discharges. For the graphite limiter scenarios using high edge fuelling rates could not be employed due to the insufficient wall pumping and lower density limit.

The variation of the effective charge for ohmic discharges as function of density is shown in Fig. 4 for the different phases including the results obtained in 1988 with a well conditioned graphite machine.

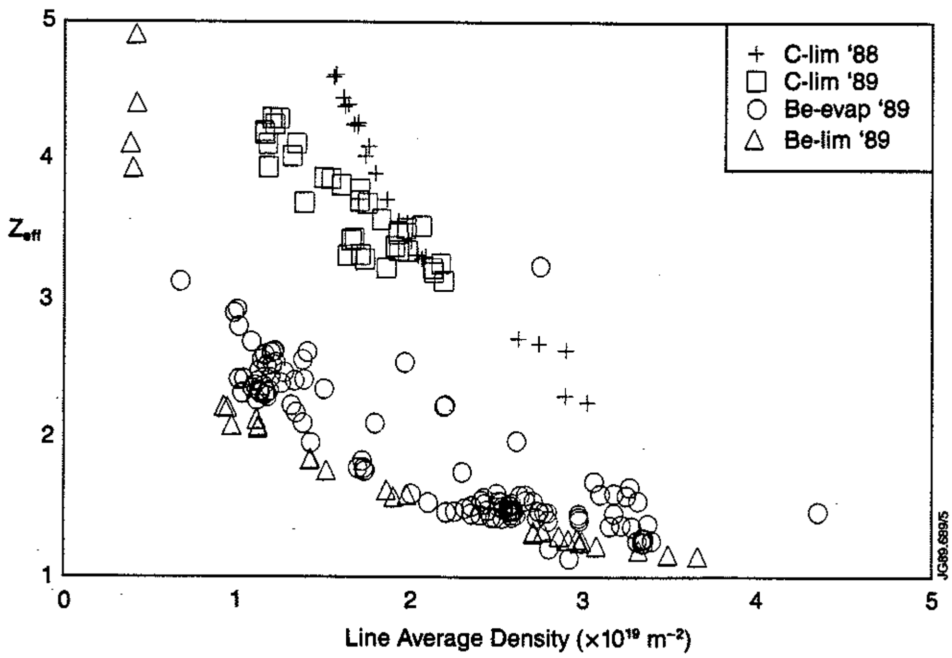

Fig. 4. Effective charge as function of density for ohmic discharges

With the reduction of the effective charge and the increased plasma purity the number of deuterons on the axis has increased for a given electron density. For $n_D(0)$ as the number of deuterons and $n_e(0)$ the number of electrons on the axis the dilution D becomes

$$D = n_D(0)/n_e(0) \tag{1}$$

For additionally heated discharges the dilution changed from 0.5 - 0.6 during the carbon phase to 0.8 to 0.9 during the beryllium phase. Table 1 summarizes the main impurities and the corresponding dilution.

Table 1. Typical impurity content and dilution for the various operation phases in ohmic discharges

	C-Phase	C/Be-Phase Limiter	C/Be-Phase X-Point	Be-Phase Limiter
Carbon (%)	5	3	1.5	0.5
Oxygen (%)	1	0.05	0.05	0.05
Beryllium (%)	-	1	1	3
Dilution	0.6	0.8	0.9	0.85

The dilution as well as the Z_{eff} values depend strongly on the power per particle. For heating powers in excess of 4 MW the Z_{eff} values and the dilution are given for the three operation phases in Fig. 5 and Fig. 6 as a function of the power per particle. For high power discharges the values were taken before the influx of impurities terminated high performance phases e.g. before the carbon bloom or high beryllium influx occurred. The points which are shown are points of existence which lie in three distinctive separate areas which represent the carbon, the C/Be and the Be phase. For the limiting cases, to stay e.g. below $Z_{eff} = 1.5$, the power per particle in the Be phase can be about a factor of 2.5 higher than in the C/Be phase and a factor 4 higher compared to the carbon phase. Similarly for the dilution, to stay above values of 0.8, the maximum tolerable power per particle relates as 1 : 1.5 : 2.5 with the Be phase again allowing for the highest powers.

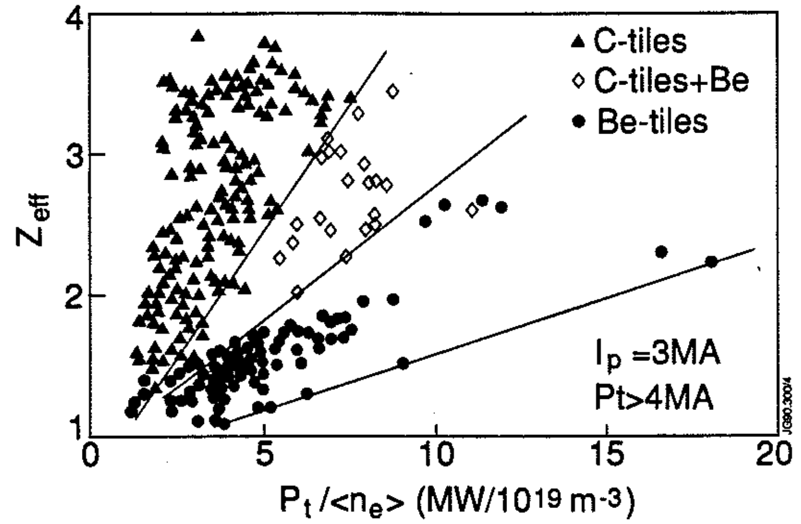

Fig. 5. Z_{eff} as function of power per particle

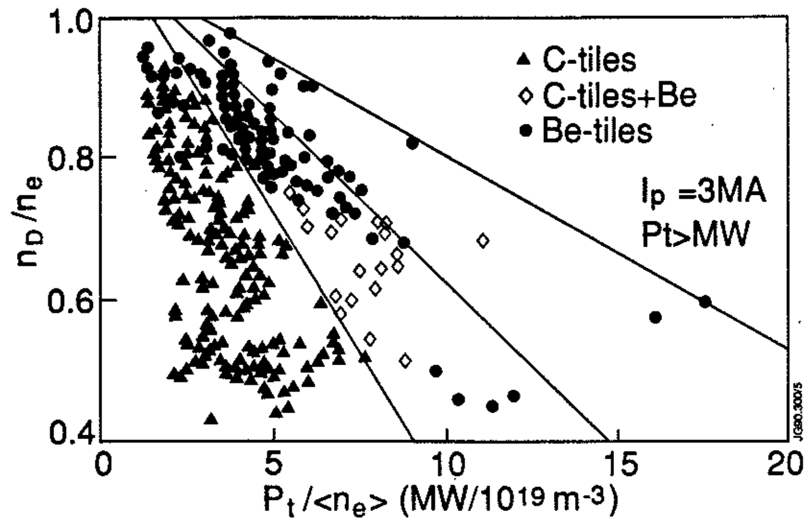

Fig. 6. Dilution as function of power per particle

Density limit and disruptions

In a carbon vessel the density limit occurs as a radiation limit. It is invariably a disruption limit. When during increasing the density the radiated power becomes comparable with the input power, the plasma edge cools and the plasma detaches itself from the limiter. An m = 2 instability grows which is destabilised by the radiative contraction of the temperature and current profiles (Wesson *et. al.*, 1986) and the plasma disrupts after times which can be as long as 1000 ms .This behaviour changed completely during the beryllium phase (Lowry *et. al.*,1990) as shown in Fig. 7 for an ohmic discharge for the beryllium limiter. As soon as the radiated power reaches about 50% of the input an asymmetric radiating structure, a Marfe, appears and after a short time time the radiated power increases well above the ohmic input in form of a short spike. Excess density is simultaneously ejected and the plasma recovers. By continued fuelling this behaviour can be repeated several times. Instead of a disruptive density limit a soft 'Marfing' limit is obtained. The decrease in density at the end of the pulse coincidences with the ramping down of the plasma current which is normally accompanied with a density pump-out.

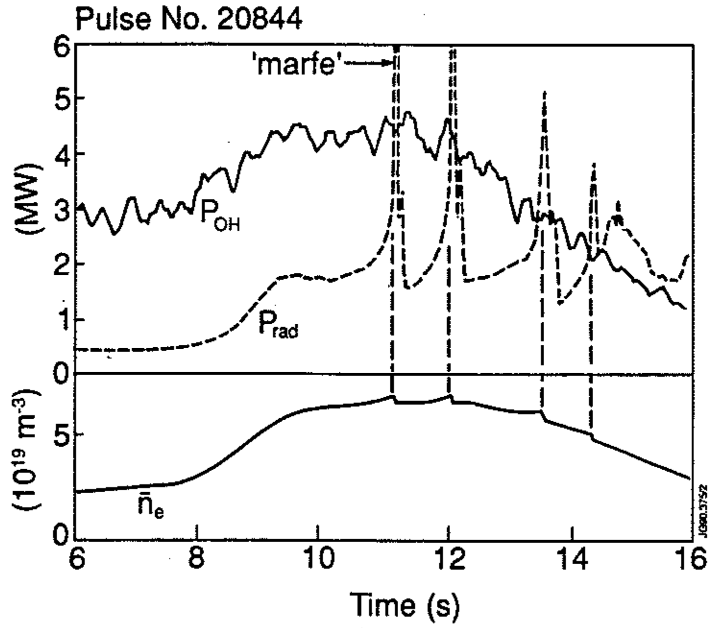

Fig. 7. Density development during an ohmic discharge with the beryllium limiter

The density limits in ohmic plasmas for a carbonized or beryllium gettered vessel are very similar despite lower impurity and radiation levels in the latter case. In ohmic plasmas with a beryllium limiter the density limit (marfing limit) inreased by a factor of about two and approached the density limit for neutral injection in a graphite vessel. The density limit for RF-heating was in the carbon phase only marginally larger than the ohmic limit. It is now identical with the neutral beam

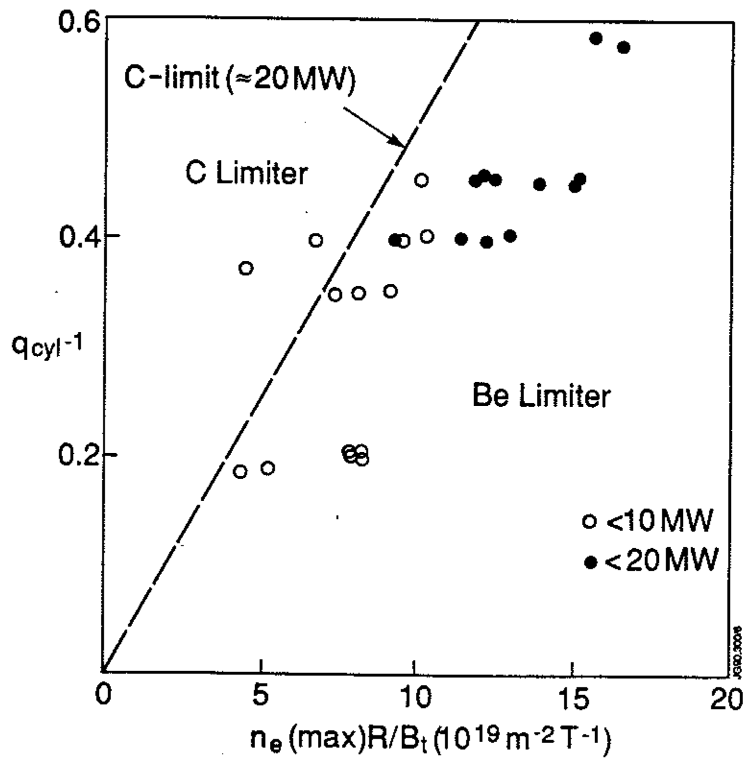

Fig. 8. Hugill diagram of operation with carbon and beryllium limiter

limit which is by about a factor of two higher. This behaviour is shown in Fig. 8 for discharges with the graphite and the beryllium limiter.

The density limit for the berylliated vessel or for the beryllium limiter depends on the power input and scales as the square root of the input power as shown in Fig. 9 for different heating methods. A better fit is obtained when for the Murakami parameter the edge density is used instead of the average density. For input powers of 10 - 20 MW a maximum value of about 33 x 10^{19} m^{-2} T^{-1} was obtained for the product of the Murakami parameter and the safety factor.

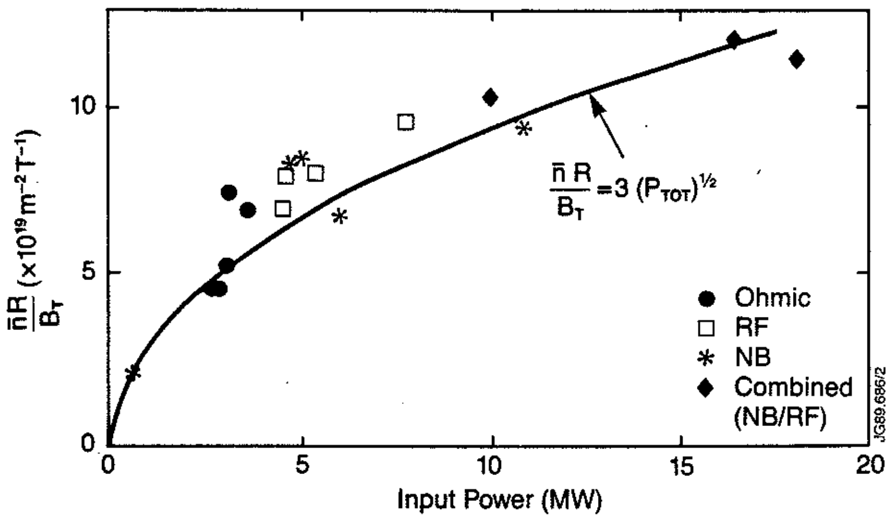

Fig. 9. Murakami parameter as function of input power during the beryllium phase for different heating methods at q_{cyl} = 2.5 and gas fuelling

The density profiles obtained during the experimental investigation of the density limit are generally flat or even hollow for high gas feed rates. Peaked profiles can be obtained for pellet fuelling, the highest central density which could be sustained was ne ~ 4 x 10^{20} m^{-3}.

Wall pumping and tritium retention

Graphite walls show hydrogen pumping during discharges (Sonnenberg *et. al.*, 1986; Ehrenberg *et. al.*, 1989). For a few seconds particle removal rates of up to a few $10^{20}s^{-1}$ can be obtained and even more after conditioning of the walls with tokamak discharges in helium. For beryllium the wall pumping can be more than one order of magnitude larger than for graphite. The mechanism appears to be the pumping of hydrogen atoms by metal walls (Ehrenberg *et. al.*, 1990; Saibene *et. al.*, 1990; Pick *et. al.*, 1985; Waelbroeck *et. al.*, 1979). With beryllium surfaces it is now possible for the first time to control the plasma density in such a way that starting from $10^{20}m^{-3}$ the density can be ramped down to $10^{19}m^{-3}$ within a few seconds.

Due to beryllium behaving as any metallic wall material, the vacuum vessel is no longer deconditioned after a high current disruption, therefore the number of useful discharges could be considerably increased.

One measure for the pumping capability of the wall is the characteristic time for deuterium pump out during a discharge after the gas supply is switched off. Typical times range from about 20 s for an unconditioned graphite machine to few seconds for a conditioned beryllium limiter. Figure 10 shows these times for the wall materials used and for different conditioning methods.

To obtain the same plasma density with beryllium as with graphite walls it is required to inject up to three times as many particles into the plasma. Therefore an area of concern is the retention of pumped hydrogen isotopes in the wall. Gas balance experiments (Sartori *et. al.*, 1990) indicate that for graphite limiters about 60% of the deuterium required to fuel the discharge is retained in the vessel, compared with about 10 - 20% for the beryllium limiter or beryllium evaporation. Taking into account the larger amount of gas which has to be used to obtain similar densities for discharges with the graphite or the beryllium limiter, the total retention (number of particles) is about equal in both cases. From these measurements it can be concluded that for 100 high density, full power discharges during the D-T operation in JET up to 3 g of tritium can be trapped in the walls. This does not pose a problem with the tritium inventory.

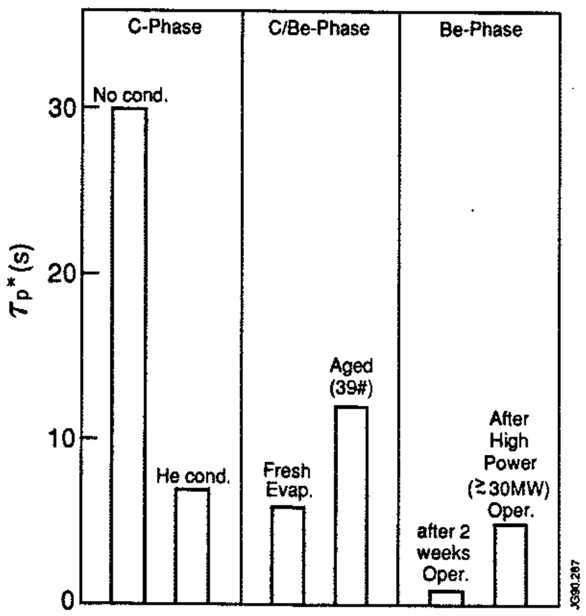

Fig. 10. Density pump out times for the different wall materials

Power handling capability

The maximum energy which can be accommodated by the high heat flux components in JET is higher than that corresponding to maximum heating power and typical maximum pulselength for which useful discharges can be sustained. The inner wall is able to survive loads of up to 400 MJ whereas useful plasmas could only be produced for a maximum of 17 MJ. At higher loads the plasma became contaminated by carbon to such an extent that with a dilution n_D/n_e of 0.7 at the start of the additional heating virtually no deuterium was retained in the plasma (n_D/n_e~0) after about one second (carbon bloom). This happens as soon as the graphite reaches temperatures in excess of 1200 C when radiation induced sublimation and self sputtering become important.

The high surface temperatures result from misaligning of individual high heat flux components so that power loads are intercepted only by small surface areas. Reducing the deviation from circularity at the inner wall from about 20 mm to 4 mm did not suppress the carbon bloom; there was no substantial change neither in the loads nor in time delay between applying additional heating and the occurrence of the bloom. Operation with the graphite belt showed that for injected energies of 50 MJ the plasma dilution was already 0.5. The maximum injected energy applied was about 120 MJ. Localized surface damage was observed with small cracks developed perpendicular to the tile edges in highly loaded and consequently eroded areas.

The beryllium belt limiter was designed for a peak heat load of 4.8 MWm^{-2} under the assumption that 100% of the injected power is conducted and evenly shared between the upper and lower ring of the belt and that the scrape-off thickness lies in the range from 7.5 to 15 mm. The power handling capability is 40 MW for 10 s and the resulting surface temperature is 1000 C.

It was found during the operation with the belt limiter that even power sharing between the top and bottom ring could not be achieved. Under the best conditions the bottom ring received only 60% of the power conducted to the top one. The observed values for the scrape-off thickness (~ 5mm) were about a factor of two lower than those assumed for the design. That means that whereas for 40 MW of power flowing to the limiter the design value for the peak load is 4.8 MWm^{-2}, the actual loading could be as high as 15 MWm^{-2}.

Operation to date has been with smaller values than 40 MW conducted but the loads applied exceed the design value by a factor of two. Moreover, in contrast to the graphite limiter, the edges of the beryllium tiles are further apart and chamfered. Due to fieldlines penetrating deeper between two adjacent tiles and their steep angle of incidence on the chamfers the beryllium tiles can receive loads of up to 100 MWm^{-2}. In addition misalignment of tiles due to mechanical inaccuracy leads as well to to increased heat loads as does the fieldripple which is about 2mm at the position of the belt limiter.

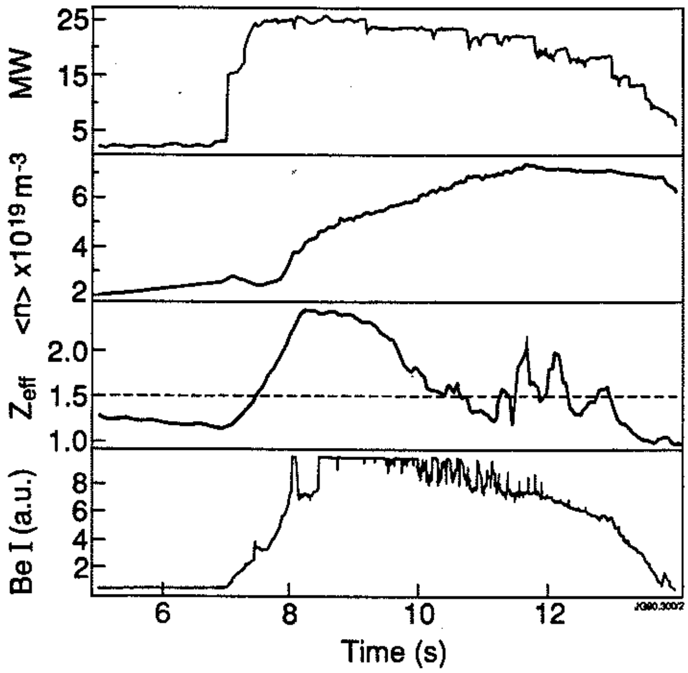

Fig. 11. Scenario for 180 MJ deposition into the torus

For the beryllium belt limiter during ohmic heating there was only moderate heating observed and no hot spots developed. The beryllium influx remained negligible. Discharges with additional heating, typically at I_p = 3 MA with heating powers from 10 MW onwards, led to the appearance of localized hot spots within 0.5 s after applying the power. This is consistent with the assessment of the power loads. Large beryllium influxes were observed. By tailoring the gas feed rates, a new operating regime was found which suppressed the build-up of the beryllium concentration in the plasma and allowed us to to apply up to 180 MJ to the plasma with Zeff values about 1.5 as shown in Fig. 11.

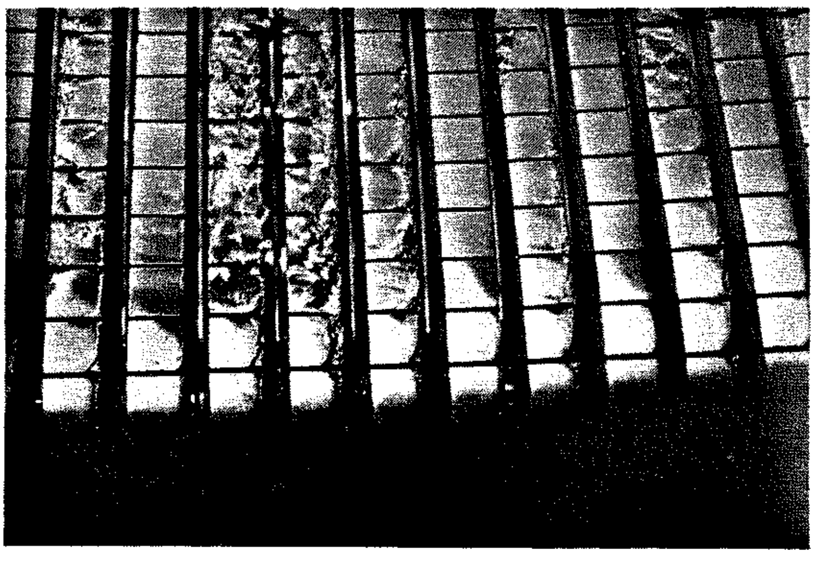

Fig. 12. Surface damage of the beryllium limiter

With increasing additional heating power hot spots were observed regularly and inspection of the tiles after opening of the vessel showed that about 5% of the surface had been melted as shown in Fig. 12, with the damaged areas centred around the minimum in the field ripple. A further 10-15% show signs of melting at the edges and additional 5% show damage due to localized abnormal loads (stress induced cracks). None of the tiles shows deep fissures, none lost substantial material from the surface or suffered any mechanical failure. A detailed description of the different types of defects is given by Deksnis *et. al.*, (1990).

Damage is localized with very severely affected tiles adjacent to those without any marks. There has been no gross mechanical failure on the more than 34000 castellations of the belt limiter. The observed failures relate more to the design features then to material problems. Therefore JET restarted operation in 1990 with essentially the same set of beryllium tiles which were used earlier. One of the aims of the present operation is to assess the behaviour of a surface damaged limiter under high power loads for long duration discharges at many repetitions.

PLASMA PERFORMANCE

The changes in the impurity and recycling behaviour resulted in improved plasma performance. Increased wall pumping allowed us to obtain the hot ion mode on the limiter, reduced radiation to investigate the beta limit, and reduced dilution to increase the fusion power from $^3He - D$ reactions to 100 kW and the fusion parameter for X-point discharges to 8 x $10^{20} m^{-3}$keV s.

Hot ion plasmas for belt limiter discharges

For the operation with the carbon limiter the low density, high ion temperature regime is not accessible because of the low deuterium pumping of the limiter. Even tokamak discharge conditioning in helium was not effective to increase the pumping capability. Poor density control and the related difficulties in obtaining low density target plasmas were the consequence. For operation at higher densities, depending on conditioning, dilution ranges from 0.4 - 0.8, typical values are 0.5. Maximum Q_{DD} values of 5 x 10^{-4} were obtained.

With beryllium evaporated on the graphite belt limiter deuterium pumping becomes sufficiently strong for operation at low electron densities (~ 10^{19} m^{-3}). High power per particle was achieved and consequently high ion temperatures were obtained (T.T.C. Jones *et. al.*, 1990). The carbon influx from the belt limiter gives n_D/n_e of about 0.6, the Q_{DD} values increased to ~ 6.5 x 10^{-4}.

The plasmas using the beryllium limiter behaved similarly to those with beryllium evaporation with respect to power per particle, ion temperature and dilution. The density profiles were however, flat in contrast to those obtained earlier with the graphite limiter or beryllium evaporation. Pellet fuelled target plasmas resulted in peaked profile hot ion discharges. The maximum fusion yield in peaked discharges was Q_{DD}~ 9 x 10^{-4}. The neutron yield increased applying RF power. In many cases the neutron output was reduced after the influx of beryllium. Table 2 summarizes the maximum obtained power per particle and the resulting dilution and Q_{DD}-values for the operation with the belt limiter.

Table 2. Dilution and Q_{DD} for limiter discharges

	Power per particle (10^{-19}MW m^{-3})	Dilution	Q_{DD} (10^{-4})
Graphite	9	0.5	5.0
Graphite plus Be evaporation	14	0.6	6.5
Beryllium Gas or NI fuelled	20	0.6	7.7
Beryllium Pellet fuelled	20	0.6	9.0

Alpha particle simulation

For the full utilisation of α-particle heating in a reactor it is necessary that the slowing down time of the α-particles is shorter than their confinement time. To study their transport α-particles were simulated by using the nuclear reaction $^3He + D \rightarrow {}^4He(3.6\ MeV) + p(14.7\ MeV)$. That was made possible by heating ^3He minority ions with ICRH in a deuterium background (Start *et. al.*, 1990). The parameters for the energetic helium ions produced by the ICRH are very similar to those expected from DT fusion reactions in JET or NET. The main difference is the ratio of parallel to perpendicular pressure, which is very anisotropic for the RF driven minority.

The experiments were carried out during Monster sawtooth discharges with plasma currents ranging from 2 to 5 MA, toraidal fields from 2 to 3.4 T, densities on axis from 2 to 6 x $10^{19}\ m^{-3}$, and central electron temperatures from 4 to 12 keV. On axis RF-heating at powers below 14 MW was used.

Previous experiments were carried out with carbon walls and limiters. The resulting maximum fusion power which could be obtained from the D - ^3He reaction was 60 kW. This power was limited by severe carbon influx. The dilution was 0.4. This problem could be eliminated by using beryllium evaporation. As a consequence of the higher plasma purity ($n_D/n_e \sim 0.7$) the reactivity increased and fusion powers of up to 100 kW were obtained. It was found that the α-particles slow down classically as predicted theoretically. Therefore it can be expected with confidence that efficient α-particle heating will occur in DT-burning tokamaks.

The beta limit

For thermal plasmas the maximum obtainable beta value is limited by MHD phenomena i.e. either by resistive kinks (Troyon-Gruber limit) or by ballooning modes. Under certain conditions the the ballooning limit may be higher by about 50%. Earlier experiments to investigate the beta limit in JET suffered from carbon influx and high dilution and the peaked density profiles became unstable at about 40% of the Troyon limit.

After beryllium evaporation the dilution was reduced and flat density profiles were obtained with neutral beam heating. The beta limit could be reached for double null X-point discharges with low toroidal fields ($B \sim 1.2$ T) during the H-mode phase (Smeulders *et. al.*, 1990). The required power to reach the beta limit was about 10 MW of neutral injection.

The beta limit in JET is a soft limit and follows the Troyon-Gruber relationship. It is characterised by beta-clipping, i.e. without becoming unstable a relaxation in mainly temperature occurs periodically whilst trying to exceed the limit. The maximum beta-values obtained were between 5 and 6% in good agreement with the prediction for JET performance.

H-mode with RF only

For a long time it appeared to be impossible to generate H-modes by ICRH alone, especially with the RF antennae at the low field side. With beryllium evaporation onto the nickel screen of the RF-antennae and dipole phasing it could finally be shown that it is possible to obtain H-modes with RF only (Bhatnagar *et. al.*,1990). That was a considerable change from the previous behaviour when for example the application of ICRH to neutral beam generated H-modes terminated them. This was caused by strong impurity influxes from the antenna screens. With advances in the antenna phasing the coupling could be improved and ICRH could be applied to neutral beam generated H-modes. There was however no improvement observed in the plasma behaviour. The radiated power was enhanced due to an increase of oxygen and nickel impurities with applying the RF power. The nickel was released from the antenna screens due to sputtering in the RF-rectified sheath in front of the antenna.

After beryllium gettering in the vessel and with an evaporated beryllium layer on the screens the impurity release during RF heating was considerably reduced due to decreased edge density. With this improvement and the simultaneous use of dipole phasing for the antennae, RF only H-modes of up to 1.5 s duration were obtained at power levels of up to 12 MW. Threshold power, edge behaviour and confinement is similar to H-modes obtained with neutral beams only. Energy confinement times are similar to NBI cases and reach two times Goldston L-mode scaling. Figure 13 shows a time trace for a typical RF H-mode shot were the signature of the transition from L to H-mode can be clearly seen in the D_α and density traces.

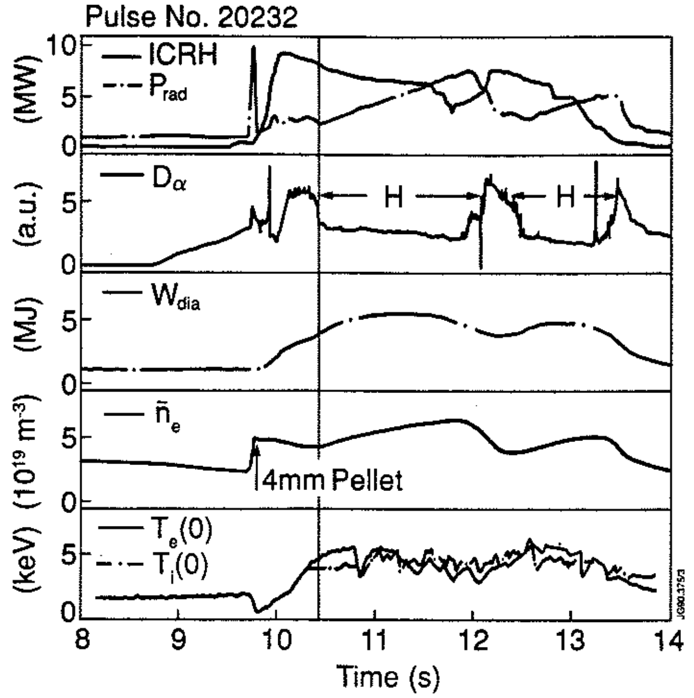

Fig. 13. Time traces for a RF only H-mode

Maximum fusion performance

Improvements in the fusion parameter $n_D\ \tau_E T_i$ were obtained in X-point discharges by using the techniques of beryllium evaporation to reduce the dilution, 140 keV beams (6 MW at 140 keV and 11 MW at 80 keV) to achieve deeper penetration and X-point radial and vertical sweeping to reduce the temperature of the target tiles and therefore to delay the carbon bloom.

For double null X-point discharges at 4 MA and a toroidal field of 2.8 Tesla the best conditions were obtained (Tanga *et. al.*, 1989; Harbour *et. al.*, 1989). The density on axis reached 4 x 10^{19} m^{-3}, the central electron temperature 8.6 keV at

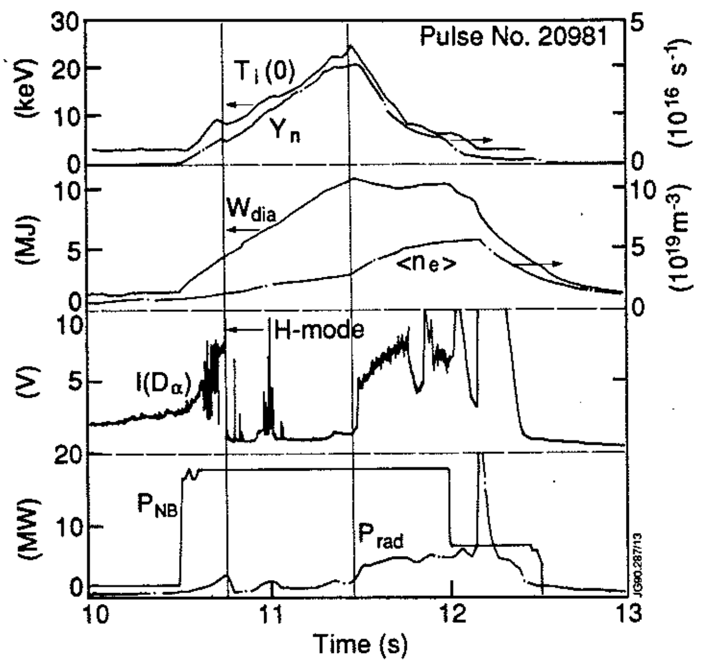

Fig. 14. Time trace for a maximum performance discharge

Zeff values of 1.4. The dilution at the maximum neutron output was ~ 0.9 and with the strong wall pumping in the resulting low density plasmas ion temperatures reached 22 keV. The confinement time (τ_E~ 1.1 s) did not change compared with previous operation with graphite. Neutron yields reached values of 3.5×10^{16} s^{-1} corresponding to Q_{DD} ~ 2×10^{-3}; the fusion parameter is increased to values exceeding 8×10^{20} m^{-3} keV s and the resulting equivalent fusion power would reach 12 MW for 18 MW of additional heating. The best conditions were only obtained transiently (~ 0.1 s), the carbon influx terminated the good performance. An example for a high performance H-mode discharge is given in Fig. 14.

SUMMARY AND CONCLUSIONS

From the experiments carried out during the beryllium assessment we can conclude that both, higher plasma purity levels and higher density operation can be achieved with beryllium limiters in comparison with graphite ones. These improvements result mainly from the elimination of oxygen and the strong wall pumping capability of beryllium. Good use was made of the widened operation regime: low density high ion temperature discharges on the limiter were possible, beta limits could be explored and RF only H-modes were obtained.

Despite very accurate alignment of the belt limiter hot spots and local melting of the limiter surface was found. Consequently plasma facing components have to be designed in such a way that alignment is not critical and that the heat load is distributed evenly.

The H-mode performance was considerably improved compared to earlier experiments, however this mode of operation was prevented from reaching its full potential by a strong influx of carbon after one second after the commencement of the high power heating. Calculations show that an equivalent Q_{DT}-value above unity would have been achieved by delaying the carbon bloom by another second. Steps are being undertaken to improve the performance. For the 1990 operation the carbon tiles on the lower X-point target plates and the nickel antenna screens are replaced with beryllium. Later the X-point target plates will be watercooled and better aligned than presently and disruption feedback coils will be installed to stabilize m=2, n=1 modes. Furthermore it is proposed to install early in 1992 a pumped divertor with the aim to study particle and power exhaust and impurity transport with power and particle loads similar to those expected in the next generation of tokamaks.

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