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Neutral Beam Heating and Current Drive Systems

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ABSTRACT.

Several Neutral Beam Injection (NBI) systems are now capable of injection of powers greater than 20MW, some for pulse lengths in excess of 5sec. These high power NBI systems have demonstrated the flexibility and utility of neutral beam heating; coupling easily to many plasma configurations; providing heating, fuelling and current drive capability and allowing access to new, improved plasma confinement regimes. All the high power NBI systems have now achieved routinely high reliability and system availability ($\gtrsim$80–90%).

The current status of NBI systems is reviewed and successful solutions to the technological problems encountered in design of high power beams, ranging from high heat transfer elements to large area cryopumps, are discussed. The interface to Tokamak operations such as the drift duct, stray field cancellation and fast interlock mechanisms are presented, emphasising the considerable progress in these areas.

The upgrading of NBI systems for tritium injection on JET and TFTR is reviewed.

The possible parameters for a 'Next Step' injector are presented, showing the choice of negative ion beams as an attractive option. With reference to the existing conceptual designs for ITER/NET it is shown that the main areas of uncertainty exist in the source and accelerator stack designs. The beamlines themselves should be able to take advantage of the properties of negative ions to lead to some marked simplifications.

1. INTRODUCTION

The past decade has seen significant advances in Neutral Beam Injection (NBI) systems. There are now several systems (on JET [1], TFTR [2], JT-60 [3] and DIIID [4]) capable of injecting more than 20MW of fast neutral atoms into the Tokamak plasma. The TFTR system has been operated successfully up to 30.5MW. These high power systems have developed into a flexible and useful experimental 'tool' for the fusion physicists. They have demonstrated the capability of neutral beam heating in terms of its ability to couple easily to many plasma configurations and to provide simultaneously heating, fuelling and current drive for the plasma.

NBI has been in the forefront of demonstrating access to new improved regimes of plasma performance such as the 'H' mode of Tokamak confinement, first seen in ASDEX [5] and subsequently in many other Tokamaks, and the 'supershot' regime in TFTR[6]. The dominance of NBI over other heating schemes is demonstrated by the list of record plasma parameters achieved by its use on the major experiments. Amongst the notable achievements are:

- record peak ion temperature of 28-30 keV in TFTR with 13.5 MW NBI into a low density plasma (Bitter et al. [6])
- record value of toroidal beta (β_{TOR}) ~ 10.7% achieved with 17-20MW NBI into a Double-Null H-mode in DIIID (Stambaugh et al. [7])
- record fusion product $n_d(0)\cdot T_i(0)\cdot\tau_E(0)) \sim 9\ 10^{20}$ m^{-3} keV sec achieved with 17 MW NBI into a Double-Null, hot ion H-mode in JET. This latter performance would extrapolate to a predicted Q_{DT}~ 0.8 if the same conditions were obtained in a 50:50 D:T plasma mixture (Keilhacker et al. [8]).

The constituent elements of the high power beam systems of 1990 are the same as those of a beamline of 10 years ago and are shown in block diagram form

in Fig.1. The performance of the elements of the beamline has improved substantially over the 10 years, however, and this has taken the technology from the most powerful beamline unit of 1980, the PLT injector delivering 0.75MW to the plasma for 0.3 seconds, to the most powerful beamline unit of 1990, the JET injector delivering up to 10.4MW to the plasma for 10 seconds.

In this paper we given an overview of important aspects of the technical status of positive ion injection systems. Due to lack of space, power supply systems will not be reviewed. The upgrades of positive ion beamlines to inject helium and tritium beams will then be discussed. We then show that the requirements for the injection systems on the 'Next Step' devices will necessitate the development of negative ion beamlines and review the status of negative ion source and accelerator development, highlighting the issues which must be addressed in the near future.

2. TECHNICAL OVERVIEW OF THE STATUS OF POSITIVE ION INJECTION SYSTEMS

2.1 Ion Sources

Ion source development has been driven by the twin needs of trouble-free source operation and high percentage yield of monatomic ions. The latter requirement arises in order both to maximise the penetration of the neutral atoms into the plasma (as shown in section 5) by maximising the full energy component (E_b) of the beam; and to limit the complication of ion dumps needed to cope with the ions of energy ($E_b/2$) and ($E_b/3$) produced by the breakup of diatomic and triatomic ions accelerated from the source.

Most present generation systems have adopted the magnetic multipole or 'bucket' source. An example of this (from the JT-60 injector [10]) is shown in Fig.2(a). A permanent-magnet cusp field surrounds the source body. The majority of the volume of the source has a low magnetic field

(< 10-20 10^{-4}T.). The characteristics motivating the choice of the bucket source are its reliable and noise-free operation at high arc efficiency (~ 0.5A/kW) at low gas pressure (~ 0.25 Pa); the high volume with low magnetic-field which enables it to illuminate large extraction areas uniformly; the ease of obtaining a high monatomic species fraction (the JT-60 source gives ~ 92% H^+ : 6% H_2^+: 2% H_3^+ at a current density of ~ 2.5×10^3 Am^{-2}).

To obtain the best monatomic fraction yield, it is necessary to use a weak 'magnetic filter' (Ehlers and Leung [11]) to prevent the primary electrons from the filaments approaching the region near the extraction grids. This prevents the production of H_2^+ (or D_2^+) in this region. As H_2^+ production is prevented, H_3+ (resulting from $H_2^+ + H_2 \rightarrow H_3^+ + H$.) should also reduce. The low temperature electrons near the grids also result in enhanced destruction of $H_2^+(D_2^+)$ by disassociative recombination. The source is thus separated into a 'driver' region, where the ions are produced and an 'extraction' region. Large volume sources operating successfully on this principle are operational on JET [12], JT-60 [10] and TFTR [2]. An example of the siting of the filaments relative to a 'supercusp' filter is shown for the JT-60 source in Fig.2(b) [13].

2.2 Accelerator Systems

The most significant development for accelerator systems has been the success of multi-aperture, water-cooled extraction systems capable of DC operation. These were realised in slot geometry during the US Common Long Pulse Source (CLPS) program [14] and also in multi-circular aperture form for the JET injectors [15]. In operational experience the JET grids have accumulated an unrivalled performance. These grids, manufactured with water cooling channels passing between the rows of circular apertures (with 262 apertures over an extraction area ~ 0.45×0.18 m^2), have accumulated

> 10^5 (accelerator.pulse.seconds) since 1984. There have been no on-load or off-load failures in the system. The grids are manufactured by a sandwich process involving copper electrodeposition onto an OHFC copper base into which the apertures and water channels are machined. Beam pulses of up to 17 secs pulse length (80 kV, 60 A, Hydrogen) have been extracted from these grids with the plasma-facing (80 kV) grid being capable of DC operation at 75 kW power loading (~ 1% of extracted power + ~ 20% arc power).

Other important development programs have contributed to the improvements in terms of the optical quality of the beams accelerated by sources, especially the joint JET/Culham Laboratory/CEA Fontenay-aux-Roses development for the JET system which produced low divergence (9-10 mrad) multiple beams in 3 and 4 grid systems [1].

For its relevance to 'Next Step' injection systems, the radiation resistant, remote handleable JET Positive Ion Neutral Injector (PINI) source/accelerator system is noteworthy. This all-ceramic/metal injector is shown in cutaway form in Fig.3.

2.3 Deflection and Dumping of Un-Neutralized Ions

On most of the large NBI systems, the residual ions are dumped by magnetic means. The neutralisation of positive ions is a very inefficient process (the equilibrium, or infinite gas target, fraction of neutrals for deuterium beams at 140 keV is only ~ 30%) and a large amount of ion power with power densities in the range 30-100MW m^{-2} (perpendicular to the beam) is common in beamline designs. Inclination of dump elements yields only partial advantage due to space restrictions.

The two successful solutions to which have been realised on operational beamlines can be separated by their differing coolant water systems. The Low Pressure approach is represented by the

'HYPERVAPOTRON' elements used in the JET Ion Dumps, calorimeter and scrapers [1]. In these elements, a low pressure (0.6 MPa) water supply is passed through channels internally-profiled to increase both the turbulence and surface area. The elements can accept power densities up to ~ 15MW m^{-2} [16] with an overall coolant pressure drop of ~ 0.03 MPa. The High Pressure approach is represented by the High Heat Flux panels used on the ASDEX Long Pulse Neutral Injection (LPNI) system [17] and also on TEXTOR NBI. The panels, with their internally-finned tubes, are based on data from the US Neutral Beam Energy Test Facility (NBETF) at LBL [18]. They require water at 1.2 MPa, with 0.6 MPa pressure drop, but are capable of a DC operation at 25MW m^{-2} (and have been to 45MW m^{-2} without burnout). There is, of course, the accompanying drawback of the high stresses in components such as bellows associated with a high pressure system.

A notable development which may yield advantages on future systems has been that of the Electrostatic Beam Dump (ESBD) pioneered by the NBI group at CEA Cadarache. The ESBD is generally an in-line system surrounding the mixed neutral-ion beam passing through it. In positive ion systems, the ions are 'peeled-off' by allowing their space charge expansion to deflect them onto the beam dump. The resulting small angle deflections lead to low power densities. The ESBD can be coupled with an Energy Recovery System (ERS). This is planned for the TORE-SUPRA NBI system and has been tested at 75keV for 0.5 secs [19]. A schematic of the combined ESBD-ERS is shown in Fig.4. The system is unusual in having an ion source at ground potential and a neutraliser at HV potential. The ions are accelerated from the source and after the neutraliser, are decelerated to land on the recovery electrode at a potential ($-V_r$) around 5% of the beam energy. In this way only a fraction of the acceleration voltage is expended on these ions and energy is recovered. The system also features an electrode

to suppress the emission of secondary electrons caused by the ions hitting the dump. Such electrons would be accelerated towards the ground potential parts of the system. The gain of the ERS is clearly higher if neutralisation of the beam is low (at high positive ion energies), also since only the full-energy ions are recovered, a source with a high monatomic fraction is required. The gains from an ERS of this type have been estimated as a function of beam energy (E_b) and monatomic fraction (α_1) and are shown in Fig.4(b). It can be seen that for the present deuterium injectors in the 100-140keV range (TFTR, JET, JT-60) the system does not lead to significant gains.

2.4 Beamline Pumping

The present large Tokamak NBI systems all use cryopumps for beamline pumping. Most designs feature conventional chevron-baffles. The requirements for low background pressure (~ 10^{-3} Pa) in the final drift region of the JET beamline [1] in order to limit re-ionization of the neutral beams (see 2.5), in the presence of high gas flows (~ 26Pa.$m^3 \cdot s^{-1}$) were too stringent for chevron-baffled cryopumps and stimulated the design of the JET 'open structure' cryopumps [20]. The JET Open Structure design shown in Fig.5 increased the transparency of the pump by having an unobstructed opening, and by shielding the Liquid Helium (LHe) panels in a nested fashion behind Liquid Nitrogen (LN_2) panels to protect the LHe surfaces from line-of sight incidence of 'thermal' gas molecules. The cryopump thus achieved ~ 47% of the theoretical 'black hole' pumping speed, ie ~ 200 m^3 s^{-1}/m^2, which is nearly twice the value achieved by chevron pumps. The installed pumping capacity on the JET beamline could thus be increased to 8 10^3 m^3 s^{-1} (H_2) for the 40 m^2 area of pump.

Probably even more relevant to the 'Next step' injectors is the development of low-loss cryo transmission lines and large, continuously

operationing cryoplants. The overall system at JET gives a good example of what can be achieved in this respect. The system features ~ 60 m long low-loss coaxial cryo-transmission lines [21] in which the LHe supply is shielded by successive vacuum; cold He boil-off gas; LN2 supply; cold N_2 boil-off gas and vacuum layers. The measured losses are ≤ 10 mW m^{-1} (LHe) and ≤ 150 mW m^{-1} (LN_2). The cryogenic plant has a LHe refrigerator for gas'boil off' recovery [22] and is capable of supplying 800 W at 4.4°K and a LN_2 storage system which delivers 1.5 m^3 hr^{-1} (equivalent to 65kW at 78°K). The plant supplies two JET beamlines and a pellet injector.

Developments such as these have allowed cryopumps to outstrip the performance of large-scale Titanium getter pumps even though there have been significant developments in regenerable versions of these [23]. Titanium getter pumps still retain the disadvantage that their pumping behaviour tends to depend on their previous history. Their capacity for H_2/D_2 (and hence T_2) retention before regeneration is also at least 2 orders of magnitude below that of the cryopump which typically can operate for two weeks between regeneration.

2.5 Drift Duct

The geometric access to the Tokamak plasma for NBI is generally very tight. The beams have to pass between the toroidal field (TF) coils of the Tokamak the distance between which must be kept small in order to minimise field ripple and the resulting loss of fast ions from the plasma. Access is also generally in a non-perpendicular direction in order to maximise beam deposition in the plasma and minimise shine-through onto the Tokamak inner wall. This situation is likely to exist in the Next Step machines and reactors where penetrations in the first-wall/blanket systems will be at a premium and hence will be kept as small as possible.

This restricted access leads to severe challenges in the design of Drift Ducts. In the drift region the beam neutrals are subject to reionisation reactions ($D_f^0 + D_2 \rightarrow D_f^+ + ...$) and high powers can accumulate from this in the long drift ducts necessary to penetrate coils, blanket etc. Long, narrow ducts have low vacuum conductance and are difficult to keep at low pressures to minimise reionisation.

The fringe field from the Tokamak has a strong spatial variation in the duct region and can act to focus the reionised power from the beams as is shown in an example from JET in figure 6. Extensive work has been done on this problem at JET [24], [25] following earlier accidents where duct protection was melted by the focussed power [24]. The recent work by de Esch et al. [25] shows good agreement between the simulated trajectories and the profiles of reionised power measured by *in situ* thermal instrumentation. This allows confident prediction of the safe operating conditions of the present (inertially cooled) duct protection which will accept peak reionised power loadings of $\leq 4.4 MW\ m^{-2}$ for pulse lengths of up to 10 seconds.

Reionised beam particles hitting the duct protection will release gas molecules. These form an increased gas target which will lead to enhanced reionisation and the situation will spiral out of control unless the gas release is kept to a minimum. Early work on this problem [26], [27] recognised the exponentially-deteriorating situation termed 'BEAM BLOCKING'. This occurs if the gas re-emission coefficient Γ (molecules per incident ion) exceeds a critical value Γ_{crit} and results in loss of the entire beam in reionisation. Γ_{crit} can be written in terms of beam and duct parameters as:

$$\Gamma_{crit} = \frac{C}{\sigma_{01}\, L\, \phi_B} \qquad (1)$$

where C is the duct conductance; L the duct length; ϕ_B is the beam flux through the duct and σ_{01} is the reionisation cross-section.

For long pulse systems where low reionisation power is required, $\Gamma << \Gamma_{crit}$ is needed and improvements in beamline pumping and materials evaluation have moved towards this situation. Bickley et al. [28] have shown in a study of the JET duct that metal (copper) ducts yield a static situation of $\Gamma \sim 0.5$ (1 atom emitted per incident ion) after conditioning with beam and that the conditioning of a metal duct continuously improves with accumulated beam-on time. The JT-60 NBI team also report a continuously improving situation with a metal duct and find that only ~ 40 beam-seconds are required to bring duct-reionisation power losses below ~ 5% of the incident power [29]. In the JET study the static situation was reached in < 1000 MW.secs of beam through the duct. For carbon or carbonised ducts on the other hand, it has been found that conditioning takes much longer (~ factor 5 ×) and the de-conditioning by H_2 or D_2 absorption onto the carbon/carbonised surfaces can take place if conditioning is suspended for a significant period (~ 1-2 days).

A measure of success in overcoming the duct problem is provided by the 3-dimensional plot in figure 7. Here the critical gas re-emission coefficient of the duct (Γ_{crit}) is plotted against the percentage duct reionised power loss (W_R) for a conditioned system and the peak neutral power transmitted through the duct (P_0). Since low Γ_{crit} is a measure of minimum disturbance to the Tokamak system, success can be defined as simultaneous achievement of low Γ_{crit}, low W_R and high P_0. The achievement of this separates those beamlines (eg. JET) which are pushing into reactor-relevant regions, from the more conservative designs (eg. JT-60).

It can be shown [28], that the percentage reionised power W_R is, to first order, proportional to pre beam gas pressure Pg and Γ/Γ_{crit}

Low Γ/Γ_{crit} can be achieved via a clean metal duct and a low Pg is achieved by good differential pumping of the beamline such (with reference to Fig.7) the ratio (S_L/C_{DD}) is as high as possible, where S_L is the pumping speed of the last beambox chamber and C_{DD} is the drift duct conductance.

2.6 Interface to Tokamak Operations

In general, the 'coupling' of NBI power into a Tokamak plasma causes few physics problems and it is now easily understood how to design out *a priori* such effects as, eg, field ripple loss of fast ions. The 'Engineering Coupling' problems which are encountered have been successfully overcome on the operational NBI systems. In addition to the Drift Duct already mentioned, there have been significant developments in handling beam shine-through and the adverse effect of the Tokamak stray magnetic field.

2.6.1 Beam Shine Through Protection

Unattenuated beam powers impinging on the Tokamak Inner Wall can be high in the absence of plasma (eg, up to 31 MW m^{-2} in JET). The protection of the inner wall has been facilitated by the development of graphite armour tiles . Progress here has often had a spin-off in the development of, eg. carbon fibre composite tiles for graphite bumper limiters or divertor targets. The large-Tokamak NBI systems have shine-through detector interlock systems to terminate the beams if excessive temperatures are detected on the Tokamak inner wall. These systems have employed a variety of technologies including pyrometers on DIIID [4] and TFTR [30], infra-red diodes on JET [31] and CCD cameras on JT-60 [29].

2.6.2 Tokamak Stray Magnetic Field

The earlier generation of plasma ion sources, such as the periplasmatron (eg. on ASDEX) and the 'field-free' source (eg. on the early TFTR NBI

system) were often susceptible to Tokamak stray magnetic fields even at levels as low as 0.5 10^{-4} T. Such systems often had their own 'bucking coils' local to the plasma source.

In general, multipole sources are less susceptible to these fields. For many of the current beamlines, shielding 'passively' by using large assemblies of soft iron has been sufficient to reduce stray fields to low levels at the sources and in the beam transport regions. Fields ~ 5 10^{-4} Tesla at a multipole source are found to cause few problems for the source's uniformity or optics.

Two Tokamak NBI systems, those of JT-60 and JET, have successfully developed active feedback controlled field compensation systems. JT-60 is an air-cored Tokamak with high stray fields at the source, accelerator and dump regions. The decision was made at an early stage to rely on local passive shielding only in the source/neutraliser region and to actively shield the transport region to the dumps using cancellation coils under feedback control from sensors measuring the Tokamak stray field. With this system the JT-60 group were able to develop the first successful large-volume cancellation system [32], the performance of which is illustrated in Fig.8

On the JET NBI system, the passive shielding surrounding the beamline proved insufficient to shield the sources and neutralisers at plasma currents (I_p) above the original JET extended performance figure (I_p = 4.8 MA). At I_p = 7MA, the stray field measured at the JET PINI sources reached ~ 17 10^{-4} T. The high fields are caused by the saturation of the iron core of JET and also arise at the onset of the H-mode when the current in the vertical field poloidal field-coils rises strongly to counteract the strong increase in plasma β. These dynamic situations cause problems of deflection of the un-neutralised beam ions in the gas neutralisers which leads to beam deflection outside the tight geometrical requirements of the

JET system (allowed deflection ~ 1 mr horizontally). This can result in excessive power being intercepted by the beam scrapers. The problem has been solved by the successful development of a feedback controlled active field compensation system (AFCS) [33]. The JET AFCS is capable of backing-off up to 20 10^{-4} T over a volume of ~40 m^3. The magnetic field in rapidly changing plasma situations can now be maintained constant to an accuracy ~ $\pm$ 0.2 10^{-4} T [33].

2.6.3 Control, Diagnostics and Interlock

There are now completely computerised control and diagnostic systems on the beamlines of JET [34], JT-60 [3] and TFTR [35] with considerable operational experience. More interestingly, for future reactor-like operations, the large beamline systems on these machines and on DIIID have extensive interlock systems which terminate the beams in the event of out-of-normal conditions.

An example of the complexity of problems which can now be considered as solved in this area would be provided by the JET NBI system. This has a two-level basic-interlock network with a fast-response (10-20 ms) Fast Beam Interlock System (FBIS) backed up by a slower timescale (125 + ms) subsystem of the JET Central Interlock and Safety System (CISS) which is devoted to NBI. FBIS is a hardwired system which operates by gating sustaining pulsetrains (1 kHz) through to the NBI power supplies [36]. In the event of an out-of-normal event (eg, a pressure excursion in the beamline or disappearance of the Tokamak plasma) the pulsetrain is terminated which switches off the power supplies. This is an inherently failsafe concept and all parts of the system will be upgraded to provide redundancy and failsafe action prior to the start of D-T operations on JET. The CISS network is Programmable Logic Controller (PLC) based and acts both to backup FBIS and to Pulse Inhibit the injectors when dangerous

conditions (eg, lack of water cooling) occur out-of-pulse [34]. Operationally, the FBIS has been called upon to act mainly on disappearance or malfunction of the Tokamak plasma itself (> 95% of trips) . The system has not failed to act correctly since its commissioning in January 1986.

2.7 Reliability and Availability

In general all the large NBI systems now report >90% availability and reliability of >80%. Extensive statistical exercises have been undertaken to justify these claims on JET [37], JT-60 [38] and TFTR [39], spurred on by the approach of the D-T phase on the large Tokamaks in which access to the beamlines for repair and maintenance will be severely restricted.

JET and TFTR have devised essentially similar performance indicators to measure reliability of their NBI systems. This 'shot' parameter (ρ_{SHOT}) [39] is essentially the ratio

$$\rho_{SHOT} = \frac{\text{NBI Energy delivered to the plasma}}{\text{NBI Energy requested for the shot}} \qquad (2)$$

Using the measure indicated in (2) the TFTR performance gave $\rho_{SHOT} \sim$ 0.83-0.98 for 1988 operations at 95-110kV, whilst JET achieved $\bar{\rho}_{SHOT} \simeq 0.82$ for 1989 operations with 10 80keV PINIs and 6 140keV PINIs.

Most of the unavailability in these large systems is still dominated by the Power Supplies (over 50% of causes on DIIID, JET and TFTR). Considerable effort has been put into removing specific causes of unreliability which is proving to be successful.

The improved availability and reliability owes much to the improved quality of the injectors themselves and to the improved techniques of accelerator conditioning. Improved quality has benefited from the adoption of the common development programs of the 1980s. The two most successful of these were the European PINI joint development for JET

(subsequently adopted by TEXTOR and ASDEX-upgrade) and the US Common Long Pulse Source (CLPS) for TFTR, DIIID and MFTF-B [40]. These have had the worthwhile spin-offs of involvement of industry, improved QA and the (often unmentioned) necessity of having to share with others (and hence really understand) the database for an injector.

Improvements in injector conditioning have followed the deployment of core snubber assemblies [41] which dissipate the stored energy in the long injector HV transmission lines and prevent it damaging the accelerator grids in the event of breakdown.

2.8 Technical Spin-offs

The data accumulating on the mechanical engineering aspects of the large, high-power beamlines is providing valuable materials information across the Fusion Technology field. The 'multiple-unit' nature of most beamline systems and the large number of elements per beamline allow the possibility of meaningful statistical analyses. The power densities handled are also in the reactor-relevant regime and this leads to the use of beamline data as evidence of the pulsed capability of Next Step Tokamak designs (see e.g: the review by Watson [42]). Ion beam dumps give the most comprehensive information on the behaviour of high heat flux structures under pulsed loads in addition to their associated components such as bellows [43]. Structures such as the JET NBI Test Bed beam dump [1] have now accumulated > 3.5 10^4 pulses without problem.

There has also been a valuable study by Papastergiou [44] of the *Reliability Growth* experienced on a beamline design (the JET line) after design changes were made to rectify early failures in the bellows and neutraliser cooling systems. The study involved the use of the 'Duane model', a curve fitting model for the time evolution of fault occurrences. Reliability improvements can be statistically quantified in this way.

Regarding specific technical spin-offs, the extensive operational experience of the JET 'Hypervapotron' dump element design (section 2.3) has provided sufficient confidence in the design for it to be chosen for the actively-cooled target elements of the JET pumped divertor [45], [46]. Cardella [47] has also performed optimisation studies of the Hypervapotron design for Next Step divertor applications.

3. HELIUM INJECTION SYSTEMS

Several of the positive ion NBI systems originally built as H/D injectors are now in the process of establishing reliable facilities for Helium injection. The injection of ^{4}He beams can provide an accurately known, centrally deposited source of ^{4}He ions which can be used to simulate the centrally produced α-particles in a reactor.

Transport studies of thermalised α's and the behaviour of pumping and exhaust systems in the removal of α- particles can thus be studied. JT-60[48] have already used long-pulse (3 sec), ^{4}He beams at the 0.4 MW level in studies of divertor 'enrichment factors' for helium. TEXTOR have a working long-pulse ^{4}He injection system (45keV, 1.1MW, 2 sec.) [49] which is producing initial results. DIIID has also injected 75keV, 0.5 sec ^{4}He beams in studies of the Isotope effect in H-mode confinement [50].

Plans are now well-advanced for the injection of long pulse ($\leq$5 secs.) helium beams on JET, with a demonstration expected in October 1990. Due to constraints imposed by the beam deflection magnet designed for D^+, ^{4}He is limited to 120kV. ^{3}He operation is limited to 155keV by the 30A limit of the accelerator power supply. Whilst the ^{4}He will be used in the thermal helium transport studies previously mentioned, the ^{3}He beams are expected to provide interesting physics in several areas. These include the efficient central heating of plasma ions using the monoenergetic, highly penetrating ^{3}He beams (ion

temperatures of 50-60keV might be possible with 10MW of ^{3}He neutrals); the realisation of high fusion yield experiments using the D-^{3}He reaction. The acceleration of injected ^{3}He ions by the JET Ion Cyclotron Radio Frequency (ICRF) could further increase the ^{3}He energy towards the maximum in the D-^{3}He cross section (at nearly 300keV ^{3}He energy), and would enhance the fusion yield still further.

3.1 Pumping of Helium

Injection of helium neutrals has been made possible by the development of 'cryosorption' techniques for the pumping of helium. Helium gas is pumped in the NBI lines by its cryosorption onto a layer of gas previously condensed onto the LHe cooled panels. The gases which can be used for this effect have been found to be Argon [49], [51], 52]; SF6 [53] and N_2 [49]. Pumping speeds ~ 20-30% of that obtained for H_2 have been achieved for He in these systems. The ice layer typically needs to be renewed between shots in order maximise the pumping speed for He. The optimum quantity of cryosorbing gas which must be pre-condensed onto the LHe panels is about 20-50 times the total quantity of helium to be pumped during the injection shot.

4. TRITIUM INJECTORS

The Tokamaks with planned D-T operation phases (JET and TFTR) have firm plans to upgrade their NBI systems to provide injection of tritium beams. On JET this is planned for 1992, although if the JET Pumped Divertor proposal proceeds, the D-T phase will be delayed until 1995-6 [54]. For TFTR the planned dates are 1993-4 [55].

The injection of tritium beams provides a method of controlling the central isotope mixture in a D-T plasma. This should be especially important for the low

recycling 'supershot' regime in TFTR where the fuelling by NBI (S_N) approaches that from recycling neutrals at the limiter (S_R) with typically $S_N/S_R \sim 0.3$.

Some of the design issues on the upgraded tritium beamlines such as integrity of interlocks and reliability of equipment inside the biological shield, were already being tackled as part of NBI designs for the planned D-T Tokamaks. Other design issues specific to the operation of T° beams are:

- tritium gas supply to the Ion Sources and neutralisers, including gas efficiency
- tritium inventory (especially stringent for TFTR)
- breakdown of HV insulators in the presence of a β-emitter
- tritium permeation through ion dumps
- water and vacuum leaks on the beamlines.

We will make a brief survey of progress in this area, much of which is being achieved in a collaborative fashion by the JET and TFTR groups.

4.1 Tritium gas supply

The TFTR and JET injectors both have separate source and neutraliser gas feeds for flexibility of operation. The source gas feed passes through an SF6 shrouded HV insulator to enter the ion source at high voltage potential. This carries the small but finite risk of breakage of the insulator leading to SF6 contamination in the T_2 cycle and concomitant poisoning of the T_2 recovery catalysts. To avoid this an efficient single gas feed has been developed for the sources which introduces gas at ground potential into enclosure where it can access the accelerator grid interspace [56] and keep the plasma source from gas starvation. The replacement of the HV gas feed should lead to at least a 30% reduction in the Tritium inventory for a shot.

4.2 Tritium Inventory

The tritium gas will be supplied to the injectors from Uranium beds. For each experiment there are site-inventory limits for tritium with a fraction of typically ⅓ to ½of these limits being allowed for in the injection systems. The beamline inventory accumulating on the cryopumps will be recovered daily by regeneration of the LHe panels. On JET the 9 gr. (~ 85 10^3 Curie) T_2 inventory per beamline would yield ~ 17 shots per day (10 sec. beam pulse length). On TFTR the requirements are more stringent with only ~ 25 10^3 Curie allowed out of the Tritium vault [57].

The TFTR Ion Sources require interpulse conditioning (unlike on JET) and much work has gone into optimising this conditioning scenario to investigate isotope exchange in the Ion Sources to enable tritium-less conditioning scenarios [84].

Tritium trapped in the ion dumps from ion beam 'drive-in' effects will form a non-regenerable component of this inventory. Tests done for TFTR with 10keV molecular deuteron beams [58] show that the trapping efficiency in copper exceeds that estimated from diffusion models by ~ 2 orders of magnitude. The excess has been attributed to lattice damage by the beams [58], [59] and agrees with results from drive-in neutron targets. It has been estimated [59] that about 50 10^3 Curie would become trapped in the JET Ion Dumps in this manner which would limit the operational scenarios. The 'pumping' due to trapping of beam ions in the dumps is illustrated by the drop in tank pressure when a beam is extracted in the JET Neutral Beam Test Bed (see Fig.9).

4.3 Breakdown of HV Insulators

The breakdown characteristics of the TFTR and JET gridstack insulators have been tested in a tritium atmosphere at Sandia Laboratories and found to pose no problem [60].

4.4 Tritium Permeation

The tritium permeation through the beam dump structures into the water cooling channels appears to cause no problem. Falter [61] has estimated only ~ 1 mg (~ 10 Curie) of tritium permeating into the water in this manner. This estimate is based on diffusion models for hydrogen in solids however, and in view of the high surface concentrations expected due to lattice damage (section 4.2) it appears to be wise to test the permeation behaviour at high concentration gradients.

4.5 Beamline Vacuum and Water Leaks

The only types of vacuum leaks on the beamlines which need to be considered as tritium release hazards are those which result in a catastrophic venting to atmosphere. In practice such leaks have not occurred on either system which gives confidence for future operations. The use of tritium makes it necessary to consider even highly-improbable scenarios however, therefore the main weak links in the vacuum system have been identified and their integrity is being discussed with the appropriate regulatory authorities.

An *in vacuo* water leak in a beamline with tritium-loaded cryopumps would risk the production of large quantities of tritiated water when the cryopumps regenerated. The JET Exhaust Detritiation System (EDS) would safely handle such quantities [62] and active drains will be fitted to the beamline. However, the probability of water leaks must be kept as low as possible. In this respect the improved reliability of the beamlines gives confidence. There have been no in-pulse water leaks on the JET beamline since February 1988 and only 1 small out-of-pulse water leak since August 1988. Previous water leak problems have now been eliminated for all practical purposes by improvements in the mechanical engineering [44] and

physics [25] designs. Similar experiences have occurred on TFTR. Nevertheless, design studies are now in progress at JET on methods for early water-leak detection. Because the water resulting from small or even modest leaks is deposited as ice/snow inside the beamline, fast regeneration scenarios of the cryopumps are being studied which would enable condensed T_2 be recovered and hence minimise the production of tritiated water.

4.6 Upgrades to the JET Beamline for T° Operation

Operation with T° beams at 160keV should yield ~ 12MW of neutral power to the JET plasma with f(E):f(E/2):f(E/3) $\simeq$ 0.7:0.18:0.12. Upgraded designs for the beamline deflection magnet [56] and the drift duct protection [28] have been necessary to safely transport this power. The JET deflection magnet (designed for 160kV D^+) will have to operate at high currents to deflect 160 keV T^+. This would have resulted in excessive stray field in the neutraliser region due to saturation of the magnet iron yoke. This has been successfully solved by shimming the magnets and beam tests with 120keV ^4He beams to simulate the 160keV T, have shown the ion dump power profiles to be safely inside the dump envelope. There is evidence that the power profiles do not exactly match code predictions however, indicating that the beams have undergone space charge expansion [63].

The upgraded duct protection is designed with integral scrapers as a more robust unit with less in vacuo bellows and pipes. Although not designed for continuous operation, it will be capable of accepting peak reionisation power loadings ~ 5MW m^{-2} for pulse lengths of up to 14 seconds. The duct will be instrumented with a full array of thermocouples which will allow accurate validation of the reionisation power profiles calculated from the measured stray magnetic field [25].

5. PARAMETER RANGE FOR NEXT STEP INJECTION SYSTEMS

'Next Step' Tokamak devices such as ITER, NET and FER are physically-large systems in order to take advantage of the size-scaling of energy confinement seen in L-mode plasmas [64] and over a more limited database in the H-mode [65]. There is also an increasing body of experimental evidence to support the view that the 'effectiveness' of additional heating, in terms of the achievement of higher central plasma parameters, is improved if the power is deposited centrally [66]. Since it can be shown that in the plasma ion temperature range from 7 to 25 keV, the fusion yield (Q) of an ignited plasma increases with ~ $\gamma_n\gamma_T$ where γ_T is the temperature profile peaking factor ($T_i(o)/<T_i>$) [67], ignited plasmas would clearly be more easily achieved by the application of central ion heating. Highly penetrating NBI systems seem prime candidates for the task of 'starting up' an ignited machine.

For DC operation of a Tokamak, non-inductively driven current is necessary with a profile peaked towards the axis. For the high plasma densities ($n_e \sim 10^{20}$ m^{-3}) foreseen for a Next Step machine, Lower Hybrid Current Drive (LHCD) would be effectively excluded from the plasma centre [68]. Penetrating NBI remains as the only experimentally-validated method of axial non-inductive current drive. Current drive studies on present generation Tokamaks have experimentally validated the theoretical efficiencies of NBI current drive [69], [70] even though present NBI systems were not optimised to drive current. On DIIID, a plasma has also been sustained entirely non-inductively by NBI [71].

Highly penetrating neutral beams require high energies because the trapping cross-sections of ionisation and charge-exchange of neutral atoms decline rapidly with energy (see Fig.10(a)). In order to obtain a beam with a mean penetration distance similar to the minor radius of the ITER machine (2.15m) in a 10^{20} m^{-3} spatially-constant plasma ($\bar{\lambda}_{20}$) one must have deuterium neutral beams with energy ~ 1MeV (see Fig.10(b)). At such energies, the neutralisation efficiency for positive ions has dropped to infinitesimal values (<0.5%, see

Fig.10(c)). The neutralisation efficiency for negative ions in ordinary gas targets remains high at ~ 60% for energies above 1MeV (Fig.10(c)). Thus it is clear that injectors on the Next Step devices must feature negative ion beamlines. The exact energy of injection has quite a wide margin of uncertainty at the moment as simulation of current drive efficiencies for typical ITER conditions show that the predicted current drive efficiency varies by $< \pm 5\%$ from a value of over the beam energy 800keV to 3MeV [72].

6. NEGATIVE ION SOURCE AND BEAMLINE DEVELOPMENT

Studies of Next Step injectors show that in order for the dimensions of the plasma ion source to be kept to a reasonable size, sources capable of supplying negative ion current densities ~ 200-500 A m^{-2} are required. Such sources are being realised, at least in small-scale experiments. Two basic lines of source development are evident; the 'surface conversion' source where negative ions are produced by double-charge exchange of positive ions incident onto a surface having low work function and 'volume production' source where the negative ions are produced within a low density plasma.

This paper will not review in detail the development of negative ion sources. For this the reader is referred to the review by Alessi [73] for surface conversion sources and by Green [74] for volume production sources. Below we give only a brief overview of the present status.

Surface conversion sources for many years relied upon the introduction of Caesium vapour to produce the low work function surface. This produces problems with voltage holdoff in accelerator stacks, and because of the general undesirability of Caesium impurities in a system attached to a Tokamak, the sources declined in status when the negative ion content of volume sources was identified. Recently, successful experiments with a Barium cathode conversion surface [75] have revived the prospects for such sources although the sputtering

of Barium still seems to be an issue. In addition, as the conversion surface is biased at ~-300 to -400V, the negative ions are emitted from the surface with fairly high transverse energies (~10eV) causing problems for the design of the focussing optics for beam transport from such systems.

Volume production sources became serious contenders for production of negative ions with the observation by Bacal et al [76] of significant H^- densities in the low pressure discharges in magnetic multipole sources. The fact that these sources were becoming the standard for positive ion beams (see section 2.1) stimulated development [74]. The negative ions are thought [77], [78] to be produced in a 2-step process via an intermediate highly-excited molecular vibrational state viz:

$$e^- + H_2 \rightarrow H_2^*(6 \leq v \leq 14) + e^- \quad (3)$$

$$e^- + H_2^* \rightarrow H^+ + H^- + e \quad (4)$$

since electron energies > 10eV are most favourable for reaction (3) but low energy electrons (~1eV) are needed for reaction (4) to proceed we are thus faced with the familiar '2 temperature region' situation as in the positive ion sources (section 2.1) and indeed the insertion of a magnetic filter into the bucket source was shown to enhance the negative ion yield in the extraction region [79]. Recently the observation by Leung et al [80] that a 'one-off' puff of Cs vapour into a volume production source enhances the negative ion yield has been adopted by the Japanese JAERI group to enable extraction of 10A of H^- from a multi-aperture source (209 holes), corresponding to ~360 A m^{-2} [81]. Volume sources still have the poorly-understood problem that the yield in deuterium (D^-) plasmas is substantially reduced from the yield in hydrogen (H^-) plasmas, where most of the present experimental work is carried out. Also the maximum current density which can be extracted from the source reduces as the size of the extraction aperture increases an effect possibly due to stripping losses near the extraction grid [74].

6.1 Advantages of Negative Ion Beamlines

In addition to the high neutralisation efficiency (η_o) *per se*, negative ion beamlines have other advantages.

Arising from the high η_o, the gas flow required for negative ion beamlines are lower. This enables more conventional 'chevron-baffled' cryopumps to be envisaged and also reduces the power loading from neutral beam reionisation ($D_f^o + D_2 \rightarrow D_f^+ .....$) which must be handled from devious particles in the beamline. The spacecharge in a negative ion system, with its reduced current densities, is lower than in the equivalent positive ion system. This leads to reduced space charge expansion near the ion dumps and also to less heating of the gas in the neutraliser. This latter effect, which reduced the effective neutralisation target in positive ion systems and increased gas loads, was first identified by Paméla [82]. It has been seen in reduced neutralisation efficiency measurements found on many positive ion systems, eg. [83], [84]. As an effect which is directly related to beam current density, it should not play a significant role in negative ion systems.

Negative ion systems are also intrinsically resistant to space charge expansion. The spacecharge neutralisation for a negative ion beam, which depends on confinement within the beam envelope of slow positive ions produced in beam-gas collisions, is more easily achieved than in the equivalent positive ion beam where it must be achieved by confinement of the highly mobile electrons which are similarly produced. Negative ion beamlines with very low divergence are hence possible and 3mr. at 90keV/nucleon energy has been achieved. [85].

Negative ion beamlines can thus, if required, be very long. This has the advantages of;

- remote positioning from the Tokamak, reduction of the neutron and tritium problems;

- long regions of beamline are available for the dumping of residual ions at low power densities;
- long pumping regions are available, increasing further the conservative aspects of the pump designs.

6.2 Disadvantages of Negative Ion Beamlines

The disadvantages depend upon the type of source adopted.

Surface conversion sources have the disadvantages associated with the effect of the sputtered/condensed low work-function material (caesium or barium) on the voltage hold-off of the accelerator stacks.

Volume production sources tend to operate at relatively high pressure (~1.3Pa) when compared to positive ion equivalents. Stripping losses of the D^-beam ($D_f^- + D_2 \rightarrow D_f^o + ...$) can occur with high probability in the long accelerator stacks leading to beam-losses. Consequently the accelerator stack pumping has to be very efficient and a working demonstration of this has not yet been produced.

For all negative ion sources, electrons would be extracted in copious quantities unless suppressed ($n_e/n(H^-) \geq 40$ in a volume source). This would severely limit beamline efficiency. Fortunately electron suppression techniques have been developed, pioneered by Holmes et al [86]. Their system is shown in Fig.11. Electrons are first suppressed by a magnetic filter field using magnets inserted in the extraction electrode. Electrons which still leak through are trapped at low potential by an electron trap in the second grid. Thus only a fraction (f) of the full accelerating voltage is wasted on their acceleration. If the current of electrons leaking through I_e is given by $\delta \times I^-$ (I^- is the negative ion current) then the beamline efficiency is reduced by a factor

$$\frac{\varepsilon}{\varepsilon_o} = [1 - \delta f] \qquad (5)$$

for practical systems f cannot be lower than ~ 0.05 to 0.1 and thus the initial filter needs to ensure $\delta \leq 1$. This has been achieved on a working system [86].

6.3 Large Scale Negative Ion Beamline Demonstrations

Clearly, large scale demonstrations of working negative ion beamlines are desirable before proceeding to a Next Step device. In the next 5 years, two demonstrations are currently funded.

i) Within the European Community's (CEC) fusion program the 'DRAGON' facility at AEA Culham will come into operation at the beginning of 1991 [87]. This will have a 4A D^- volume production source, coupled to a 200kV pre-accelerator stack (4 grids). The design aims to produce 2.4 A equivalent of D° atoms (110A m^{-2} equivalent) from a 288 aperture system.

DRAGON will represent the first demonstration of D^- extraction and will demonstrate the following aspects of the CEC's ITER NBI proposal (see below)

- operation of a grounded plasma source
- operation of a neutraliser at high voltage with segmented construction
- operation of an electrostatic beam dump
- operation of the 200kV pre-accelerator
- the predicted stripping losses for the CEC D^- accelerator.

ii) Within the Japanese fusion program, the negative ion development is aimed at the realisation of a negative ion injector for JT-60U. This system will culminate in a 500keV accelerator feeding a 10MW (neutrals), 10 second pulselength beamline [88], shown in Fig.12. Notable features are the cryopumps at the accelerator stack position and the extremely long (~10m) gas neutraliser. The beamline is

conservative, using a source at HV potential and magnetic deflection for the ions. The dumping field is feedback-controlled in strength depending on the Tokamak stray field but the source and neutraliser are shielded passively. The system aims to demonstrate 2MA NBI current drive.

This system will test many of the features of the Japanese ITER NBI proposal (see below). In its original conception it does not rely on the enhancement in negative ion yield coming from caesium vapour puffing. An intermediate step system (350keV, 0.1A, 9 apertures) will test the H^- yield enhancement from Cs vapour on a large system and is due to come into operation at the end of 1991. The full JT-60U negative ion system is foreseen for 1995 operation.

7. NBI PROPOSALS FOR ITER

Within the context of the ITER proposals [89], the Technology phase foresees a steady state Tokamak with a 2 week burn period. This machine would have a plasma current up to 11 MA and an installed current drive capability of up to 150MW. Approximately 50% of the current drive capability would come from NBI.

The common parameters for the NBI system for ITER are given in Table 1. Of particular note are the requirements for the system to operate at 50% and 75% of full energy during the start-up phase of the plasma, and the ambitious requirement to have a vertical power profile control at the plasma so that control of the current profile in the Tokamak could be achieved.

On the basis of this parameter choice, 4 design proposals for the NBI system have been received from the CEC [90],Japan [91], USA [92] and USSR [93]. At the overview level all the proposals are essentially similar and an overview drawing of the Japanese proposal is shown in Fig.13.

7.1 Issues in the ITER NBI proposals

In general, most of the contentious issues in the proposals involve the source/extraction and accelerator stack section of the beamline. A reflection of the uncertainties is given by the fact that the 4 proposals differ radically in this area and each has unproven features.

All designs feature electrostatic d.c. accelerators and the problems of avoiding electrical breakdowns in these structures must be addressed. A particular problem for negative ion systems is, as already indicated, the acceleration of larger quantities of electrons than in the positive ion case. The impact of these electrons on grid structures and the resultant X-ray and UV production with its impact on further ionisation in the grid volume poses severe challenges. The designs are probably approximately 2 orders of magnitude from previous successful demonstration in terms of their specific X-ray and U.V. production.

For the CEC concept [90] which proposes a high pressure (0.9 Pa) volume production filter source, the stripping losses and electron acceleration problems are postulated to be minimised by the operation of the source at ground potential, with the accelerator stack and neutraliser being at increasingly positive voltages. Operating a source at ground will enable stronger pumping in this area as pumping access can be achieved up at relatively high pressures without the risk of long path breakdowns. The operation of the source at ground potential also simplifies source replacement enormously. (The source is identified in all the proposals as the item for most frequent replacement). As already stated, the DRAGON facility will test the source/extraction to 200kV and the concept of the HV potential neutraliser including the 'onion skin' type shielding around this to prevent acceleration of field-emission electrons from the beamline walls towards the neutraliser [87].

The Japanese concept [91] proposes a 'Cs-seeded' volume production source operating at HV potential with a gas neutraliser at ground potential. The Cs is proposed to enable source operation at lower pressures (~0.5 Pa) and hence reduce the gas load stripping and x-ray production problems. The voltage hold-off in the presence of caesium and the long term behaviour of the caesium D^- enhancement are the major issues. Unlike the EC concept, which accelerates beamlets in rows to keep an open structure for pumping, the Japanese concept forms a sheet beam after the 4 grid extraction has acce;erated the ions to ~ 180kV. Plasma may form in the centre of this sheet beam and increase the space charge effects with deterioration of beam optics [94]. As already stated, a successful demonstration of the JT-60U negative ion system would answer many of the uncertainties about this concept.

The US concept [92] proposes to use a Barium converter, surface conversion source based on the work at FOM [75] now being pursued further at LBL. The accelerator in the US concept differs from the others in being an Electrostatic Quadrupole (ESQ) accelerator [95]. The long term behaviour of the Barium in the converter and the handling of the large quantities of sputtered barium (over 1 kg. in the 2 week pulse) and its effects on the extraction system voltage holdoff are clearly issues here. The optics problems arising from the initial high transverse negative ion temperatures (section 6) also need to be overcome. The ESQ is claimed to have strong advantages in preventing breakdowns. The ESQ transverse focussing fields offer the possibility of steering electrons born in the accelerating stack off to the side of the assembly. X-ray production and breakdown problems should thus be reduced. The transverse focussing enables longer acceleration stacks thus reducing the voltage gradient and the probability of breakdown. It is essential for this concept that a demonstration of ESQ acceleration of currents ~1A is performed. At present 45mA of H^- has been accelerated to 200keV. [96].

The USSR concept [93] proposes a 'ring-extraction' hollow-cathode Cs-assisted source. Source uniformity and its effects on optics are the key issues here in addition to all the usual problems associated with caesium.

7.2 General Comments on the ITER Beamline Design

In many aspects the ITER beamline designs are all conservative and this gives confidence in the practicability of realisation.

In comparison to the equivalent '10MW module' of the JET NBI system, a lower quantity of gas (about ⅓) is required leading to a lower installed pumping requirement (~1 - 1.2 10^3 m^3 sec^{-1}) and the adoption of conventional chevron baffled cryopumps. The ion dump power densities are also much lower. A quantity of ion power ~ 45% of that dumped on the 80keV JET beamline is spread over a much larger area. This increases the safety margins against burnout and reduces the erosion and surface damage problems on these dumps which will help to offset the problems inherent in the long pulse operation. The drift duct problems also appear to be less severe. On the JET beamline 10.8 MW of neutral power is transmitted through a 0.22 m^2 aperture. On ITER the equivalent figures are ~ 32.4MW transmitted through 2.72 m^2.. An approximate evaluation of the beamline parameters shows Γ_{crit} ~ 13 - 14 for ITER compared to ~ 4 for JET. Reionisation power loading and duct conditioning should therefore be less of a problem.

Reionisation focussing studies similar to those carried out on JET by de Esch et al [25], have not yet been performed for the ITER design. This seems an important and timely topic now that the design is more firm.

The Tokamak stray field from the 22MA, air cored ITER will be very high at the sources/neutralisers (70-120 10^{-4}T) and has been recognised as a problem. Heavy double-layer passive magnetic shielding is foreseen in all the design proposals. In view of the success of feedback-controlled active

compensation on JT-60 and JET (section 2.6.2) it seems advisable to review its use in the ITER beamline. Savings in weight, cost and radioactive material inventory can be identified and the active compensation systems are more easily upgradable and changeable to meet parameter extensions of the system.

8. FURTHER DEVELOPMENTS FOR NEXT STEP INJECTORS

Two developments which lack comprehensive funding programs but which appear to offer significant technical advantages are those of the plasma neutraliser and the RF excited multipole source.

8.1 Plasma Neutralisers

It is well known that the charge stripping interaction between negative hydrogen and positive ions (A^{+}) in a plasma ($H^- + A^{n+} \rightarrow H^\circ + A^{(n-1)+}$) proceeds with a high probability. Recent measurements [97] have shown that the cross-section enhancement coming from having multiple-charged ions from high-Z elements is not as high as previously thought, scaling approximately as $Z^{1.3}$ instead of Z^2. Thus it now appears that plasma neutraliser cells which concentrate on the efficient production of dense hydrogenic ions are competitive with the previous heavier gas designs. Hydrogenic neutralisers are also more compatible with Tokamak NBI systems.

Plasma neutralisers using deuterium can result in a substantial increase in the conversion probability of D^- to D° above the gas-neutraliser case from ~ 60% to about 85% as shown in Fig.14. The gas load is also substantially reduced in these devices.

Given these advantages, the demonstration of a working full-scale plasma neutraliser would seem a high priority. Moses and Trow [98] have

experimented successfully with r.f. driven Argon plasma neutralisers attaining $n_e \sim 5\ 10^{19}\ m^{-3}$ over a 0.5m length in a 50kW discharge. There is now growing evidence that r.f. excited multipole sources can obtain very high efficiencies (see section 8.2). Using this Holmes et al [90] have proposed an r.f. driven plasma neutraliser with a cusp containment field. A schematic of this proposal is shown in Fig.15. The containment is nearly completely 6-sided as the D^-/D° beams are able to pass through the rows of containment magnets at the neutraliser entrance and exit. Using the figures from the RF buckets, Holmes et al calculate only ~ 0.2MW is needed to drive such a neutraliser at nearly 95% fractional ionisation.

Such a neutraliser inserted in the EC ITER concept would increase the overall beamline efficiency from an estimated 45% to 66%. The increased efficiency could either be used to decrease cost (omit one of the beamlines) or increase safety margins and decrease power supply provision and would be more than enough to compensate the lost current drive efficiency if the accelerator voltage had to drop below (say) 1MV because of breakdown considerations.

8.2 RF Excited Plasma Sources

One major problem in long-pulse injectors would be filament lifetime and it would clearly be disadvantageous to keep replacing ion sources regularly in the remote-handleable conditions of the ITER beamline. Also the de-excitation reactions which are thought to destroy negative ions in volume production sources are enhanced [77], [78], by the presence of dissociated atomic neutrals which produce associative detachment of the negative ions:

$$H^- + H \rightarrow 2H + e \qquad (6)$$

Wall materials which enhance the recombination of neutrals to molecules are hence desirable. Production of the vibrationally excited molecules from which negative ions can be formed (reaction (3)) are also

thought to be enhanced by differing wall materials. The problem with the conventional tungsten-filamented volume source is that after a period of operation the walls are covered with Tungsten and surface optimisation cannot be pursued.

These problems would be overcome by the successful large-scale development of r.f.-excited multipole sources. Successful r.f.-excited, multi-pole containment sources have been run with very high efficiencies obtaining positive ion values as high as ~4 A/kW in the case of Di Vergilio et al [99] using a multipole source with internal r.f. antenna. Successful positive ion rf sources have also been developed at IPP Garching [100] where recently a JET PINI source has been converted to r.f. excitation and a factor 4 efficiency enhancement over the filamented version has been achieved [101].

With this background a clear case exists for substantial funding being channelled into r.f. excitation sources to demonstrate their efficiency as negative ion generators. Work is already under way at AEA Culham Laboratory on small scale demonstration projects with this aim [102].

9. CONCLUSIONS

Positive ion NBI systems now represent a mature and robust technology which has provided an invaluable tool in the progress of research of Fusion Physics in recent years.

The operation of high power NBI systems (> 20MW) composed of high power beamline units (~ 10MW) has shown that the design issues surrounding the realisation of such beamlines can now be considered solved. With current technology, it is possible to design a working d.c. beamline at these high power levels. Indeed it is only the non-d.c. aspects of the JET drift duct which prevent the most powerful extant beamline unit (the JET system capable of delivering

10.4MW) from being such a beamline. The 'engineering-coupling' problems of interfacing such beamlines to Tokamak operations are also now solved in essence.

The upgrade of NBI systems to inject helium and tritium beams is well advanced maintaining the importance of the fuelling aspects of current systems. Helium pumping by cryosorption has been successfully realised on large systems for long pulses. The design issues for tritium beams have been addressed, in the main, and solutions established.

The success of positive ion beamline realisation gives confidence in proceeding to the Next Step where negative ion systems will be required because of the high beam energies needed to penetrate large Tokamak plasmas. With their high neutralisation efficiency, low spacecharge and inherent resistance to spacecharge expansion, negative ion systems can be remote from the Tokamak. This gives advantages in radioactive inventory and also allows dumped power densities to be low. Pumping requirements are also lower than for the positive ion beamlines. Such conservative tendencies in the design enhance the prospects of success.

Negative ion sources are being developed to the stage where they can yield current densities in the range 200-500 A m^{-2}, suitable for the Next Step device. Demonstration of large scale extraction system remains to be performed however and stripping losses in extraction systems could be high.

The most serious design issues to be resolved in the 4 proposals for the ITER NBI system are all in the area of the ion source/extraction system/accelerator stack. All 4 proposals have potential major problems here. Demonstrations on a large scale are in prospect for most of the crucial design concepts in the CEC and the Japanese proposals.

A considerable upgrade in the overall efficiency of a negative ion beamline (from ~45% to ~66%) could be achieved by the successful realisation of large

plasma neutralisers. A similar comprehensive program should also be instigated to develop r.f. excited multipole sources for negative ion production.

Finally, one should observe that, in the space of a review article, it is not possible to indicate all the pitfalls which have been overcome on the route to the success of the positive ion systems. In all cases, in retrospect it has been shown that the building of Test Bed units, of the maximum possible realism, is invaluable. In this way the 'surprises' of the positive ion physics program (e.g. spacecharge effects in multiple beams, neutraliser gas heating effects) have been identified and accommodated. The engineering coupling problems (e.g. stray field on JT-60) have also been solved. It would be very unwise to economise in the provision of a full beamline Test Bed unit for ITER, in spite of confidence in the outcome!

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Table 1

- Common parameters for the 1990 ITER NBI system.

INJECTED POWER (D°)	:	75MW
ENERGY (MAXIMUM)	:	1.3MeV
PULSE LENGTH (MAXIMUM)	:	350 hrs
OVERALL SYSTEM EFFICIENCY	:	> 40%
ENERGY OPERATION POINTS	:	50%, 75%, 100% FULLENERGY
ENERGY STEP CHANGE RATE	:	10 secs.
POWER PROFILE CONTROL AT PLASMA*	:	HOLLOW, FLAT or PEAKED
POWER STEPS IN PROFILE CONTROL	:	10%
POWER CHANGE RATE	:	10MW S^{-1}
AIMING TANGENCY RADIUS	:	6.2m (cf. R_o = 6.0m)
PORT DIMENSIONS	:	0.8m(W) x 3.5m(H)
TRITIUM ACCUMULATION (TOTAL)	:	< 10 grams
NO. OF PORTS 3		BEAMLINES MODULES/PORT : 3
INJECTED POWER/MODULE	:	10 MW @ 1.3 MeV
BEAMLINE DIMENSIONS	:	4M (OD) x 15M (L)

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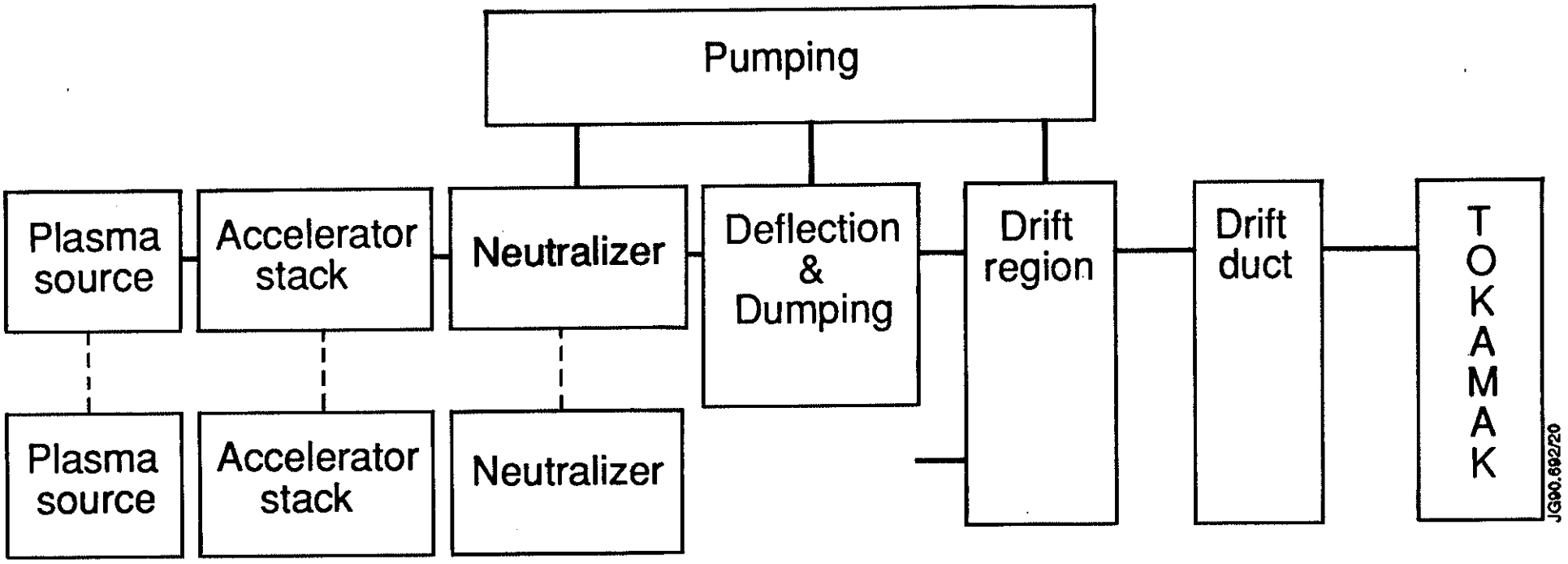


Fig.1 Block Diagram of the Elements of a Neutral Injector.

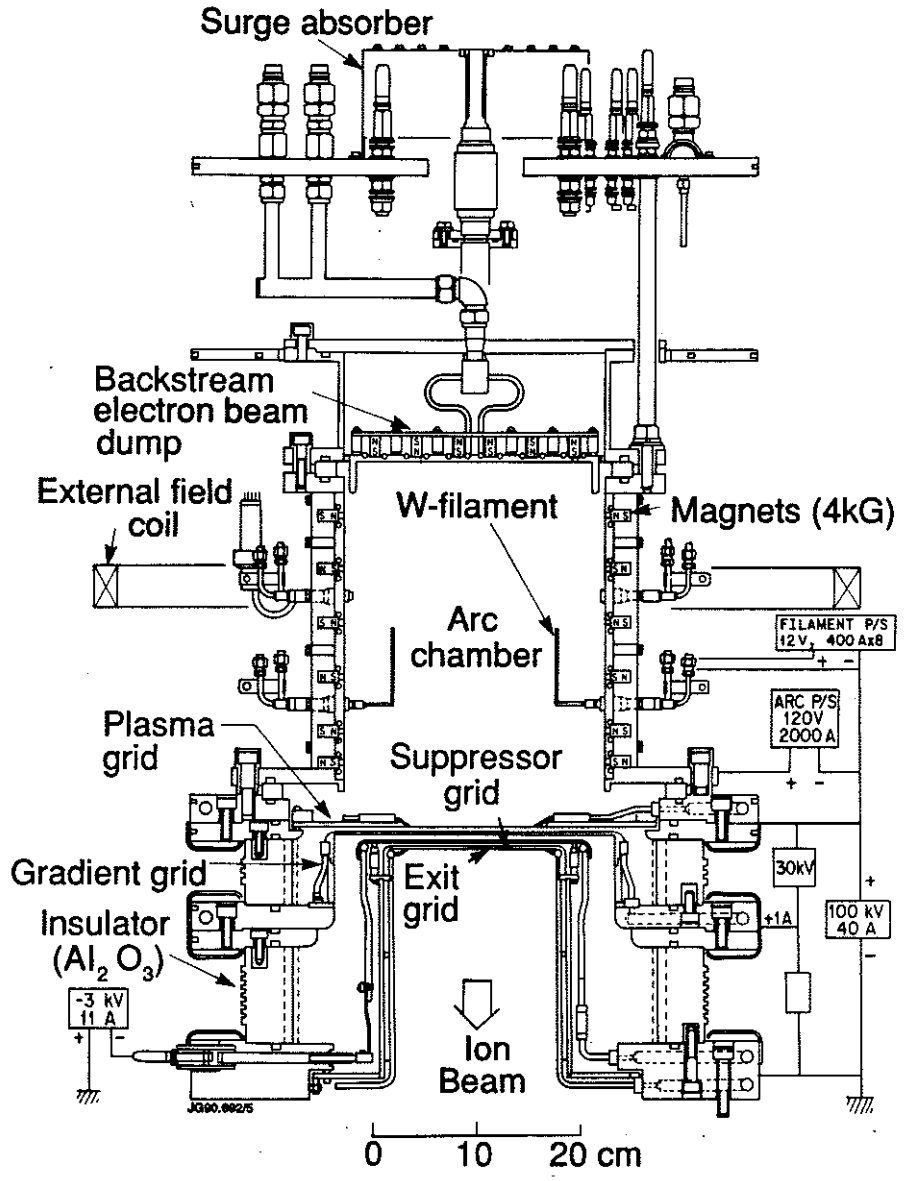
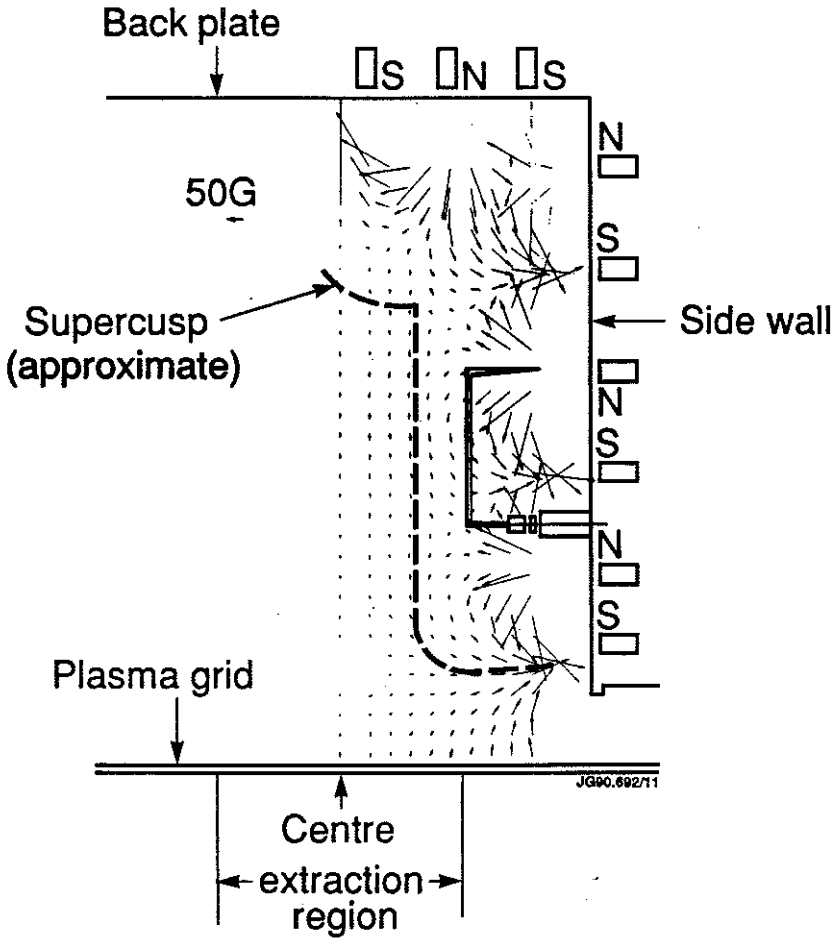


Fig.2 (a) Engineering drawing of the Magnetic Multipole source and extraction system for JT-60 NBI. The source illuminates an extraction area 0.12x0.27 m^2 and the accelerator unit is capable of delivery 70A at 75keV when operating in hydrogen (from ref [10]).



(b) Magnetic field distribution plot in a plane of the JT-60 magnetic multipole source, measured ***without*** the filaments energised (taken from ref.[13]). Also shown is the approximate position of the 'supercusp' field relative to the filaments.

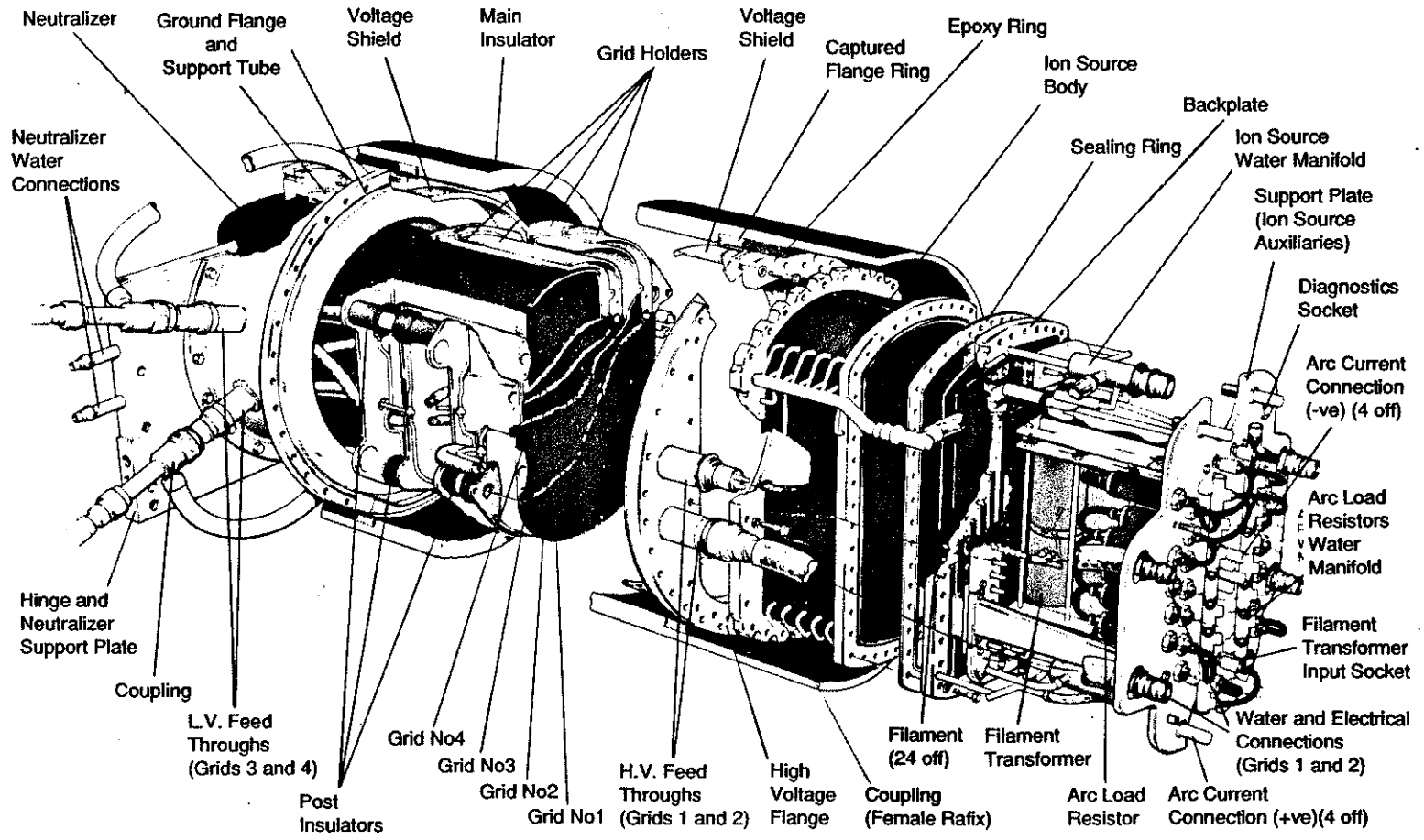
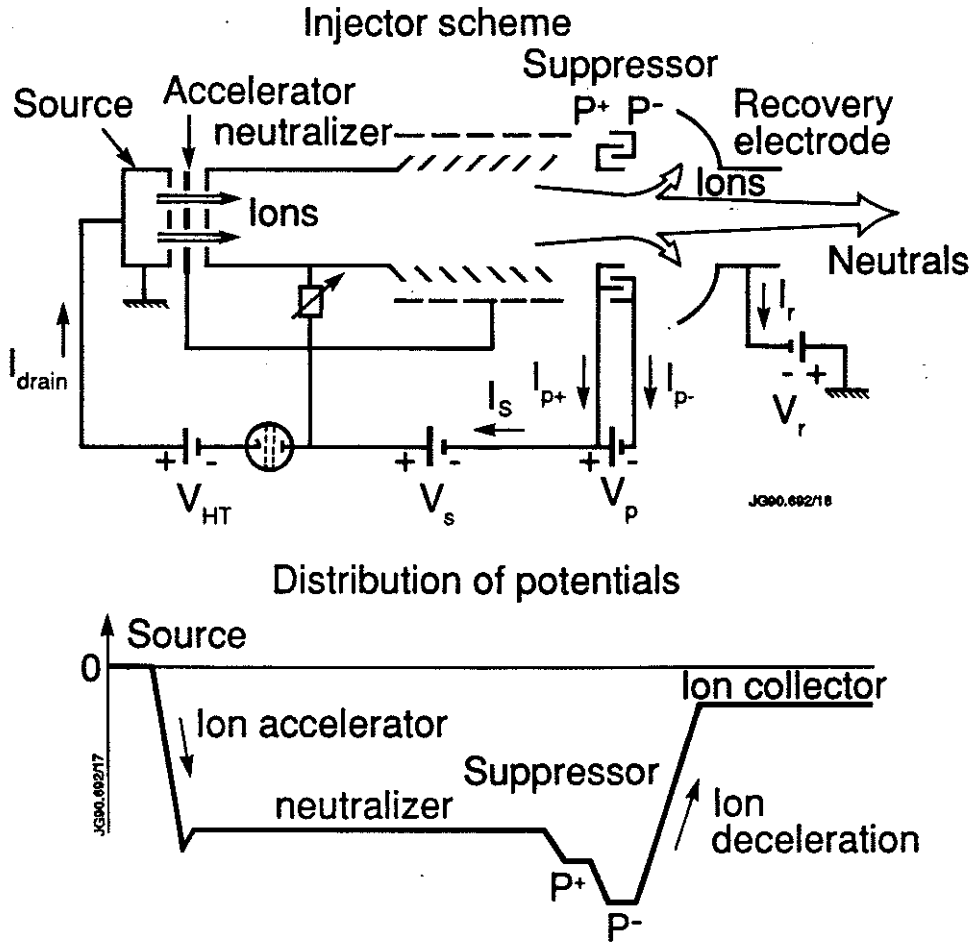


Fig.3 Cutaway view of the JET Positive Ion Neutral Injector (PINI) in its 4 grid 80kV form.
This 4 grid unit is capable of operation at currents of 60A (Hydrogen) or 42A (Deuterium) from an extraction area of 0.18x0.45 m^2. The higher voltage version has a revised 3 grid accelerator stack and has currently been operated up to 160kV, 37A in deuterium.



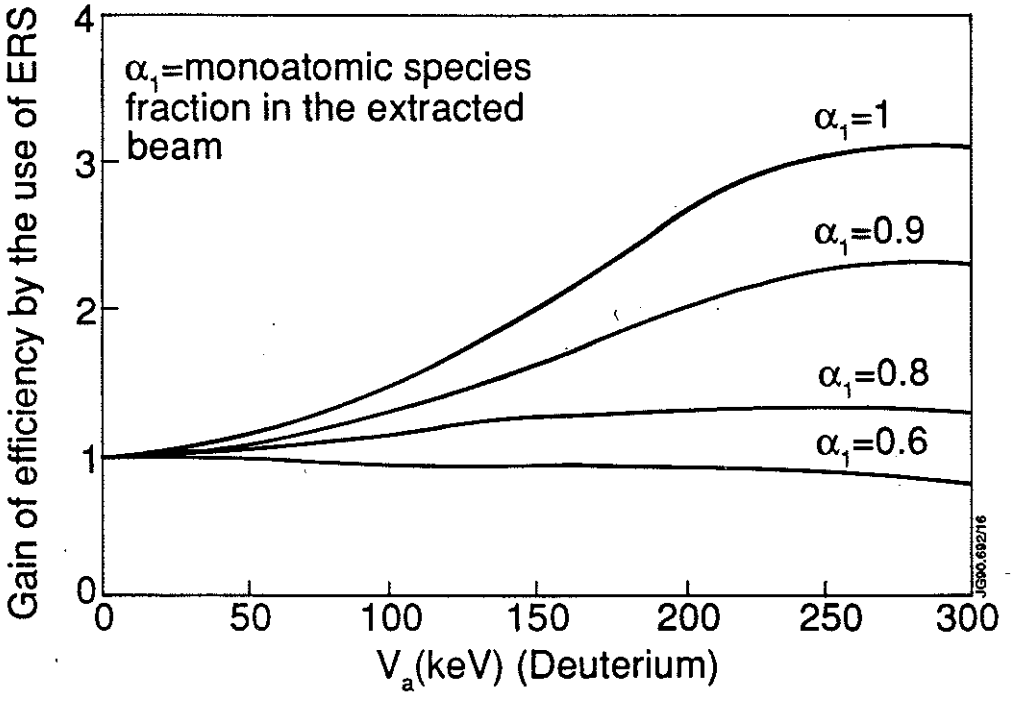


Fig.4(a) Schematic of the TORE-SUPRA injector including the Electrostatic Energy Recovery Electrode. Also shown is the distribution of potentials along the beamline (taken from ref.[19[).

(b) Calculated gain in energy efficiency using the Energy Recovery Electrode for a positive ion-derived neutral deuterium beam. The efficiency is plotted for different monatomic beam fractions (α_1) from the ion source assuming a neutraliser gas target of 10^{20} m^{-2}.

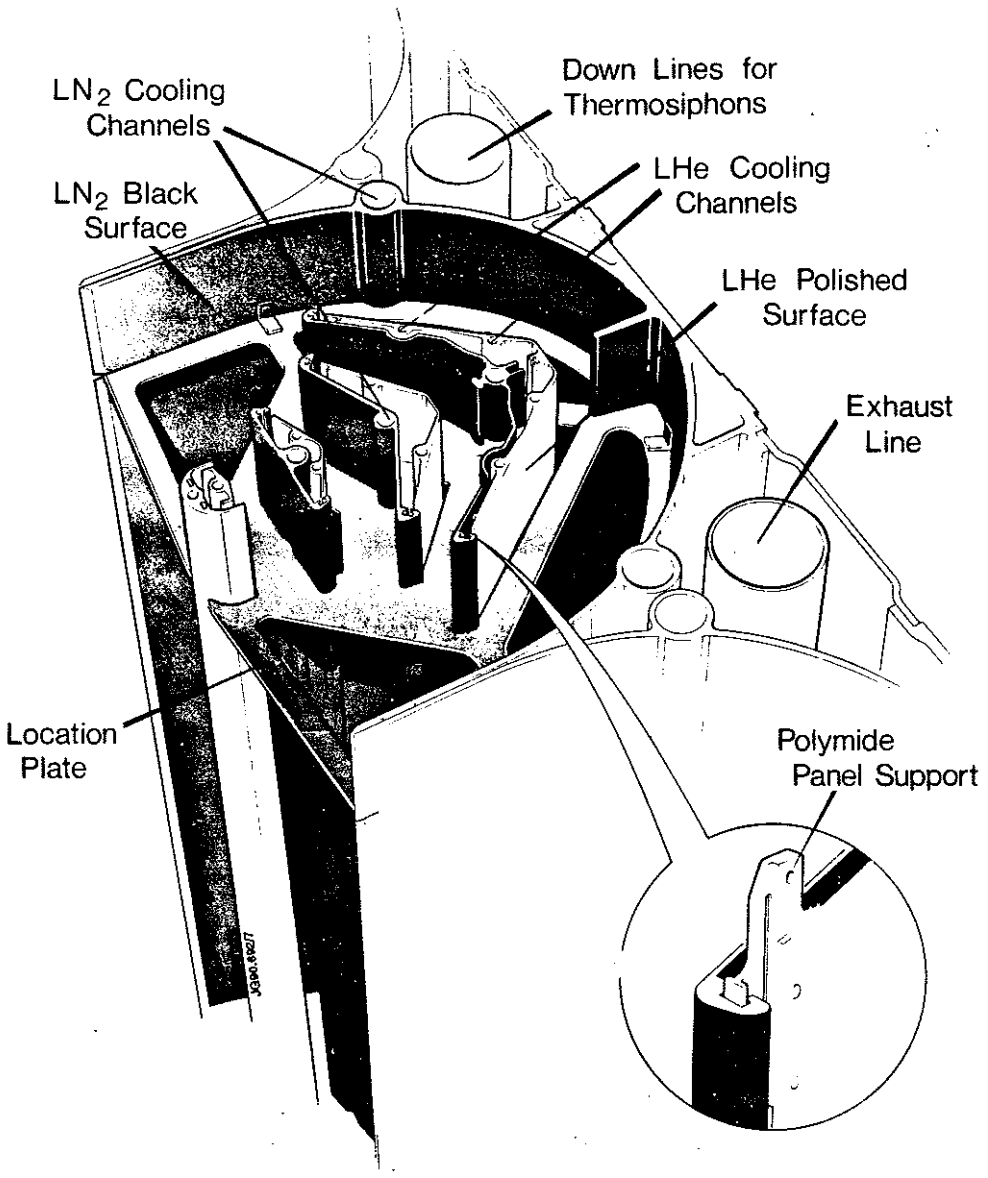
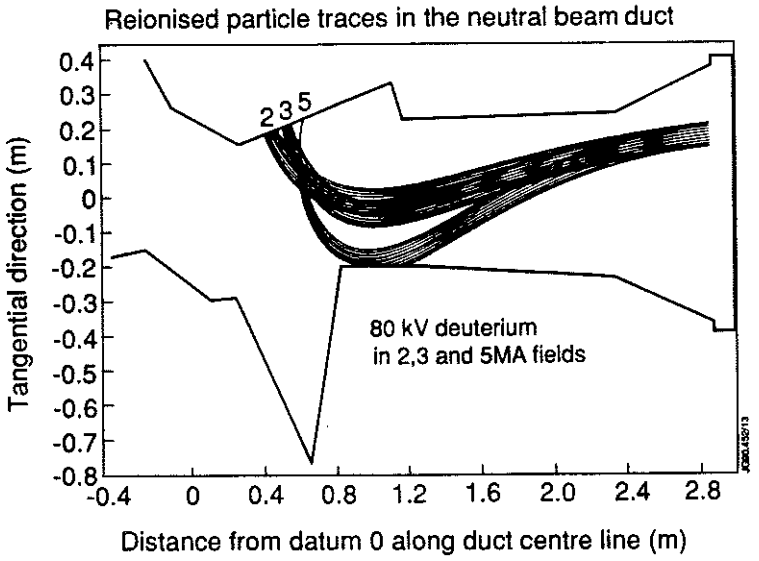


Fig.5 Detail of the JET Open Structure Cryopump module. The JET beamline has 10 such modules lining each side wall of the Neutral Injector Box.



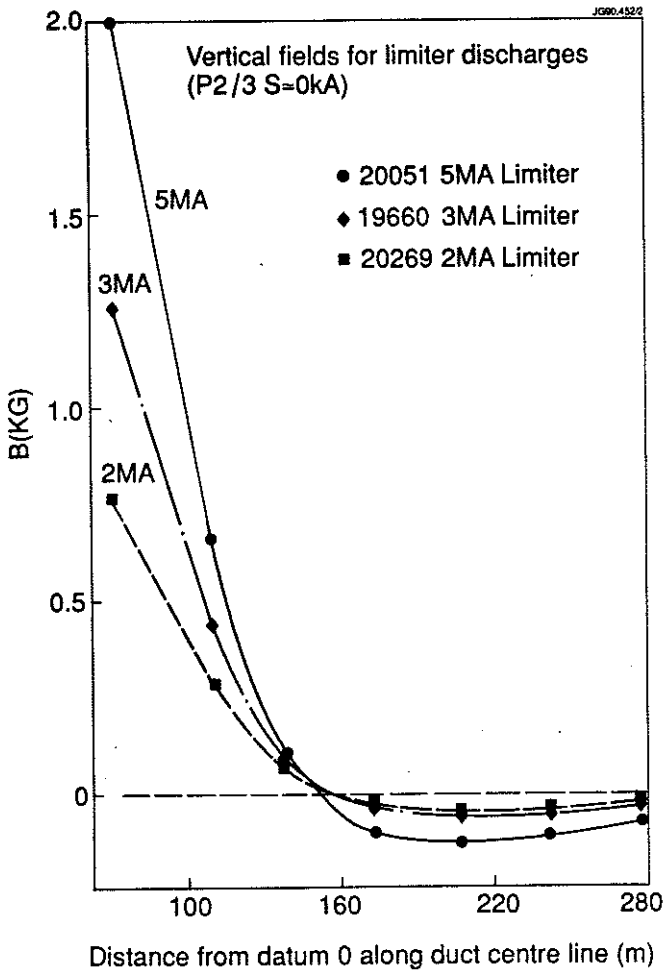


Fig.6(a) Reionised particle trajectories from 1 PINI simulated in the JET drift duct (plan view). The simulation has been done for field maps for 2,3 and 5MA JET plasmas (Fields as in 6(b))

(b) Measured vertical fringe fields in the JET drift duct for JET limiter plasmas of various currents. The vertical fringe fields arise mainly from the stray field of the vertical position control coils of the JET poloidal field system. Extra fields come into play when the poloidal shaping coils (P2/P3) are also energised.

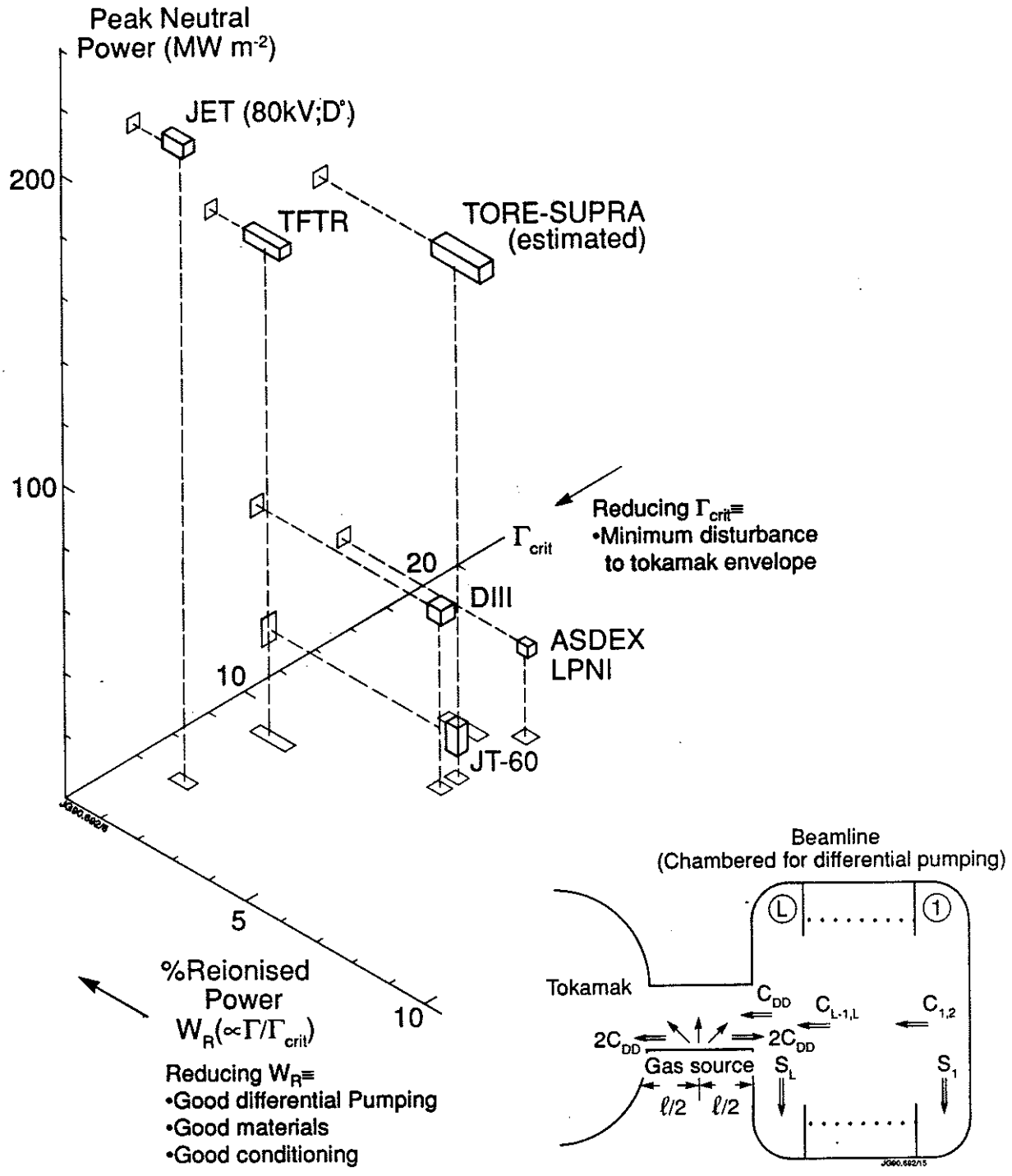


Fig.7 3-Dimensional plot showing comparative performance of different NBI Drift Duct designs for various Tokamak NBI systems. The positions of the ducts in the 3-D space are represented precisely. In reality for all systems the error bars on a particular co-ordinate are of the order ± 10-20% of the given value. Duct conductances used in the determination of Γ_{crit} (via equation (1) of the text) are multipled by 4 from the measured values on the approximation that the reionised gas source is produced mainly at the longitudinal ***centre*** of the duct and hence, can be pumped out by the 2 halves of the duct acting in parallel (see inset).

As described in the text, a low gas pressure in the duct (p_g) can be achieved by maximising the ratio of the last drift chamber pumping speed (S_L) to the duct conductance (C_{DD}).

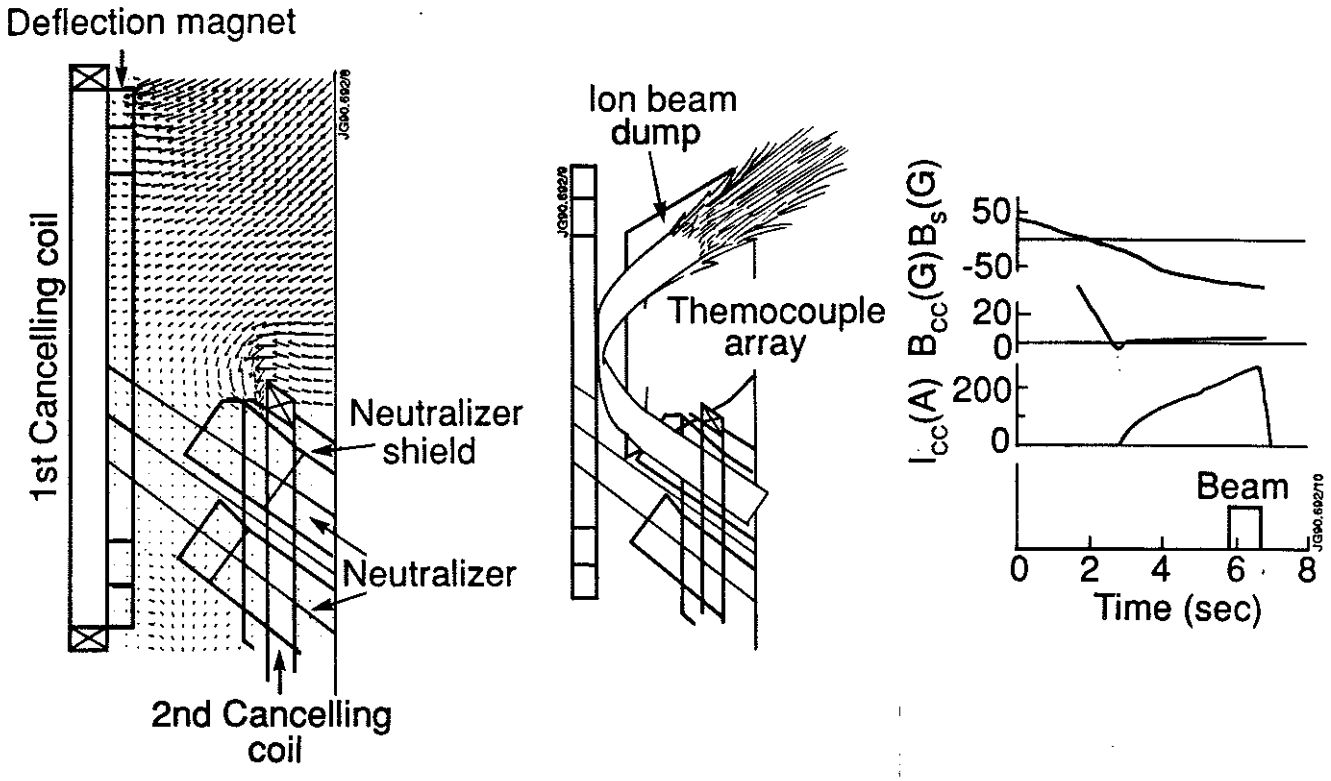


Fig.8(a) Schematic of JT-60 NBI beamline showing a magnetic field vector plot with the cancellation coils energised. The large volume of zero field is evident (from ref.[32]).

(b) As Fig.8(a) but indicating the ion beam transport envelopes and their relationship to the cancelled field volume.

(c) Waveforms showing the cancelled field (B_{cc}) and the current in the cancellation coils (I_{cc}) during a JT-60 discharge with NBI. The stray field in the absence of cancellation (B_s) is also shown.

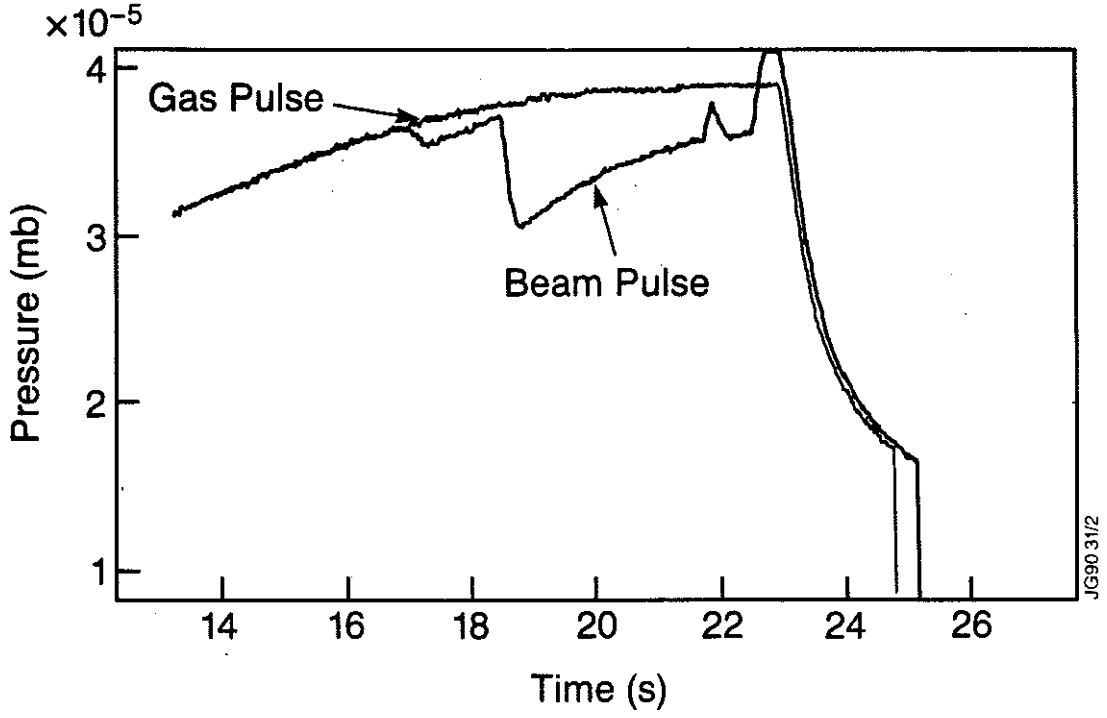


Fig.9 Measured pressure variation in the JET Test Beam Line showing the effect during a beam pulse. The reduction in pressure indicates 'pumping' of the plasma source and by beam trapping in the ion dumps.

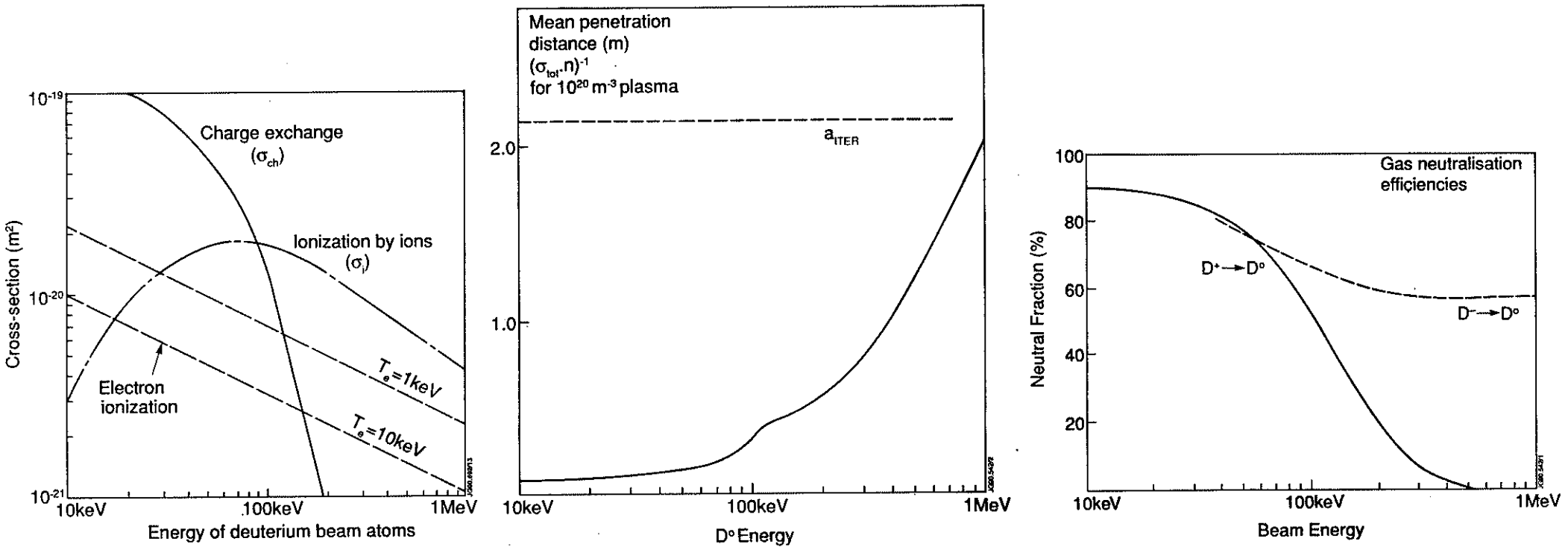


Fig.10(a) Cross sections for charge exchange (σ_{ch}) and ionisation (σ_i) of deuterium beam neutrals by plasma ions and the effective cross section $\langle\sigma_{ei}\upsilon\rangle$ for ionisation by electron impact in a Maxwellian plasma of varying temperatures (T_e).

(b) Mean penetration distance ($\lambda = (\sigma_{tot} n_e)^{-1}$) for a deuterium neutral in a spatially constant plasma with density (n_e) = 10^{20} m^{-3} and Maxwellian average electron temperature = 10keV. a_{ITER} indicates the minor radius of the proposed ITER Tokamak.

(c) Neutral fractions for positive (D^+) and negative (D^-) beams as a function of ion beam energy emerging from:

- an infinite gas target (D^+)
- the finite optimum gas target (D^-)

Note the optimum gas target for D^- varies from ~0.4 10^{20} m^{-2} at 160keV to ~ 2 10^{20} m^{-2} at 1MeV.

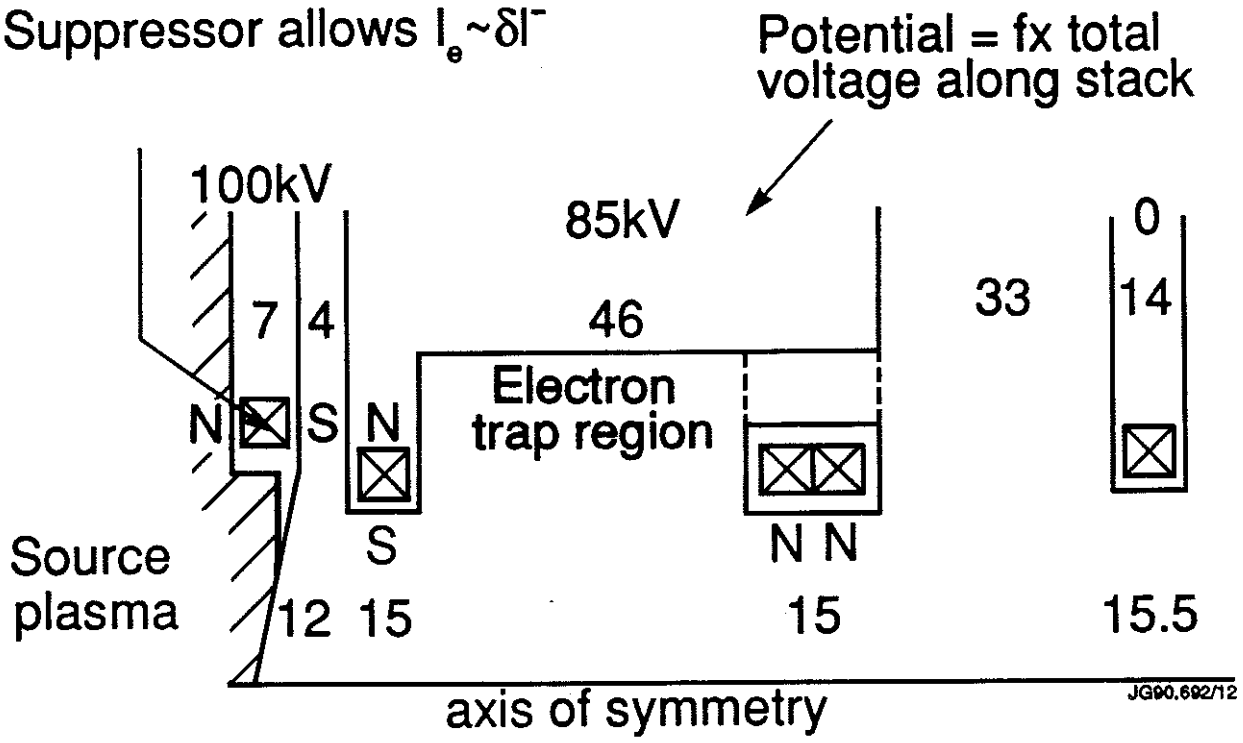
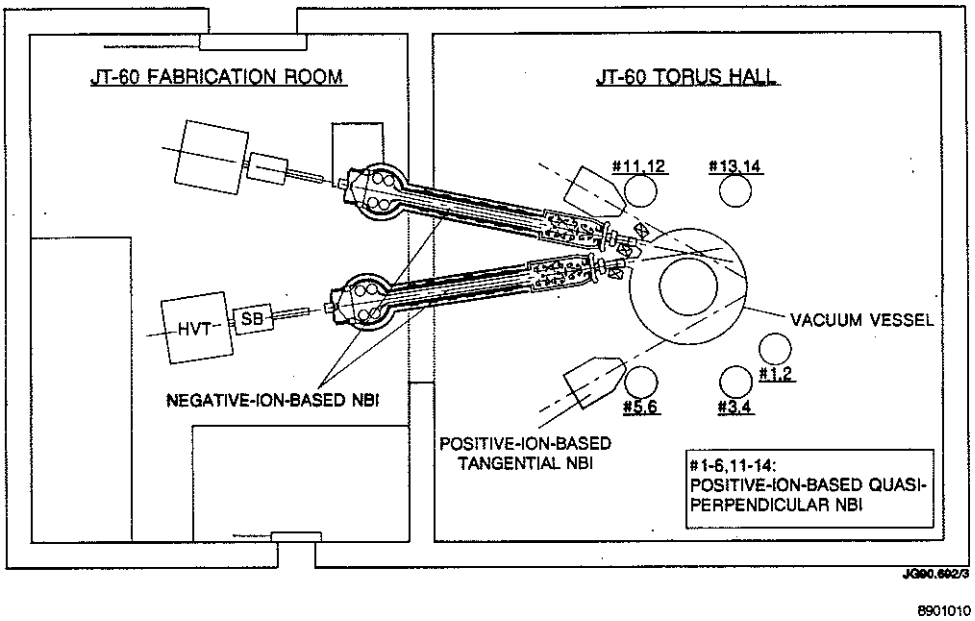


Fig.11 Schematic of the extraction system for the prototype negative ion accelerator of Holmes et al. used for development of electron suppression and electron trap concepts [86] (Dimensions in mm).



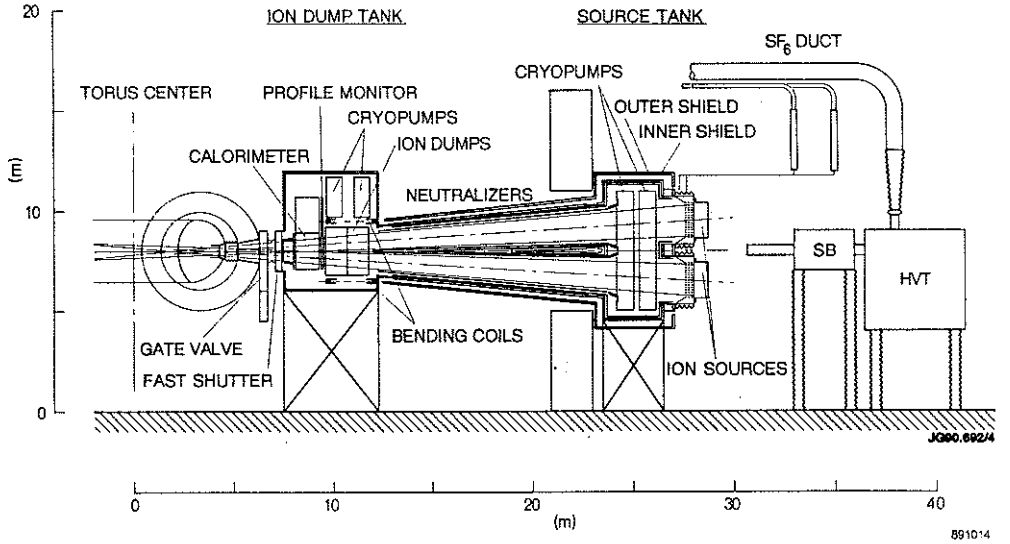
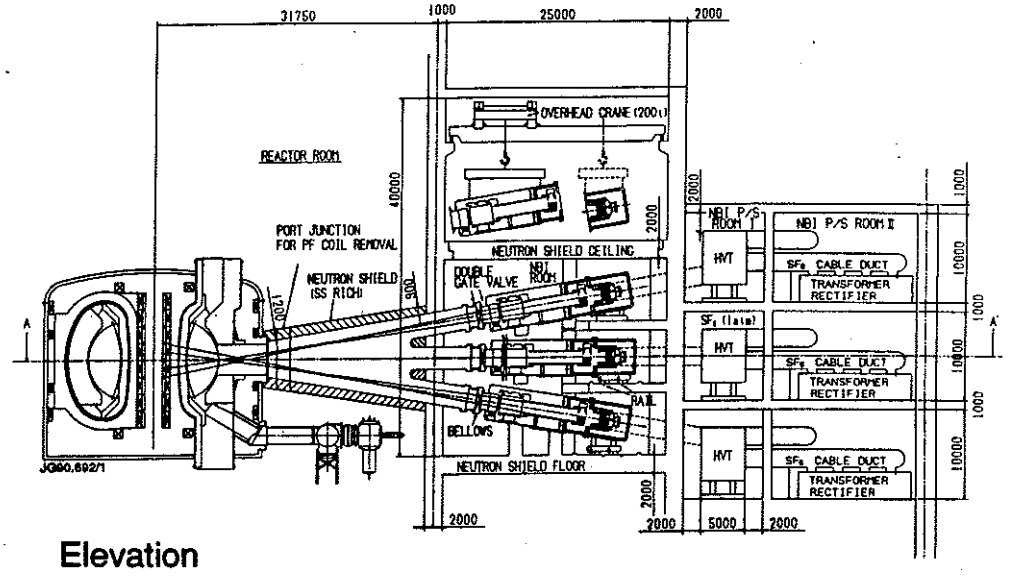


Fig.12 Overview of JT-60U negative ion system.
(a) Plan view indicating the system's position relative the Positive Ion injectors on JT-60. (b) Elevation view.



Elevation

VIEW OF SECTION B-B'

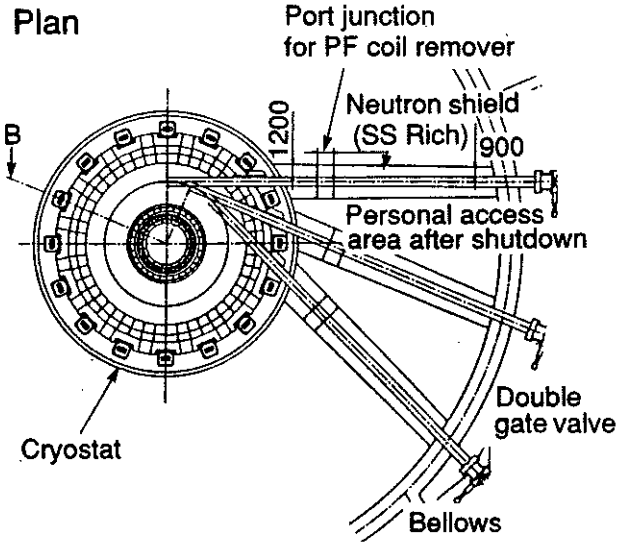


Fig.13 Overview of the Japanese ITER NBI proposal showing the relationship to the ITER plasma (Ohara et al. ITER-IL-HD-4-0-11).

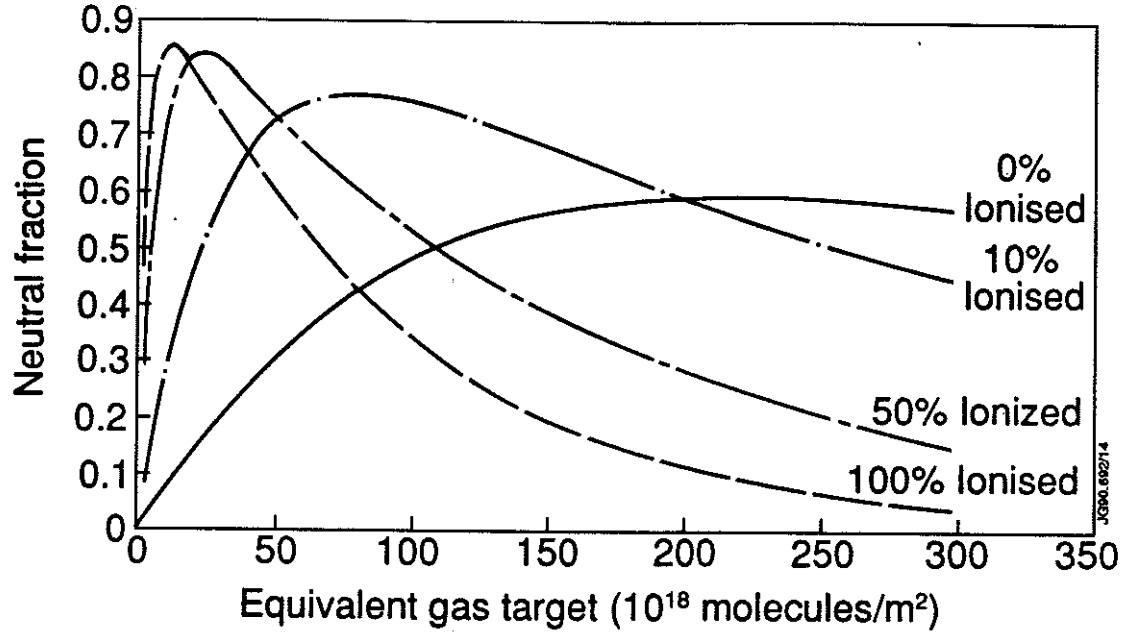


Fig.14 Neutral fraction obtained from a D^- beam at 1.3MeV as a function of equivalent gas target of a plasma neutraliser. The curves are drawn for each value of the percentage ionisation in the neutraliser (0% corresponds to a conventional gas neutraliser). At high equivalent gas targets for high ionisation percentages the neutral fraction falls rapidly due to D° reionisation to form D^+.

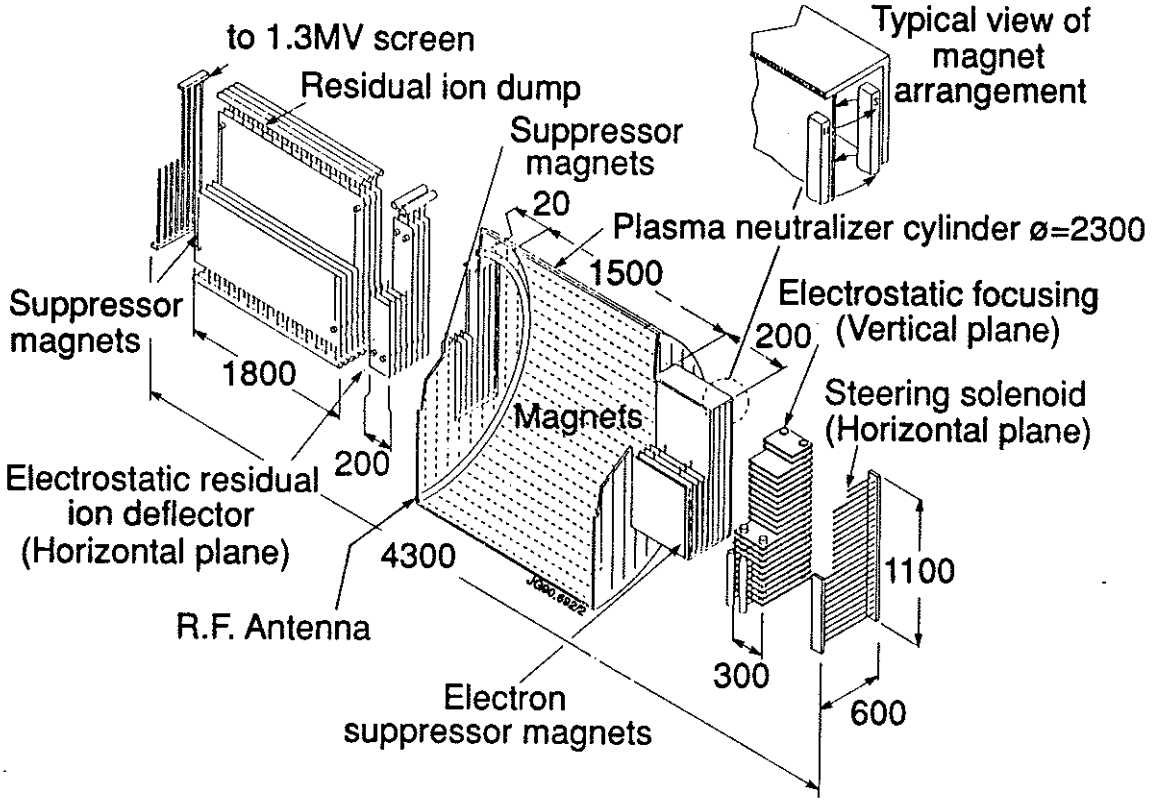


Fig.15 Schematic of r.f. excited plasma neutraliser with cusp-field containment as proposed by Holmes et al for the CEC ITER NBI beamline [90] (Dimensions in mm).

APPENDIX 1.

THE JET TEAM

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