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Proceedings of the Plasma Heating Requirements Workshop

Held at the Holiday Inn, Gaithersburg, Md.
December 5,6 and 7, 1977

U.S. Department of Energy

Office of Fusion Energy

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Proceedings of the Plasma Heating Requirements Workshop

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Edited by H. Stanley Staten
Office of Fusion Energy

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FOREWORD

On December 5, 6, and 7, 1977, a meeting was held at the Holiday Inn, Gaithersburg, Maryland, to discuss the anticipated plasma heating requirements and the corresponding development program needs for existing and future magnetic fusion confinement facilities. The goal was to reach a consensus on the technology development requirements and priorities.

In the past, the Magnetic Fusion Energy program has been characterized by efforts to confine plasmas and understand the basic plasma properties inherent in the particular confinement approach. This effort has been very successful in the Tokamak, Mirror and Bumpy Torus programs; enough so that the emphasis has shifted from simply providing good plasma confinement to heating the confined plasmas to reactor-like conditions. In this new role, the Magnetic Fusion Energy program pace has been limited by the heating technology available or being developed. Neutral beam and radiofrequency heating systems have been or are being developed for PLT, Doublet-III, PDX, TFTR, 2X-IIB, MFTF, ISX and EBT. These systems are well defined and, in most cases, adequate development activities are in place.

It is equally important for the technology development program to perceive the future plasma heating needs of the program and to accomplish the necessary development to provide the technology base for timely program advancement. It was the purpose of this meeting to peer into the future and to provide guidance to the technology program.

The meeting consisted of three sessions on successive days as outlined in the Agenda on pages iv, v, and vi. During the first session the user groups defined their future requirements. During the second session, the status and plans for the development programs were presented. The third session reviewed the needs and status of development and attempted to draw conclusions on future directions which the development activities should take. This document consists of the written reports, discussions, figures and summaries as consecutively presented at the meeting.

The Office of Fusion Energy wishes to than all of the participants for making this meeting a success.

Franklin E. Coffman, Acting
Assistant Director for
Development and Technology
Office of Fusion Energy

AGENDA

PLASMA HEATING DEVELOPMENT
REQUIREMENTS WORKSHOP

December 5, 1977

Requirements Definition

9:00	Welcome	J. Williams[1/], DMFE[2/]
9:05	Purpose of Workshop	C. Henning[3/], DMFE
9:15	DMFE Overview on Requirements Definition	S. Dean, DMFE
9:30	ORNL-TNS Heating Overall Requirements	M. Peng, ORNL
10:20	BREAK	
10:40	GA/ANL/TNS Heating Overall Requirements	J. Rawls, GA
11:30	PPPL-TFTR & TFTR Upgrade Heating Requirements	D. Jassby, PPPL
12:15	LUNCH	
1:30	MFTF and MFTF Upgrade Requirements	C. Damm, LLL
2:00	Field Reversed and Tandem Fusion and Hybrid Reactor Beam Requirements	R. Moir, LLL
2:30	Alternate Concepts Heating Requirements	R. Krakowski, LASL
3:00	Reactor Heating Requirements - Neutral Beams/RF	J. Scharer, Wisc.
3:30	Heating and Fueling Requirements Up to and Beyond JET Scale Experiments	J. Sheffield, ORNL
4:30	Discussions and Adjournment	

[1/] Presently employed with Los Alamos Scientific Laboratory.

[2/] Now the Office of Fusion Energy.

[3/] Presently employed with Lawrence Livermore Laboratory.

December 6, 1977

State of Technology and Developments Plans

9:00	Positive Ion Systems - State of the Art and Ultimate Potential	H. Haselton, ORNL
9:30	" "	B. Pyle, LBL
10:00	Positive Ion Systems with Direct Recovery	R. Moir, LLL
10:20	BREAK	
10:40	Negative Ion Systems - State of the Art	
10:45	• Direct Extraction Systems	K. Prelec, BNL
11:25	• Double Charge Exchange System	B. Hooper, LLL
10:05	LUNCH	
1:00	Negative Ion Development Program Plans	
1:05	• BNL Program	T. Sluyters, BNL
1:25	• LLL/LBL Program	B. Pyle, LBL
1:45	• TRW Program	D. Arnush, TRW
2:05	• ORNL Program	W. Stirling, ORNL
2:25	BREAK	
2:45	Negative Ion Systems for TFTR/U, MFTF/U, etc.	B. Hooper, LLL
3:10	Negative Ion NB Systems - Engineering Characteristics (e.g. gas load, total input power, etc.)	J. Fink, LLL
3:30	Control, Protection, Limitations, Problems - Technology of High-Voltage Systems	D. Hopkins, LBL
4:00	Ion Cyclotron Heating Technology	J. Hosea, PPPL
4:15	Electron Cyclotron Heating Technology	O. Eason, ORNL
4:30	Lower Hybrid Heating Technology	R. Motely, PPPL
4:45	Discussions and Adjournment	

December 7, 1977

Direction (Where to From Here) Discussions

9:00	Implications/Effects of Reactor-Like Environment	Moderator: J. Sheffield, ORNL
	• Viability of Neutral Beam Heating for Reactors	
	• Viability of RF Heating for Reactors	
10:15	BREAK	
10:30	Potential Ideas to Reduce Heating Requirements	Moderator: D. Jassby, PPPL
	• Perpendicular Injection	
	• Ripple Injection	
	• α Heating Effects	
	• Profile Effects	
	• Startup Scenarios	
12:00	LUNCH	
1:00	Summary of Requirements	L. Stewart, Exxon
1:15	Summary of Requirements - Round Table	Moderator: E. Kintner, DMFE
	• Short Term (TNS, MFTF, etc.)	
	• Long Term (Reactors)	
3:30	Adjournment	

ORNL-TNS/PEPR OVERALL HEATING REQUIREMENTS

Y-K. M. Peng and J. A. Rome
Fusion Energy Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

Workshop on Plasma Heating Development Requirements
Division of Magnetic Fusion Energy
December 5-7, 1977

INTRODUCTION (Slides 2-6)

The ORNL TNS/PEPR studies have the objectives[1,2] of 1) leading to a system that demonstrates the fusion reactor core in the mid-to-late 1980's and extrapolates to an economic tokamak power reactor, and 2) providing a near-term focus for the scientific and technological programs toward the power reactor. This discussion of the overall heating requirements for the ORNL TNS/PEPR is concerned with the neutral beams as the primary heating method, the electron-cyclotron resonance (ECR) heating at a lower power level for profile control, and the upper hybrid resonance (UHR) initiation and preheating of currentless plasmas to reduce current start-up loop voltage (V_ℓ) requirements.

NEUTRAL BEAM HEATING SCENARIO WITH CONTROLLED DENSITY BUILD-UP (Slides 7-14)

Neutral beams giving an estimated net power of 50-75 MW into the plasma provide the main plasma heating. To avoid relying on deuterium beam energies around 300 keV for full penetration at averaged densities of $\sim \bar{n} = 2 \times 10^{14}$ cm^{-3}, a neutral beam heating scenario[3] with controlled density buildup is proposed to permit the use of beams of 150-200 keV.

In this scenario, the initial density of the ohmically heated plasma is assumed to be near 0.5×10^{14} cm^{-3}, consistent with the Murakami scaling[4] and permitting full penetration with 150 keV, nearly perpendicular D° beams ($Z_{eff} \simeq 1.5$). Since the increased plasma heating power can support higher plasma density, the plasma density and temperature can be allowed to increase in a controlled fashion. When the volume averaged beta ($\bar{\beta}$) reaches 2% at $B_T = 5.3$ T, the maximum beta (β_{max}) at the plasma magnetic axis reaches 7%. With a peak density of $n_o = 2 \times 10^{14}$ cm^{-3}, the peak temperature becomes 12 keV. The local α-particle heating power density is above 0.8 MW/cm^3, with the particle drift orbit effects included.

Assuming empirical energy confinement, this centralized α-particle heating can be shown to compensate[5] for the lack of beam penetration expected for the 150-keV beam at $n_o = 2 \times 10^{14}$ cm^{-3} and $Z_{eff} \simeq 1.5$. When $\overline{\beta}$ increases beyond 2.5%, the plasma is ignited at the center and can reach D-T burning steady states with the magnitude of fusion power output determined by the plasma density. The plasma heating scenario with controlled density buildup may be tested in JET if $\overline{\beta}$ above 5% can be achieved with $B_T = 3.5$ T.

This heating scenario and other scenarios such as ripple injection,[6] expanding plasma,[7] and compression boosting[8] tend to increase the probability of successful heating with D° beam energies of 100-200 keV. This suggests the use of positive ion beams based on present-day beam technologies.[9] However, because of the relatively low overall efficiency ($\lesssim$20%) of positive ion D° beams when E_b = 150-200 keV, efficient technology of direct recovery[10] of the ion energy should be developed to substantially reduce the neutral beam power supply requirements.

Neutral beam injection heating has one important limitation that calls for some attention, regardless of positive or negative ion beams. Recent 1½-dimensional beam deposition calculations[11] reveal that, for fixed beam energy, the operating window in plasma density is only a factor of two to three. More than 10% of the neutral beam particles pass through the plasma at lower densities, and hollow beam deposition results at higher densities. To increase the operating window in density, independent control of the beam deposition profile during the approach to ignition is required as $n_o a$ increases. This might be achieved by using a variable beam energy or a variable beam species mix. The technologies for achieving this should be studied.

MICROWAVE HEATING NEAR THE ELECTRON CYCLOTRON RESONANCE (ERC) (Slides 15-20)

Recent MHD stability calculations[12] of ballooning kink modes in D-shaped flux conserving tokamak (FCT) equilibria[13] have revealed significant dependence of critical beta ($\overline{\beta}_c$) for stability (and hence the fusion power density) on the plasma pressure profile. This suggests the need for external modification of the pressure profile in a fashion more flexible and localized than the neutral beam deposition profile.

On the other hand, it has been shown that the toroidal current profile in a high β FCT equilibrium[13] is significantly different from the current profile of an ohmically heated equilibrium. The FCT high β equilibrium, left alone, is expected to evolve toward a low β, ohmic equilibrium in a plasma skin time, which is typically a few tens of seconds in a reactor. One conceivable way of maintaining the high β equilibrium beyond the plasma skin time is to reduce the plasma resistivity in regions where high current density is required for high β. Localized heating is one way to make this possible.

The ECR heating, because of its narrow resonance zone, provides for such a mechanism. Recent calculations of ECR heating in ISX shows that the heating profile width can be typically one-fifth of the plasma minor radius. Assuming that the local plasma density and heat conductivity are not altered substantially by the local ECR heating, a 20% increase in local temperature then seems desirable. This would then require a total microwave power of roughly 20 MW, one-twenty-fifth of the total α-particle power in a D-T tokamak reactor. For the ORNL TNS/PEPR with B_T = 5.3 T at the major radius and an aspect ratio of 4, a microwave frequency range between 120-150 GHz is required.

MICROWAVE PLASMA PREHEATING NEAR THE UPPER HYBRID RESONANCE (UHR) (Slides 21-26)

The ohmic heating power supply required to start up the plasma current over the full bore in a tokamak reactor has been estimated[15] to be above 1 GW for a duration of around one-tenth of a second. This pulsed power supply requirement implies substantial technology development and cost. The fundamental reason for this high requirement is the estimated loop voltage (V_ℓ between ∿300-500 V) needed to bring the initial plasma current and temperature through the relatively large barrier of energy and volt-second losses due to minute amounts of low Z impurities, when $T_e < 50$ eV.[17] There are strong technology and cost inc-ntives to reduce this start-up loop voltage requirement in a tokamak reactor.

Based on recent experimental observations by Anisimov et al.,[18] plasma initiation and preheating near the UHR are proposed[19] to obtain $T_e \simeq 250$ eV at least in a small volume in the toroidal chamber before the toroidal current is initiated by applying V_ℓ. In this new start-up scenario, anomalous absorption of the extraordinary microwaves introduced from the high field side heats the electrons. Energy and particle losses are assumed to be due to magnetic curvature and parallel drifts, ionization of the neutrals, cooling by ions, and radiation by low Z impurities. To estimate the V_ℓ required for current start-up from the microwave plasma, T_e is assumed to remain constant at 250 eV when the plasma current increases to full value after the application of constant V . The plasma safety factor, q, plasma resistive voltage, and Z_{eff} are also assumed to be constant during this process.

Our rough estimates are encouraging in that, using a microwave power of 0.6 MW at 120 GHz, V_ℓ can be reduced to about 30 V without increasing the start-up loss of volt=seconds. Given sufficient support, this start-up scheme can be tested in ISX using the 28-GHz, 200-kW power sources developed for the EBT-S experiment.[20] If verified, the impact of this method on the start-up cost of tokamak reactors is expected to be dramatic.

DISCUSSION (Slide 27)

The ORNL TNS/PEPR studies on the plasma heating requirements form good examples of efforts that contribute strongly toward an economic tokamak reactor. Our considerations have naturally led to the following foci for the technological and scientific programs on plasma heating:

1. Direct recovery of positive ion energy in deuterium beams at E_b = 150-200 keV.

2. Schemes to alter neutral beam energy deposition independently of plasma density during heating, e.g., by varying E_b or energy species mix.

3. CW microwave power around 20 MW at a range of frequencies between 120 and 150 GHz for plasma profile control.

4. CW microwave power around 1 MW at a frequency between 120 and 150 GHz to reduce current start-up loop voltage by an order of magnitude.

Whether these results could benefit the tokamak fusion program depends strongly on whether the necessary developments and experimentations are carried out in a timely fashion.

ACKNOWLEDGEMENT

This write-up has benefited from useful discussions with J. D. Callen, R. A. Dory, H. C. Howe, M. Roberts, and D. Steiner.

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Y-K. M. Peng and J. A. Rome
TNS Plasma Engineering
ORNL TNS PROGRAM

THE ORNL TNS PROGRAM AIMS TO REDUCE COST AND RELIANCE ON ADVANCED HEATING TECHNOLOGIES

WORKSHOP ON
PLASMA HEATING DEVELOPMENT REQUIREMENTS
DIVISION OF MAGNETIC FUSION ENERGY
December 5 – 7, 1977

Slide 2

Slide 1

THE ORNL TNS OBJECTIVES ARE TWO-FOLD

- Lead to a system that
  - Demonstrates reactor core in next decade
  - Is extrapolatable to an economic tokamak power reactor

- Provide near-term focus for the scientific and technology engineering programs towards the above objectives

PRE-CONCEPTUAL   CONCEPTUAL   SELECTION

↑

NOW

BASED ON THE FY77 TNS STUDIES
WE CHOOSE THE FOLLOWING BASELINE PARAMETERS

- 20 Nb_3Sn COILS, $B_{MAX} = 10.9$ T, $B_T = 5.3$ T
- $R_0 = 5$ m, $a = 1.2$ m, $\sigma = 1.6$, D–SHAPE
- COMPACT POLOIDAL DIVERTOR,

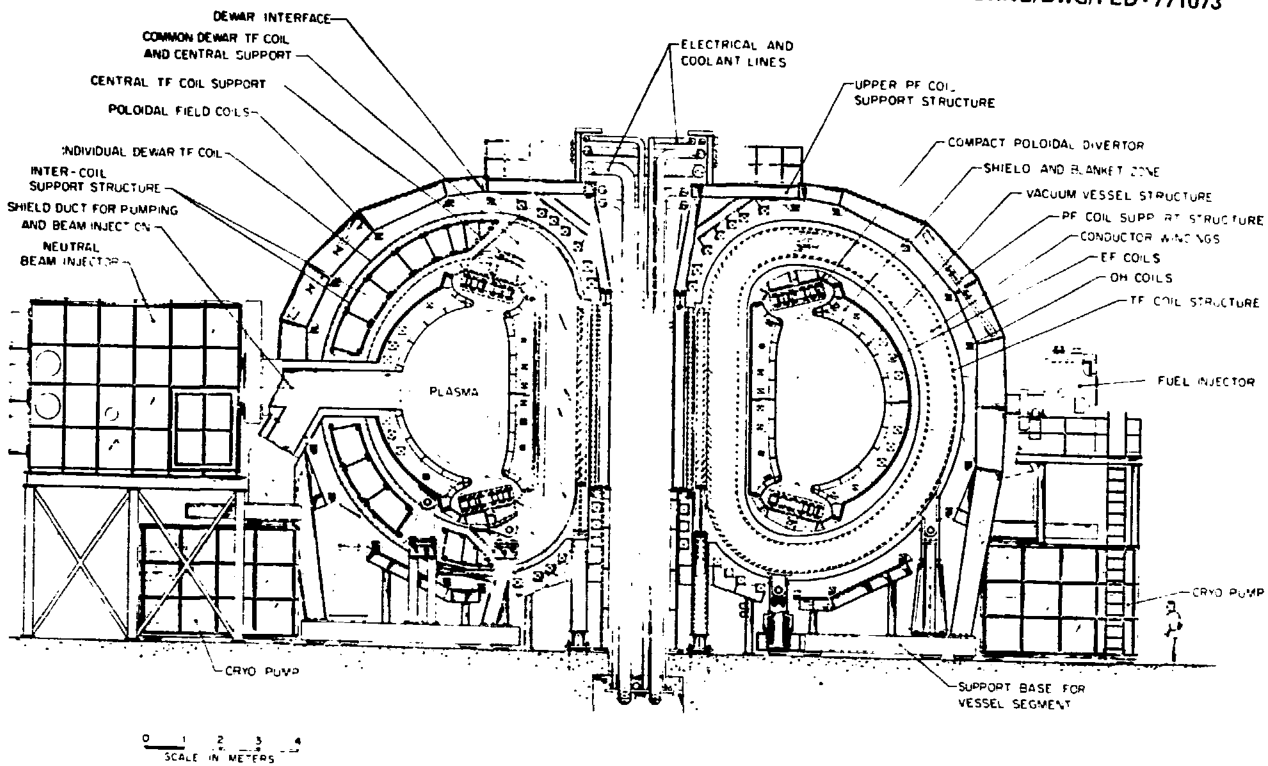
  INTERIOR COPPER EQUILIBRIUM FIELD COILS
- 5 NEUTRAL–BEAM INJECTION LINES (50-75 MW)

  9 CRYO–PUMPS

  1 ORNL PELLET INJECTOR
- COMMON DUCTS FOR INJECTORS AND PUMPS

Slide 3

ORNL/DWG/FED-771073



ELEVATION (CUTAWAY)

Slide 4

A CONSISTENT SET OF PHYSICS ASSUMPTIONS ARE USED FOR THE OAK RIDGE MEDIUM FIELD IGNITION REACTOR

PARAMETERS REFINED WITH 1-D TRANSPORT, MHD EQUILIBRIUM AND STABILITY CALCULATIONS

	LOW β	IGNITION	BURN
$\bar{\beta}$ (%)	1.0	3.0-5.0	3.5-10.0
I_P (MA)	4.0	4.5-5.5	
$\bar{N}$ (CM^{-3})	$0.3\text{-}0.5 \times 10^{14}$	$0.6\text{-}2.5 \times 10^{14}$	
$\bar{T}$ (keV)	1-2	4-7	5-10
COLLISIONALITY	0.14-0.98	0.01-0.15	
$\overline{n\tau}$ (CM^{-3} SEC)			$0.6\text{-}3 \times 10^{14}$
$P_{D\text{-}T}$ (MW)			100-2000
$P_{D\text{-}T}/V$ (MW/M^3)			0.4-8.0
W_L (MW/M^2)			0.3-5.0

Slide 5

TNS STUDIES IDENTIFY PHYSICS ASSUMPTIONS AND OPERATING SCENARIOS TO REDUCE IMPLIED TECHNOLOGY REQUIREMENTS

→ • Neutral beam heating scenarios
  — Reduced beam energy

• 1–D transport evaluations of pellet requirements
  — Reduced pellet velocity

• Reassessment of toroidal field ripple requirements
  — Improved machine access

• Hybrid equilibrium field coils
  — Reduced engineering difficulty and power supply

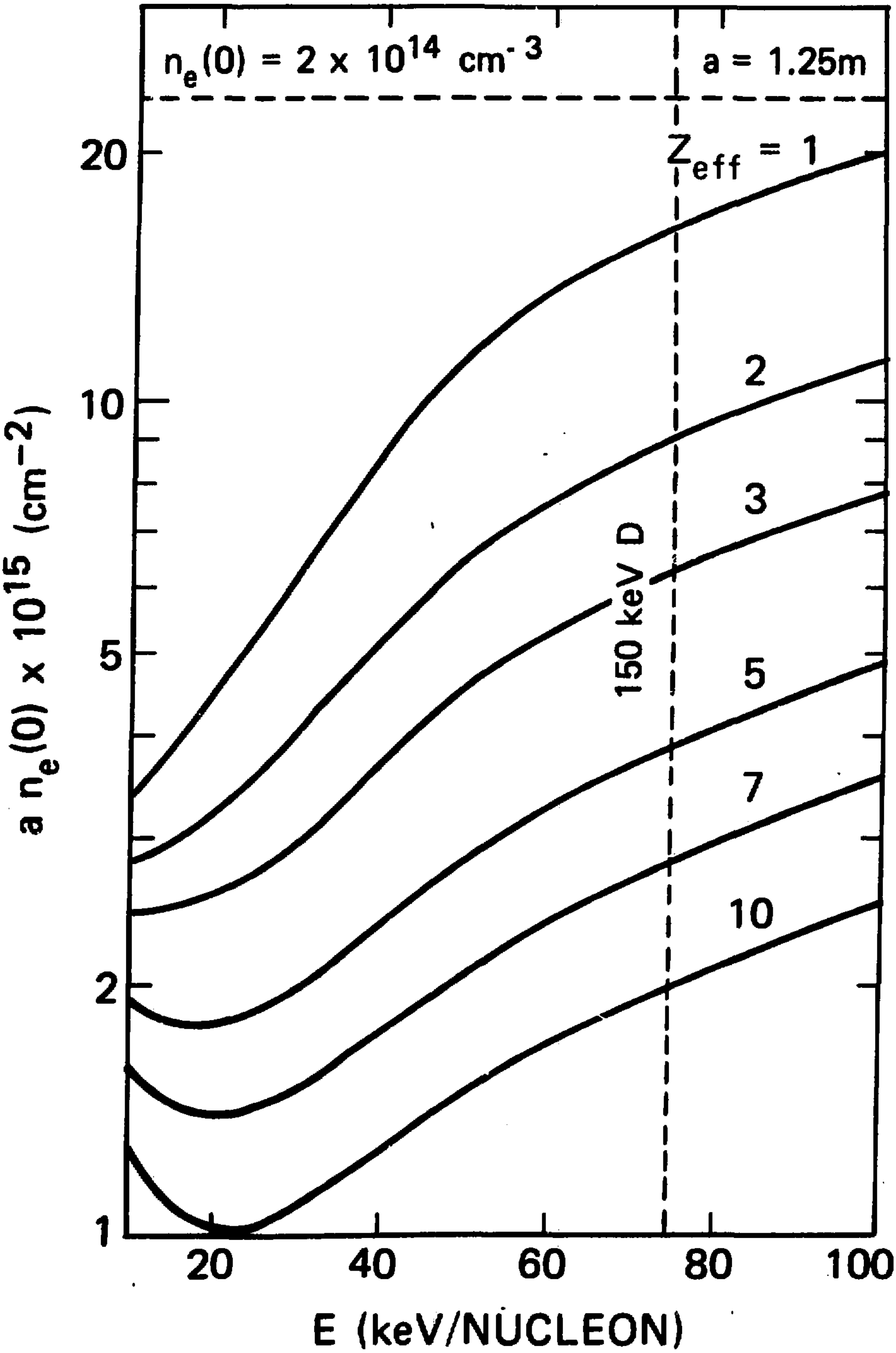
→ • Microwave startup near upper hybrid resonance
  — Reduced startup costs

• Iron core option
  — Improved overall engineering feasibility and reduced power supply

• Compact poloidal divertors

Slide 6

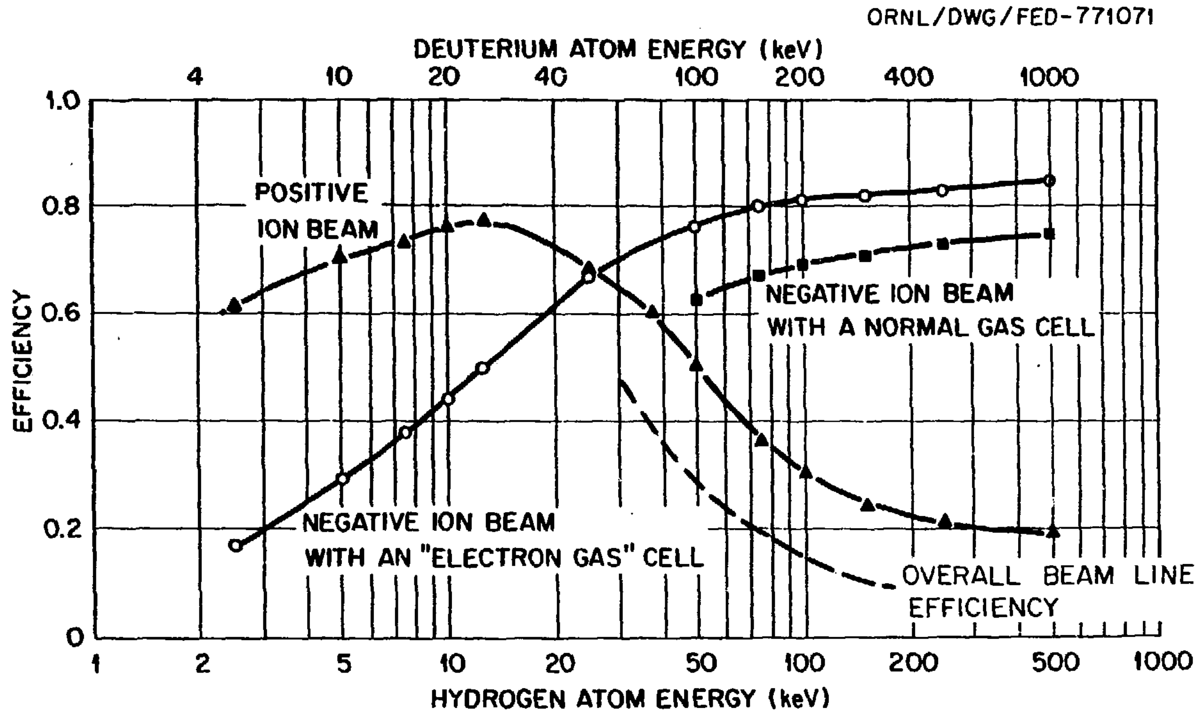
ORNL/DWG/FED–77837A



Slide 7

POSITIVE ION NEUTRAL BEAM PRODUCTION

EFFICIENCY IS LESS THAN .3 WHEN E_b (D^+) > 200 keV



Neutral Beam Production Power Efficiency

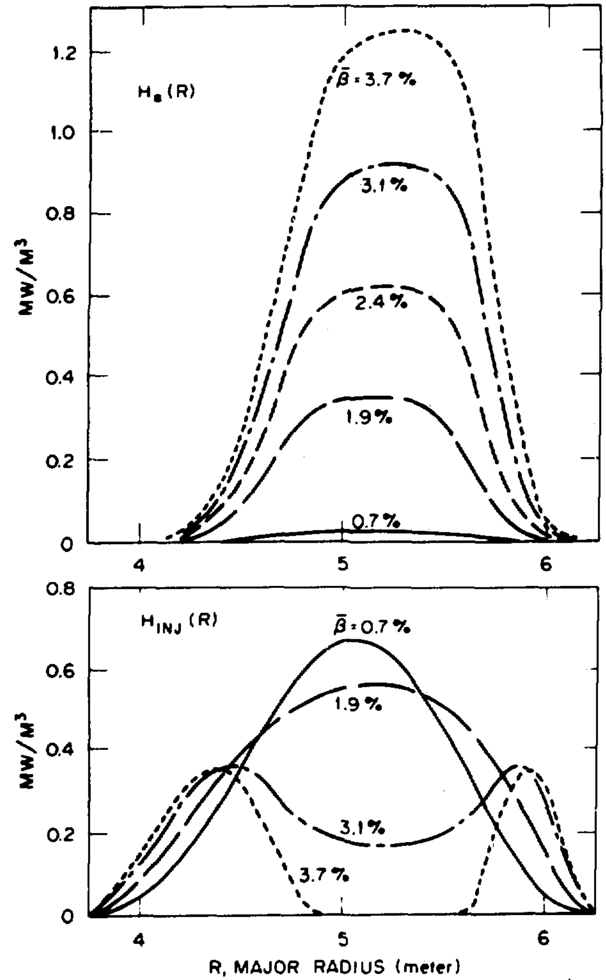
Slide 8

REQUIRED H_{INJ} (R) RECEDES FROM CENTER

WHERE H_α (R) peaks when $\beta > .025$

($\bar{n} \simeq 0.7 \times 10^{14}$ cm^{-3})

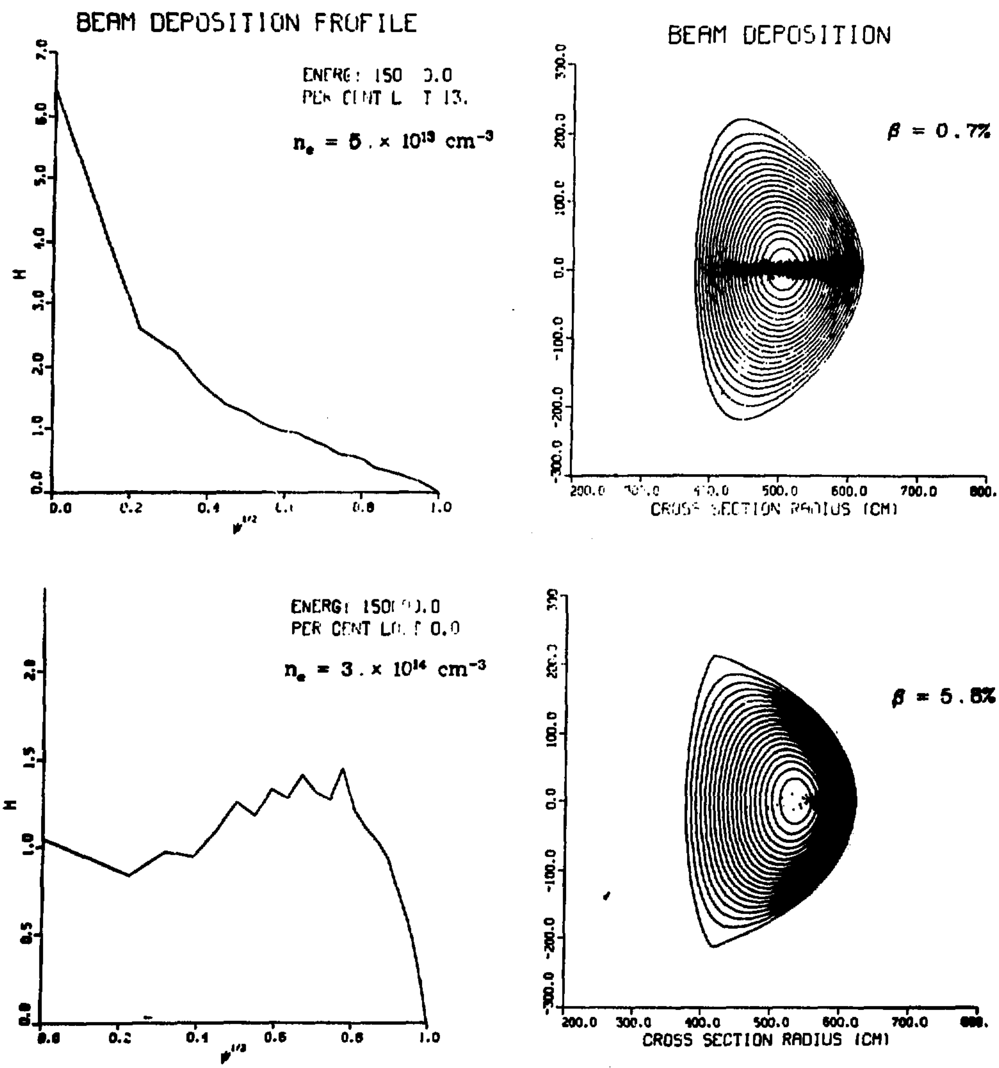
ORNL/DWG/FED-77267



Slide 9

ORNL/DWG/FED-771074

2-DIMENSIONAL BEAM DEPOSITION CALCULATIONS ARE USED



OTHER INJECTION SCENARIOS MAY ALSO SUCCESSFULLY REDUCE REQUIRED E_b

- Ripple injection ⟶ $E_b < 150$ keV

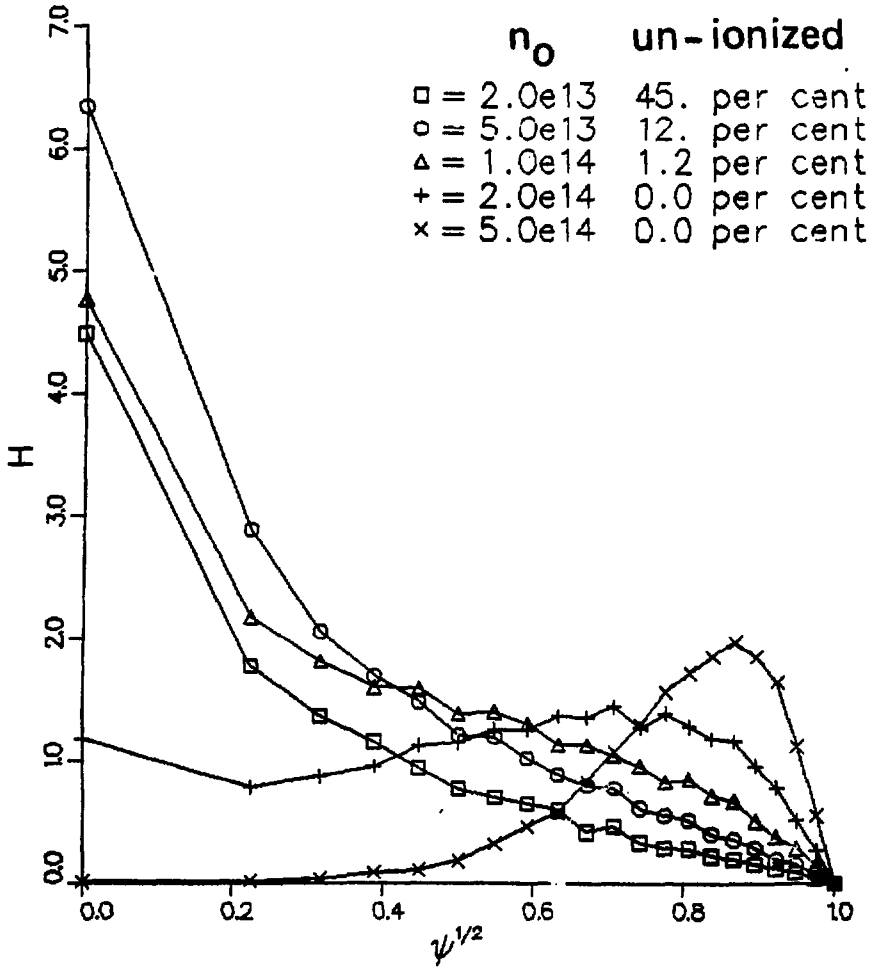
  ripple coils

- Expanding radius ⟶ low E_b

  $\tau_R \sim \tau_E$

ORNL/DWG/FED-771072

FOR A GIVEN BEAM ENERGY, THE ALLOWED DENSITY WINDOW IS ONLY A FACTOR OF 2 – 3



Slide 12

IMPROVEMENTS IN POSITIVE ION BEAM TECHNOLOGY ARE POSSIBLE

- Control H_{INJ} (R) at fixed density to enlarge density window
  - ? Variable E_b
  - ? Varaible species mix
  - ? Switch from D to H
- Increase injector efficiency near 200 keV
  - ? Direct recovery
  - ? Higher D^+ species yield

Slide 13

THE ORNL TNS NEUTRAL INJECTION SCENARIO CAN REDUCE COST AND RELIANCE ON ADVANCED BEAM TECHNOLOGY

- Injection heating scenarios suggest $E_b < 200$ keV acceptable
  - ---► Injector reliability, flexibility, and remote maintainability
- Small density window for efficient heating
  - ---► Variable E_b or species mix
- Improvements in efficiency of + ion injectors should be considered

Slide 14

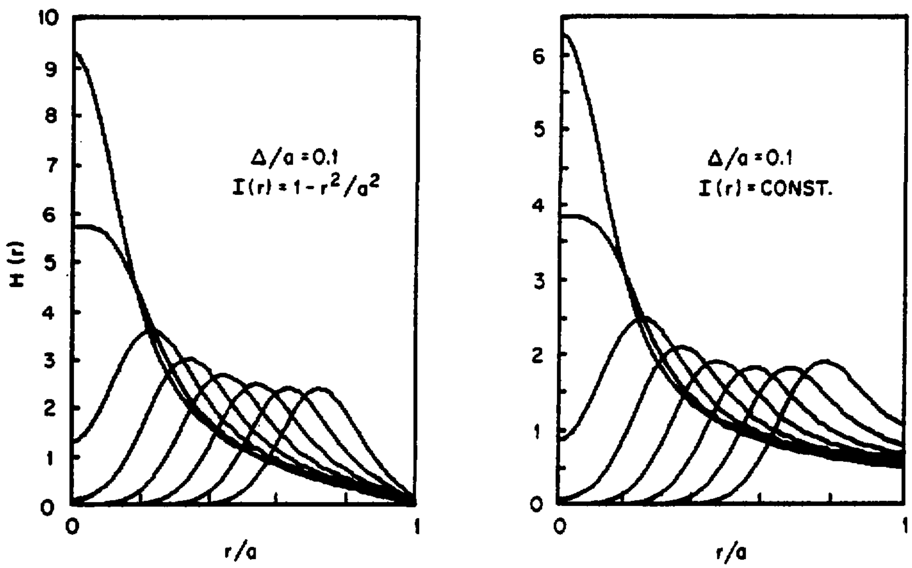
MICROWAVE HEATING OFFERS FLEXIBILITY

- Proven heating scheme
- Good accessibility
- Localized energy deposition profile, H_{μ} (R)
- Potential need of plasma profile control

$\underline{H_\mu \text{ (r) CAN BE WIDELY VARIED BY}}$

VARYING THE LOCATION OF THE ECR ZONE

ORNL-DWG 77-8814

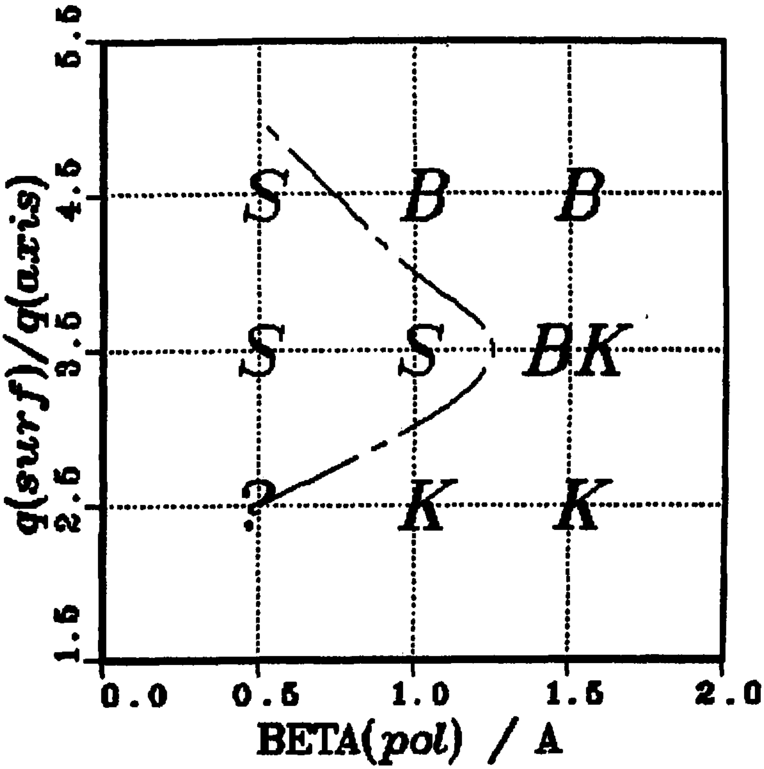


- Location of $H_\mu(r)$ can be altered by varying B_T or F_μ.

Slide 16

ORNL/DWG/FED-771075

PROPER PLASMA PROFILE CONTROL CAN SUBSTANTIALLY INCREASE MHD STABLE $\bar{\beta}_c$, LEADING TO HIGH FUSION POWER DENSITY



(R. A. Dory et al , DPP APS 5F4)

Slide 17

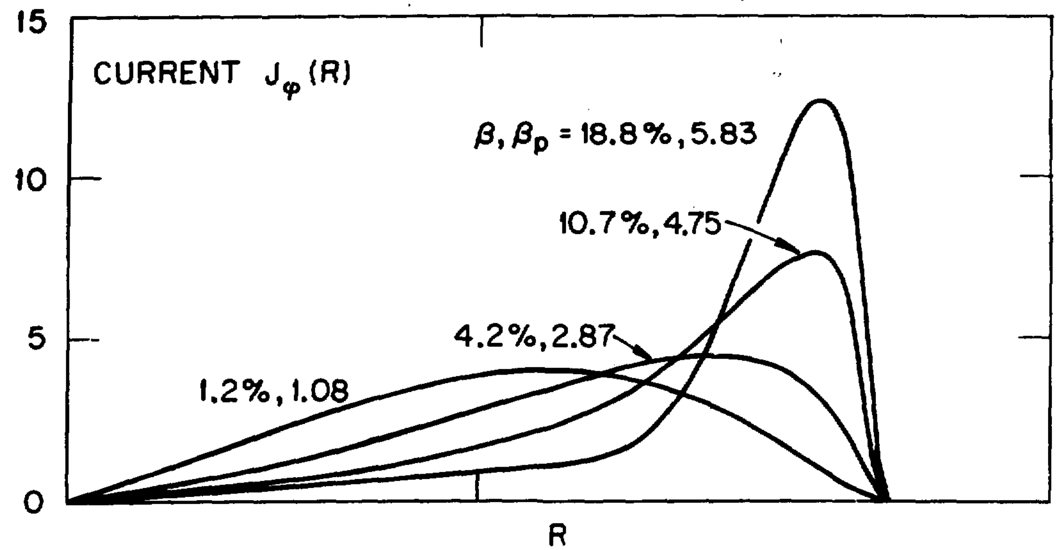
THE REQUIRED P_μ FOR PROFILE CONTROL CAN BE ESTIMATED TO BE ~ 20 MW AT 120 – 150 GHz

- $p_\alpha \sim \partial(\chi n\ \partial T/\partial r)/\partial r$
  $p_\mu \sim \partial(\chi n\ \partial \Delta T/\partial r)/\partial r$, local T change
- Assume
  n, χ unchanged, $\Delta T \sim T/5$
  over $\Delta r \sim a/5$ at $r \sim a/2$
- Then $P_\mu \sim P_\alpha/25 \sim 20$ MW, $\tau \sim \tau_E$
- $B_T = 4.3 - 7.0$ T Within plasma
  $\longrightarrow F_\mu = 120 - 150$ GHz
- Technology yet to be developed

LOCALIZED HEATING IS ONE POSSIBLE WAY TO MAINTAIN HIGH β FCT EQUILIBRIUM PROFILES BEYOND THE PLASMA SKIN TIME

- Assume that proper plasma shape maintained
- High β J_φ (R) profile not compatible with

$$E_\varphi = \eta J_\varphi \text{ for steady state}$$



Slide 19

OTHER WAVE HEATING METHODS MAY ALSO BE EFFECTIVE

- Lower hybrid (~ 1.5 GHz)
- Ion cyclotron resonance (25 – 100 MHz)
- Low frequency magnetosonic wave (1 – 10 MHz)
- Low frequency shear Alfven wave (1 – 2 MHz)

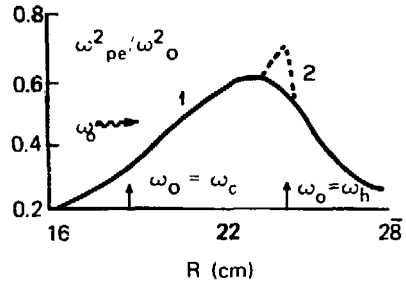
Slide 20

ANOTHER APPLICATION OF MICROWAVES MAY BE TO PREHEAT TOROIDAL PLASMAS TO $T_e \sim 200$ eV BEFORE CREATING LARGE I_p

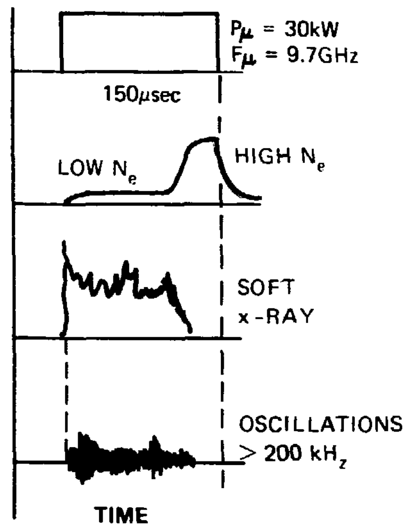
- Startup $V_L \doteq 300 - 500$ volts for ~ 50 msec
- OH power supply cost up to 150 M dollars
- Large energy and volt–second barriers by low Z impurities when $T_e < 50$ eV
- Desirable to preheat electrons to above 100 eV in a small volume

Slide 21

EXPERIMENTS BY ANISIMOV, ET AL. HAS PROVIDED SOME INTERESTING POSSIBILITIES USING EXTRAORDINARY MICROWAVES

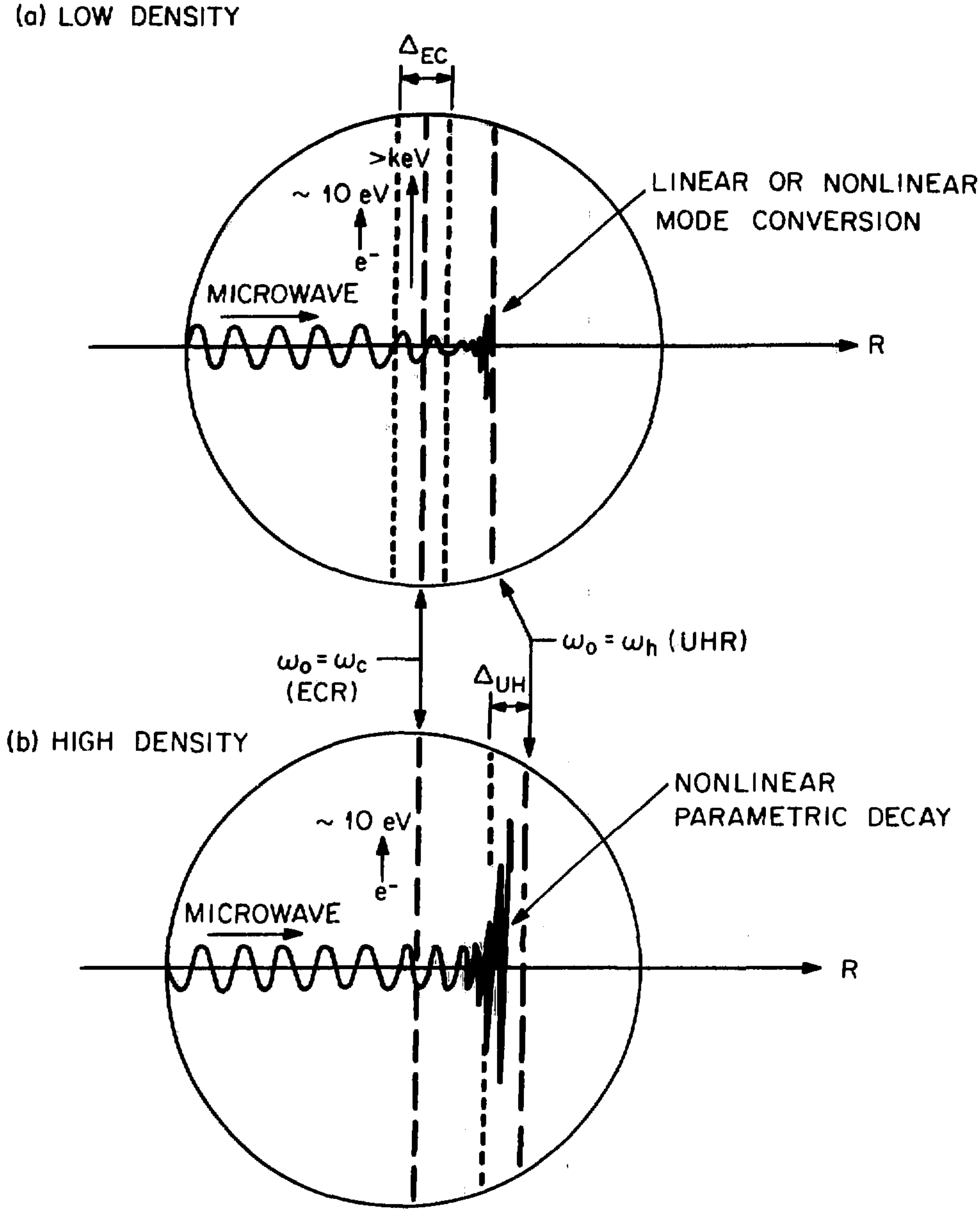


- H_2 AT 2×10^{-4} Torr
- $B_T \sim$ 3kG. $F_\mu \sim$ 9.7GHz
  1) Cw, $P_\mu \cong$ 200W
  2) + PULSED, P_μ = 300W
- EXTRAORDINARY MODE



- AT LOW DENSITY, $W_e > 20$ keV RUN AWAY ELECTRONS AND LOW FREQUENCY OSCILLATIONS ARE INDICATED.
- AT HIGH DENSITY, Te ESTIMATED TO BE A FEW 100 eV, WITH NO INDICATIONS OF >> keV ELECTRONS AND LOW FREQUENCY OSCILLATIONS FOR ABOUT 30 μsec

THE ASSUMED WAVE – PLASMA INTERACTIONS HAVE TWO DENSITY REGIMES



Slide 23

THE FOLLOWING ASSUMPTIONS ARE USED IN OUR ESTIMATES

(ORNL/TM – 6112)

- Anomalous absorption near UHR, heating electrons
- Energy and particle balance:
  - o Curvature and parallel drifts
  - o Ionization of neutrals
  - o Cooling by ions
  - o Radiation by low Z impurities
- For startup V_L requirements
  - o Fixed $T_e \sim 250$ eV,

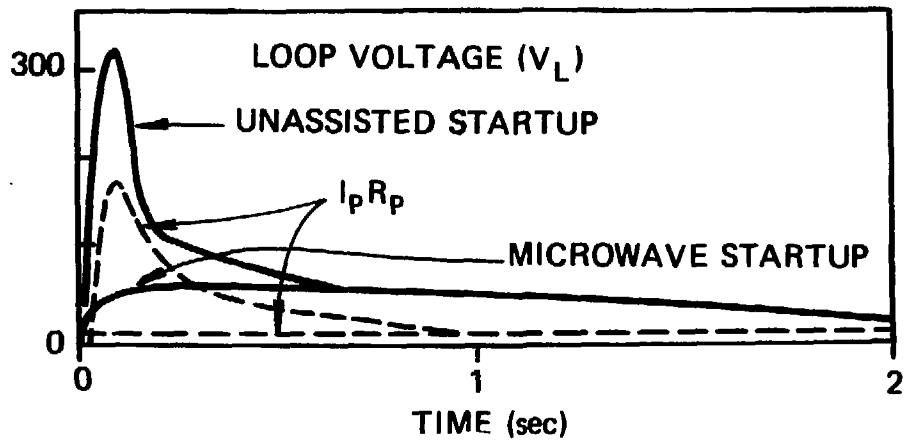
    V_L , q, V_{RES} , Z_{EFF}
  - o a $\sim I^{1/2}$

Slide 24

UPPER HYBRID RESONANCE PREHEATING SIGNIFICANTLY REDUCES START-UP REQUIREMENTS

- V_L CAN BE REDUCED BY A FACTOR OF 5-10
- STARTUP PULSE (ΔT) SMOOTHED BY A FACTOR OF 2-5
- MUCH REDUCED LOSS OF VOLT SECONDS DURING STARTUP
- $P_\mu \lesssim$ MW AT 120 GH_Z

ORNL/DWG/FED-77833A



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MICROWAVE HEATING SCENARIOS CAN ENHANCE $\overline{\beta}_C$ AND REDUCE STARTUP V_L

- Proper plasma profile control may
  - Increase stable $\overline{\beta}_C$ substantially
  - Maintain high β equilibrium over long times
- Microwave startup near UHR may reduce startup V_L by factor of 10
- Points to the development of

  CW power = 1 MW – startup

  ~ 20 MW – profile control

  ? Tunable frequency, multiple frequencies
- Requirements similar to those of EBT-II

Slide 26

THE ORNL TNS PROGRAM AIMS TO REDUCE COST AND RELIANCE ON ADVANCED HEATING TECHNOLOGY

ORNL TNS STUDIES

- Lead to economic tokamak power reactor
- Identify physics assumptions and operating scenarios to reduce implied technology requirements

NEUTRAL INJECTION

- Neutral beam heating scenario suggests

  $E_b = 150 - 200$ keV

  ⟶ Development of variable beam deposition and improved + ion injector efficiency

MICROWAVE HEATING

- ECR profile heating and UHR pre–heating scenarios may result in large benefits

  ⟶ development of sources

  $F_\mu = 120 - 150$ GHz, CW

  $P_\mu = 1$ MW – startup

  20 MW – profile control

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GA-A14760

AUXILIARY HEATING REQUIREMENTS FOR A D-T BURNING TOKAMAK: THE GA/ANL TNS DESIGN

by

John M. Rawls

DECEMBER 1977

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Work supported by
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GENERAL ATOMIC PROJECT 3235.871.001 DECEMBER 1977

1. GENERAL CONSIDERATIONS FOR LONG BURN TIME DEVICES

The design of the auxiliary heating system is of primary importance for any D-T burning tokamak. It is, first and foremost, the means of reaching ignition. It is also a major cost item, very likely in excess of \$$10^8$ for either the neutral beam or the rf approach. If neutral beams are employed, there are also substantial indirect cost implications resulting from the increased size of the facility, additional remote maintenance and shielding requirements, and a substantial R&D program. The auxiliary heating system also impacts the choice of site through the power requirements, safety aspects, and in the case of rf heating, potential interference problems.

The system requirements are, of course, defined by the objectives of the device in question. As presently envisioned, the next large tokamak after TFTR will be designed to achieve ignition and long enough burn times to study the major elements of burn dynamics: alpha particle physics, profiling, impurity control, fueling, and ash accumulation. The technological objectives are less firm, but may include the generation of high grade heat and the capability of operation at a high (> 50%) duty cycle. In addition, the design will employ, whenever possible, technologies and operating procedures which can be extrapolated to the reactor regime.

The long burn time demands superconducting field coils, a decision which dramatically impacts the overall design in many ways. A number of the direct implications, some of which help to define the auxiliary heating requirements, are listed below.

1. The facility power requirements are dramatically reduced by such a choice, making siting easier and reducing both capital and operating costs.

2. Minimum cost designs for a fixed field ripple at the plasma exhibit fewer, but larger, coils than the corresponding normal coil designs; this increases the size of the tokamak, but alleviates the access problems for heating, diagnostics, and maintenance.

3. The adoption of NbTi will minimize the technological risks inherent in the timetable for device fabrication. This limits the peak field at the magnet to about 10 T; the achievement of economically feasible power densities ($\propto \beta^2 B^4$) then necessitates the adoption of high ($\simeq$ 10%) beta plasma configurations. Alternatively, use of Nb_3Sn can result in lower beta designs at the expense of increased development costs and accompanying scheduling delays.

4. The nuclear shielding required for the superconducting magnets both reduces the field at the plasma and increases the aspect ratio, hence reducing the amount of ohmic heating and increasing the auxiliary heating requirements.

While auxiliary heating may ultimately play important roles in plasma initiation and profile control and perhaps even in impurity control and diagnostics, its principal function remains to supply the energy needed to raise the plasma to ignition temperatures and this will be the thrust of the subsequent material. The task at hand revolves around three primary problem areas. First, the ignition point must be determined; this serves to specify the bulk heating requirements. Second, the heating scenario itself must be examined to ascertain if the ignition point is dynamically accessible. Finally, a thorough study must be made of the technological feasibility of the heating scheme employed. This exercise is carried out for two heating options: neutral beams and lower hybrid heating or, more precisely, electron Landau damping. The latter has been chosen from among the plethora of rf possibilities by virtue of its technological advantages.

2. DETERMINATION OF THE IGNITION POINT: BULK HEATING REQUIREMENTS

Assuming for the moment that an adequate auxiliary heating system exists, the principal physical ingredients in determining the ignition point are the energy confinement, the alpha confinement, the applicable beta limits, and the efficiency of the impurity control procedures. It is conceivable that any or all of these key ingredients may be significantly impacted by the choice of auxiliary heating method. To proceed, assume classical alpha confinement, perfect impurity control, and sufficient plasma shape optimization that beta at ignition can be as high as 5% and that the higher beta value achieved after the thermal excursion to the burn equilibrium point is also MHD stable. Turning now to energy confinement, the thresholds for thermal breakeven (thermal fusion power produced balances auxiliary power supplied) and for ignition (thermal fusion power produced and thermalized in the plasma balances the sum of all plasma energy losses) are functions only of the temperature T and the quality of confinement $n\tau$. Using central values of density and temperature and the global value of τ, the important tokamak data points are displayed in Fig. 1 for comparison with these objectives.

The determination of the ignition point thus requires a scaling law expressing τ in terms of plasma and machine parameters. Existing data for confinement exhibits a quadratic dependence on plasma size and a linear dependence on density until very high densities are reached; above a certain density threshold, neoclassical confinement becomes more restrictive. (This transition point is predicted to be moved to higher densities by the use of noncircular cross sections.) The resulting scaling law, $n\tau \propto (na)^2$, so-called Alcator scaling, is adopted in what follows. It should be noted that existing data exhibits a tendency for larger, hotter tokamaks to have somewhat better confinement than is predicted by Alcator scaling. While this is a comforting trend from the point of view of reactor design, it has not been incorporated into the sizing studies on the grounds of design conservatism.

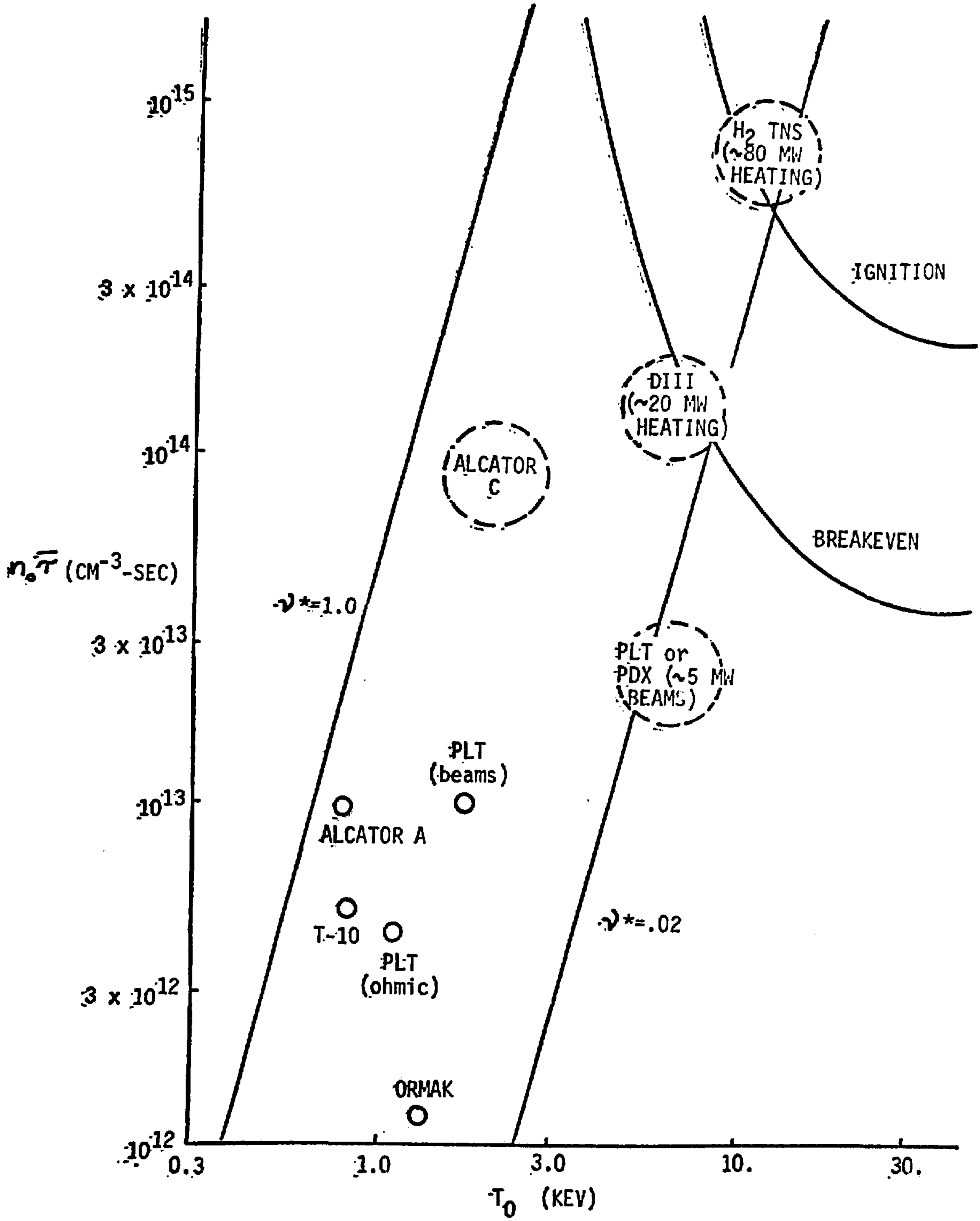

Fig. 1. Comparison of the values of nτ and T achieved by existing tokamaks with those needed for thermal breakeven and for ignition. The ν^* = constant lines and the predictions for other devices are based on Alcator scaling.

The fact that $n\tau$ is a function only of the line density na leads to some important conclusions. One such conclusion follows from the fact that the effective collision frequency ν^*(ν^* = electron-ion collision frequency times aspect ratio/bounce frequency), generally regarded to be the most important parameter to determine the applicable transport mechanism, is also a function only of na and T, namely $\nu^* \propto naT^{-2}$. Existing experiments span the range $0.1 \lesssim \nu^* \lesssim 1$ and it is anticipated that the regime $\nu^* \lesssim (m_e/m_i)^{1/2} \approx 0.02$ (for hydrogen discharges) will exhibit severe degradation of confinement due to the onset of trapped particle modes. The philosophy adopted is that Alcator scaling can be extrapolated with a reasonably high confidence level provided $\nu^* > 0.02$; hence the ignition design point is chosen to satisfy this criterion. Incidentially, some deterioration in confinement with increasing temperature is required to prevent a thermal runaway after ignition is achieved; in the picture presented here, this is provided by the dissipative trapped ion mode, which gives a contribution to proportional to $T^{-7/2}$ for $\nu^* < 0.02$.

Figure 1 also illustrates the design points for a number of tokamaks presently under construction in addition to that for the GA TNS design. Precisely the same set of assumptions were used to compute the fully heated (20 MW) Doublet III design point and the TNS ignition point. Since these correspond to the same values of ν^* and β and very nearly the same wall loadings (and hence comparable impurity control problems), the TNS design basis has in many respects a comparable physics confidence level to that for DIII. Unfortunately, as we will see, this conclusion does not apply to the question of heating.

Having chosen the $n\tau$ - T ignition point, the plasma can be sized on the basis of beta considerations. Typical layouts for D-T burning superconducting tokamaks give a toroidal field at the plasma of about one-half the maximum field at the magnet; hence B_T = 5 T is assumed for the purpose of this simple treatment. Demanding beta at ignition to be less than 5% then gives an upper bound on the density and hence a lower bound on the plasma size for a fixed $n\tau$. Figure 2 illustrates the case for a = 1.1 m. The shaded region is defined by $\beta < 5\%$, $\nu^* > 0.02$, and the ignition curve.

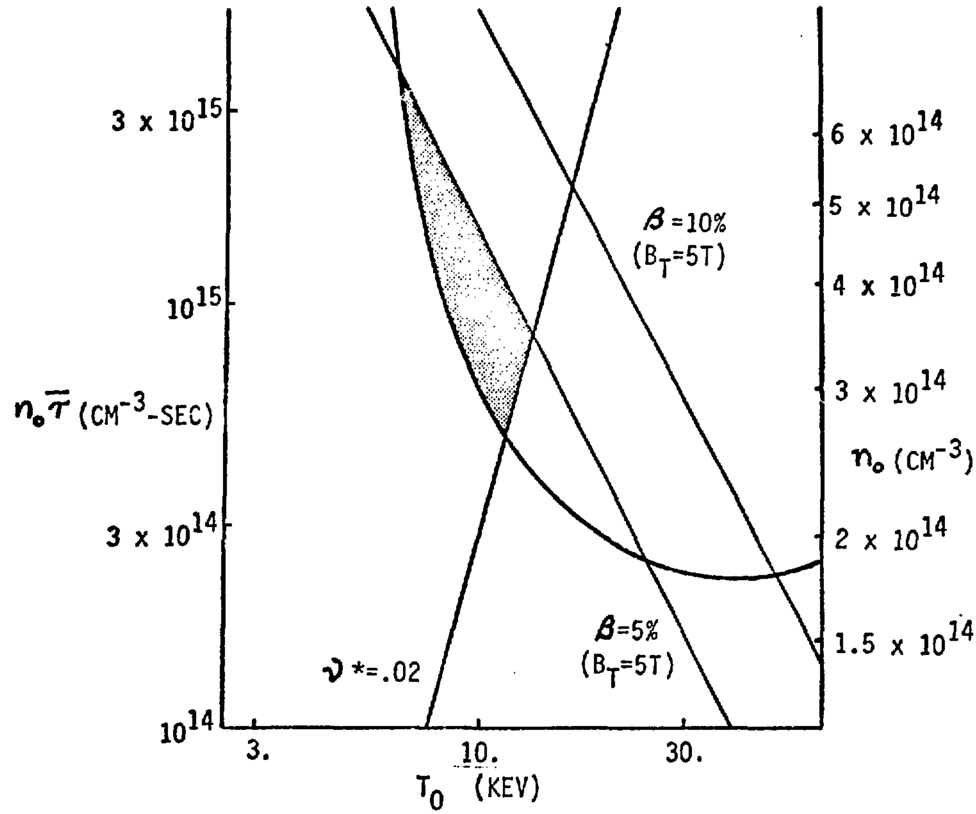

Fig. 2. The shaded region satisfies global ignition, $\beta < 5\%$, and $\nu^* > 0.02$ for a tokamak with $B_T = 5$ T, a = 1.1 m, and a vertical elongation of 2.7.

Subsequent to achieving ignition, the plasma temperature rises at fixed density until the confinement deteriorates to the extent needed for thermal equilibrium. Higher beta values may then be achieved by gas puffing, but in any case, MHD activity will limit the achievable values of beta.

The auxiliary power required to sustain ignition temperatures in hydrogen operation in a transport-dominated regime is given by

$$P = \frac{3nT}{\tau_E} V \propto TR$$

and is about 80 MW for the TNS design. But in D-T operation, alpha power at near ignition temperatures provides some of the heating; it is found that 60 MW is sufficient in the D-T case. These predictions have been verified by means of 1-D and 2-D transport codes.

3. NEUTRAL BEAM SYSTEM REQUIREMENTS

While many of the systems and much of the physics can simply be scaled up from existing devices to an ignition device, this does not apply to a neutral beam injection system. That the technological problems become far more difficult is seen by computing E_b, the beam energy required for adequate penetration. Because E_b is proportional to the line density, it can be computed directly in terms of $n\tau$ for the case of Alcator scaling:

$$n\tau \propto (na)^2 \propto E_b^2 \quad .$$

This gives $E_b \approx 150 - 200$ keV for the case of hydrogen beams. For the case of deuterium, the result is 300-400 keV, a frightening technological prospect for positive ion beams; in this energy range, the neutralization efficiency is only 5 - 8%. Note that this prediction for E_b is consistent with the choice of 120 keV D° beams for TFTR because ignition requires an order of magnitude increase over the $n\tau$ values needed for breakeven beam driven operation.

It is important to realize that at least the penetration problem does not get worse in the progression TNS-DEMO-power plant provided one need only heat to ignition. This follows because one can operate at the same ignition point in $n\tau$-T space. However, this requires smaller densities at ignition for the larger devices and consequently greater burdens on the system needed to fuel such devices to economically competitive power densities.

Although Alcator scaling leads to the seemingly inescapable conclusion that the energy required for adequate penetration by neutral beams of ignition grade plasmas is in excess of 300 keV, in actuality, the situation is not quite so bleak. One can envision optimistic dynamic scenarios in which fueling is judiciously mixed with auxiliary heating and the onset of alpha heating in such a way that one need never penetrate the ignition $n\tau$. Alternatively, there is hope that clever heating techniques such as ripple injection or heating followed by compression, schemes suggested by Jassby, may effectively lower E_b. Another possibility involves the use of direct conversion; in this case, one may operate with comparable efficiencies and use only the full energy component. But, the required energy will very likely remain in excess of 150 keV and the associated R&D program will be substantial.

The ion sources must retain a good species mix to enable a substantial fraction of the energy to penetrate, they must operate at high gas efficiencies to minimize the developmental problems in the vacuum pumping system, and they must be in excess of 100 amps to provide the needed power with a reasonable number of sources. The beam optics requirements will likely be more stringent because of the larger B-coils. Multi-megawatt beam dumps and calorimeters will certainly be needed. The beams must be on for about 5 sec at a repetition rate of one shot every five minutes. This system must also be capable of remote operation and maintenance.

The severe radiation environment is also a formidable design consideration for a neutral beam injection system. Cryopumping becomes impractical if too much heat must be extracted; this limits the use of cryopanels to systems with beam duct diameters $d \lesssim 50$ cm. For duct sizes in this range, the damage to the source caused by neutron streaming increases as d^4 so there is strong motivation to work at high power densities. Such a duct size is probably adequate for positive ion sources, but this saddles the negative ion source development program with a real challenge: even with a power density of 1 kW/cm^2, delivering 60 MW of power requires 30 such injectors, more than can possibly fit around the device.

Let us return to the penetration question armed with somewhat more sophisticated machinery since this question holds the key to the viability of the neutral beam heating approach. The tacit assumption in the entire analysis that $\beta \sim 5$-10% is achievable translates into the statement that the plasma cross section must be noncircular. With noncircular flux surfaces, one must be careful when defining "adequate penetration." For example, the radial distribution of the hot ions produced by injection, the H(r) profile, is no longer an appropriate measure of penetration. The real quantity of interest is the energy deposition profile, and if the flux surfaces are not similar (in the geometrical sense), the H(r) profile does not accurately measure this quantity. More suitable is the condition $R = 1$, where R is the ratio of the total power density in the core of the plasma (flux surfaces with $r < 0.5$ a at the elliptic axis) to the volume averaged power density. Specifically,

$$R = \frac{4}{P_{TOT}} \int_{0}^{a/2} P(r)\, dV \quad ,$$

where P(r) is the radial profile of beam power density resulting from <u>all</u> components of the beam and P_{TOT} is the total injected power.

The R-factor is plotted in Figs. 3, 4, and 5 as a function of deuterium beam energy for various values of the central density assuming $R = 4.2$ m, $a = 1.2$ m, a vertical elongation of 2.7, $Z_{eff} = 1$, and perpendicular injection. The three cases shown are for three different source mixtures: 75-15-10, 75-0-0 (corresponding to removal of the D_2^+ and D_3^+ after extraction from the source), and 90-5-5. The case with only a full energy component of course exhibits the best penetration, but even in this case, 250-300 keV beams are necessary to penetrate the densities needed for ignition. These curves are normalized (for each energy) to the same value of injected power. Normalizing instead to the same value of the power off the line gives rise to Fig. 6 for the 75-0-0 source case. (In this way of presenting the data, the numerical value of R no longer has special significance.) The conclusion to be drawn is that the efficiency of power delivery to the plasma core is maximized for 80 keV $< E_b <$ 150 keV, and, unfortunately, the energy deposition profiles for such energies are strongly peaked toward the plasma edge. This is expected to cause deterioration of confinement and enhanced plasma-wall interaction so such energies are simply not high enough to heat ignition grade plasmas.

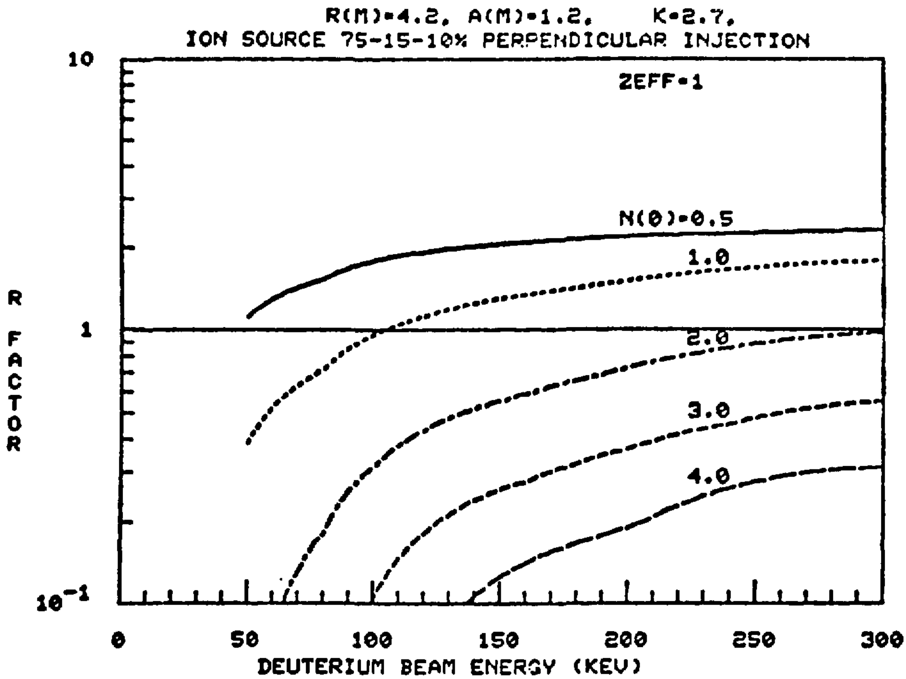

Fig. 3. Power density factor inside r = 0.5 a (from injector)

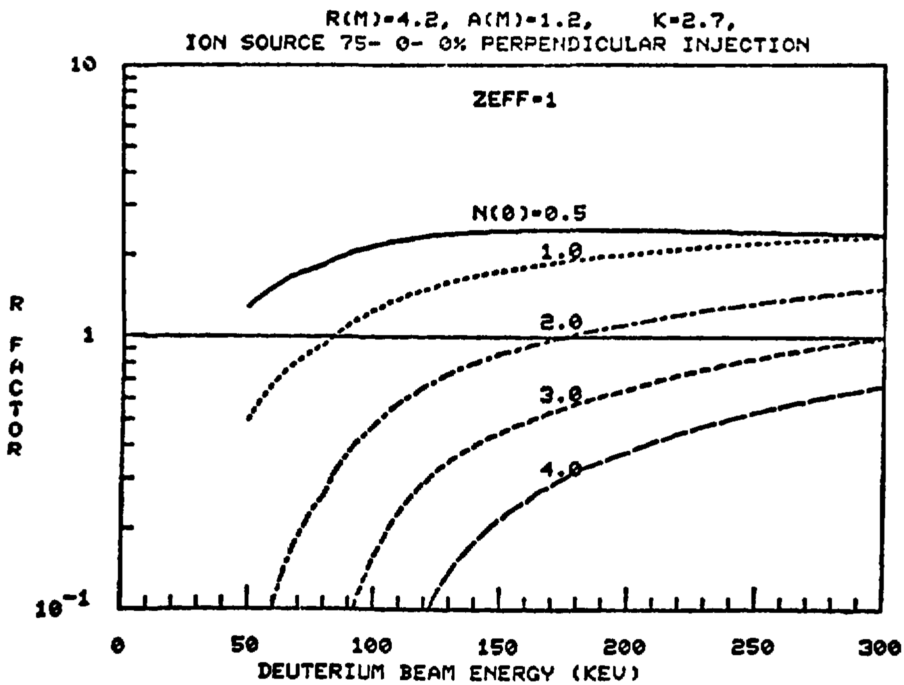

Fig. 4. Power density factor inside r = 0.5 a (from injector)

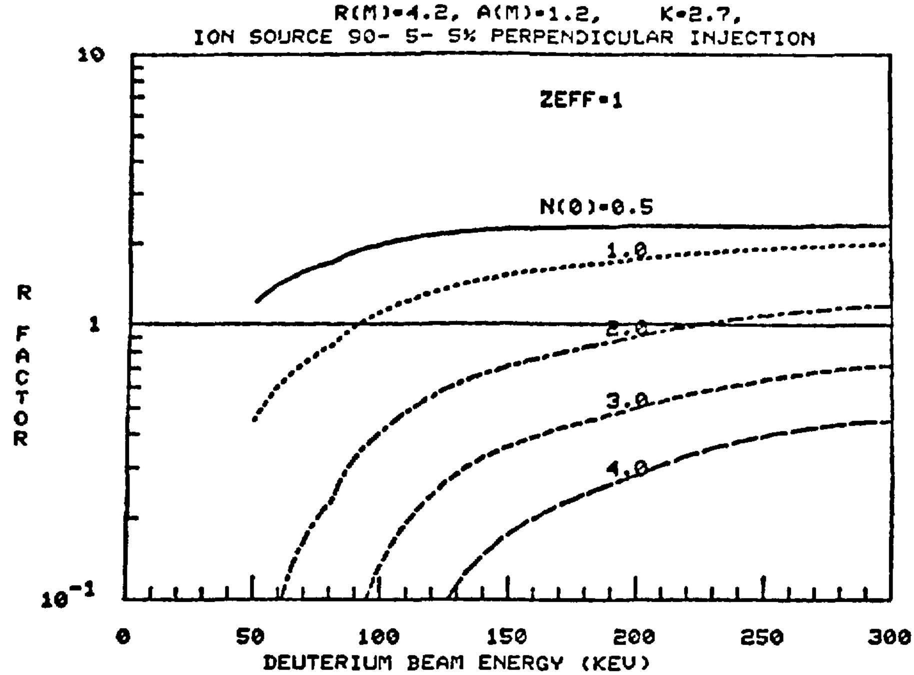

Fig. 5. Power density factor inside r = 0.5a (from injector)

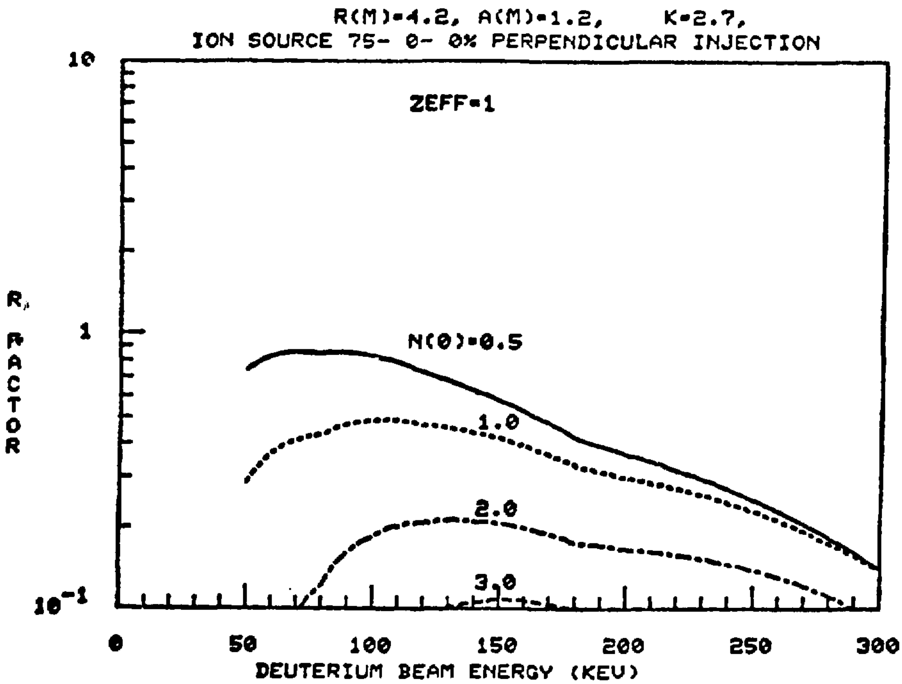

Fig. 6. Power density factor inside R = 0.5 a (from source)

4. HEATING TO IGNITION VIA LANDAU DAMPING OF LOWER HYBRID WAVES

The problems faced by neutral beams are clearly technological ones although ingenuity in the physics can lessen this burden. The situation with rf heating is quite different. The technology, for the most part, is well in hand and the principal concern is the physics. The tack employed in the rf analysis then will be to try to optimize the physics under the assumption that the technology can supply the needed flexibility.

But technology does provide some constraints. The combination of only moderate efficiency power generation and low efficiency power transmission makes frequencies of $\gtrsim$ 50 GHz unlikely candidates for bulk plasma heating of large tokamaks; this rules out electron cyclotron and upper hybrid resonance heating. Frequencies between $\approx$ 1 kHz and $\approx$ 50 MHz are also problematical in that the wave lengths are so long that coil or loop antennas are required, but the response time of the (conducting) wall is too long to permit such structures to be located outside the vacuum vessel. Such coils or loops inside the vacuum chamber can provide excellent coupling of the wave energy to the plasma, but the materials problems, particularly those involving the insulators, are formidable.

The technologically most promising range of frequencies is thus $\approx$ 50 MHz to $\approx$ 50 GHz, a range in which waveguides provide a convenient and efficient means for transmission of the wave energy to the plasma. At the low end of this frequency range is ion cyclotron resonance heating. High ($\approx$ 65%) efficiency, high power, CW systems in this frequency range can be built using off-the-shelf components which will deliver the energy to the plasma chamber via a ridged waveguide. The principal worry about such systems is the control problems associated with tracking the eigenmodes of the system as the density evolves. This may be especially troublesome in conjunction with gas puffing.

We have instead concentrated primarily on frequencies somewhat above the maximum lower hybrid frequency in the plasma

$$\omega_{LH} = \omega_{pi} \left(1 + \omega_{pe}^2/\omega_{ce}^2\right)^{-1/2} \quad ,$$

which is 1.4 GHz for $n = 2 \times 10^{14}$ cm^{-3} and $B_T = 5$ T. Again, high power, high ($\approx$ 50%) efficiency, CW operation is possible with existing technology and even higher efficiencies can be expected with a modest development program. Lower hybrid heating has an advantage of being able to deliver high power densities so that the required access is less than for beam heating. In addition, bends in the waveguide reduce neutronics problems and low transmission losses permit the oscillators themselves to be far removed from the tokamak.

Lower hybrid waves are launched into the plasma by means of grill antennas such as the one shown in Fig. 7. The narrow dimension (the E-field direction) of each waveguide in the antenna is parallel to the magnetic field direction. Experimentally verified codes exist which describe the coupling of this energy to the plasma; very high efficiencies ($\approx$ 90%) have been obtained.

The two variables characterizing a lower hybrid heating system are the frequency ω and the phase velocity v_{ph} parallel to the magnetic field lines. These are determined, respectively, by the klystron and by a set of phase shifters applied to each of the four horizontal sets of waveguides. The physics considerations which enter into the specification of these quantities are summarized in Fig. 8, where the two variables employed are the parallel index of refraction $n_{\parallel} = ck_{\parallel}/\omega = c/v_{ph}$ and $\omega/\omega_{LH/max}$, the ratio of the pump frequency to the maximum lower hybrid frequency (essentially ω_{LH} at $r = 0$). If $n_{\parallel} < 1$, the wave reflects from the $\omega = \omega_{pe}$ layer near the plasma edge. If $\omega/\omega_{LH/max} < 1$, the lower hybrid resonance layer occurs within the plasma and the wave energy will be strongly damped before penetrating to this layer. In fact, linear mode conversion of the wave energy

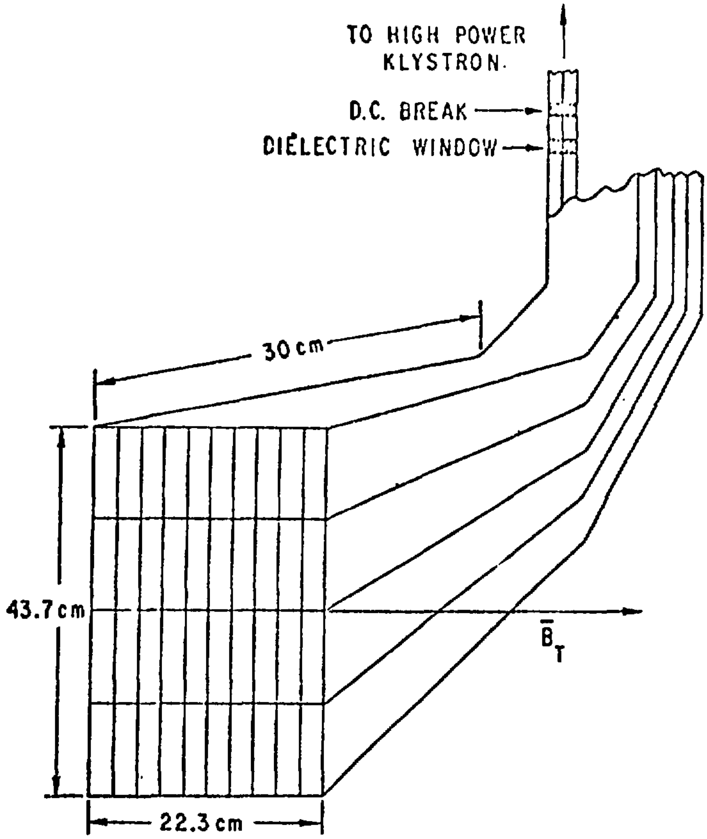

Fig. 7. Lower hybrid grill antenna

to thermal modes, a phenomenon that characterizes resonant absorption, is believed to occur for $\omega/\omega_{LH/max}$ typically up to 2, even for n = 2. Experimentally, parametric instabilities also do considerable damping in the frequency range up to $\omega/\omega_{LH/max} = 2$. This suggests operating at $\omega/\omega_{LH/max} \gtrsim 2$ in order to achieve maximum penetration of the wave energy. In this regime, electron Landau damping, which is of course always present, is believed to be the dominant absorption mechanism. Operating in such a regime has the added advantage of greater theoretical tractability compared to a parametric instability dominated regime.

Independent of the absorption mechanism, however, the lower hybrid waves can penetrate only to a given maximum density n_1 before either encountering

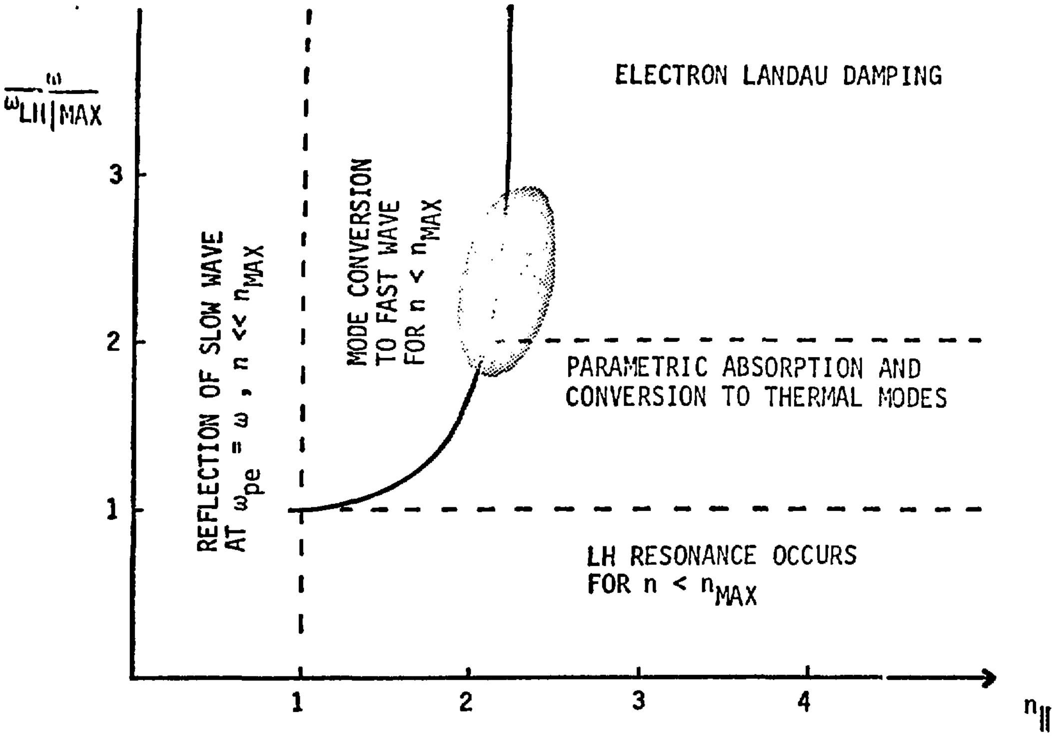

Fig. 8. The key physical mechanisms operating for various choices of $n_{\|}$ and ω. The specific example chosen is $n_{max} = 2 \times 10^{20}$ m^{-3}, $B_T = 5$ T, and a 50-50 D-T mixture. The shaded region is the anticipated operating regime.

the lower hybrid resonance layer or mode converting to the fast wave, thereby being reflected to lower densities. The mode conversion is shown in Fig. 9; the condition $n_{\|}^2 > 0$ must be satisfied for propagation into the plasma. The maximum accessible density n_1 is a function of $n_{\|}$ and $\omega/\omega_{LH/max}$; Fig. 9 illustrates the case for $n = 2 \times 10^{14}$ cm^{-3}. Clearly, smaller $\omega/\omega_{LH/max}$ and larger $n_{\|}$ are preferable to access the high densities needed for ignition. On the other hand, if $n_{\|}$ is large, the parallel wave phase velocity $v_{ph} = c/n_{\|}$ will be so slow that considerable damping occurs on electrons at temperatures much less than those needed for ignition. In this case, the wave energy will be deposited near the plasma edge. In fact, to achieve central temperatures of the order of 7 keV, one needs $n_{\|}$ of the

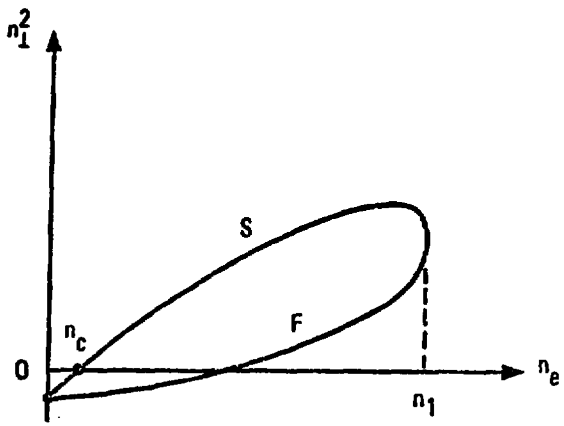

Fig. 9. Propagation of the slow (S) and fast (F) modes

order of 2. Hence, the constraints on heating to ignition via electron Landau damping force operation in the relative narrow window shown as the shaded region in Fig. 8.

If a specific frequency within this Landau damping operating region is chosen, it is in fact possible to derive a condition on the machine parameters to determine whether lower hybrid heating can raise the plasma to the required ignition temperature at the required nτ values. In what follows, the pump frequency is chosen to be twice the lower hybrid frequency, the lowest frequency consistent with dominance of electron Landau damping

$$\omega = 2\omega_{LH/max} \quad .$$

At this frequency, the accessibility condition reads

$$n \le n_1 = 10^{19} \, B_T^2 \left\{ 28 \, n_{\parallel}^2 + 3 - 8 \, n_{\parallel} \left[3 \left(4 \, n_{\parallel}^2 + 1 \right) \right]^{1/2} \right\} \quad ,$$

where n is in m^{-3} and B_T is in tesla. The penetration condition requires that the phase velocity is sufficiently large that the wave may propagate at the given temperature without being heavily damped,

$$v_{ph} \gtrsim 2.8 \, v_{Te} \quad .$$

This estimate includes enhanced wave penetration due to quasilinear plateau formation and the concommitant reduction of Landau damping. Dependencies on minor radius, plasma density, and applied rf power are logarithmically weak. These two conditions may be combined into the form

$$n/B_T^2 \lesssim \text{function of T only.}$$

In fact, a condition of this form may be derived for any fixed value of $\omega/\omega_{LH/max}$. This inequality restricts the region of n-T space that can be heated to the region to the left of the curve labelled RF in Fig. 10.

In order to heat to ignition, this region of n-T space must have a non-null intersection with the region in which the fusion alpha power exceeds the sum of the transport and radiative losses, i.e.,

$$\frac{1}{4} n^2 \langle\sigma v\rangle (T)*3.5 \text{ MeV} > \frac{3nT}{\tau_E} + C_B n^2 T^{1/2} \quad ,$$

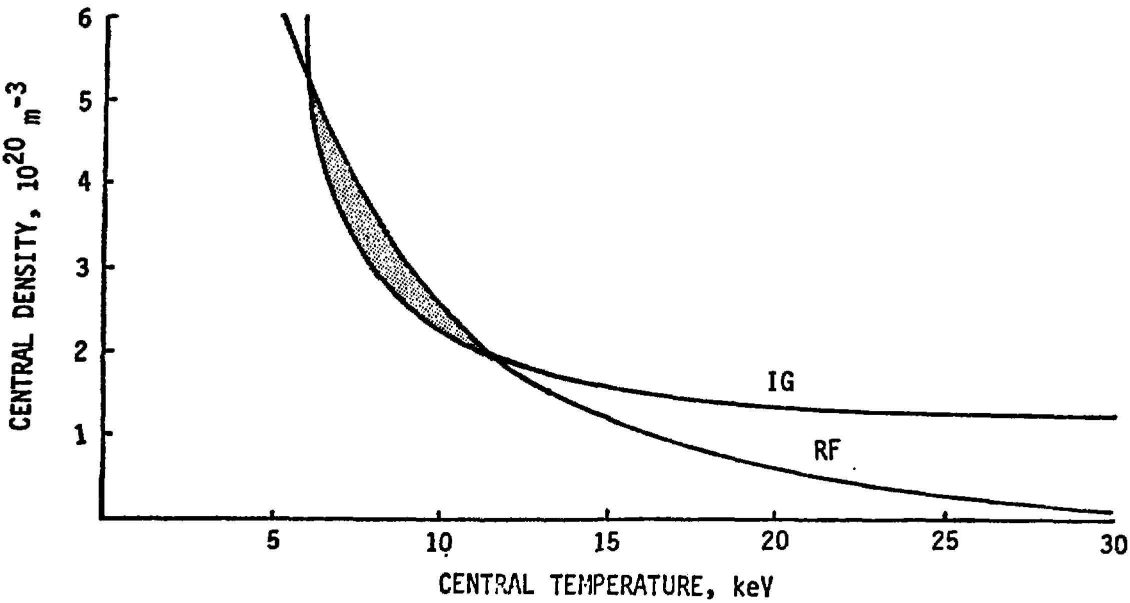

Fig. 10. Constraints on Landau damping as a means of attaining ignition in a Tokamak with a = 2 m, B_T = 7 tesla, and $\omega = 2\omega_{LH/max}$ using Alcator scaling

where the only radiative process considered is Bremsstrahlung. For any Alcator-like scaling law, i.e., $\tau_E = C_T na^2 f(T)$, the global ignition condition amounts to a lower bound on the line-average density, namely,

$$na > \text{function of T only.}$$

The case for true Alcator scaling [f(T) = constant] is exhibited in Fig. 10; ignition is realized to the right of the curve denoted IG. In this representation, n and T are the central values and the ignition condition has been evaluated by averaging over parabolic profiles.

The criterion that lower hybrid heating is capable of heating to ignition is simply the condition that these two curves intersect, i.e., that

$$a\ B_T^2 > \text{function of T only,}$$

obtained by taking the ratio of the above rf and global ignition conditions. Minimizing the right hand side of this inequality gives a simple geometrical constraint on heating to ignition. Numerically,

$$a(m)\ B_T^2\ (T) > 80$$

a constraint which, when taken literally, is difficult to satisfy for TNS designs.

However, this criterion is not as prohibitive as it might appear at first sight. For one thing, the plasma size enters the derivation only through the $\tau \propto a^2$ relation and it is believed that the correct treatment for noncircular cross sections involves the replacement

$$a \rightarrow \left(\frac{2\kappa^2}{1 + \kappa^2}\right)^{1/2} a$$

where κ is the height-to-width ratio. This gives a factor of 1.18 for $\kappa = 1.5$ and 1.34 for $\kappa = 3$, typical doublets. Secondly, one has the freedom to position and phase the grill such that the wave spends most of its time on the high field side, thereby enhancing the effective B_T. Further, it may be possible to optimize the penetration by choosing a frequency somewhat different from $2\omega_{LH/max}$. There is also the distinct possibility that the very powerful quasilinear damping ($\approx 10^4 \times$ collisional effects) may so deplete the resonant portion of velocity space that Landau damping is suppressed, resulting in greater penetration. This effect may be enhanced by propagating only waves antiparallel to the net electron motion associated with the plasma current.

As a final remark, note that the quantity aB_T^2 increases as one progresses in the scenario TNS-DEMO-power plant so that electron Landau damping is very promising as a reactor heating scheme. In addition, if transport is in fact reduced at higher temperatures in conformity with a scaling law of the form $\tau \propto na^2\ T^{1/2}$, as suggested by existing data, then the condition on aB_T^2 is eased by about a factor of 2.

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DIRECTIONS FOR POSSIBLE UPGRADES OF THE MIRROR FUSION TEST FACILITY (MFTF)

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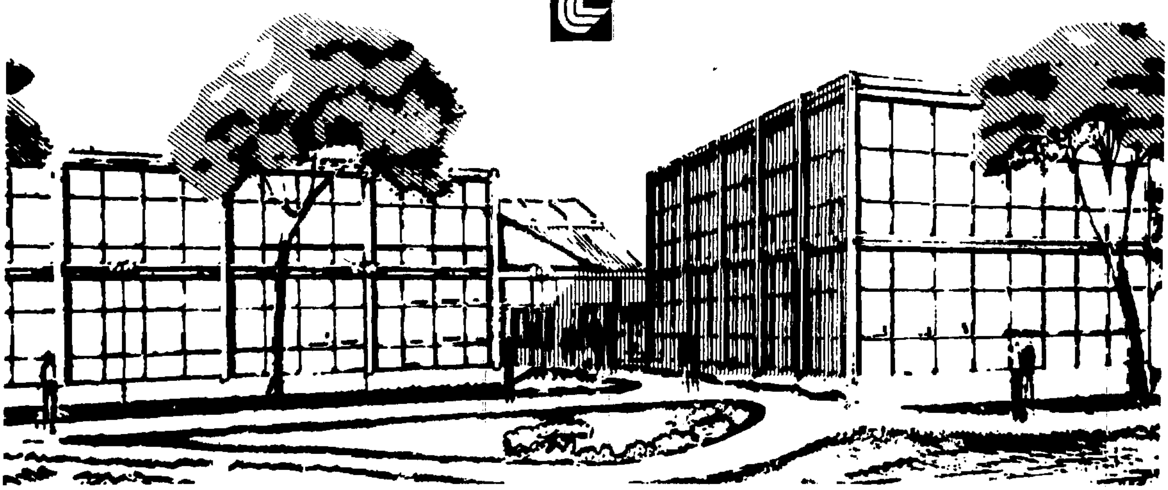

DIRECTIONS FOR POSSIBLE UPGRADES OF THE MIRROR FUSION TEST FACILITY (MFTF)*

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A. W. Molvik, G. D. Porter, J. W. Shearer, and B. W. Stallard
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ABSTRACT

The Mirror Fusion Test Facility (MFTF) may be upgraded by extending the time of plasma sustenance in an approach to steady-state operation and/or by increasing the neutral-beam injection energy. Some parameter bounds for these upgrades are discussed as they relate to a definition of the required neutral-beam development.

INTRODUCTION

We have made a preliminary study of possible Mirror Fusion Test Facility (MFTF) upgrades. Some results of this study are presented here to help define the neutral-beam development required to achieve these improvements in the capabilities of MFTF.

Preliminary design of the MFTF, based on the original Proposal,[1] has been underway for more than a year, and line-item funding for the construction project began in FY1978. Completion is expected in FY1981. The heart of the facility is a large, superconducting "Yin-Yang" magnet that confines a plasma fed by an array of energetic neutral beams (Fig. 1). The main machine parameters and the "base-case" plasma parameters are listed in Table 1. These parameters were specified in the Proposal [1] to achieve the main scientific objective of the facility, which is to extend the theoretical scaling laws for the microinstabilities characteristic of the mirror loss-cone ion distribution. The technological objectives of

* Work performed under the auspices of the U.S. Department of Energy under contract No. W-7405-Eng-48.

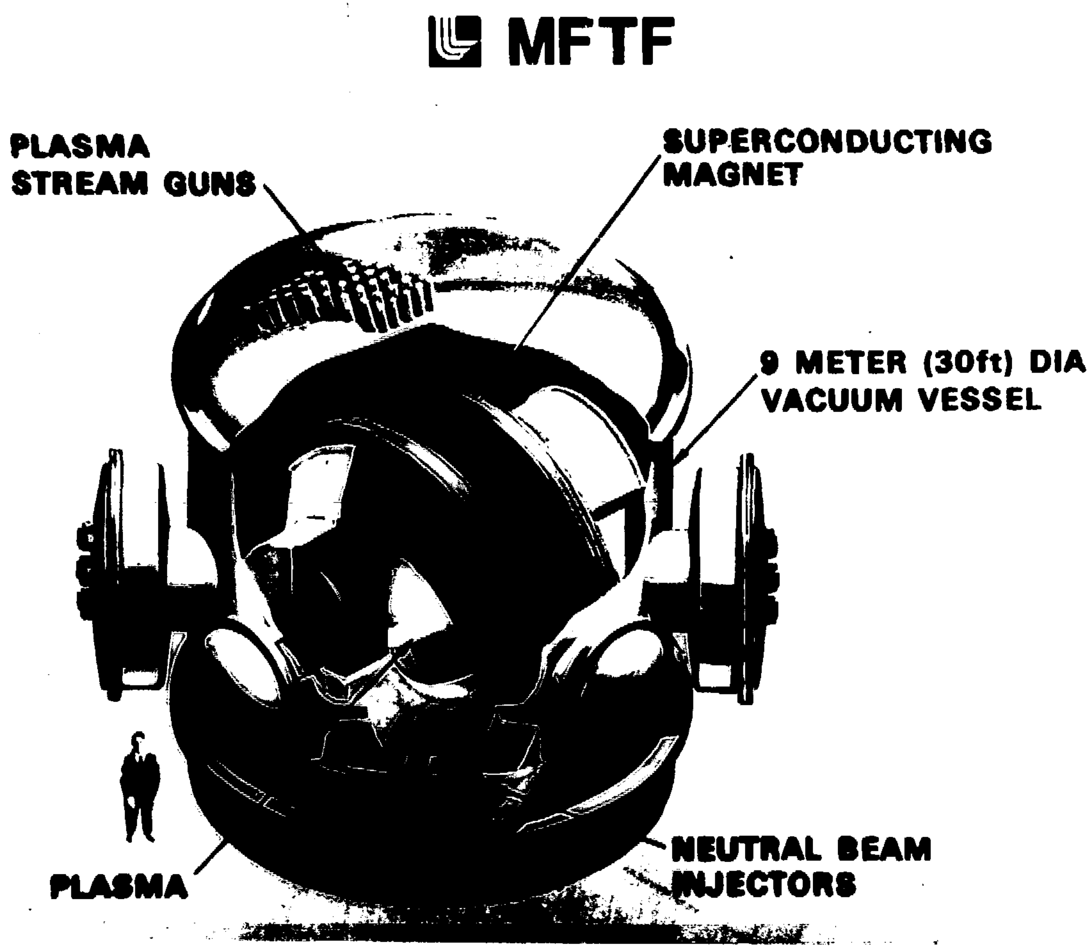

Figure 1. Conceptual drawing of MFTF experiment.

TABLE 1 MFTF PARAMETERS

Plasma (goals)	
$n\tau$ ($cm^{-3} \cdot s$)	10^{12}
T_i (keV)	50
T_e (keV)	1.0
R_p/ρ_i	13
L/ρ_i	100
β	0.5
Machine	
$B_{central}$ (T)	2.0
$L_{between\ mirrors}$ (m)	3.4
Mirror ratio (R_m)	2
Startup	Plasma stream
Startup beams	1000 A, 20 keV
Sustaining beams	750 A, 80 keV

MFTF as described in the Proposal[1] are oriented by our conception of what a mirror fusion reactor would be like. In all of its variations, the mirror reactor involves superconducting magnets, neutral-beam fueling and heating, and steady-state operation. The directions for possible upgrade of MFTF which we have looked at are

- An approach to steady-state operation, and
- Injection at higher beam energy.

These upgrades address an extension of both the scientific and technological goals of the experiment towards the regime of interest for mirror reactors.

APPROACH TO STEADY-STATE OPERATION

With the 0.5-s injection pulse now specified for initial MFTF operation, the getter surfaces used for vacuum pumping during startup are not in equilibrium with the impinging particle flux and continue to absorb these particles after the buildup is completed. If the injection time is extended to tens of seconds, the surfaces will saturate and begin to re-emit an atom for each atom incident, allowing us to evaluate plasma confinement with equilibrium vacuum conditions. Since the MFTF magnet is continuously energized and the vacuum cryopumping is adequate for more than 9 h of continuous injection before renewal is necessary, an extension of the beam injection time would enable us to test all aspects of steady-state plasma sustenance.

We have recently completed a study[2] of the effect of extended injection pulse length on MFTF operation and have identified a need for improved high-power-density beam dumps, in addition to the basic requirement for steady-state beam sources. The 80-kV sustaining-beam power supplies are capable of 30-s operation, at a 10% duty cycle, so that beam sources designed for true steady-state operation can be tested.

One additional limitation inherent in extended-pulse injection was revealed by our study.[2] With deuterium injection, neutron activation of the MFTF apparatus poses a radiation hazard during maintenance when personnel must work within the vacuum chamber. For standard plasma parameters and a typical cycle of machine operation, injection time would be limited on this account to about twenty 30-s injection pulses each day. Although this may be acceptable, any increase in plasma volume or density would aggravate the problem. We therefore desire to retain a hydrogen option with beam sources designed for steady-state injection.

Interestingly, the activation problem is eased with increased beam energy, (assuming a constant plasma volume), primarily because the plasma density decreases at constant β. This is illustrated in Fig. 2, where the normalized neutron production rate is plotted against T_i.

INJECTION AT HIGHER BEAM ENERGY

The motivation for injecting at high energy in mirror systems arises from the opportunity to improve the confinement, which is basically limited by ion-ion scattering into the loss cone. Provided no other loss processes (such as electron drag or nonclassical diffusion) contribute, the confinement parameter is a strong function of ion energy:

$$n\tau_{ii} \approx 2 \times 10^{10} E_i^{3/2} \log R_m,$$

where R_m is the mirror ratio. The ion energy is ultimately limited, in a given machine, by nonadiabatic losses arising from the large orbit dimensions.

We have estimated the nonadiabatic limits for MFTF[1] and reproduce the result here in Fig. 3. The base-case parameters from Table 1 lie well below the limiting curves, even for a short plasma characteristic of the beam size (half length $L_p \approx 34$ cm). For such short plasmas, the maximum ion energy could be increased to $\sim$200 keV at $\beta = 0.5$; for a longer plasma (which could be achieved in MFTF with proper beam aiming) the adiabatic limit is raised to $\sim$440 keV at $\beta = 0.5$. Higher β values provide a greater restriction. Lower β values lead to rapid increase in the adiabatic limit, but stability questions become more serious, as we shall see below.

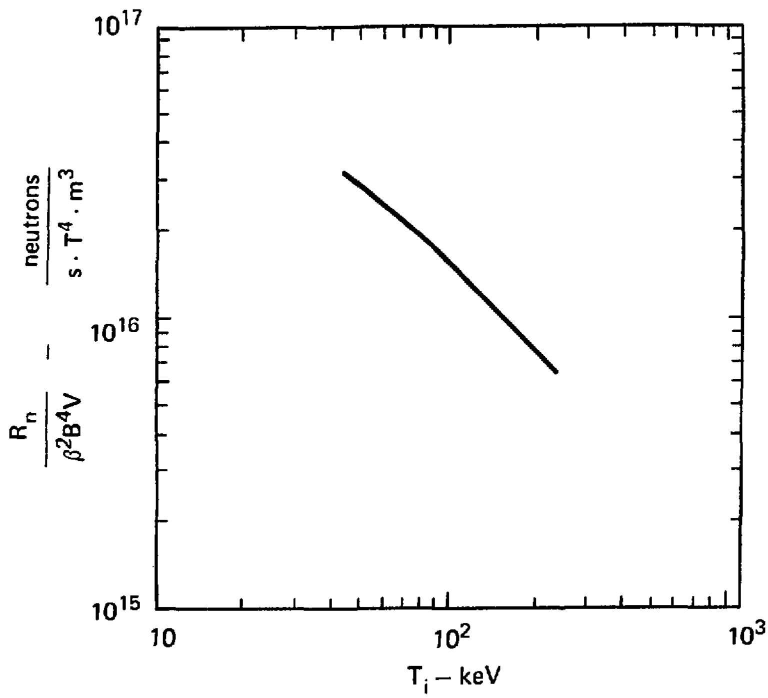

Figure 2. Normalized neutron production rate vs ion temperature.

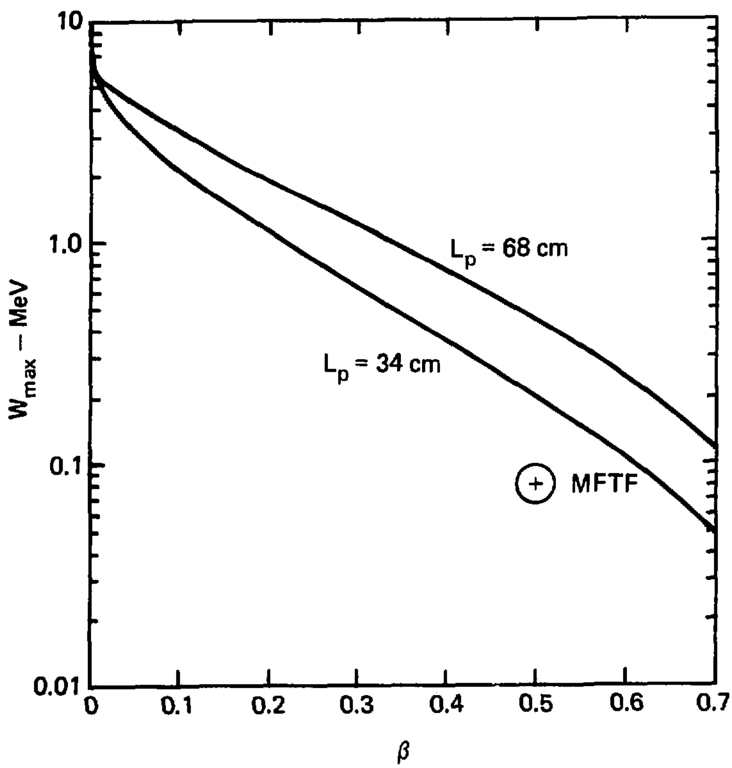

Figure 3. Maximum deuteron energy for adiabatic confinement in MFTF vs plasma β, at a central vacuum field of 2.0 T. L_p is the plasma half-width. The "base-case" MFTF point is shown.

The design of MFTF is oriented towards the study of microinstability scaling laws and, in particular, the scaling of the requirements for stabilization of the drift-cyclotron loss-cone (DCLC) mode. This has been discussed extensively[1,3]; for our present purposes, we reproduce in Fig. 4 a simplified version of a stability boundary curve from Ref. 3. The DCLC mode is driven by the radial density gradient of the plasma and can be stabilized by partial filling of the loss-cone ion distribution with warm plasma. The amount of warm plasma required is determined by the scale length of the radial density gradient measured in ion gyroradii, as seen in Fig. 4. The 2XIIB plasma, with $R/\rho_i \sim 2\text{-}3$, requires substantial warm plasma, whereas our MFTF base case, which is scaled at $R/\rho_i \approx 13$, is expected to require about an order of magnitude less for stability. In 2XIIB, the energy confinement is dominated by electron drag on the ions, with a confinement parameter given by

$$n\tau_E \approx 1.4 \times 10^{12}\ T_e^{3/2}.$$

Reduction of the warm-plasma component in MFTF would allow T_e to rise from the 2XIIB value of $\sim$140 eV to $\sim$1 keV (Fig. 4), improving the confinement correspondingly. Further approach to the ion-ion scattering limit then depends on increasing the scale length of the radial gradient. In a plasma of fixed radius, we can "flat-top" the density distribution so that the plasma core has a very large effective radial scale size. The warm-plasma requirement in the core would then be reduced, and T_e and confinement would increase. Successful flat-topping would then permit us to increase the ion energy in the core even though the number of orbits across the plasma would be reduced. The sharp plasma edges would, of course, be maintained by substantial beam injection to overcome the relatively poor edge confinement.

That such a configuration can be realized is illustrated in Fig. 5. Using our numerical rate-equation code,[4] we have calculated an equilibrium distribution with an injection of 30 A of D^0 at 200 keV distributed uniformly to 25 cm, with 100-A injected beam at 80 keV maintaining the plasma edge. Charge-exchange, wall refluxing of gas, and finite orbit dimensions

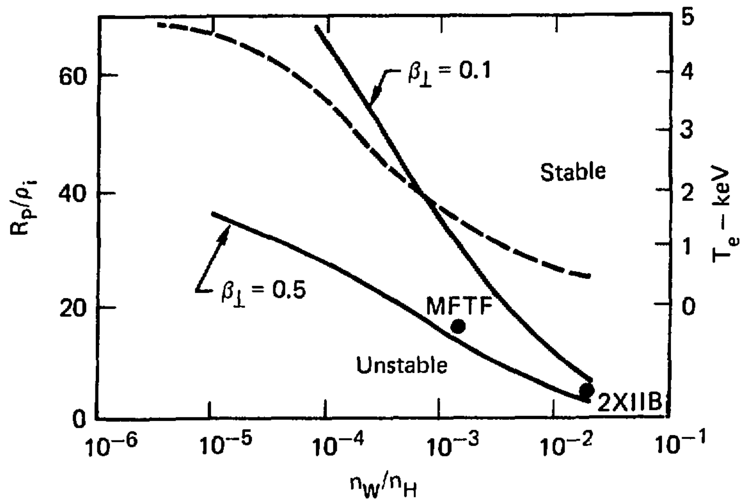

Figure 4. Simplified stability boundaries (solid curves) for the DCLC mode. R_p/ρ_i is the scale size of the radial density gradient, n_W/n_H is the fraction of warm plasma. The dashed curve gives an estimate of T_e for various fractions of n_W in MFTF.

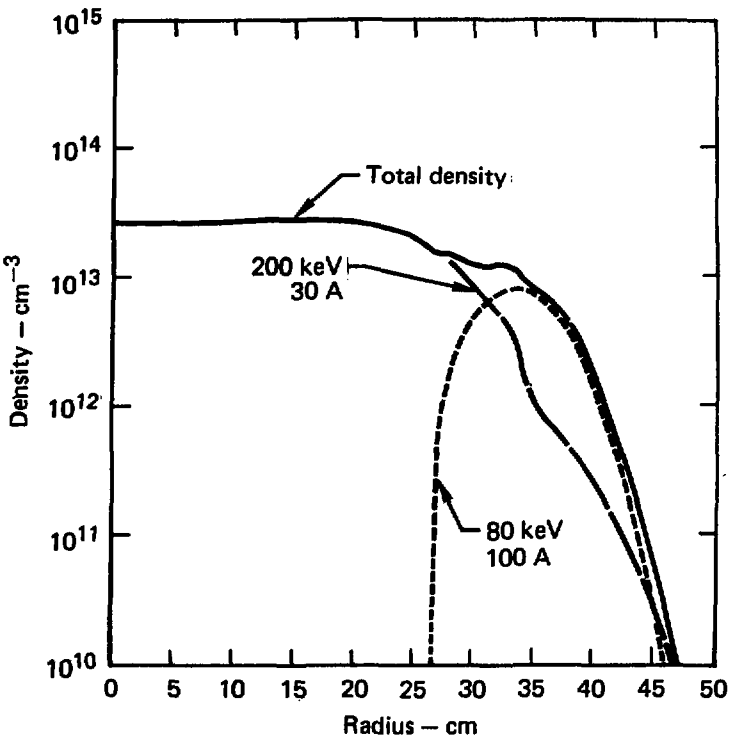

Figure 5. Results of numerical injection calculation. Dashed curves show radial distributions of trapped 200-keV and 80-keV deuterons; solid curve shows the resultant "flat-topped" distribution.

are included in this calculation, with confinement specified as classical (ion-ion scattering) for the core and with $n\tau = 3 \times 10^{11}$ cm^{-3}-s for the edge (the latter is scaled from 2XIIB parameters). The main point of this calculation is that a flat-topped distribution can be achieved in MFTF with reasonable beam parameters.

We cannot predict with certainty the degree of success to be achieved by the flat-topping technique. Consequently, to estimate the beam required to sustain the plasma core at higher energy and the $n\tau$ that could be realized, we have retained T_e as a parameter in our calculations. These estimates have been made[5] for a plasma at constant $\beta = 0.5$ and core radius = 30 cm, with a half-length of $L_p = 60$ cm; we assumed a core stabilized by warm plasma and an edge maintained by other beams (such as the 80-keV beams of Fig.5). Loss processes are ion-ion scattering and electron drag, combined in these calculations with appropriate integration and averaging over the ion distribution.

The $n\tau$ to be expected is shown in Fig. 6, with 2XIIB and MFTF base-case points blocked in. Above ~400 keV, curves are dashed to indicate failure of adiabaticity at $\beta = 0.5$. The neutral-beam current required to maintain the core can be determined after calculating the average ion energy and density (at constant β). Curves for constant core radius, R = 30 cm, are shown in Fig. 7. The current requirement levels off at energies above ~300 keV because the increasing $n\tau$ is balanced by the decreasing beam absorption (from lower density at constant β).

For these curves, the number of gyro-orbits across the core decreases as E_B is increased, perhaps altering the effectiveness of the flat-topping. If instead of keeping the core radius constant, we maintain R/ρ_i above some limit, the plasma size and required current will increase. An example for $R/\rho_i \geq 8$ is shown by dashed lines in Fig. 7. For $T_e = 4$ and 8 keV, these curves terminate at ~500 keV, indicating a plasma size exceeding the maximum radius we can support in MFTF, $R \approx 60$ cm. However, this energy is already above the adiabatic limit. The important result from Fig. 7 is that the current requirements drop to very reasonable values as E_B is raised above 100 keV, <u>if</u> T_e can be increased above ~2 keV. When these data are translated to beam power requirements (Fig. 8) we observe a broad minimum in the range of E_B between 100 and 300 keV.

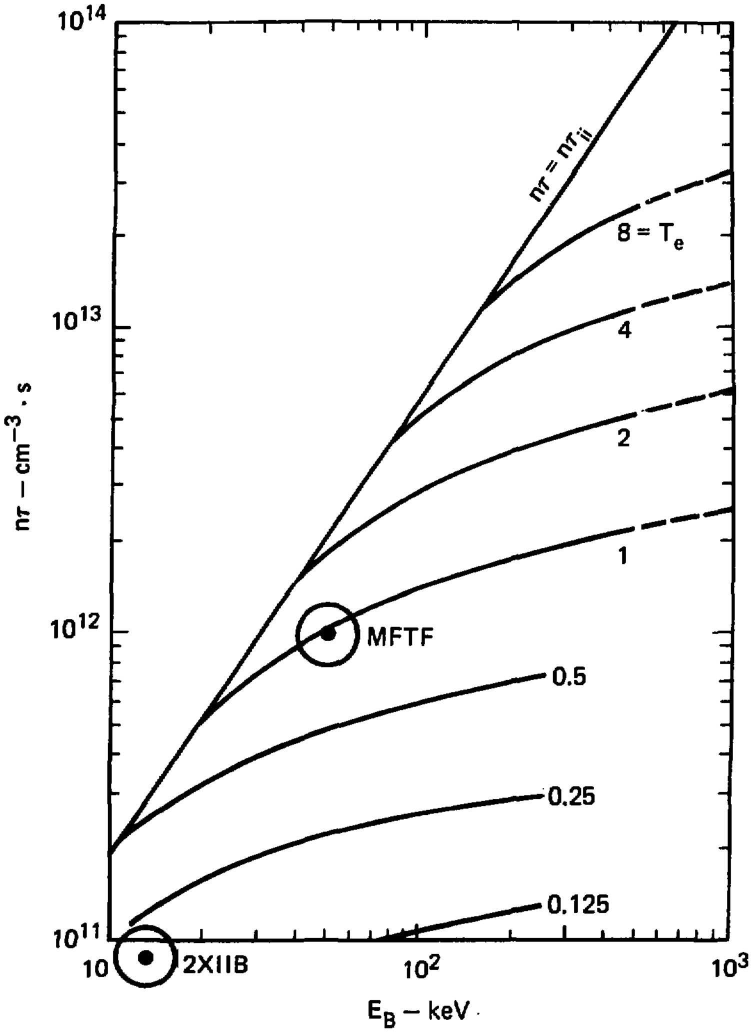

Figure 6. Estimate of confinement parameter, $n\tau$ vs beam energy E_B.

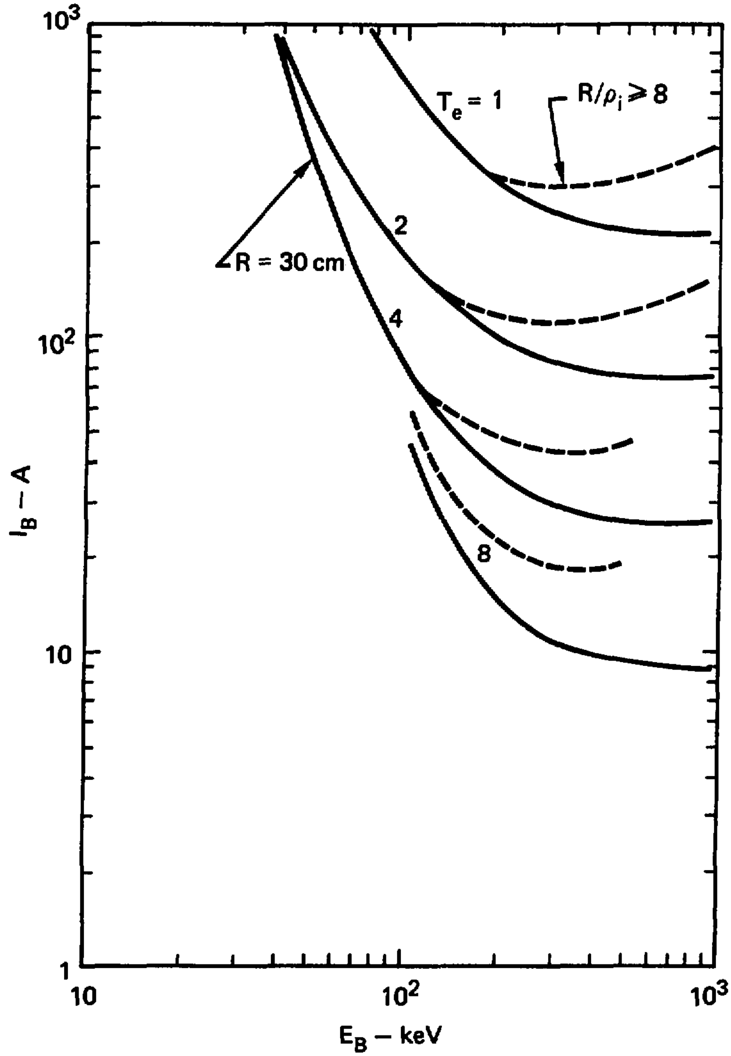

Figure 7. Neutral-beam current I_B required to sustain core density at β = 0.5 vs beam energy E_B. Solid curves are for constant core radius R = 30 cm; dashed curves maintain $R/\rho_i \geq 8$.

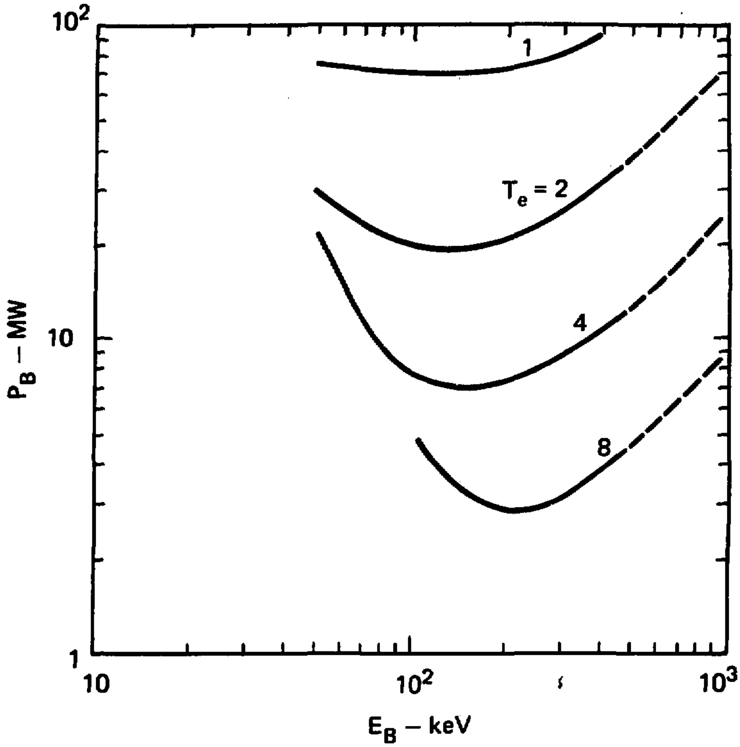

Figure 8. Neutral beam power P_B vs beam energy, E_B. Curves are for core radius constant at R = 30 cm.

CONCLUSIONS

Extension of the beam injection pulse to $\sim$30s in MFTF would enable us to test plasma sustenance in steady state, including the relevant technology required for mirror reactors. For this test, we require neutral-beam sources designed for steady-state operation with an option for either hydrogen or deuterium extraction.

Increasing the beam injection energy to the range of 200-300 keV could increase the confinement parameter by an order of magnitude (to $n\tau \approx 10^{13}\ \mathrm{cm^{-3}\cdot s}$) above the MFTF base case. This upgrade is dependent on a successful flat-topping of the density profile, leading to an increase in T_e in accordance with our understanding of the DCLC stabilization requirements. Neutral-beam currents of 10 - 50 A incident on the plasma would be required, depending on the degree of success in raising T_e.

Following the MFTF construction completion date in late 1981, and planning for a year of initial operation to explore the base-case regime, our need for beam upgrade could be as early as 1983. We expect to refine and extend these studies during the present FY1978.

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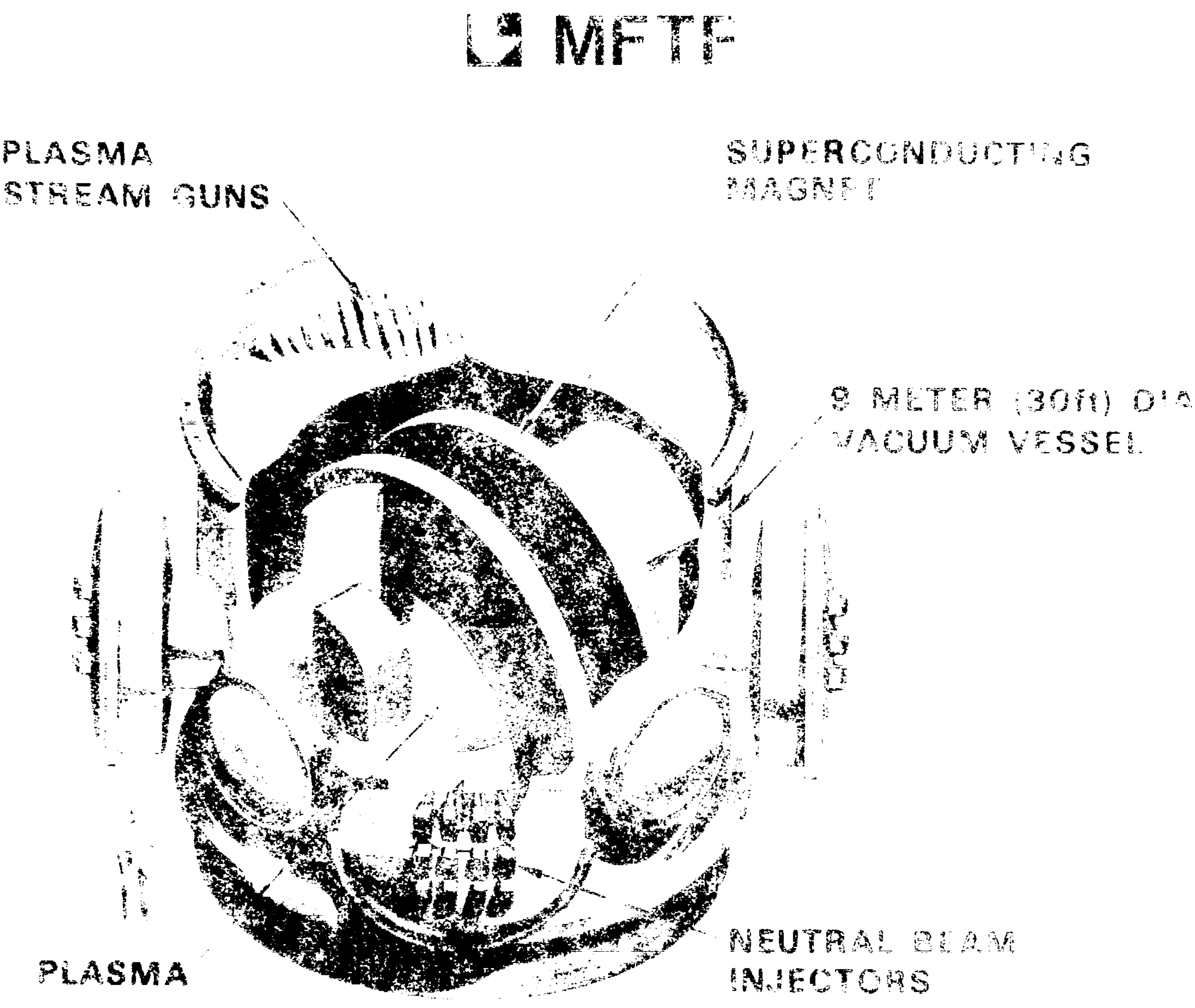
MFTF
PLASMA
STREAM GUNS
SUPERCONDUCTING
MAGNET
9 METER (30ft) DIA
VACUUM VESSEL
NEUTRAL BEAM
INJECTORS
PLASMA

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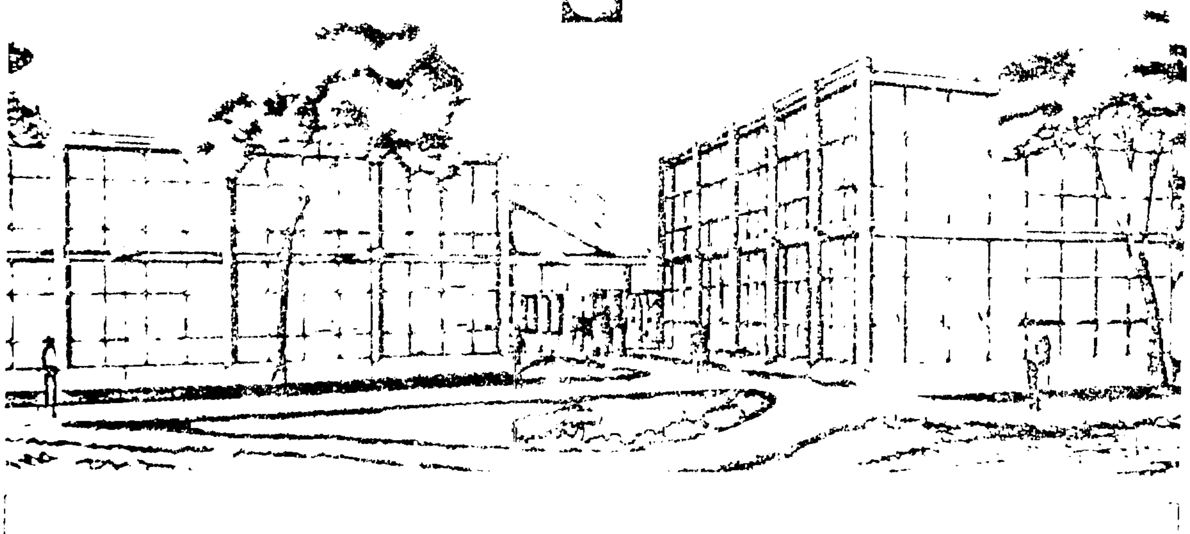

7

NEUTRAL BEAM REQUIREMENTS FOR MIRROR REACTORS*

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December 1, 1977

ABSTRACT

The neutral beam requirements for mirror reactors as presently envisioned are 200 keV for the Field Reversed Mirror (FRM) and 1200 keV for the Tandem Mirror (TMR). The hybrid version of the Standard Mirror, FRM and TMR require 100-120 keV. Due to the energy dependence of atomic processes, negative ions should produce neutrals more efficiently than positive ions above some energy and below this energy, positive ions are probably more efficient. This energy is probably somewhere between 100 and 150 keV for D°, and 150 and 225 for T°. Thus we conclude that hybrid reactors can use D^{+} ions but all of the fusion reactor designs call for D^{-} ions to make the neutral beams. Trends in the energy requirements are discussed. The hardening of neutral beams against neutron and gamma radiation is discussed.

* Work performed under the auspices of the U.S. Department of Energy under contract No. W-7405-Eng-48.

1. Introduction

Mirror reactors depend on non-thermal ion energy distributions which in all the reactor designs to date are produced by neutral beam injection. The electrons can be in thermodynamic equilibrium but the ions must be maintained in a nonthermal steady state. Because the ions cannot be allowed to come into thermodynamic equilibrium, heating methods which first heat electrons, which in turn heat ions, will not do. Ion cyclotron R.F. heating of ions in principle might be effective but so far has not been of much interest. The neutral injection in the mirror designs to date are the only heat source other than alpha particles; however, auxiliary heating could reduce the neutral beam energy and power requirements if that were desirable but could not eliminate their use.

In the discussion to follow, the beam requirements such as energy and power, are reviewed. The shielding requirements of the neutral beam components from neutron and gamma radiation are discussed, as well as a status of the past and future shielding design efforts.

2. Mirror Reactor Conceptual Designs

The early reactor designs done at LLL (1970-73) were based on the standard mirror configuration (Yin-Yang coil, steady-state neutral beams, and direct energy conversion) and employed 600 keV injection due to the end loss direct converter understanding at that time. The hybrid reactor used 100 keV injectors. The next reactor design done in 1973-74 was on the FERF (Fusion Engineering Research Facility) whose purpose was primarily as a neutron source for material testing. It did not have a blanket or energy recovery as power production was not its purpose. The injection requirement was 65 keV D° and 97 keV T°.

The next design done in 1975 was a standard mirror hybrid with 100 keV D° injectors. Next was the standard mirror reactor design of 1976 with a careful optimization of all parameters to minimize the cost of power. Q came out to be only 1.1 and the injection energy, which was best, was 150 keV. This was essentially the first negative ion injector on a mirror reactor design. In 1976 and 1977, we designed the Field Reversed Mirror reactor (FRM) and the Tandem Mirror Reactor (TMR). The energy for the FRM was 200 keV and 1200 keV for the TMR. The hybrid version of the standard mirror, FRM and TMR were 120 - 125 keV. These parameters are summarized in Table 1. The ratio of gross electrical power to net electrical power (second column of Table 1) is an economic indicator discussed in the next section. The standard mirror is clearly uneconomical but still shown for completeness.

The standard mirror reactor is shown in Fig. 1, the FRM in Fig. 2, the TMR in Fig. 3, the standard mirror hybrid in Fig. 4, and the TMR hybrid in Fig. 5.

The injection energy for each reactor listed in Table 1 was arrived at by folding in the physics model current at LLL at the time, with the engineered system current at the time, in such a way that all free parameters and, in particular, injection energy were optimized to minimize electrical power. In the hybrid most of the electrical power came from fission reactors that burned the hybrid-produced fissile fuel.

The physics and engineering models which led to the quoted injection energies for the three cases is complex for each case, and will not be discussed here. However in Sec. 4 the trends are discussed which are likely to affect injection energy as mirror reactor concepts evolve.

TABLE 1

Injector Parameters for Mirror Reactors

	Q	P_{GROSS}/P_{NET}	η_i	$W_{D^{\circ} INJ}$	POWER PER UNIT	UNITS
STANDARD MIRROR	1.1	4.4	.8	150 keV	270 MW	4
FIELD REVERSED	5	1.5	.7	200	4	12
TANDEM	5	1.7	.7	1200	120	4
STANDARD MIRROR HYBRID	.7	3.2	.7	120	60	4
FRM HYBRID	~2	~1.4	.7	~120	~4	12
TMR HYBRID	1.8	1.4	.7	125	70	2

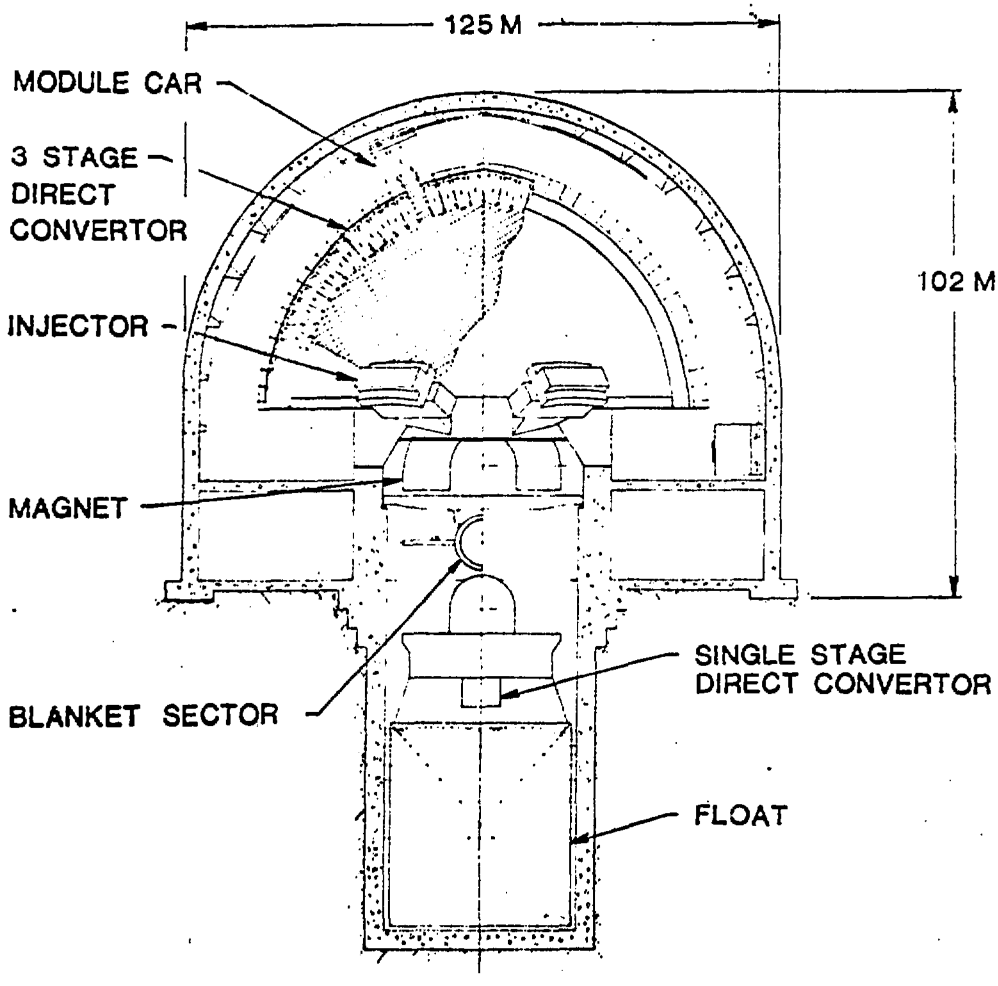


Figure 1. Mirror Fusion Reactor

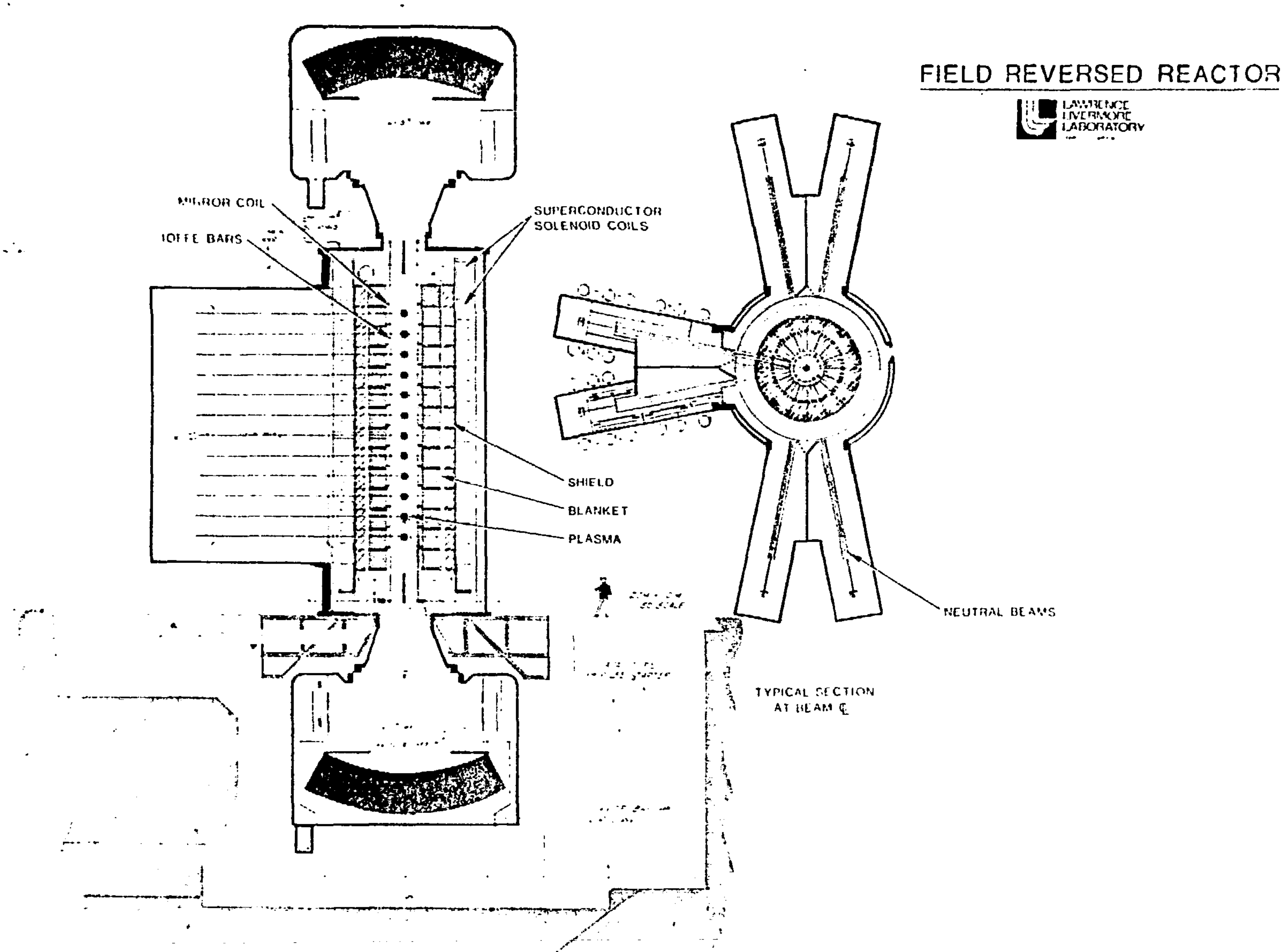

FIELD REVERSED REACTOR
LAWRENCE LIVERMORE LABORATORY
MIRROR COIL
IOFFE BARS
SUPERCONDUCTOR SOLENOID COILS
SHIELD
BLANKET
PLASMA
NEUTRAL BEAMS
TYPICAL SECTION AT BEAM ℄

Figure 2.

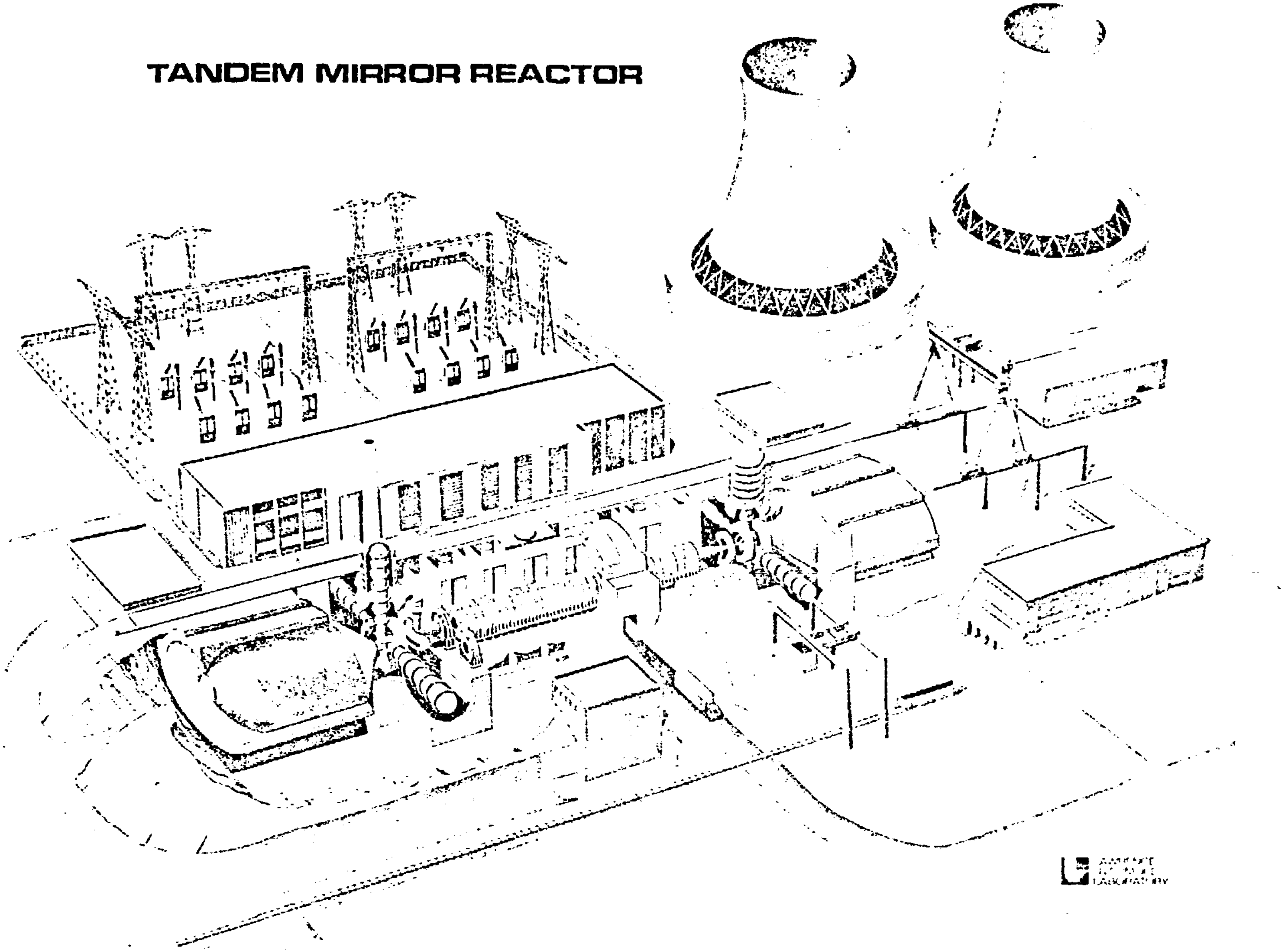


Figure 3.

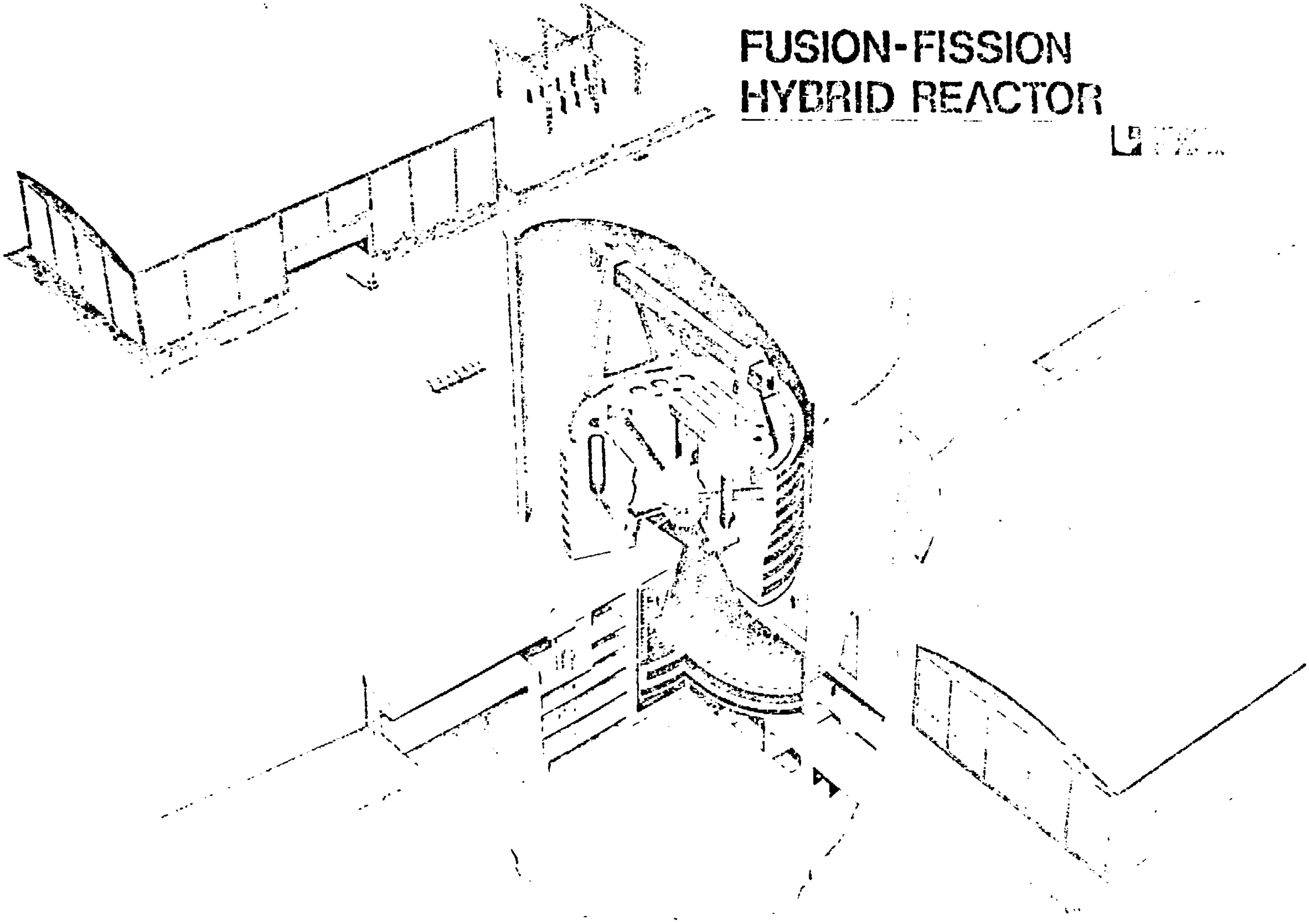


Figure 4.

TANDEM MIRROR HYBRID REACTOR

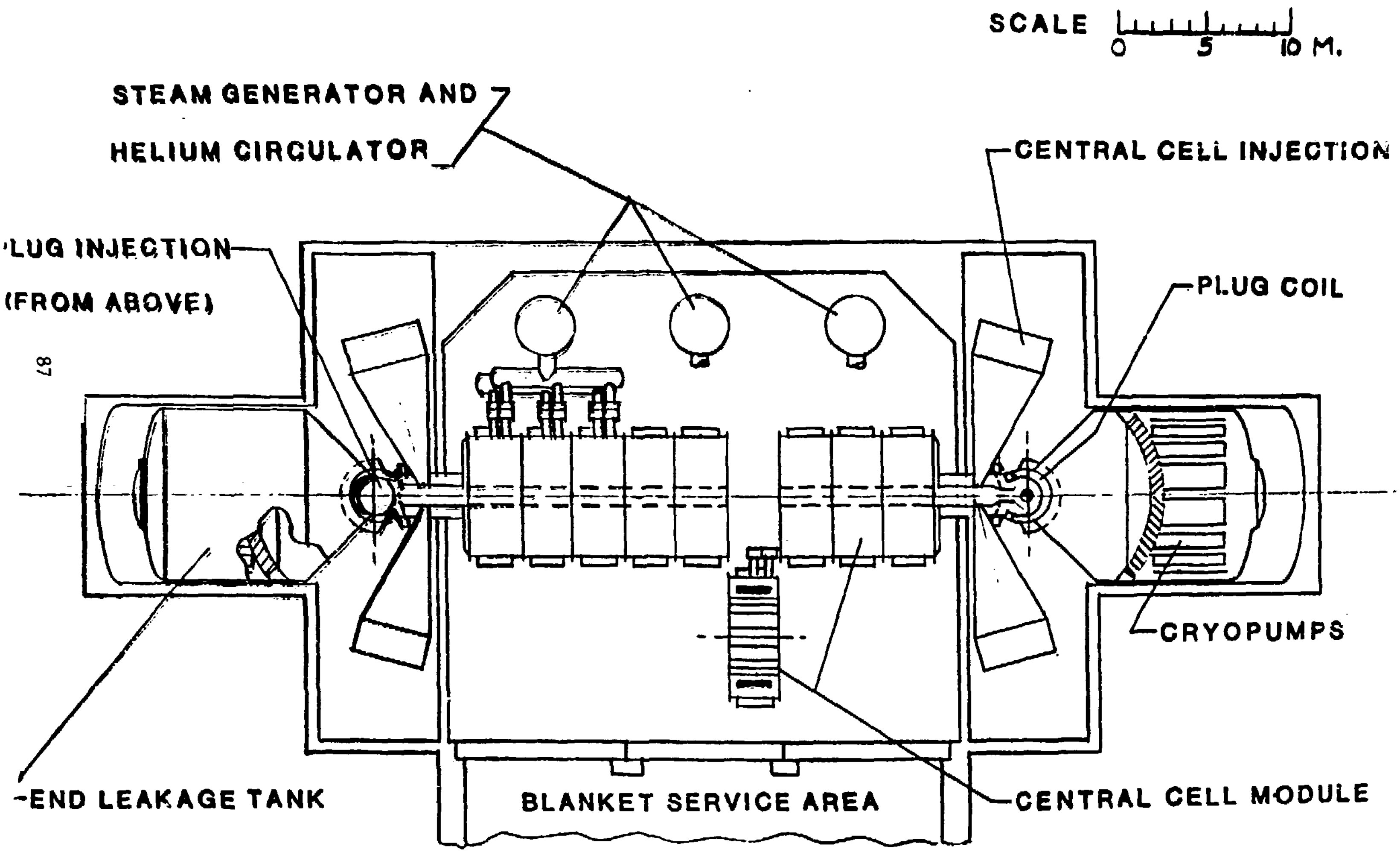


Figure 5.

3. Q Requirements for Mirror Reactors

The recirculation of power in power plant tends to degrade the economic competitiveness. A plant which can sell 0.8 unit of power for 1.0 unit of power generated will enjoy an overwhelming competitive edge over a plant that can sell only 0.5 units of power, if everything else is the same. We can quantify the above example and then make several observations.

We take an injected reactor which amplifies injected power by a factor of 1 + Q, where Q is the fusion power divided by the injected power. We assume the neutrons deposit M-times their kinetic energy in the blanket. The direct converter recovers the injected power plus the alpha particle power with an efficiency, η_{DC}. The undirect converted power and the blanket power is converted to gross electrical power, P_{Gross} with an efficiency, η_{th}. A fraction of the gross power, $f_{recirculation}$, is fed back to the injector which converts this electrical power to plasma energy with an efficiency, η_i. The ratio of gross-to-net electrical power, G is given below:

$$G = \frac{P_{gross}}{P_{net}} = \frac{1}{1 - f_{rec.}}$$

Based on judgement of the kind of performance that seems likely, we have chosen the following parameters as typical:

$\eta_i = 0.7$, $\eta_{DC} = 0.5$, $\eta_{th} = 0.4$. Under the above simplifying assumptions the G versus Q values for three cases are plotted in Fig. 6. . Case 1 is a fusion reactor where M is chosen to be 1.2. Case 2 is a hybrid reactor designed to produce ^{233}U as well as some ^{239}Pu with M = 5. Case 3 is a hybrid designed to produce ^{239}Pu with M = 10.

The curves each have a vertical and horizontal asymptote. The vertical asymptote occurs at break-even values for Q. The horizontal asymptote shows the idea of diminishing returns for further increases in Q.

For example, a fusion reactor, under the above assumptions which are felt to be reasonable by the author, must have Q ≥ 2 to break even and Q values above 10 result in small further improvements. For a Pu producing hybrid, the break-even Q is about 1/4 and Q above 1.8 results in small further improvements. For ^{233}U, the Q values are about 1/2 and 3.4.

For G > 2 the reactor is uneconomical. For G < 1.2, the Q value is high enough so that it is not a major issue in economics. The value of 1.2 is, of course, an arbitrary cutoff of a continuous variable.

Based on the Q values for the conceptual designs to date, as shown in Table 1, we conclude:

- The mirror fusion reactor Q of 5 seems somewhat low and 10 is probably needed*.
- The hybrid Q value of 2 is already high enough.
- The standard mirror hybrid with Q of 0.7 has a somewhat large economic penalty.

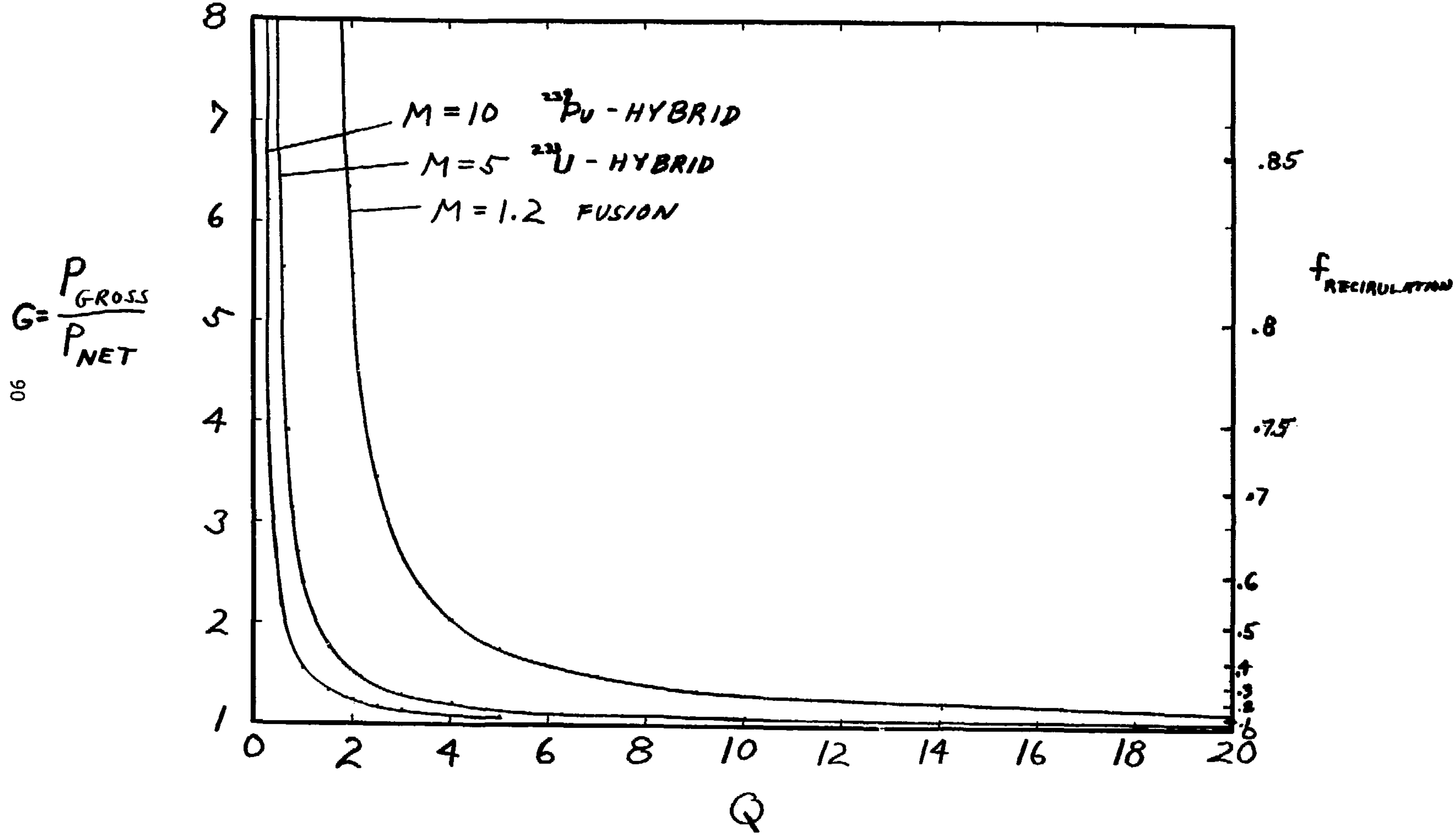
The hybrid because its saleable product is fissile fuel as well as electricity can perhaps tolerate a somewhat lower Q than shown above, but not by much due to the incipient rise of the curve for falling Q values.

Ways should be found to increase Q to about 10 for the Tandem and Field Reversal concepts which do not sacrifice too much other economic factors like power density. Recently there is encouragement for larger size FRM's when measured in gyroradii. Reduced power density and increased injection energy for penetration may result. Heating electrons (ECRH, RF, e-beams) in the TMR may result in increased Q values.

* A conclusion D. Steiner came to previously.

P_{GROSS} VERSUS Q

FIG 6



4. Trends likely to effect injection energy

a. Higher Q

As discussed in Sec. 3, Q as high as 10 may be needed for an economical reactor. Stating this goal and obtaining it are two different things; however, new ideas and improvements in old ideas seem to be yielding results as exemplified in the TMR. Assuming a way can be found to achieve $Q \sim 10$ without giving up other factors, like power density, then considerable alpha particle heating will occur and, perhaps, almost completely remove the heating role of neutral beams. Then the neutral beam would play the more single-purpose role of maintaining the non-thermal ion energy distribution, such as circulating current in the FRM and end plugs for the TMR. What effect this will have on injection energy is not clear, but the likelihood of the injection energy dropping much below 150 keV seems unlikely.

b. Beam Penetration

In order to penetrate a thick plasma, the beam energy must be high. The FRM, which is 5-ion orbits across, requires as high as 200 keV due in part to penetration. There is some indication that plasma stability may permit a larger plasma size (10 orbits across). This will have the beneficial effect of substantially raising Q from its present 5 but, at the same time, force the injection energy up to permit adequate beam penetration. On the other hand, other means of penetrating plasmas should be explored, such as ion cross-field flow, for example, which would allow more optimal energies like $\sim$100 keV.

c. TMR - auxiliary electron heating

The plug injection as now designed requires 1200 keV beams. Logan thinks auxiliary electron heating could reduce this injection energy to as low as 400 keV. The heating could be R.F., ECRH, or e-beam, but must be efficient 50-70% and low-cost, $\sim$0.3-0.5 \$/W.

d. Power density versus energy

The power density is proportional to $\frac{\langle\sigma v\rangle}{T^2}$ and peaks at about 20 keV and falls rapidly above 20 keV. The reaction rate parameter $\langle\sigma v\rangle$ peaks at about 100 keV and falls as $W^{-1/2}$. From this one observes that the ions have no energetic need in the plasma to be at energies above 100 keV. If injection is at energies much above 100 keV, the reason is to heat electrons or to penetrate or maintain non-thermal velocity distributions as in FRM and TMR for confinement. The above observation is,in my opinion, profound but meaningless if one does not have the freedom to apply it, e.g. TMR vitally needs high-injection energy for the end stoppering.

e. Ignition

Ignition trades a heating problem for a fueling problem (a bargain!) but usually results in low-power density which is a serious economics penalty and must be dealt with. The TMR, as we now understand it, can ignite ($Q \sim 10$ for 2000 MWe) but the minimum cost power occurs for the driven TMR with $Q \sim 5$ due to a tradeoff with power density. Similar tradeoffs have been discussed by Jassby for the slightly-driven Tokamak.

5. Hardening of neutral beams for neutron and γ radiation

An important requirement in design of neutral beam injectors is the protection of the injector components against the hostile radiation environment. Because neutrals must have line-of-sight to the fusing plasma, the neutrons can stream up the beam line. In principle, the ion source and accelerator structure can be located out of line-of-sight with bending magnets to guide the beam around a corner. This seems bulky and likely to run into severe beam-optics problems. Thus the line-of-sight injectors will necessarily be in a rather intense radiation environment. The vulnerable components are insulators, semiconductor devices, and cryopanels. Proper shielding designs can adequately protect the vulnerable components and the metal electrodes that see the highest radiation loads are not expected to be a problem, because the flux is low (100 times lower than at the blanket) due to geometry and distance from the source, causing a dilution.

Insulators: Dielectric breakdown due to high levels of ionizing radiation must be avoided. Structural damage due to accumulated radiation doses will determine replacement time.

Semiconductors: Solid-state lasers and rectifiers used in some injector designs must be well shielded.

Cryopanels: These are made of metal and although not subject to radiation damage, will suffer from nuclear heating which must be shielded to reduce refrigeration power to practical values.

Over the past 5 years at LLL increasing attention by the reactor-study group has been given to the effects of radiation on neutral injectors. The evolution of neutral beam hardening will be given briefly.

In 1973, the first reactor injector design was done by G. Hamilton for a FERF (Fig. 7). Shielding was provided primarily for the magnet as shown in Fig. 8. T. Wilcox then made a detailed analysis using Monte Carlo codes, to calculate neutron and gamma fluxes at many locations. The model is shown in Fig. 9. The point labeled '9' is the location of the source with its vulnerable high-voltage insulators. The neutron dose rate there is 2×10^{8} rem/h(1.3×10^{12} n cm^{-2} sec^{-1}), and the gamma rate is 1×10^{6} rem/hr. This corresponds to 0.05 W cm^{-3} in stainless steel.

The machine was designed with the idea that the machine itself including the sources were part of neutron-damage studies.

The next injector was designed by Fink, Hamilton, and Barr in 1975 for the hybrid reactor. The individual beams were separated just enough to put shielding in between the individual beams as shown in Fig. 10. This added shielding essentially eliminated line-of-sight bombardment of cryopanels and direct converter insulators and somewhat attenuated the radiation seen by the ion sources. The neutron flux at the sources was estimated to 10^{13} n $cm^{-2}sec^{-1}$, however no detailed Monte Carlo calculations were made.

The next injector design done in 1976 by Fink, Barr, and Hamilton, was a 150 keV D^{-} neutral injector shown in Fig. 11. The major change in this design from the point of view of shielding, was to recess the high-voltage insulators into the shielding block, thus eliminating any line-of-sight (14 MeV neutrons) and greatly attenuating the radiation environment. Low-voltage insulators which are essentially non-load-bearing and can take relatively high doses remain in the source. This design uses a photodetachment neutralizer. The solid-state lasers are recessed into the shielding also. Again no Monte Carlo shielding calculations were done but rather estimates made. The next injector study, done in 1977 by Fink, Bender, and colleagues (Fig. 12) further develops the shielding. A Monte Carlo calculation on a ^{3}He injector is underway.

An assessment of "Electrical Insulator Requirements for Mirror Fusion Reactor" has been made by R. H. Condit and R. A. VanKonynenburg. The table of contents follows which gives an idea of the substance of this study.

Future Work

In FY 78 we plan to carry out two injector studies; one based on D^+ of about 120 keV and the other based on D^- at ~1 MeV. Both will place heavy emphasis on shielding design and analysis with the extensive use of Monte Carlo codes.

"Electrical Insulator Requirements for Mirror Fusion Reactor"
by R. H. Condit and R. A. Van Konynenburg.

Contents

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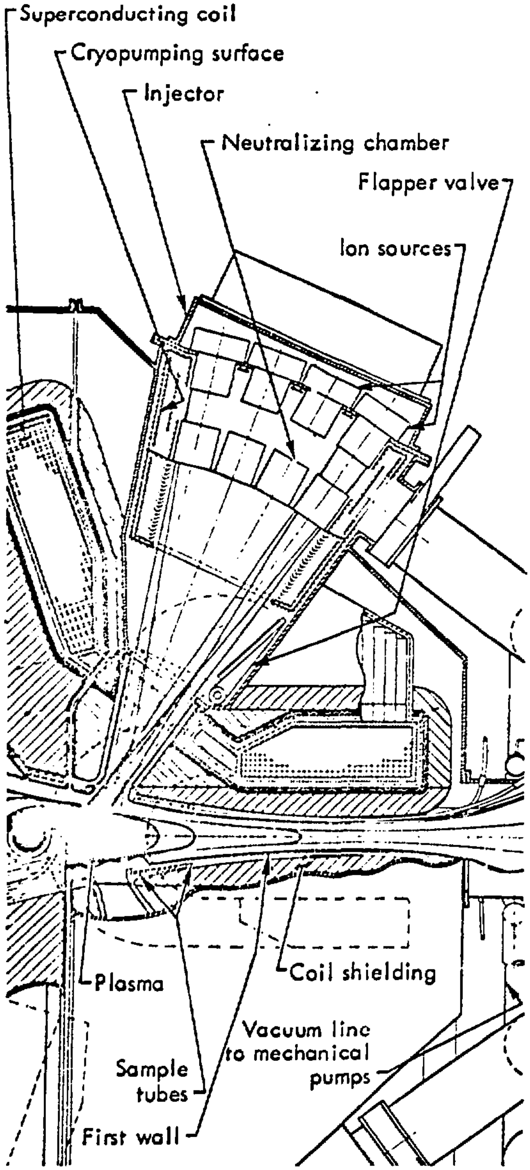
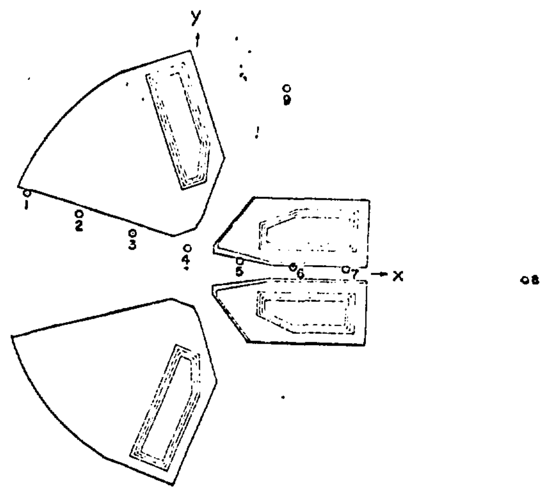


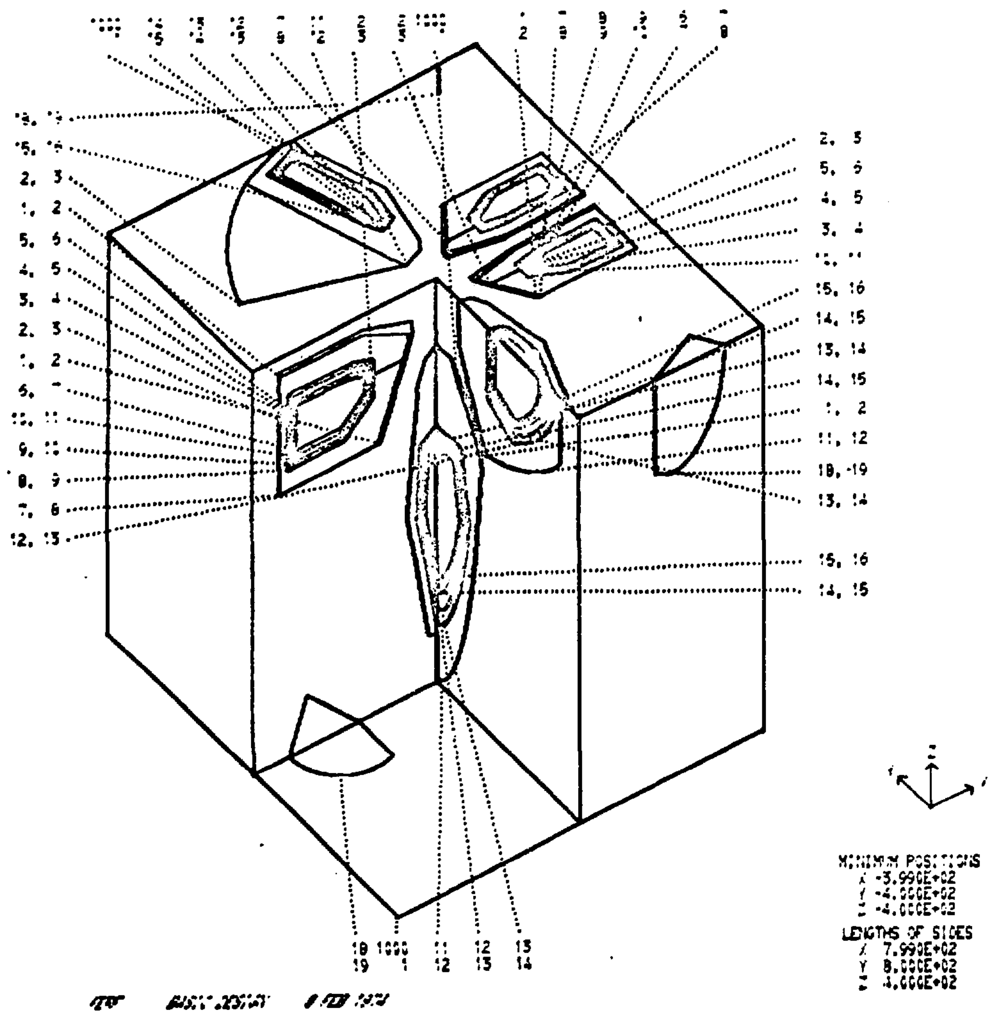
Detail from Fig. 1 showing typical section of one of the injector points.

Figure 7.



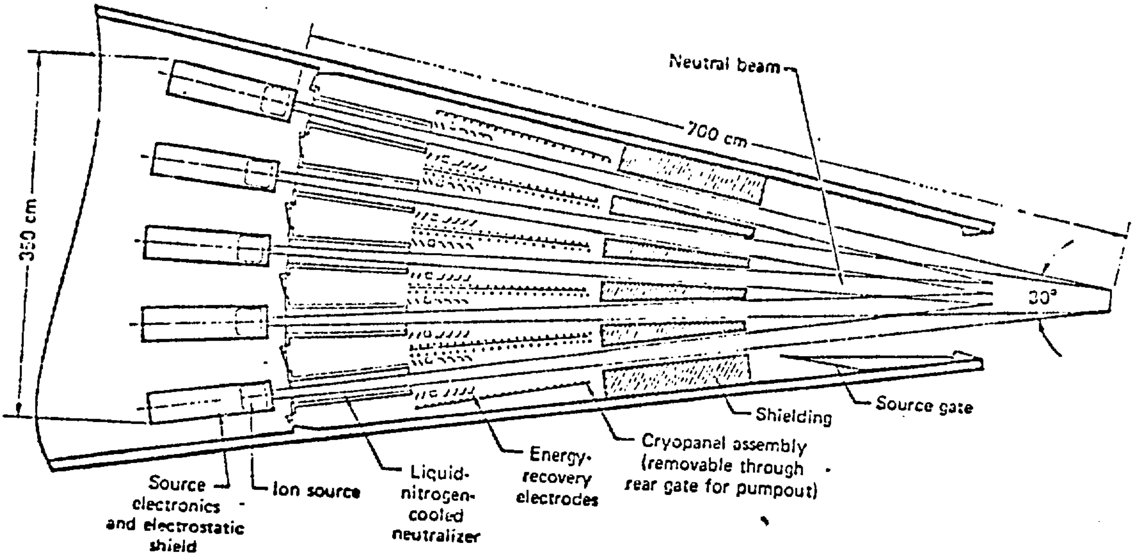
Cross-sectional view of the geometry model used for shielding calculations (z = 0 plane).

Figure 8.



Computer illustration of geometry model used in shielding calculations. Numbers refer to material compositions and statistical regions.

Figure 9.



100 keV D° INJECTOR FOR HYBRID (1975)

Figure 10.

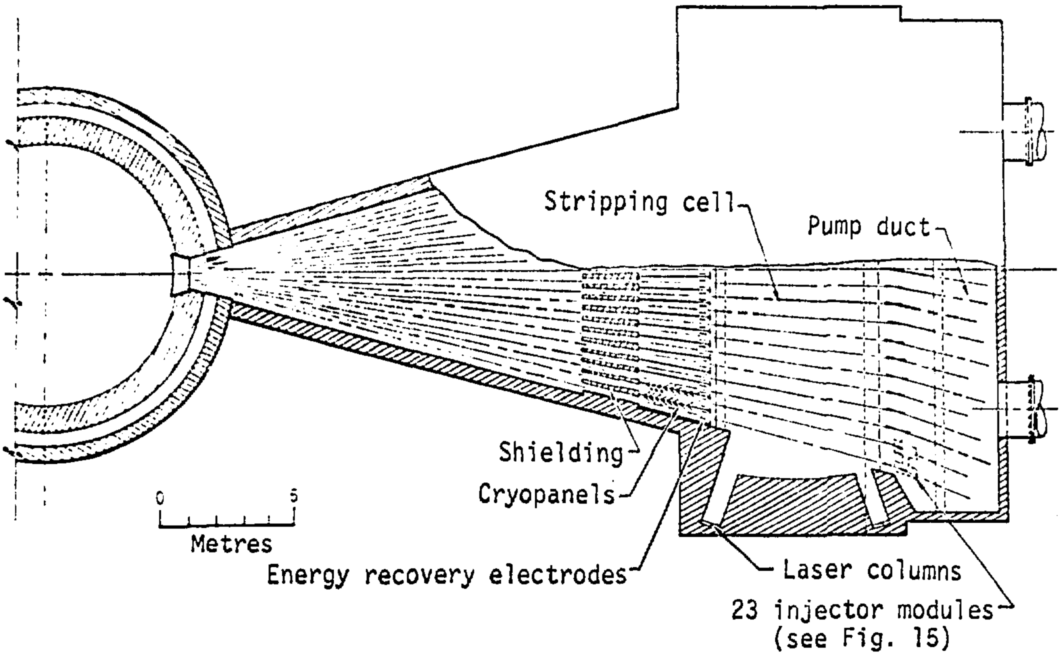


Fig. 11a. Proposed neutral beam injector delivering 1800 A of 150-keV deuterium and tritium atoms.

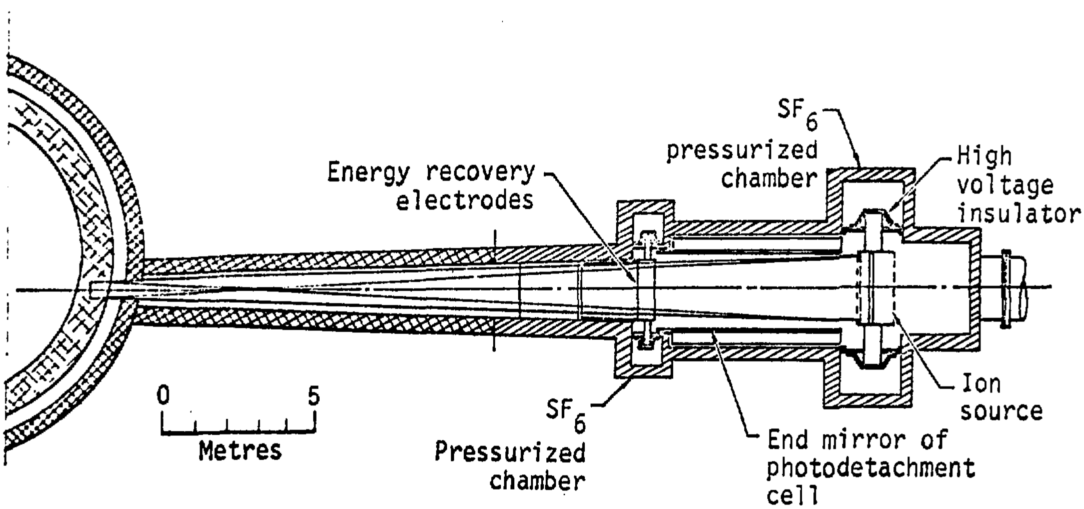


Fig. 11b. Vertical cross section of proposed neutral beam injector.

POSITIVE ION INJECTOR SOURCE DETAILS

120 keV HYBRID INJECTOR (1977)

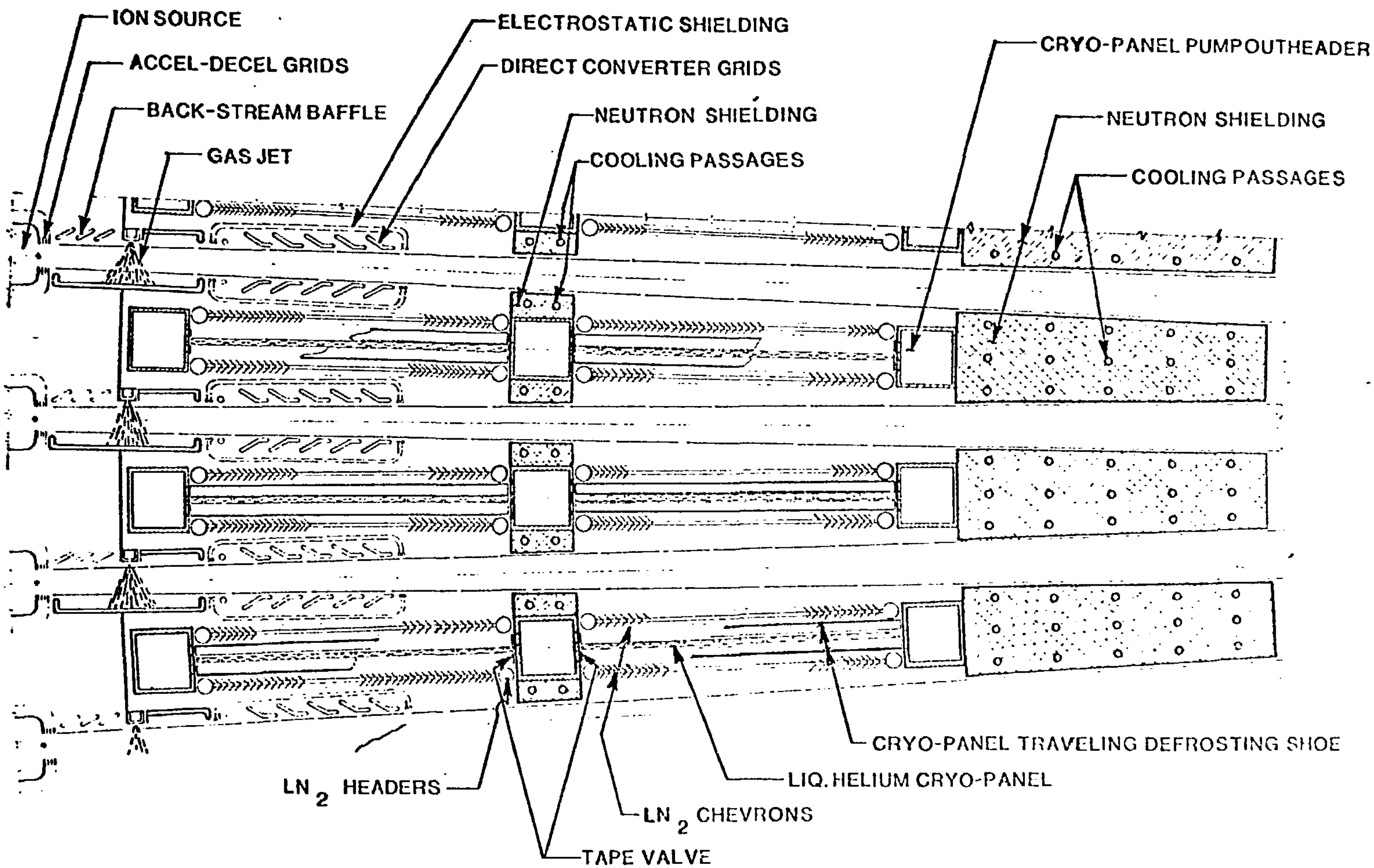


Figure 12.

6. Conclusions

Mirror fusion reactor designs carried out to date call for high-injection energy (standard mirror -150 keV, FRM - 200 keV, Tandem mirror - 1200 keV) which can be met by D^- beams but due to low efficiency, not by D^+ beams. Hybrid mirror reactors (standard, FRM, Tandem) use 100 - 125 keV injectors and can use D^+ ions.

If a reactor concept calls for injection above ~100 keV, the reason is not based on maximizing the reaction rate parameter $\langle\sigma v\rangle$, or the power density ($\propto \frac{\langle\sigma v\rangle}{t^2}$) but rather on some other requirement, such as penetration, heating, end plugging, maintaining plasma currents. If the beam is used for both heating and fueling simultaneously, then injection over 100 keV incurs disadvantages. On the other hand, heater beams seem to prefer high energy, requiring less current (especially to preferrentially heat electrons); in fact, the higher, the better, and 3.5 MeV He is an excellent heater; that is, ignition or near ignition is desirable.

If the confinement concept calls for high-injection energy, effort should be placed on evolving the concept towards lower energies as well as figuring out how to supply such high energies.

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TITLE: MAGNETIC FUSION ALTERNATE CONCEPTS HEATING REQUIREMENTS

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MAGNETIC FUSION ALTERNATE CONCEPTS HEATING REQUIREMENTS

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Abstract

A review is made of the plasma heating options and needs for several representative magnetic fusion energy (MFE) alternate reactor concepts. While these alternate concepts often share heating technology with the mainline approaches (i.e. tokamaks and mirrors), significant differences can arise which can lead to either relaxed or more difficult technology development requirements.

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1. INTRODUCTION

Alternate magnetic fusion reactors are distinguished from the "mainline" tokamak and mirror approaches by differences in plasma/energy confinement physics, thermonuclear burn cycle and research emphasis. At the same time and in various combinations, the alternate concepts share with mainline approaches the advantages and disadvantages of the several plasma heating techniques invoked to achieve thermonuclear temperature ($T \sim 5$ keV). Therefore, as technological progress is made on the leading heating options (i.e. neutral beam and rf) suggested for tokamaks and mirrors, many of the alternate concepts will benefit. Additionally, should "fatal flaws" in these approaches arise, alternate concepts using other heating options can then be emphasized.

The compatability of the several alternate reactor concepts with a number of available heating options is indicated in Table I. This matrix is not inclusive of all suggested fusion reactors, nor of all possible heating methods, nor probably of all possible combinations of those entries that are included; Table I does serve to suggest, however, the rich range of options available as magnetic fusion research advances. Tokamaks are included for comparison with the first group of toroidal devices, and "mirrors" are included with the second group of nominally linear devices. The heating technology required of mirror and tokamak confinement per se will not be addressed here.

2. HEATING REQUIREMENTS FOR SPECIFIC CONFINEMENT SCHEMES

The methods and technology required of each magnetic confinement scheme is briefly discussed. The ordering employed in Table I is followed.

2.1. Stellerators.

The recent encouraging results from stellerator confinement physics[1] has led to a call[2] for a reassessment of this concept as a potential fusion reactor.[3] Although detailed, contemporary reactor designs for stellerator confinement are non-existent, neutral-beam and/or high-frequency heating similar to that envisaged for tokamaks will probably be required. Since the stellerator equilibrium and stability will be established entirely by external conductors, significant ohmic heating of the stellerator plasma appears more difficult than that for the tokamak.

2.2. Surmac.

The Surmac device[4] represents a class of axisymetric multipole devices which until recently have received little consideration as a power reactor. Because superconducting coils may have to thread a hot steady-state plasma, neutron-producing fusion reactions cannot be used, and, hence, conceptual reactor designs have considered primarily "advanced" fusion fuels (e.g. p-^{11}B). It is contended[4] that because of the highly desirable surface magnetic configuration, both heating and refueling by neutral beams should be possible. Furthermore, the use of partially stripped ion beams (i.e. $^{11}B^{+}$) has been proposed to be injected with large orbits across the magnetic surface layer.

2.3. Tormac.

The use of an axisymmetric bi-cusp geometry to confine a steady-state thermonuclear plasma on closed field lines against mirror-like cusp losses has been proposed in a reactor context.[5,6] Heating of the Tormac bicusp would be achieved by either neutral-beam injection along the open field lines (i.e. along the cusp field lines) or by magneto-acoustic ("shaker") heating. Like the Surmac this device would operate in a steady-state mode and provides an attractive open system for the use of neutral beams. Also, like the Surmac, the question of startup as well as the quantitative needs of neutral-beam or rf-heating technology for a reactor-like device remain unanswered, primarily because of the low level-of-effort being devoted to reactor studies of these concepts.

2.4. Elmo Bumpy Torus (EBT).

The EBTR proposes a steady-state toroidal reactor of high aspect ratio (30-60) which confines a plasma in a minimum average-field configuration at high beta (~ 0.25). The toroidal multiple-mirror configuration would be formed by 24-48 hot-electron annuli that are sustained by 120 GHz microwave heating. This bulk heating will supply 200 MW to the plasma for ignition and 50 MW at steady state. Continuous-wave profile heating will be required at 70-90 GHz and a power to the plasma of 5-10 MW. Although many important aspects of the EBTR startup (and refueling) are not resolved, 150 keV neutral beams at 200 MW for ~ 3 sec may also be needed to achieve ignition at this voltage, and positive ion injection may still be practical. Similar beam energies at 50 MW may also be necessary for supplementary heating and burn sustenance. These neutral beams must penetrate a $1.2(10)^{20}$ m^{-3}, 15 keV plasma. Detailed beam penetration and trapping calculations have not yet been made. It is claimed[8] that a 28-GHz, 200-kW cw Gyroklystron is under development, which should serve as a scale device for the above-mentioned rf power supplies.

2.5. Reversed-Field Pinch Reactor (RFPR).

The Reversed-Field Pinch (RFP) achieves ideal MHD equilibrium and stability in a toroidal plasma by inducing a high shear at the plasma-vacuum boundary when the toroidal field is reversed in the vacuum region. Like the tokamak, the poloidal field is generated by an induced toroidal plasma current, but in the RFP case the plasma pressure is supported almost totally by poloidal field, the effective "tokamak q-value" can be considerably below unity (higher toroidal current density and aspect ratio), and ohmic heating to ignition conditions is possible. Hence, ohmic heating by toroidal currents is the only heating mechanism proposed for the two relatively similar RFPR designs that have been reported.(9,10) The Los Alamos RFPR design proposes a relatively short burn ($\sim$1-2 s), the $\sim 2(10)^{21}$ m^{-3} dense plasma being brought to ignition by 30-40 MA of toroidal current in the initially 0.8-m radius plasma after a tokamak-like startup.(9) For this 750 MWe (net) power plant $\sim$10 GJ is transferred to the room-temperature poloidal field coils in 100 ms from 10-15 homopolar motor/generator sets operating at a nominal 10 kV. A total poloidal flux swing of $\sim$ 800 V-s is required. The Culham RFPR design burns for 30 s, requires approximately the same energy to be transferred in 500 ms to drive 20 MA of toroidal ohmic-heating current at a voltage of $\sim$1 kV. These values are similar to those envisaged for a comparable tokamak reactor design(11) (9 GJ, 15 MA, 10-s transfer time, 150 V-s at $Z_{eff} = 1$). Because of the larger flux swing envisaged for the RFPR, iron-core transformer coupling in the ohmic-heating circuit may be a necessity.

2.6. Toroidal Theta-Pinch Reactor (RTPR).

The 0.4-s pulsed burn for the RTPR would be achieved by a staged sequence of rapid ($\sim$1-μs, 0.7 kV/cm) implosion heating followed by a slow ($\sim$ 30 ms, 7.0 T) adiabatic compression.(12) For an aspect ratio of 200, which assures acceptable levels of fast-feedback-stabilization power, and a pulse frequency of 0.1 Hz, this 2 GWe (net) design(12) would require 1.3 GJ of fast implosion energy (2.0 MJ/m) and 61.0 GJ (97.1 MJ/m) if slow adiabatic-compression energy, capacitors and homopolar motor/generators, respectively, being used. Since each unit of net electrical energy generated by the RTPR requires 3.1 units of transferred electromagnetic energy, the latter must be recovered after each burn with an overall efficiency of 87% (including joule losses in the room-temperature coils) in order to achieve the 17% recirculating power fraction reported in Ref. 12; this amounts to a 95% energy transfer efficiency for a lossless load. The relatively low level but expensive fast-implosion energy store and the overall switching requirements present the major problems for this otherwise proven heating scheme. The scale of the above energy technology requirements is set by the growth rate of the m = 1 MHD instability, the desire for finite-Larmor radius stabilization of $m \geq 2$ modes, and the need to minimize the reactive m = 1 feedback power.[12] The storage and transfer on a ~30 ms time scale of large quantities of energy by homopolar motor/generator sets appears technologically and economically feasible.[13-15]

2.7. Dense Z-Pinch (DZP).

Although the use of a straight, self-constricting current channel or Z-pinch for the generation of thermonuclear plasma represents one of the first magnetic confinement schemes proposed, early experimental experience with catastrophic "sausage" (m = 0) and "kink" (m = 1) MHD instabilities have precluded its consideration as a reactor concept. The possibility of MHD stabilization by axial flow, finite Larmor radius (FLR) effects, and/or embedding the simple Z-pinch into a dense, cold gas has created renewed interest in this concept.[16,17] Reference 17, in particular, presents results from a preliminary examination of the DZP reactor potential. For the reference calculation based on a pulsed, gas-embedded operation, a high-energy electron beam forms the initial current channel, after which a low inductance (50 nH, 38 MV) capacitor bank delivers 1.0 MA and 73 MJ to the 2.2-mm diameter x 100-mm long pinch. This heats the plasma region to 360 eV ($3.2(10)^{26}$ m^{-3}) in 2.3 μs. The ohmically pre-heated plasma is then brought to ignition by switching an additional capacitor bank into the Z-pinch, thereby bringing the discharge current to the ~30 MA level (~ 6 TA/m^2). In addition to requiring a low-inductance, high-voltage capacitor bank ($1.3(10)^{-3}$ μH/MV required compared to 1 μH/MV presently achievable), the need to recover a major portion of the transferred magnetic energy, to guarantee an acceptable energy balance as well as the high cost of capacitive energy store appear as difficult problems. The possible use of steady-flow systems, which achieve MHD stability by FLR effects, axial flow, or a combination thereof promises to alleviate greatly these heating/confinement related energy problems.[17]

2.8. LINUS.

The LINUS version[18,19] of imploding-liner fusion envisages adiabatically compressing to ignition a pre-heated D-T plasma in cylindrical geometry by means of a gas-driven liquid-metal shell of 2.5-m initial inner radius and 1.7-m initial thickness. For a length of 10-m, initial (maximum) radial velocity of $\sim 10^3$ m/s and an azimuthal rotational speed (needed to stabilize the inner surface against Raleigh-Taylor hydrodynamic instabilities at peak-compression and "turn-around") of 130 rpm, the liner will require 1.14 GJ (114 MJ/m). Since the gross electrical output amounts to 16.7 MJ/m (a pulse frequency of 3 Hz is proposed[19]), the liner implosion cycle must be very reversible and efficient. The primary energy storage for the LINUS, however, is inertial/hydrodynamic. The formation of the $\sim 10^{21}$ m^{-3}, ~300 eV initial plasma in a closed field configuration (axially-reversed magnetic fields at 0.5 T are proposed to inhibit particle endloss from the 10-m long plasma column) is to be created by a pulsed, annular and rotating relativistic electron beam.[20] Approximately 10 MJ will be required of the electron beam, which is assumed to create the plasma with 50% efficiency.

2.9. Fast Liner Reactor (FLR).

The FLR approach to fusion power[21,22] proposes magnetically driving a thin (~1-mm) cylindrical (0.2-0.3 m radius and length) metal shell into a warm (100-200 eV), dense (10^{23}-10^{24} m^{-3}) plasma. The $\beta \gg 1$ plasma pressure is supported by the $\sim 10^4$ m/s metal wall, but a small (5-10 T) poloidal magnetic field threads the plasma to inhibit heat conduction to both the cylinder walls and solid end plugs. Approximately 5-10 MJ is required to form the initial plasma and would be supplied on a ~5-μs time scale from a capacitive store via either a co-axial plasma-gun injector, laser or electron-beam heating, or ohmically heating solid D-T threads located on the liner axis. In either case, heating efficiencies on the order of 50% are expected (with the exception of the laser-beam case). To circumvent the need for rotational stabilization of

Raleigh-Taylor modes in the liner large liner masses, and reversible recovery of postburn liner energy, the very fast ($\sim 10^4$ m/s) radial implosion velocities are selected. Reactor scaling predicting by both simple analytic expressions[21] and detail MHD computations[22] dictate large liner energy requirements for acceptable (<20%) recirculating power fractions. Hence, the initial liner energy amounts to 1-2 GJ, of which 30-40% can be effectively transferred to the plasma, the difference being lost to joule heating and compression of the liner shell. Since > 50% transfer efficiencies of energy between the liner and the energy store are desired, a total of ~ 2-5 GJ must be stored and transferred to effect the adiabatic compression to ignition and burn. Since this transfer must occur in a 5-10 μs time scale, only capacitive or capacitive (homopolar)/inductive storage devices are considered.

2.10. Electron-Beam Heated Solenoid (EBHS).

In the EBHS concept[23] a linear D-T plasma column is heated to ignition by a relativistic electron beam (REB) that is injected along an axial guide field. Although a small amount of energy must be supplied to break down and preionize the plasma, the bulk of the heating is provided by the REB. Additional energy must also be supplied to a pulsed solenoid in order to move the plasma off the wall, but the associated compression heating is neglibible. The axial confinement is provided by high-beta (~ 0.9) multiple mirrors, and at ignited densities in the range 10^{22}-10^{23} m^{-3} steady-state (superconducting) solenoidal fields of ~ 15 T are needed. The plasma diameter and length are projected[23] to be 20-40 mm and 100-500 m, respectively, and the burn occurs for 10's of milliseconds.

Hence, the major part of the EBHS heating function will be provided by an 8-10 MeV REB delivered in ~ 10 μs at 200 kA and a total energy of 18 MJ/pulse[23] (reactor length 364 m, plasma radius 18.4 mm and burn time 8.2 ms). This operating point would require three REB generators with a cathode radius of 16 mm. For a first-wall radius of 57 mm, a fusion neutron flux of 1 MW/m^2 and a gross electrical power of 500 MWe, the REB generator must be pulsed at a frequency of 7 Hz. Recirculating power fractions on the order of 0.15-0.20 are expected for the EBHS.

2.11. Laser-Heated Solenoid (LHS).

The overall heating cycle envisaged for the LHS[24,25] is similar in principle to that described for the theta pinch. The laser energy requirement to heat a D-T plasma completely to ignition would be too severe, approaching a GJ. Consequently, the laser replaces the implosion-heating function described for the theta pinch, and the laser initiated and heated plasma is adiabatically compressed to ignition. Using high compression/confinement fields (28 T) and an arbitrary endloss-reduction factor (2-4 over free streaming), viable LHS reactor designs have been proposed[25] that are 500-m in length (compressed plasma radius 26-mm). For this case the required CO_2 (10.6- m) laser energy is 80 MJ. For this laser energy, 5-10 passes through the 500-m long plasma column are required for complete absorption by classical inverse-bremsstrahlung processes. The ignited $9.0(10)^{22}\ m^{-3}$ plasma is confined for 15 ms. Within a 50-mm radius first-wall magnet bore 1.2 GJ of magnetic energy is stored, and this energy must be pulsed into and out of the solenoid each burn cycle. The energy stored by the superconducting coils at a nominal 2.0-m radius, this energy only being pulsed out of the volume encompassed by the 50-mm radius first-wall pulsed coil, amounts 490 GJ if the superconductor operates at half the compression field (i.e. 14 T).

2.12. Linear Theta-Pinch Reactor (LTPR).

The heating mechanism for the LTP[26] is identical to that proposed for the RTPR[12]: fast implosion heating to 800 eV (1.0 kV/cm) followed by adiabatic compression to ignition and burn. To achieve an acceptable energy balance at an acceptable reactor length (size, power level), a "re-entrant" end plug condition was modeled,[26] which predicts a reactor length of 100 m for a recirculating power fraction of 0.27 and a net power of 490 MWe (pulse frequency of 0.1 Hz for a 2 MW/m^2 wall loading). For these conditions the fast (~ 1-μs) implosion energy store amounted to 530 MJ and the adiabatic-compression power supply requires a transfer of 41 GJ in 30 ms. Many of the comments for the RTPR made in Sec. 2.6. apply equally as well to the LTPR.

3. SUMMARY

Since many of the technological requirements of neutral beam injection or rf heating for alternate magnetic fusion reactors are shared with the "mainline" approaches, it can be expected that "spin-off" from the "mainline" D/T effort will be sufficient to support the continuation of the current level of alternate concepts research for the Surmac, Tormac and EBTR. However, the higher allowed beta values (i.e. higher density) which make these alternate concepts more attractive than tokamaks, tend to aggravate the problem of beam penetration to the central plasma, unless the plasma radius is correspondingly reduced. The development of high-energy, repetitively pulsed relativistic electron beams and long-wavelength lasers by the inertial fusion program will benefit the EBHS and the LHS.

Those alternate concepts that do not require neutral beams or rf heating will nonetheless require substantial advances in the technology of energy transfer and storage. These requirements are summarized in Table II.

4. AUTHORS NOTE

The notice given for the preparation of this summary was not sufficient to permit consultation with each of the individual design teams for purposes of both obtaining more current information and verifying our interpretation of the reported literature.

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TABLE I

HEATING OPTIONS FOR

MAGNETIC FUSION ALTERNATE CONCEPTS

REACTOR CONCEPT \ HEATING OPTION	BEAMS: ELECTRON	ION	NEUTRAL	LASER	PLASMA GUNS	MAGNETIC IMPLOSION	MAGNETIC COMPRESSION	OHMIC HEATING	RF
TOKAMAK	(A)	(A)	(P)		(S1)		(A)	(S2)	(A)
STELLERATOR			P					(S2)	(A)
SURMAC		(A)	P		(S1)			S2	
TORMAC			P						A
ELMO BUMPY TORUS			P						A
REVERSE-FIELD PINCH			(A)					P	(A)
THETA PINCH						S1	S2		(A)
"MIRROR"			P	(S1)					
DENSE Z-PINCH	S1							P	
LINUS				(S1)			P		
FAST LINER	(S1)			(S1)	(S1)		P		
EBHS	P								
LHS	(S1)			P			A		
THETA PINCH	(A)			(A)		S1	S2		(A)

P: PRIMARY S1: STAGED HEATING A: AUXILLIARY ( ): OPTIONAL
S2

Table II ENERGY TRANSFER AND STORAGE REQUIREMENTS FOR FUSION POWER

Reactor Concept	Net Power (MWe) (Hz)	AUXILIARY HEATING Method	Energy (GJ)	Transfer Time (μs)	Peak Power (MW)	Storage	PRIMARY HEATING Method	Energy (GJ)	Transfer Time (μs)	Peak Power (MW)	Storage
STELLERATOR	?						NB, RF	?			
SURMAC							NB, IB				
TORMAC							NB, RF				
EBTR	1000	RF NB	(Bulk Heating) (150 keV)	(120 GHz)	50-200 50-200	L L	RF	(Profile Heating)	(70-90 GHz)	5-10	L
RFPR	750(.1)	——					OH	10	1.0(105)	——	H
RTPR	2000(.1)	IH	1.3	1	——	C	AC	61	$3.0(10)^3$	——	H
DZP	100	REB OH	? 0.07	1 1-2	? ——	C C	OH	?	1-2	——	C
LINUS	500(3)	REB	0.01	>1	——	C	AC	1.14	$10\text{-}20(10)^3$	——	I
FLR	1000(.1)	OH, PI REB, LH	0.01	1-5	——	C	AC	2-5	5	——	C
EBHS	500(7)	——	——	——	——	——	REB	.018	10.	——	C
LHS	1000	LH	0.08	>1	——	C	AC	1-2	>1	——	C
LTP	490(.1)	IH	0.5	1	——	C	AC	41	$3.0(10)^3$	——	H

RF = radiofrequency heating
NB = neutral-beam heating
IH = implosion heating
REB = relativistic-electron-beam heating
OH = ohmic heating
PI = plasma injection
LH = laser heating
IB = ion beam
AC = adiabatic compression
C = capacitor
H = homopolar M/G
I = inertial
L = line

TOKAMAK REACTOR STARTUP STUDIES UTILIZING RF AND NEUTRAL BEAM HEATING

J.E. SCHARER
UNIVERSITY OF WISCONSIN

RADIOFREQUENCY HEATING SCHEMES

1. LOW FREQUENCY ALFVÉN WAVES ($F \sim 0.5 - 1$ MHz)
2. ICRF ($F \sim 20 - 100$ MHz)
3. LHRH ($F \sim 0.5 - 10$ GHz)
4. ECRH ($F \sim 30 -$ ~~120~~ 200 GHz)

RF APPLICATIONS TO FUSION REACTORS

1. SUPPLEMENTARY ION HEATING IN TOKAMAKS (FUNDAMENTAL & HARMONIC CYCLOTRON HEATING IN ICRF, LHRH)
2. SUPPLEMENTARY ELECTRON HEATING IN TOKAMAKS (LANDAU & TRANSIT TIME DAMPING IN ICRH AND LOW FREQUENCY ALFVÉN WAVES, LANDAU DAMPING FOR LHRH, AND CYCLOTRON RESONANCE HEATING FOR ECRH)
3. ION TAIL PRODUCTION; ENHANCING NEUTRAL BEAM-PLASMA FUSIONS IN A TCT DRIVEN REACTOR; Q ENHANCEMENT (ICRF, LHRH)

ELECTRON TEMPERATURE AND CURRENT PROFILE CONTROL (ECRH, LHRH, ICRH)

GAS BREAKDOWN ON TOKAMAK AXIS FOR CURRENT CHANNEL DEFINITION IN OHMIC HEATING STARTUP (ECRH)

TRAPPING AND HEATING OF IONS IN THE END CELLS OF A TANDEM MIRROR DEVICE (ICRF)

ELECTRON HEATING IN THE CENTRAL CELL OF A TANDEM MIRROR DEVICE FOR FUSION Q ENHANCEMENT (LOW FREQUENCY ALFVÉN WAVES, LHRH, ECRH)

REDUCTION OF NEUTRAL BEAM ENERGY REQUIREMENTS IN A ~~ICT~~ DEVICE (ICRF, LHRH)
REACTOR

SCALING OF $2\omega_{ci}$ HEATING TO A TOKAMAK REACTOR

1. RF POWER LEVELS REQUIRED ~100 MW @ ~100 MHz FOR ~1 SEC FOR A 800 MW_e D-T REACTOR. $n \approx 10^{14}/cm^3$, $B_o \approx 66$ kG, AND $T_i \approx 6$-10 keV.

2. WAVE PROPAGATION AND HEATING MECHANISMS.
   A. FINITE ION GYRORADIUS ($k_\perp \rho_i > 0$). E_+, $\dfrac{\omega - 2\omega_{ci}(R)}{k_{\parallel} v_i} \lesssim 1$
   B. ELECTRON LANDAU (E_Z) AND TRANSIT-TIME (B_Z) DAMPING $1 < \omega/k_{\parallel} v_e < 10$.
   C. FAST ALFVEN WAVE EIGENMODES AND MODE CONVERSION PROCESSES DUE TO THERMAL EFFECTS NEAR $2\omega_{CD}$, ω_{ii}.

3. WAVE COUPLING (COIL OR APERTURE).
   A. WAVE MODE SPECTRUM (r-RADIAL, m-POLOIDAL, n-TOROIDAL).
   B. LAUNCHING STRUCTURE SPECTRUM (m,n).
   C. IMPEDANCE SEEN BY LAUNCHING STRUCTURE, DEPENDENCE ON EIGENMODE AND MODE CONVERSION ABSORPTION PROCESSES.
4. REACTOR ENVIRONMENTAL COMPATIBILITY OF LAUNCHING STRUCTURE.
   A. TECHNOLOGY-ARCING, MATERIALS, COOLING, RADIATION DAMAGE.
   B. LAYOUT, MINIMIZE WALL SURFACE AREA OF STRUCTURE.

$$P_{RF} = \omega E^* \cdot (\bar{\bar{K}}^a \cdot E)/2 \quad (\text{watts/cm}^3)$$

$$P_{RFE} = \frac{\pi^{1/2}}{4k_{\parallel}} \left[\frac{\omega_{pe}}{\omega_{ce}}\right]^2 k_\perp^2 |E_y|^2 v_e e^{-(\omega/k_{\parallel} v_e)^2} \text{ (CGS)}$$

$$\langle P_{RFI} \rangle_{\substack{\text{flux}\\\text{surface}}} = \left[\frac{\omega_{pi}^2}{\omega_{ci}}\right] \frac{R}{r} \lambda_i^{n-1} |E_+|^2/4n$$

WHERE $\lambda_i = \dfrac{k_\perp^2 v_{thi}^2}{2\omega_{ci}^2}$

AND SINCE $\omega = 2\omega_{CD} = 3\omega_{CT}$ WE HAVE

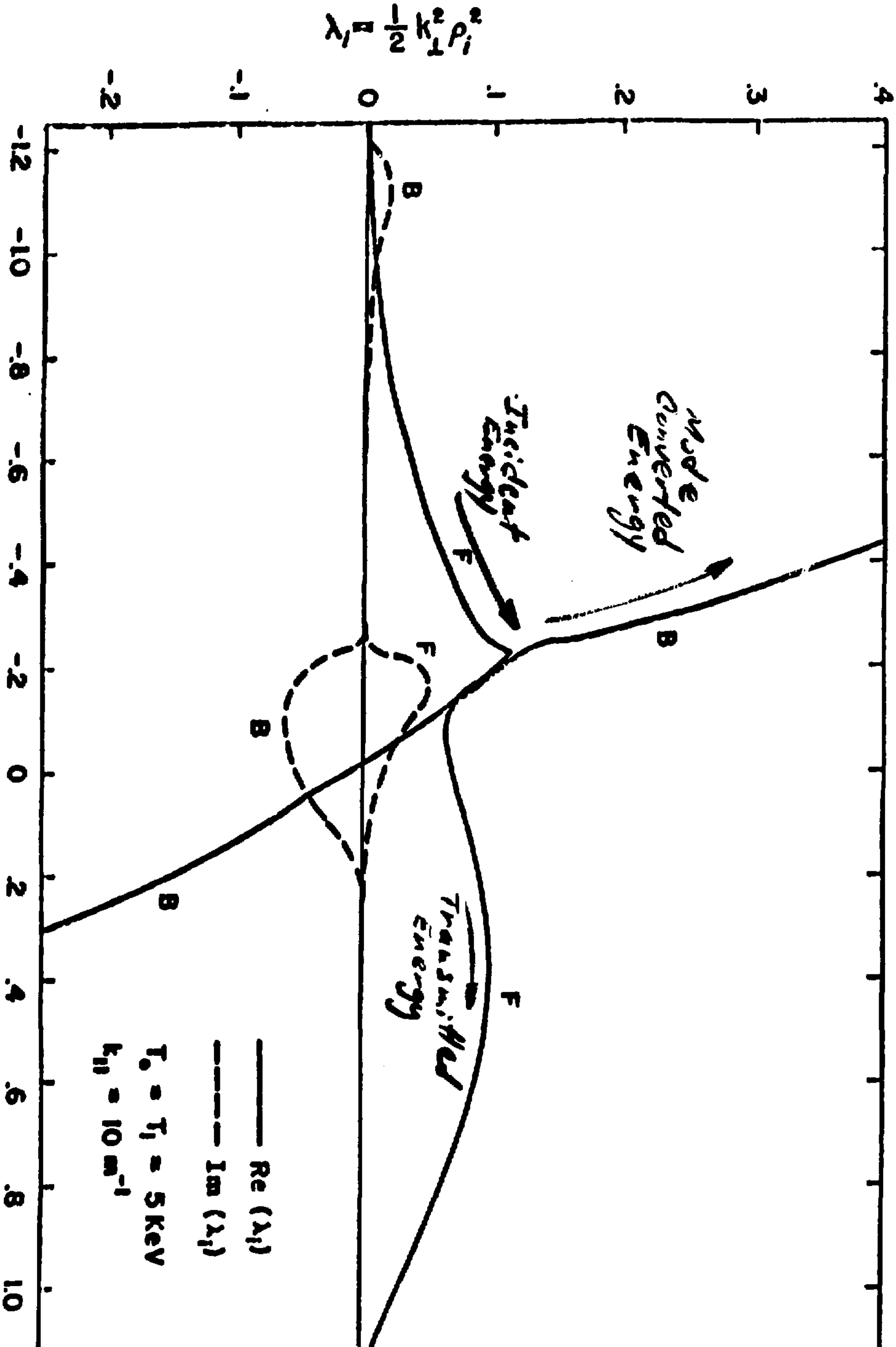
$n = 2$ FOR DEUTERIUM

$= 3$ FOR TRITIUM

$$|E_+|^2 = \left|1 - \frac{iE_x}{E_y}\right|^2 |E_y|^2$$

$$\frac{iE_x}{E_y} \simeq \frac{-iK_{xy}}{K_{xx} - n_z^2}$$

FOR $\omega = 2\omega_{CD}$ and $k_z^2 \ll k_\perp^2$



$\lambda_I = \frac{1}{2} k_\perp^2 \rho_i^2$
RADIUS (m.)
$2\omega_{cD} = 3\omega_{cT}$
Mode Converted Energy
Incident Energy
Transmitted Energy
Re (λ_I)
Im (λ_I)
$T_e = T_i = 5$ KeV
$k_\parallel = 10$ m^{-1}

TRANSPORT CODE

ENERGY - ELECTRONS

$$3/2 \frac{\partial}{\partial t}(n_e T_e) = \eta_{NC} J^2 + n_D n_T \langle \sigma V \rangle_{DT} U_{\alpha e} + P_{inj}\left(U_{Be} + f_{TCT}\left(\frac{E_\alpha}{W_B}\right) U_{\alpha e}\right)$$

$$+ \frac{1}{r}\frac{\partial}{\partial r}\left(r n_e \chi_e \frac{\partial T_e}{\partial r}\right) - 3/2 \frac{1}{r}\frac{\partial}{\partial r}(r n_e T_e V) - 3/2 \frac{n_i}{\tau_{ei}}(T_e - T_i)$$

$$- P_{rad} + P_{RFe}$$

ENERGY - IONS

$$3/2 \frac{\partial}{\partial t}(n_i T_i) = n_D n_T \langle \sigma V \rangle_{DT} U_{\alpha i} + P_{inj}\left(U_{Bi} + f_{TCT}\left(\frac{E_\alpha}{W_B}\right) U_{\alpha i}\right)$$

$$+ \frac{1}{r}\frac{\partial}{\partial r}\left(r n_i \chi_i \frac{\partial T_i}{\partial r}\right) - 3/2 \frac{1}{r}\frac{\partial}{\partial r}(r n_i T_i V)$$

$$+ 3/2 \frac{n_i}{\tau_{ei}}(T_e - T_i) + P_{RFi}$$

+ Continuity, Maxwell's, Ohms Law, and Momentum Eqns.

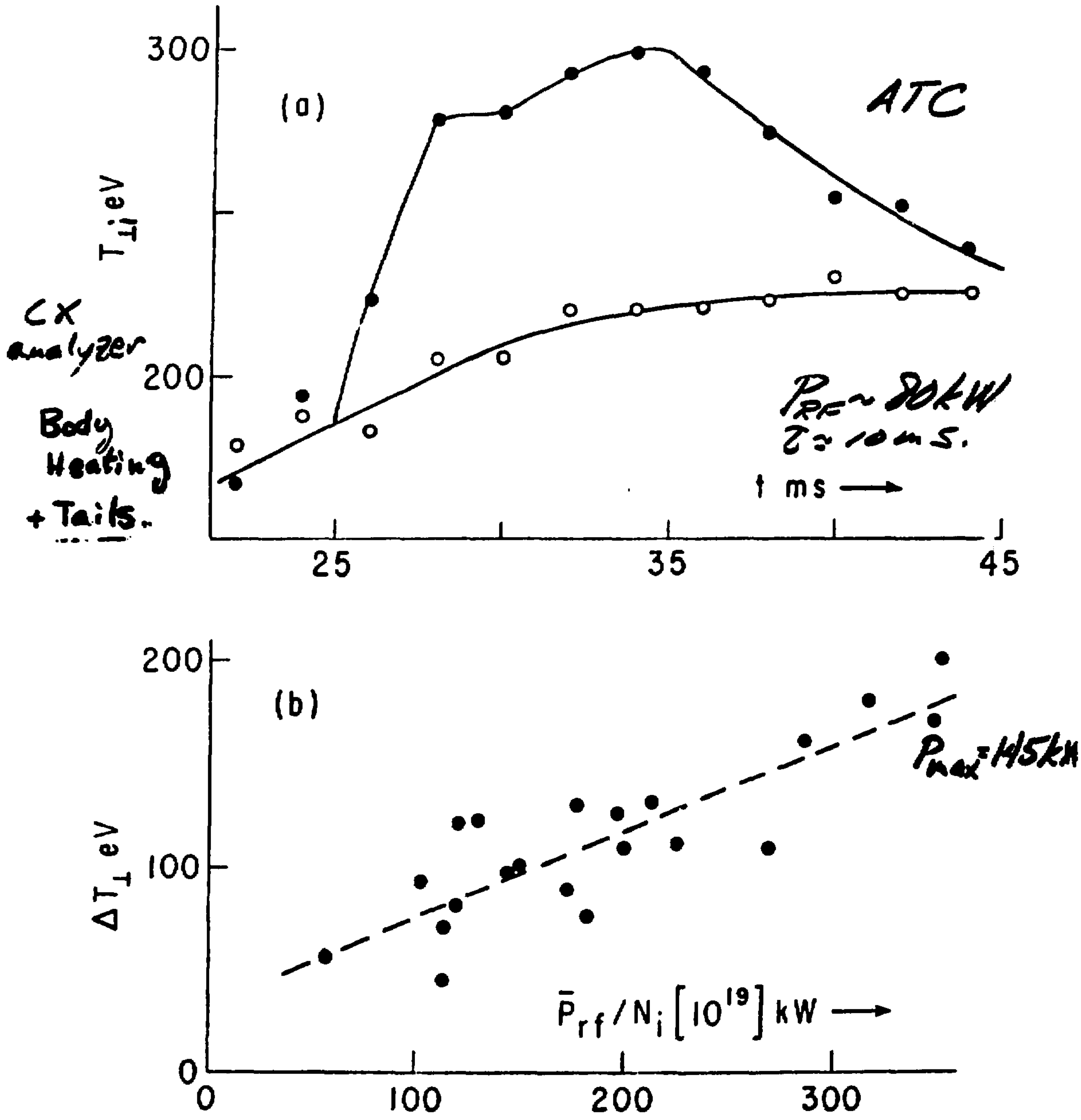
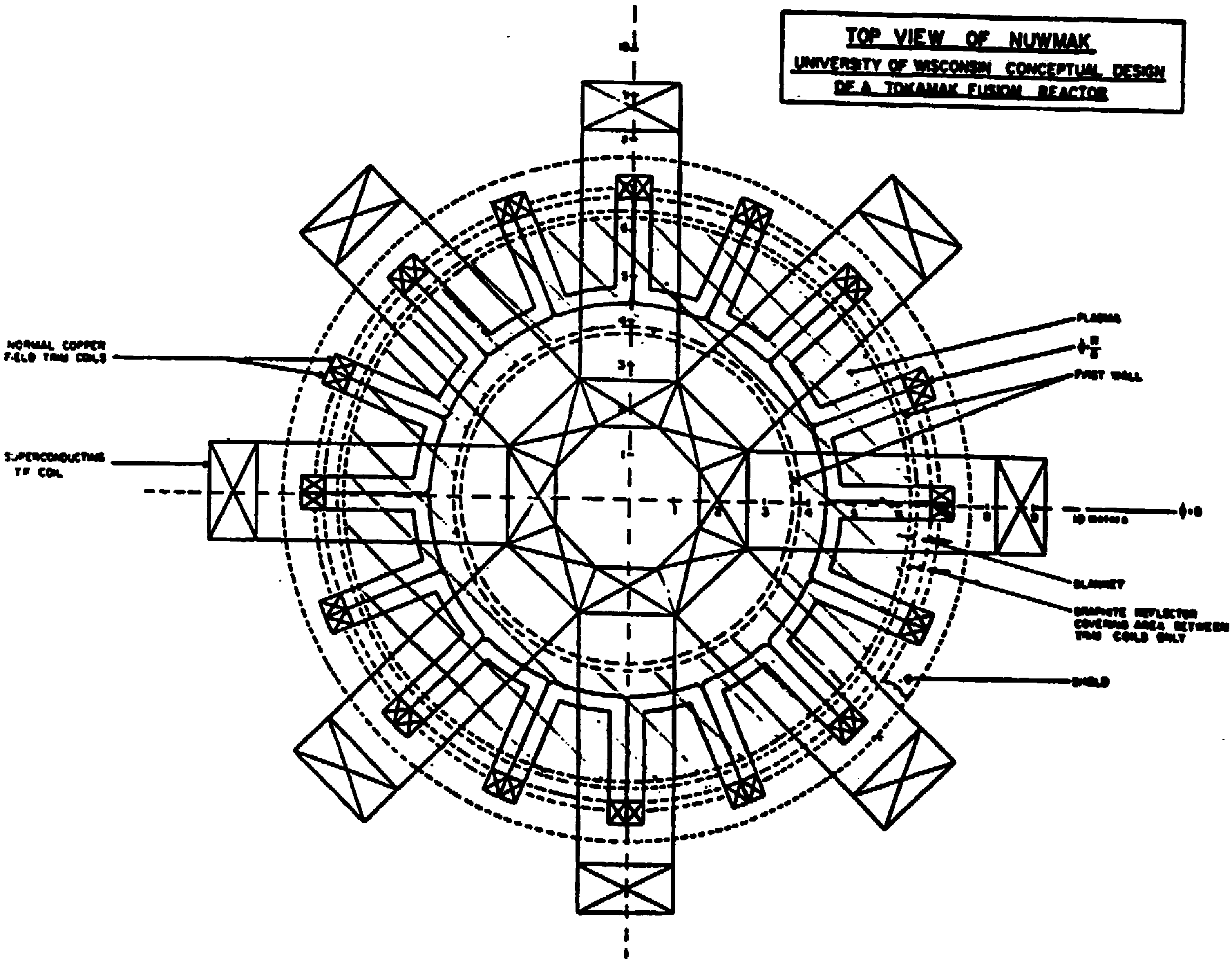

300
200
(a)
ATC
$T_{\perp i}$ eV
CX analyzer
Body Heating + Tails.
$P_{RF} \sim 80$ kW
$\tau \simeq 10$ ms.
t ms →
25
35
45
200
100
0
(b)
$P_{max} = 145$ kW
$\Delta T_{\perp}$ eV
$\bar{P}_{rf}/N_i\,[10^{19}]$ kW →
0
100
200
300

Fig. 4



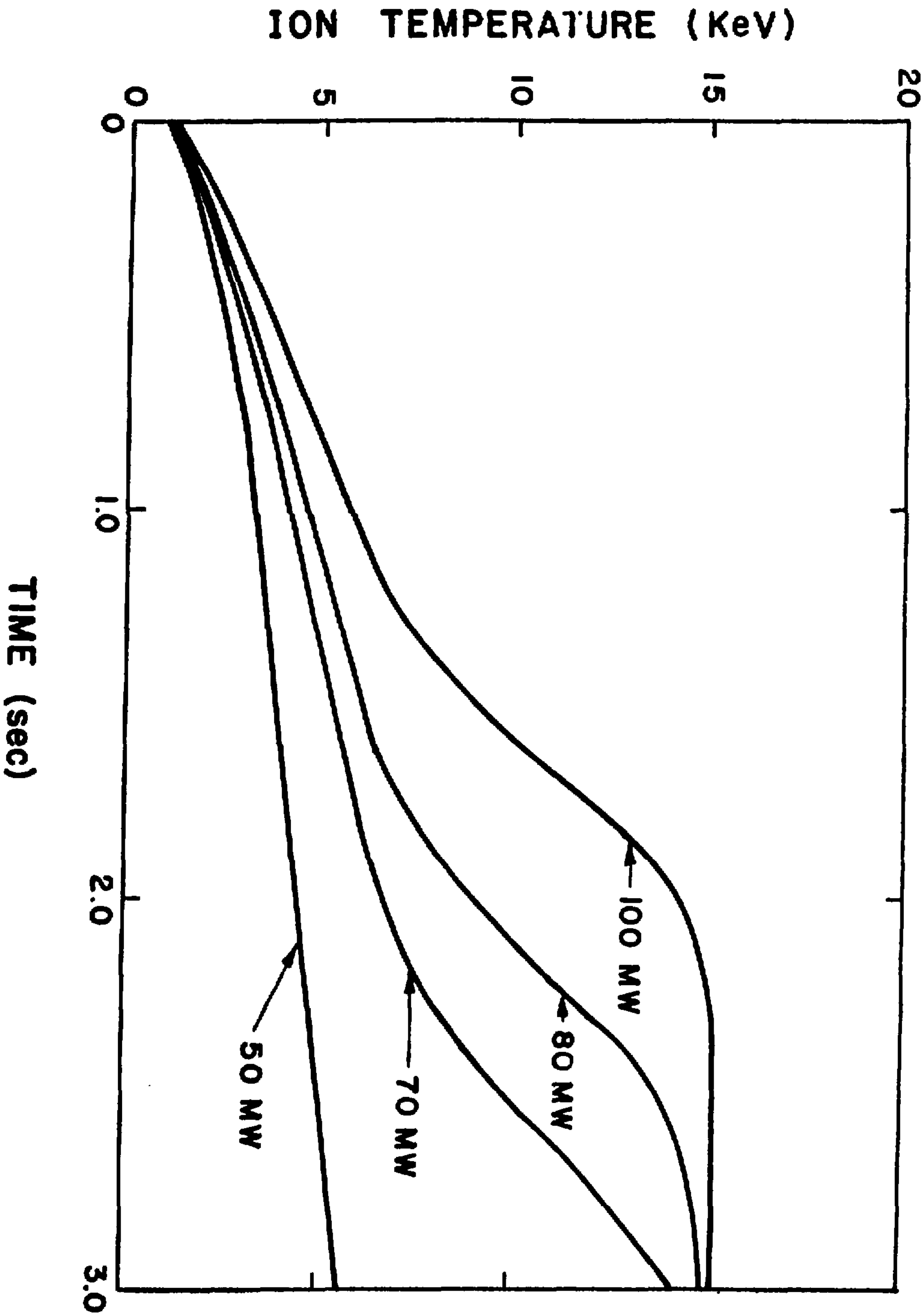
TOP VIEW OF NUWMAK
UNIVERSITY OF WISCONSIN CONCEPTUAL DESIGN
OF A TOKAMAK FUSION REACTOR

PLASMA AND FAST MAGNETOSONIC WAVE PARAMETERS

$a = 125$ cm	$\overline{n_e} = 1.95 \times 10^{14} cm^{-3}$
$R = 500$ cm	$\overline{n_D} = \bar{n}_T = 0.975 \times 10^{14} cm^{-3}$
$A = 4$	$\frac{n_{edge}}{n_{max}} = \frac{T_{edge}}{T_{max}} = 0.01$
$S = 1.33$	$T_i = T_e = 1.$[illegible]
$I = 6.48$ MA	$f = 92$ MHz
$B_o = 60$ KG	$k_{\parallel} = .1\ cm^{-1}$
$P_{th} \approx 2300$ MW	$k_{\perp} = 1.6\ cm^{-1}$

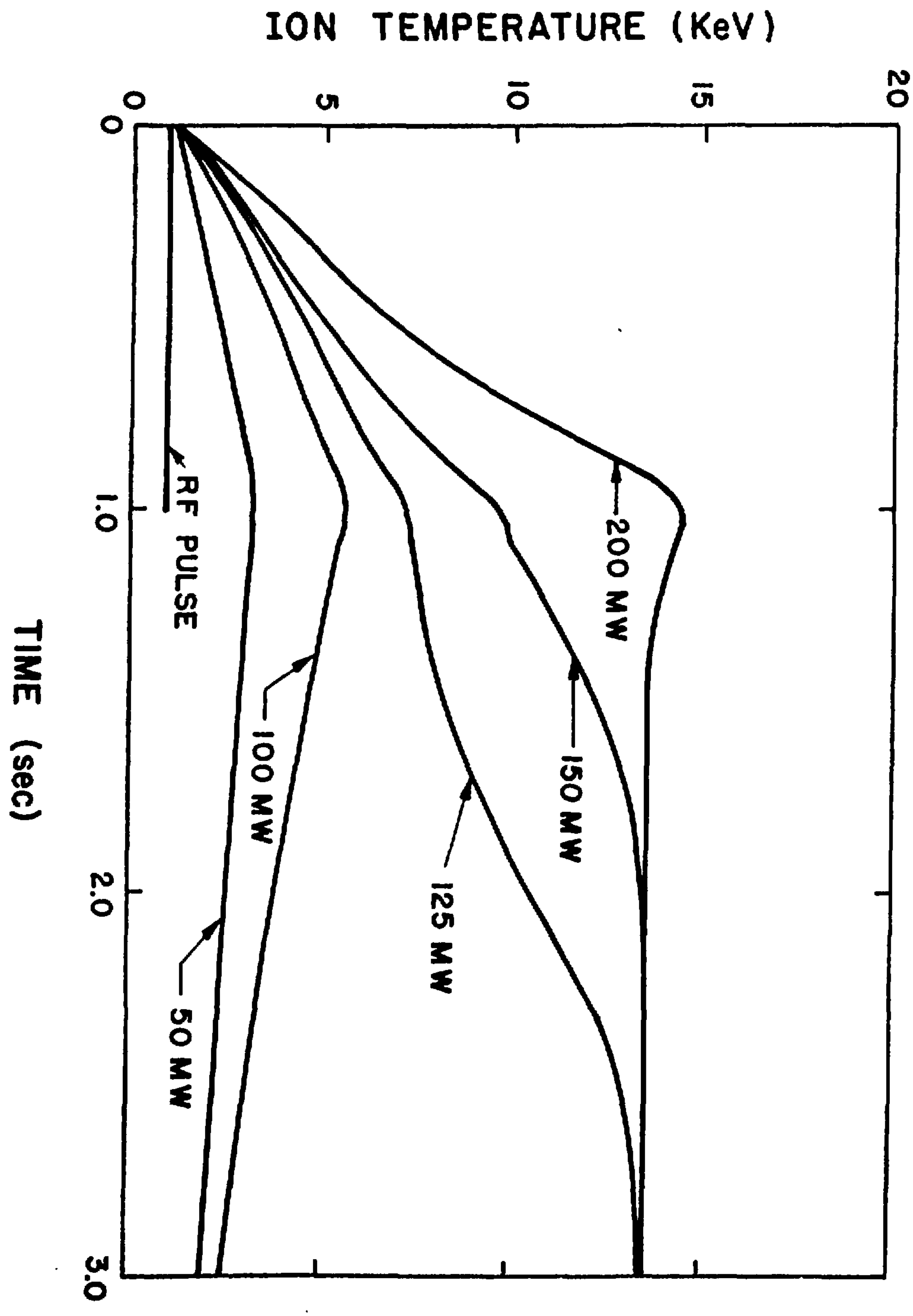
LONGITUDINAL MODE = 50

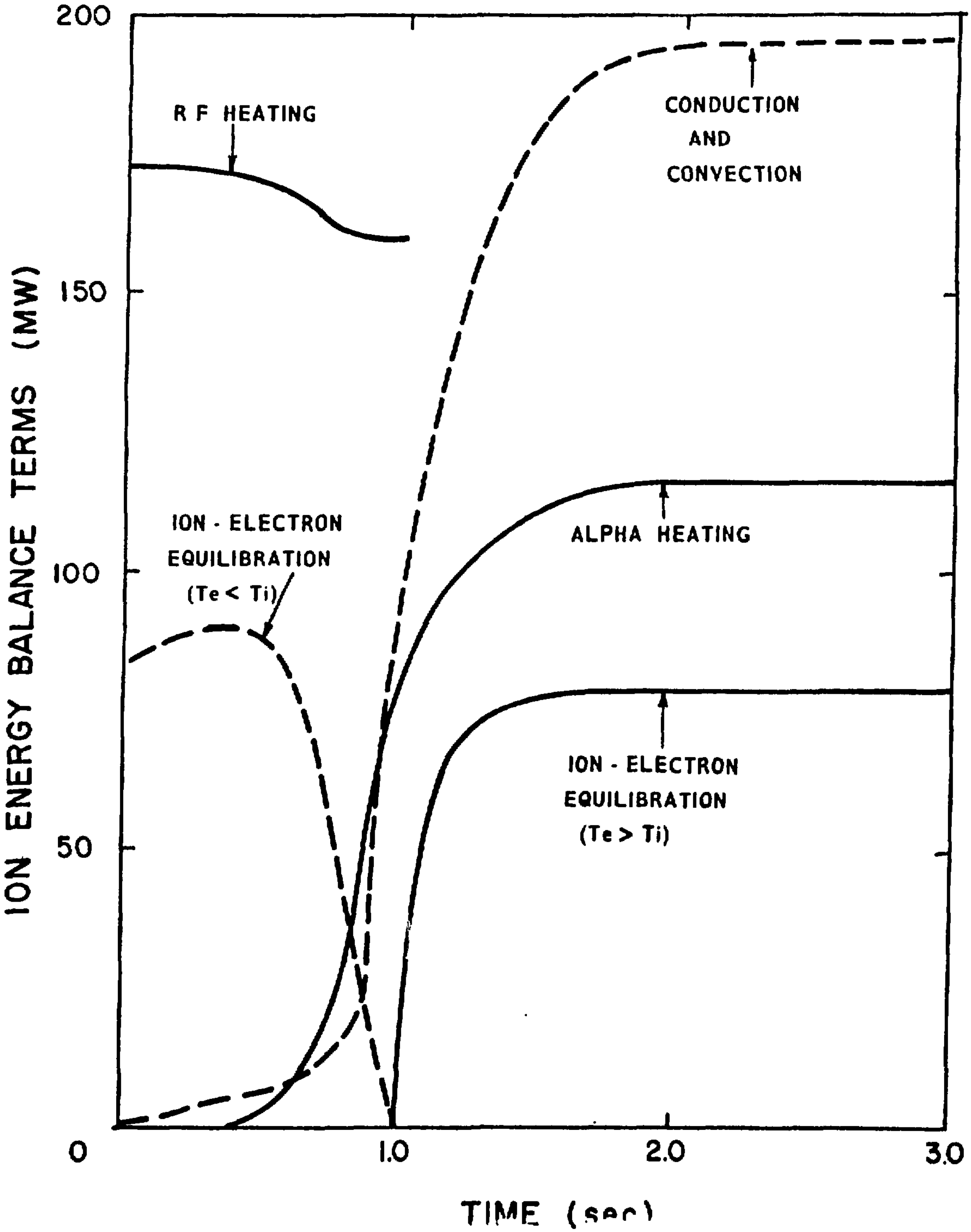
$$Q_p = \frac{\omega U_s}{P_{abs}} \sim 80$$

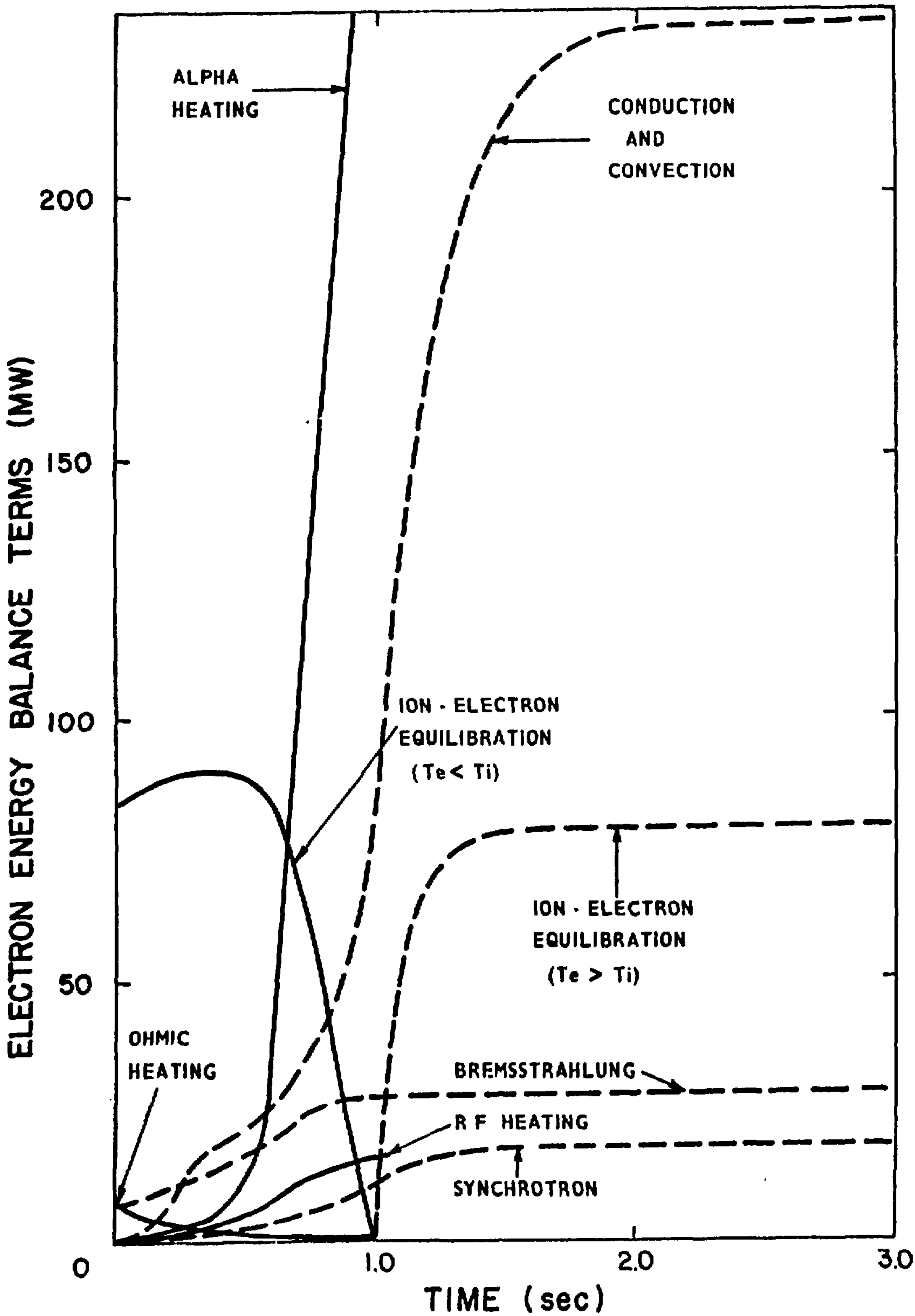


ION TEMPERATURE (KeV)
0
5
10
15
20
0
1.0
2.0
3.0
TIME (sec)
100 MW
80 MW
70 MW
50 MW

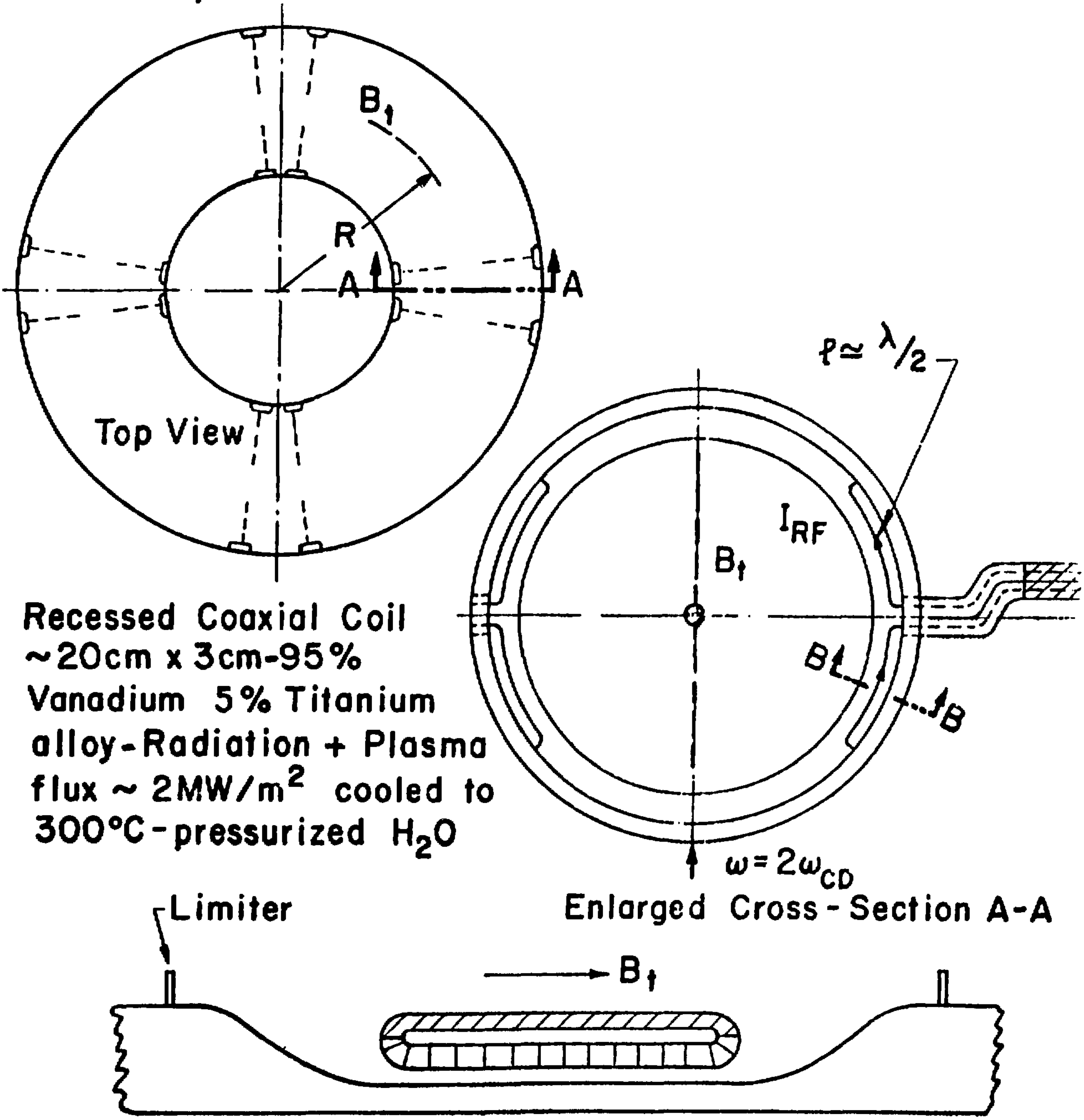
SUBJECT


ION TEMPERATURE (KeV)
0
5
10
15
20
0
1.0
2.0
3.0
TIME (sec)
RF PULSE
200 MW
150 MW
125 MW
100 MW
50 MW

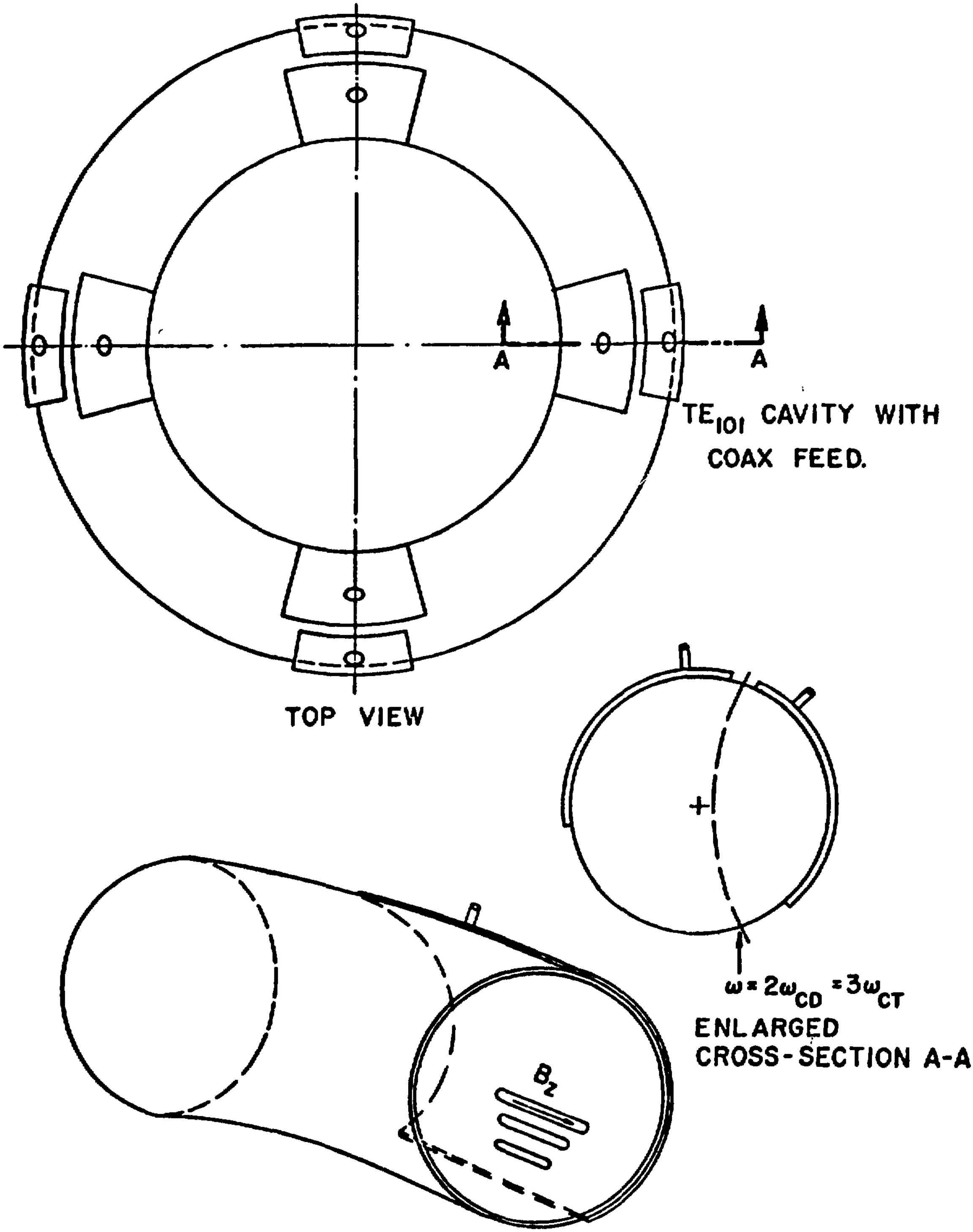

200
150
100
50
0
1.0
2.0
3.0
ION ENERGY BALANCE TERMS (MW)
TIME (sec)
R F HEATING
CONDUCTION
AND
CONVECTION
ION - ELECTRON
EQUILIBRATION
(Te < Ti)
ALPHA HEATING
ION - ELECTRON
EQUILIBRATION
(Te > Ti)


ALPHA
HEATING
CONDUCTION
AND
CONVECTION
ION - ELECTRON
EQUILIBRATION
(Te < Ti)
ION - ELECTRON
EQUILIBRATION
(Te > Ti)
OHMIC
HEATING
BREMSSTRAHLUNG
R F HEATING
SYNCHROTRON
ELECTRON ENERGY BALANCE TERMS (MW)
0
50
100
150
200
1.0
2.0
3.0
TIME (sec)

NUWMAK Base design of RF Coil Coupling Array
80 MW for 3 sec. start up to ignition.
8 RF Units, 10 MW/Unit. $f = 90\,MHz = 2f_{CD}$
$R = 5\,m$, $a = 1.25\,m$



Enlarged Cross-Section B-B



SURFACE APERTURE COUPLING - 90MHz, 10MW/UNIT

LIST OF DEVELOPMENT PROGRAMS IN THE RF HEATING AREA

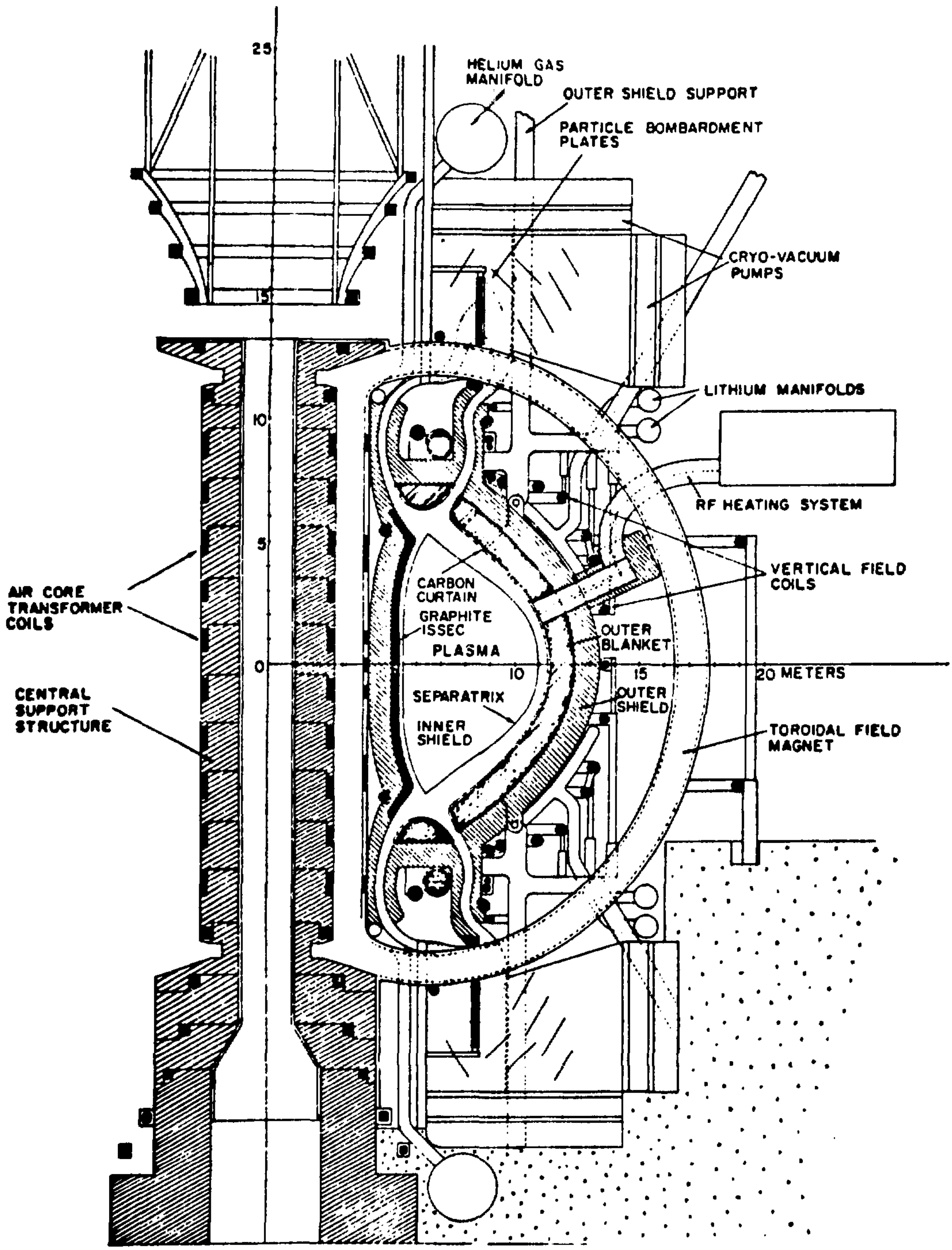
1. RF POWER ENGINEERING IN THE 10 - 100 MW RANGE (CONTROL, COUPLING, MATCHING, SAFETY)

2. MATERIAL DEVELOPMENT FOR RF COUPLING STRUCTURES
   - A) EFFECT OF NEUTRON BOMBARDMENT ON MATERIAL CONDUCTIVITY
   - B) SURVEY OF PROPERTIES OF MATERIALS SUCH AS VANADIUM, MOLYBDENUM, OR OTHER ALLOYS
   - C) COOLING OF COUPLING STRUCTURES: THICKNESS OF MATERIAL, STRUCTURAL INTEGRITY, CHOICE OF COOLANT AND HEAT PIPE EFFICIENCY
   - D) STUDY OF DIELECTRIC PROPERTIES NEAR REACTOR ENVIRONMENT, RF WINDOWS, CERAMIC AND TiO_2 PROPERTIES

3. COMPATIBILITY OF COILS AND APERTURES IN A REACTOR

4. HIGH POWER ARCING STUDIES AT COUPLING STRUCTURE - PLASMA INTERFACE

CROSS SECTION VIEW OF UWMAK III



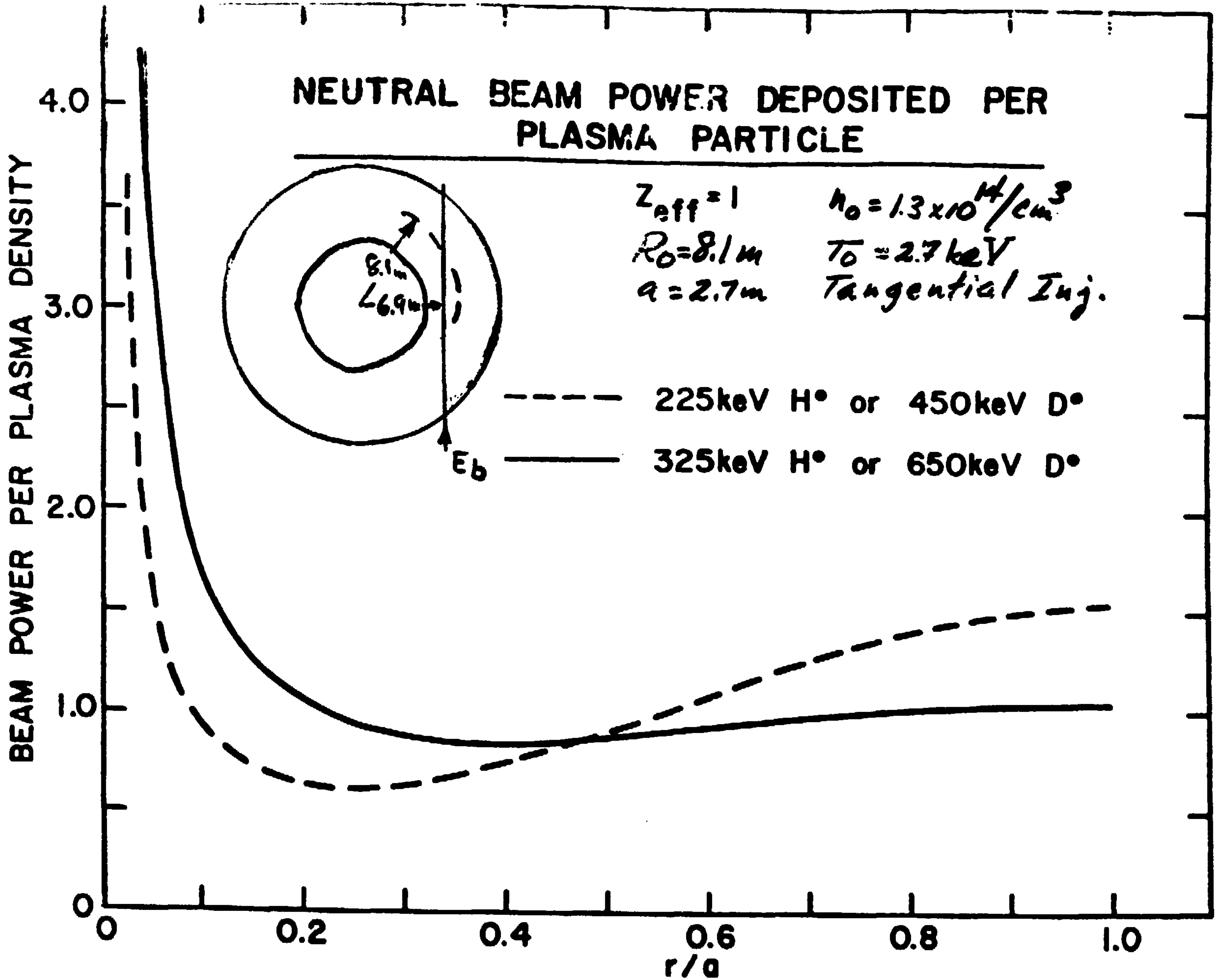


Figure 5-7 - Radial profile of beam deposition per unit volume divided by the plasma density profile for UWMAK-

TALK TO DEPARTMENT OF ENERGY WORKSHOP ON

PLASMA HEATING DEVELOPMENT REQUIREMENTS

December 5, 1977

HEATING AND FUELING REQUIREMENTS UP TO AND BEYOND JET SCALE EXPERIMENTS

BY JOHN SHEFFIELD

HEAD OF TOKAMAK EXPERIMENTAL SECTION

OAK RIDGE NATIONAL LABORATORY

WHAT ARE THE HEATING AND FUELING REQUIREMENTS?

- $P \simeq \frac{abR\ n\ T}{\tau}$ (MW)

  SO THE ANSWER TO THE QUESTION DEPENDS ON THE n,T GOALS FOR GIVEN DIMENSIONS, AND ON THE ENERGY CONFINEMENT TIME τ

- BUT WE DO NOT KNOW τ FOR CONDITIONS OF INTEREST $T > 2$ keV

- THUS ALL WE CAN SAY IS "IF I ASSUME SOME VALUE FOR (τ), THEN A GIVEN POWER (P) WILL LEAD TO A CERTAIN (T), GIVEN (n)."

- WE HAVE SOME KNOWLEDGE THAT (n) IS LIMITED, MURAKAMI SCALING $n \,\alpha \sim \frac{B}{R}$ (OHMIC HEATING) $\sim \frac{P}{A}$ EDGE POWER DENSITY IN GENERAL?

- WE HAVE SOME KNOWLEDGE OF (τ) FOR $T \lesssim 2$ keV, EMPIRICAL SCALING, BUT FOR $T > 2$ keV WE MUST MAKE SOME ASSUMPTION ABOUT THE SCALING WITH (T)

- FOR EXAMPLE IF TRAPPED PARTICLE SCALING APPLIES $P_{LOSS} \,\alpha\, T^{4.5}$ THEN FOR A GIVEN POWER $T \,\alpha\, (P)^{0.22}$, AND THE BENEFITS OF INCREASING THE POWER ARE LIMITED.

A SELF-SUSTAINED PLASMA IS THE MAIN GOAL OF THE TOKAMAK PROGRAM

- SELF SUSTAINING REQUIRES $P_\alpha = P_L$

  THIS AMOUNTS FOR REPRESENTATIVE PROFILES TO THE REQUIREMENT

  $n_s T_s \tau_s \gtrsim 300$ (x 10^{19} m^{-3}, keV, s)

  WITH A WEAK DEPENDENCE ON T

- IN ADDITION SINCE P_α MUST EXCEED P_b (BREMSSTRAHLUNG), WE NEED ($Z_{eff} = 1$)

  $T_{\alpha b} > 4.4$ keV, WITH A WEAK DEPENDENCE ON Z_{eff}

- FOR $Z_{eff} > 1$, LINE RADIATION INCREASES THIS TEMPERATURE.

  IN ADDITION CONDUCTION LOSSES LEAD TO A HIGHER TEMPERATURE.

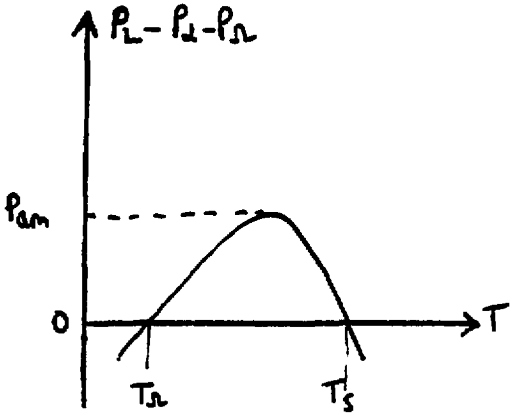
- THE REQUIREMENT THAT MOST OF THE α's ARE CONTAINED AMOUNTS TO

  $I \gtrsim 2.5$ MA

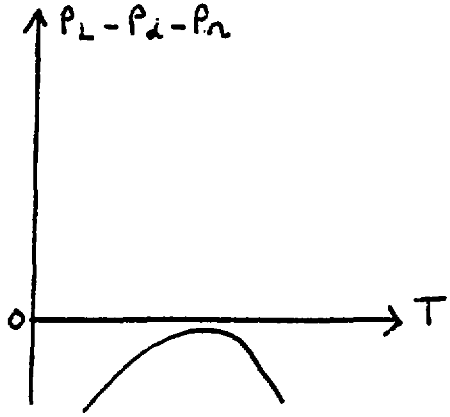
IGNITION IS A GOAL OF THE NEAR-TERM PROGRAM

- IGNITION REQUIRES THAT $\underbrace{P_\alpha}_{\text{alpha}} + \underbrace{P_\Omega}_{\text{ohmic}} + \underbrace{P_a}_{\text{additional}} > P_L = \underbrace{P_b}_{\text{brems}} + \underbrace{P_c}_{\text{"conduction"}}$

THERE ARE ESSENTIALLY TWO MAIN CATEGORIES



$P_a > P_{am}$ REQUIRED FOR IGNITION



IGNITION BY OHMIC HEATING
+ ALPHA HEATING ($P_a = 0$)

SINCE $P_\alpha < P_b$ FOR $T < T_{\alpha b}$ WE CERTAINLY REQUIRE $(P_\Omega + P_a) > P_c$ at $T_{\alpha b}$

FOR IGNITION

WHAT IS THE ENERGY CONFINEMENT SCALING τ?

VARIOUS EMPIRICAL SCALING LAWS HAVE BEEN ESTABLISHED FROM STUDIES OF EXISTING TOKAMAKS, TWO OF THESE ARE:

ALCATOR EMPIRICAL SCALING $\tau_{ae} \sim 4 \times 10^{-2}\, n\, a^2$ (s)

HUGILL-SHEFFIELD SCALING $\tau_{HS} \sim 4 \times 10^{-3}\, n^{0.6}\, a^{1.6}\, B^{0.9}$ (s)

(A DEPENDENCE ON ATOMIC MASS HAS BEEN SUPPRESSED AND THE MOST OPTIMISTIC VERSION HAS BEEN TAKEN.)

THESE SCALINGS REPRESENT, IN A SENSE, THE BEST THAT HAS BEEN ACHIEVED. MANY "UNOPTIMIZED" RESULTS ARE WORSE.

THERE IS ESSENTIALLY NO KNOWLEDGE OF THE SCALING FOR T > 2 keV

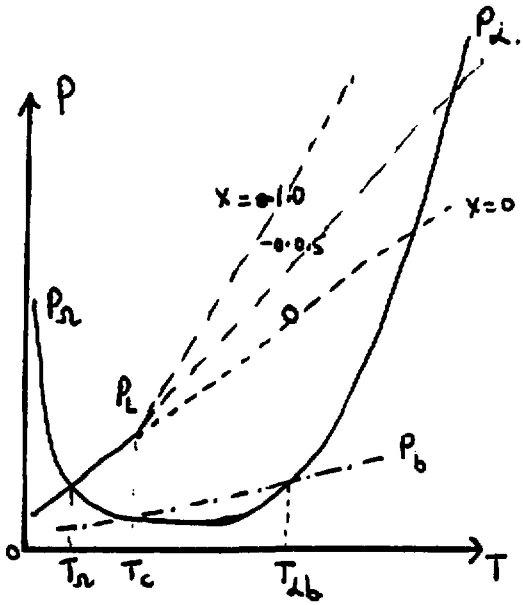
LET US ASSUME THAT $\tau = \tau_{em} \left(\frac{T}{T_c}\right)^x$ IN THE UNEXPLORED REGION.

IT SEEMS PRUDENT TO TAKE T_c = 2 keV AND TO ALLOW THAT $X \leq 0$.

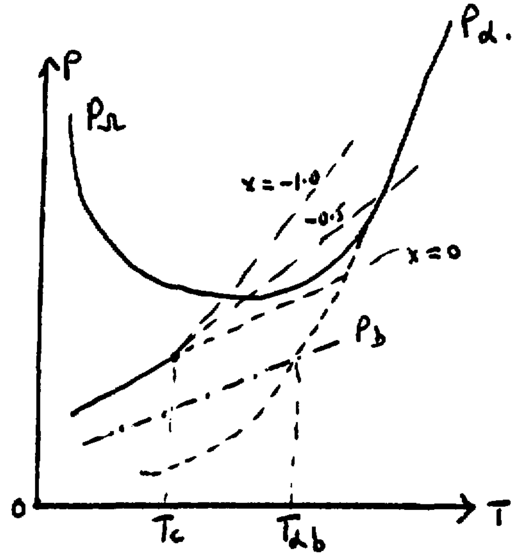
EMPIRICAL (X = 0); PLATEAU SCALING (X = -1.5); TRAPPED PARTICLE (X = -3.5)

IT IS ASSUMED THAT THERE IS NO SUDDEN STEP IN CONFINEMENT TIME AS (T) IS INCREASED AT CONSTANT n, a, B, etc.

WHAT DOES THE POWER BALANCE LOOK LIKE?



ADDITIONAL HEATING NEEDED



OHMIC HEATING MAY BE SUFFICIENT

IGNITION REQUIRES $\quad P_{Ta} = P_\Omega + P_a > P_c \qquad T \sim T_{\alpha b}$

ALCATOR EMPIRICAL SCALING $\quad P_{ae} = 2.5\ R\ T_c \left(\frac{T_{\alpha b}}{T_c}\right)^{1-x}$ MW

HUGILL-SHEFFIELD SCALING $P_{HS} = \frac{25\ a^{0.4}\ n^{0.4}\ R\ T_c}{B^{0.9}} \left(\frac{T_{\alpha b}}{T_c}\right)^{1-x}$ MW

OHMIC HEATING TO IGNITION (B. Coppi et al) $P_a = 0$

EQUATING $P_{ae} = P \rightarrow \frac{I}{a} \gtrsim \left(\frac{T_{\alpha b}^{2.5-x}\, T_c^{\,x}}{0.2\, Z_{eff}}\right)^{0.5}$

- FOR x = 0 and $Z_{eff} = 1$, need $\frac{I}{a} > 15$ MA/m

OR FOR $I \gtrsim 2.5$ MA $a \sim 0.2$ m

- ALTERNATIVELY, SINCE $\frac{I}{a} = \frac{0.5}{q_a}\frac{B}{R/a}$ WE NEED

$B \gtrsim 150$ kG EVEN FOR $q_a \sim 2$ $R/a \sim 2.5$

THUS FOR MOST TOKAMAKS, OHMIC HEATING IS ONLY AN INITIAL HEATING MECHANISM, FOR TYPICAL VALUES OF q_a, R/a, B, and a $T_\Omega \lesssim 1$ keV

$P_a \lesssim P_\Omega$

- THE USE OF MODEST ADDITIONAL POWER LEADS TO A LOWER $\frac{I}{a}$ REQUIREMENT BY THE FACTOR $\left(\frac{1}{1 + \frac{P_a}{P_\Omega}\Big|_{T_{\alpha b}}}\right)^{0.5}$ BUT ONLY A SMALL SUBSET OF DEVICES FALL INTO THIS CATEGORY.

IGNITION WITH ADDITION POWER $P_a > P_\Omega \big|_{T_{\alpha b}}$

- THE ADDITIONAL POWER REQUIRED FOR IGNITION IS

$$P_{am} \simeq 2.5\ R\ T_c \left(\frac{T_{\alpha b}}{T_c}\right)^{1-x} \quad \text{MW FOR ALCATOR EMPIRICAL SCALING}$$

- IF WE APPLY THIS LEVEL OF ADDITIONAL HEATING AND ASSUMING THAT THERE IS A MAXIMUM IN $P_L - P_\alpha - P_\Omega$ at $T \sim T_{\alpha b}$ THEN THE PLASMA WILL EVENTUALLY REACH A TEMPERATURE $T_s \sim T_{\alpha b}$ AT WHICH

$$P_{\alpha s} + P_{as} = \frac{0.1\ a^2\ R\ n_s\ T_s}{\tau_s} \quad \text{WHERE } P_{\alpha s} \simeq \frac{a^2\ R\ (n_s T_s)^2}{3 \times 10^3}$$

- BUT THE MAXIMUM ALLOWED LEVEL OF $n_s T_s$ IS SET BY THE β LIMIT

$$\beta_s = \frac{0.8\ n_s T_s}{B^2} \rightarrow \quad P_{\alpha s} \simeq \frac{a^2\ R\ \beta_s^2\ B^4}{1.9 \times 10^3}$$

- ALTERNATIVELY THIS MAY BE WRITTEN IN TERMS OF THE POLOIDAL β

$$\beta_p = \frac{0.2\ a^2\ n_s T_s}{I^2}$$

- IF THE VALUE OF τ_s IS LESS THAN THE VALUE NEEDED TO GIVE $n_s T_s \tau_s \gtrsim 300$ THEN ADDITIONAL POWER WILL BE NEEDED TO ACHIEVE T_s

THE CONDITION IS $n_s T_s \tau_s \gtrsim \dfrac{300}{(1+f_a)}$ WHERE $f_a = \dfrac{P_{as}}{P_{\alpha s}}$.

- $f_a >> 1$ CORRESPONDS TO THE KIND OF BEAM-PLASMA SCENARIOS PROPOSED FOR TFTR AND JET

- $f_a \lesssim 1$ IS REALLY THE ONLY CASE UNDER WHICH WE COULD STILL TALK ABOUT "IGNITION" IN THE CONVENTIONAL SENSE

- WHILE n_sT_s IS A LIMIT SET BY STABILITY, IT MAY NOT BE REALIZABLE AT RELEVANT TEMPERATURES BECAUSE OF DENSITY LIMITATIONS.

- IF WE INTERPRET MURAKAMI SCALING AS A LIMIT SET BY THE POWER PER UNIT AREA FLOWING TO THE PLASMA EDGE, THEN THIS LIMIT APPEARS TO BE, ASSUMING FLAT PROFILES IN THE LARGER TOKAMAKS,

$$n_m \sim 1.5\ P_w\ (\mathrm{W/cm^2})\ (\times\ 10^{13}\ \mathrm{cm^{-3}}).$$

- CONSIDER THE IMPLICATION FOR TFTR (β_s = 0.05) AND JET EXTENDED (β_s = 0.10) WITH P = 25 MW, FOR T_s = 6.

TFTR, $n_sT_s = 169$, $n_s = 28$, $P_{\alpha s} = 17$ MW
$P_w = 30\ \mathrm{W/cm^2}$, $n_m \approx 45$,

JET (ext) $n_sT_s = 149$ $n_s = 25$ at $T_s = 6$ $P_{\alpha s} = 57$
$P_w = 13\ \mathrm{W/cm^2}$ $n_m = 20$

- THUS ALLOWING FOR THE α-POWER, EVEN JET ON THIS ASSUMPTION WOULD OPERATE AT THE LIMITING DENSITY (n_s). HOWEVER IF WE ASSUMED THAT MURAKAMI SCALING RELATED TO POWER PER UNIT VOLUME, BOTH DEVICES WOULD HAVE $n_m < n_s$.

COULD JET IGNITE?

- ASSUMING THE FULL BENEFITS OF USING NONCIRCULAR PLASMAS PERMIT $\beta_s = 0.1$ AND
  LET THE RELEVANT RADIAL SCALE LENGTH BE $a^* = \sqrt{ab} = 1.62$ m.

- JET

$a = 1.25$	extended	$\beta_s = 0.1$	$\tau_{ae} \gtrsim 2.6$ s
$b = 2.10$	$\beta_s = 34.5$	$\beta_p = 3.4$	($\tau_{HS} \simeq 1.4$ s)
$R/a = 2.37$	$I = 4.8$	$n_s T_s = 169$	
$b/a = 1.68$	$q_a = 6.0$	$P_{\alpha s} = 57$	

- ASSUME ALCATOR-EMPIRICAL SCALING $\quad T = \tau_{ae}\left(\frac{T}{T_c}\right)^x$

LET $T_c = 2$ keV

$T_s = 6$ keV $\rightarrow n_s \simeq 25$

ASSUME $Z_{eff} = 1 \rightarrow P_b \sim 10$ MW

	$n_s T_s \tau_s$	P_{am}	P_{as}	f_a
$x = 0$	387	33	0	0
$x = -0.5$	223	48	20	0.35
$x = -1.0$	129	72	77	1.35

- ON THE <u>MOST OPTIMISTIC ASSUMPTIONS</u> JET COULD "IGNITE." IT REQUIRES
  NO IMPURITY PROBLEMS, $Z_{eff} \sim 1$.
  VERY LITTLE DEGRADATION OF CONFINEMENT WITH INCREASING T: $x \gtrsim -0.5$
  MAXIMUM β AND $\beta_p > R/A$
  THE FULL BENEFITS OF NONCIRCULARITY, I.E., RADIAL SCALE $\sim \sqrt{ab}$
  THAT THE MAXIMUM DENSITY IS PROPORTIONAL TO P_w.

PROPOSED ADDITIONAL HEATING IN JET

- BASIC OPERATION → 25 MW DC SUPPLY → $\sim$ 10 MW HEATING
- EXTENSION TO $\gtrsim$ 63 MW DC SUPPLY → $\gtrsim$ 25 MW HEATING
- INITIAL HEATING — NEUTRAL INJECTION H $\lesssim$ 80 keV
  DEPENDING ON PLASMA SOURCE SPECIES
  — RF HEATING — LHRH AND/OR ICRH
- EXTENSION TO — NEUTRAL INJECTION D $\lesssim$ 160 keV
  DEPENDING ON SPECIES
  — FOR COLLIDING BEAM STARTUP D, T $\lesssim$ 80 keV.
- TO ALLOW FOR BAD CONDITIONS — ADIABATIC COMPRESSION
  COMPRESSION TIME $\tau_c \sim 0.04$ s
- POWER SUPPLIES OF FLEXIBLE FORM TO ALLOW FOR A VARIETY OF COMBINATIONS OF THE HEATING SYSTEMS
- UNIT SIZE. $\sim$ 80 keV, SPLIT SECONDARY 25 → 50 A

ALLOWS $\lesssim$ 40 kV, $\lesssim$ 80 kV, $\lesssim$ 120 kV, $\lesssim$ 160 kV

ICRH LHRH

NBH ⟷

SOME POSSIBLE HEATING AND FUELING SYSTEMS FOR TOKAMAKS

SYSTEM		HEATING	FUELING	COMMENT
GAS PUFFING		NO	YES	FUELING
PELLETS		NO	YES	ONLY
CLUSTERS		YES $\varepsilon \gtrsim T$	YES	
				HEATING AND
NEUTRAL BEAMS	$\lesssim$ 100 keV	YES $\varepsilon > T$	YES	FUELING
H,D,T	> 100 keV	YES	LIMITED	
OHMIC		YES	NO	
RF	$\lesssim$ 10 kH_z CANNOBIO	YES	NO	
SYSTEMS	$\sim$ 100 kH_z $TTMP_i$	YES	NO	HEATING
	$\sim$ 1 MH_z $TTMP_e$	YES	NO	ONLY
	MAGNETIC-ACOUSTIC RESONANCES	YES	NO	
	$\sim$ 100 MH_z ICRH	YES	NO	
	$\gtrsim$ 1 GH_z LHRH	YES	NO	
	$\gtrsim$ 30 GH_z ERH	YES	NO	
ADIABATIC COMPRESSION		YES	NO	

COMMENTS ON FUELING SYSTEMS

- IN DEVICES WITH $na > \dfrac{10\ \varepsilon_b}{f(Z_{eff})\ A_b}$ $(\times\ 10^{19}\ m^{-3} \cdot m)$

  THERE ARE APPARENTLY NO FUELING SYSTEMS WHICH GIVE PENETRATION DIRECTLY (I.E., IN A FEW TRAPPING LENGTHS, TO THE PLASMA CENTER).

- WHILE IT WOULD APPEAR THAT THIS PROBLEM COULD BE OVERCOME BY INCREASING ε_b/A_b, IN FACT AT ACCEPTABLE POWER LEVELS THE CURRENT OF PARTICLES FOR ε_b > 100 keV WOULD NOT BE ADEQUATE FOR FUELING IN TYPICAL CONDITIONS

- RIPPLE INJECTION IS A TECHNIQUE THAT MIGHT ALLOW MEDIUM ENERGY IONS TO REACH THE PLASMA CENTER

- IN PRESENT, SMALL EXPERIMENTS, IONS, FROM GAS PUFFING DO APPEAR TO REACH THE PLASMA CENTER SURPRISINGLY WELL. IF THIS IS DUE TO THE WARE PINCH EFFECT, THEN IT MAY NOT WORK IN LARGER DEVICES WITH SMALLER E_ψ

- IF, HOWEVER, IONS CREATED ON THE OUTSIDE STILL DO PENETRATE, THEN IT SHOULD NOT BE DEPENDENT ON WHETHER THEY COME FROM PUFFING, CLUSTERS, PELLETS, OR MEDIUM ENERGY NEUTRAL INJECTION. THE LATTER THREE OFFER ADVANTAGES IF A DIVERTOR IS USED, AS THEY MAY CROSS THE SCRAPE-OFF LAYER.

HEATING SYSTEMS MAY BE DIVIDED INTO TWO MAIN CATEGORIES:

- CENTER HEATING SYSTEMS - INCLUDE

  OHMIC HEATING

  NEUTRAL INJECTION WITH $\varepsilon_b > 0.1\ f(Z_{eff})\ A_b$ na

  NEUTRAL INJECTION AT LOWER ε_b + RIPPLE INJECTION?

  SOME RF SYSTEMS

  ADIABATIC COMPRESSION

- EDGE HEATING SYSTEMS - INCLUDE

  CLUSTERS

  NEUTRAL INJECTION WITH $\varepsilon_b < 0.1\ f(Z_{eff})\ A_b$ na

  SOME RF SYSTEMS

START UP IN FUTURE LARGE DEVICES WILL TAKE SECONDS

- THE MESSAGE OF EMPIRICAL SCALING IS THAT τ INCREASES WITH (a) and ($\sim$na)

  LARGER DEVICES WILL BE NEEDED TO MAINTAIN GOOD CONFINEMENT IF (τ) DEGRADES WITH (T)

- IN THE LARGE DEVICES THE PLASMA ENERGY CONTENT NEAR IGNITION WILL BE VERY LARGE

  E.G., $a \simeq 2.0$ $b = 3.0$, $R = 6.0$ $n = 20$, $T = 5 \rightarrow W_p = 360$ MJ

  GIVEN THAT HEATING WILL STILL BE RELATIVELY EXPENSIVE, THIS ENERGY MUST BE SUPPLIED AT "MODEST" POWER <100 MW

  THIS INDICATES THAT START UP WILL TAKE A FEW SECONDS

- THE USE OF SUPERCONDUCTING COILS WILL IMPOSE FURTHER RESTRICTIONS ON THE RATE OF RISE OF I AND β, OWING TO LIMITATIONS ON THE RATE OF RISE OF MAGNETIC FIELDS CROSSING THE COILS

- SCENARIOS SUCH AS THOSE PROPOSED FOR $\left\{\begin{array}{l}\text{JET}\\ \text{TFTR}\end{array}\right.$ WILL BE NEEDED

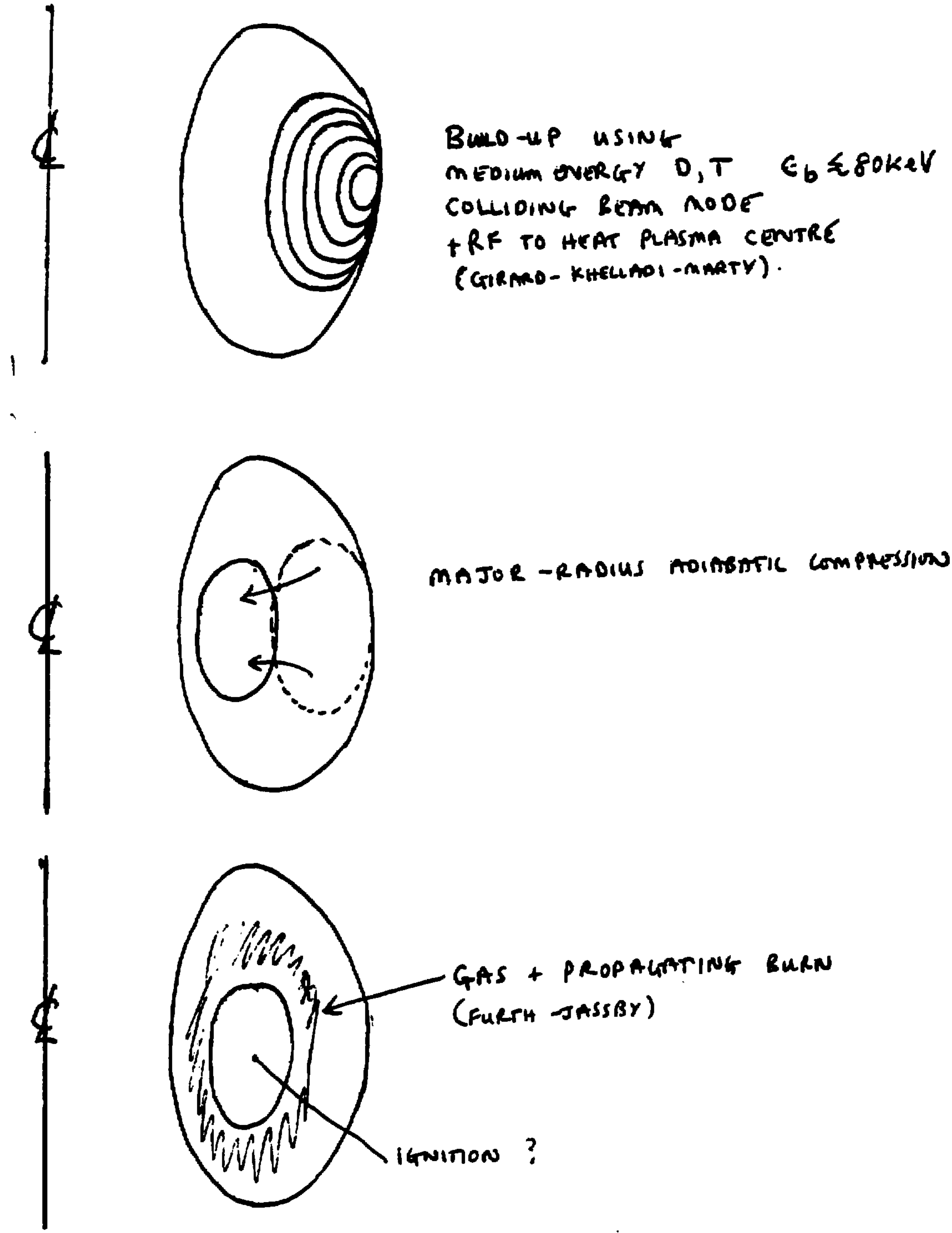
  PLASMA BUILD UP LAYER BY LAYER

  COMPRESSION TO IGNITION FOLLOWED BY A PROPAGATING BURN

  RIPPLE INJECTION

  USE OF BEAM-PLASMA, BEAM-BEAM EFFECTS TO REDUCE ADDITIONAL POWER

  A COMBINATION OF HEATING AND FUELING TECHNIQUES



A PROPOSED SCENARIO FOR JET.

REQUIREMENTS FOR FUTURE HEATING AND FUELING SYSTEMS

- OPERATION IN AN ACTIVE ENVIRONMENT - REMOTE HANDLING
- SYSTEMS REQUIRING ROUTINE MAINTENANCE MUST BE EASY TO MAINTAIN
- SYSTEMS MUST BE VERY RELIABLE
- SYSTEMS MUST BE COST EFFECTIVE

FINAL COMMENTS ON HEATING AND FUELING SYSTEMS FOR THE FUTURE

- OHMIC HEATING - EXCEPT IN THE IGNITOR CONCEPT THIS IS ONLY A PLASMA PREHEATING SYSTEM
- GAS PUFFING - AN IMPORTANT SIMPLE TECHNIQUE, IT MAY NOT BE VIABLE IF DIVERTORS ARE USED
- PELLET INJECTION - A RELATIVELY SIMPLE TECHNIQUE THAT MAY ALLOW PENETRATION BEYOND THE SCRAPE-OFF LAYER IF DIVERTORS ARE USED
- CLUSTER INJECTION - IN THEIR PRESENT FORM IT IS NOT CLEAR WHETHER THESE DEVICES ($\sim$MeV) SATISFY THE CONSTRAINTS OF EASY HANDLING IMPOSED BY OPERATING IN AN ACTIVE ENVIRONMENT. FURTHER THEY SHOULD BE COMPARED FOR COST EFFECTIVENESS WITH LOW ENERGY NEUTRAL INJECTORS

- RF SYSTEMS - THESE ARE AN EXTREMELY IMPORTANT OPTION WHICH SHOULD
  - GIVE HEATING OF THE CENTER AND WILL GIVE HEATING DECOUPLED FROM THE FUELING
  - THE ACTIVE ENVIRONMENT PLACES CONSTRAINTS ON LAUNCHING STRUCTURE AND IT IS NOT CLEAR WHETHER SOME OF THE COIL LAUNCHING SYSTEMS WILL BE POSSIBLE IN REACTORS
  - THE HIGH FREQUENCY SYSTEMS HAVE THE ATTRACTIVE FEATURE THAT THEY USE WAVEGUIDES LAUNCHES

- ADIABATIC COMPRESSION (MAJOR-RADIUS)
  - COMPRESSION PLACES CONSTRAINTS ON THE TOKAMAK DESIGN AND IN TURN IT IS CONSTRAINED BY THE TOKAMAK DESIGN
  - A COMPRESSION TIME $\tau_C \lesssim 0.4\ \tau_E$ IS REQUIRED FOR IT TO BE EFFECTIVE, AND THE CAPABILITIES ARE LIMITED BY THE PERMISSIBLE VOLTAGE ON THE POLOIDAL COILS, THE PENETRATION OF THE FIELDS THROUGH THE MACHINE STRUCTURE AND THE PERMISSIBLE RATE OF CHANGE OF FIELD. IN JET THE MINIMUM $\tau_C \sim 0.04$ s
  - THE EFFECTIVENESS OF COMPRESSION DEPENDS UPON THE SCALING LAW. IT IS VERY EFFECTIVE WITH TRAPPED PARTICLE AND RADIATION LOSSES

NEUTRAL INJECTION

- THE MAJORITY OF SCENARIOS INVOLVING NEUTRAL INJECTION USE INJECTION AT MEDIUM ENERGIES, I.E., $\varepsilon_b \lesssim 80\ A_b$ keV

  EXAMPLES FOR D-T SYSTEMS ARE:

FUELING AND HEATING	BY BUILDUP	$\varepsilon_b \lesssim 100$ keV
	BY RIPPLE INJECTION	$\varepsilon_b \lesssim 100$ keV
BEAM-PLASMA FUSION		$\varepsilon_b \lesssim 80\ A_b$ keV
BEAM-BEAM FUSION		$\varepsilon_b \lesssim 100$ keV

- HIGHER ENERGY BEAM $\varepsilon_b > 80\ A_b$ keV (D,T) ARE RELEVANT ONLY AS HEATING SYSTEMS. AT RELEVANT POWER LEVELS THEY DO NOT GIVE SIGNIFICANT FUELING

  AS HEATING SYSTEMS THEY MUST BE COMPARED FOR COST EFFECTIVENESS WITH ALTERNATIVES SUCH AS RF SYSTEMS

- THE SIGNIFICANCE OF THESE COMMENTS IS THAT THE HIGHER ENERGY BEAMS WOULD PROBABLY BE PRODUCED BY NEGATIVE ION SYSTEMS WHILE FOR THE LOWER ENERGY SYSTEMS $\varepsilon_b \lesssim 80\ A_b$ keV, WE MAY USE EITHER POSITIVE OR NEGATIVE ION SYSTEMS

- A_b IS THE ATOMIC MASS NUMBER

- ANALYSES FOR JET SUGGEST THAT FOR 80 keV, 60 keV (H) and 120 keV (D) WITH SPECIES CONTENT 1:2:3 = 80:10:10 OVERALL EFFICIENCIES COULD BE ACHIEVED WITH POSITIVE ION SYSTEMS OF

22% NO RECOVERY	34% WITH RECOVERY	80 keV H
31% NO RECOVERY	45% WITH RECOVERY	60 keV H
31% NO RECOVERY	45% WITH RECOVERY	120 keV D

- FOR NEGATIVE ION SYSTEMS THE NEUTRALIZATION EFFICIENCY IS ∿70% HOWEVER IF WE ALLOW SIMILAR LOSSES TO THOSE INCLUDED ABOVE, OF PLASMA SOURCE, INTERCEPTION AND REIONIZATION THEN AN OVERALL EFFICIENCY ∿50% MIGHT POSSIBLY BE REALIZED.

- IN DECIDING THE RELATIVE IMPORTANCE TO PLACE UPON THE DEVELOPMENT OF POSITIVE AND NEGATIVE ION SYSTEMS, WE SHOULD ASSESS TOGETHER

  THE NEEDS OF THE PROGRAM

  THE REQUIREMENTS OF SIMPLICITY
  RELIABILITY
  EASY MAINTENANCE

  THE OVERALL SYSTEM EFFICIENCY

  AND THE COST/WATT DELIVERED TO THE PLASMA

CONCLUSIONS ON INJECTION DEVELOPMENT REQUIREMENTS FOR TOKAMAKS

- NEUTRAL INJECTION SYSTEMS REQUIRED FOR TOKAMAKS ARE MAINLY IN THE RANGE $\varepsilon_b \lesssim 80\ A_b$ keV (H,D,T)

- ONLY IF GOOD BEAM PENETRATION IS A NECESSITY FOR HEATING IN LARGE TOKAMAKS, AND RIPPLE INJECTION IS NOT POSSIBLE WILL THERE BE A NEED FOR $\varepsilon_b > 80\ A_b$ keV

- NEGATIVE ION SYSTEMS WILL BE NEEDED IF THERE IS A REQUIREMENT FOR EFFICIENT SYSTEMS WITH $\varepsilon_b > 80\ A_b$ keV

- FOR $\varepsilon_b < 80\ A_b$ keV (H,D,T) EITHER POSITIVE OR NEGATIVE ION SYSTEMS MAY BE USED

  THE PRESENT COST EFFECTIVE DESIGNS WITH THE NEAR TERM POTENTIAL FOR GOOD HANDLING CHARACTERISTICS ARE BASED ON POSITIVE ION SYSTEMS

- A SIZABLE EFFORT SHOULD BE DEVOTED TO MAKING MORE EFFICIENT, CHEAPER, MORE EASILY HANDLEABLE (REMOTELY HANDLEABLE), AND MORE RELIABLE POSITIVE ION SYSTEM WITH $\varepsilon_b \lesssim 80\ A_b$ keV

- A SYSTEMATICALLY INCREASING EFFORT ON NEGATIVE ION SYSTEMS SHOULD BE UNDERTAKEN SO THAT IT IS POSSIBLE TO TAKE ADVANTAGE OF THE POTENTIALLY HIGHER OVERALL SYSTEM EFFICIENCIES

  THIS WORK, HOWEVER, SHOULD NOT BE ALLOWED TO PREJUDICE THE ESSENTIAL WORK REQUIRED FOR THE POSITIVE ION SYSTEMS

POSITIVE ION SYSTEMS — STATE OF THE ART AND ULTIMATE POTENTIAL*

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December 6, 1977

Neutral beams formed by positive ion extraction have proven to be a viable means of heating fusion-like plasmas. State of the art in the Plasma Technology Section at ORNL is depicted in Fig. 1, where positive ion energy per pulse is used as a measure of performance and plotted as a function of time of achievement or expected completion. The PLT or ISX-B ion source has been operated at 40-keV, 60-A, and 0.3-sec pulses with H(D) neutral injected power of 750 kW ($\sim$ 1000 kW) on the PLT device. In the near future these systems will be expanded to 50 keV, 100 A, and 0.5 sec for application on ISX and PDX. Plans have been made to continue the positive ion development in accordance with the requirements for TNS.

A schematic of the PLT or ISX-B ion source is shown in Fig. 2. The extraction plasma is contained by placing permanent magnets arranged about the periphery of a cylindrical anode.[1] The cathode region of the plasma generator serves as an intense source of electrons and feeds the anode plasma. The impedance of the arc is controllable by adjusting the gas feed in the cathode and anode regions and by tuning the coil which surrounds the cathode chamber. High plasma uniformity and arc efficiency are realized,[1,2] and a 70-A current of ions at 35 kV is extracted from a 22-cm grid pattern.

The ion source is coupled to a transport system or beam line with care taken to make the entire system compact, simple, and inexpensive. The injection solid angle is optimized by utilizing the maximum injection orifice and by coupling the

*Research sponsored by the Department of Energy under contract with Union Carbide Corporation.

ion source as tightly as possible to the fusion device. The ion optics are then matched to this injection solid angle. The requirements on the ion optics are kept as conservative as possible to reduce the requirements on critical hardware and thus improve reliability and functionality. Figure 3 depicts the PLT beam line as it was tested and developed on the Medium Energy Test Facility at ORNL. A plasma ion temperature rise of approximately 2 eV per kW of injected power is realized.

The Research and Development program has provided us with several options for focusing and controlling ions. ORMAK optics[3] (1° HWHM) have been improved upon by shaping the extraction apertures (0.6° HWHM). Precel optics, placing a potential between plasma generator anode and plasma grid, provide an improvement of 30% in the beam transmission through a PLT beam transport system. A coated plasma electrode with high precel and large gap provided the best optics yet obtained of 0.33° HWHM at 30 kV.[4] Multistage optics at high energy[5] reveal an improvement of 50% to 70% over ORMAK-type injectors. Care is taken to match the injection requirements to the appropriate optics such that the most reliable and economical system is realized.

A sophisticated monitoring and data analysis system is now available for developmental work on these systems. Figure 4 represents an example of the data taken on a PLT beam line. The power is monitored on each element of the system, and from this information, the ion optics are discernible and the system functionality is easily optimized. Similar data acquisition techniques have been used to measure the beam species in the ion dump region, and then the beam species at the source is calculated from well-known cross sections(see Fig. 5). The atomic yield of 85% is a valuable asset when considering ion optics, neutral beam penetration, and energy recovery schemes. The dominance of the atomic species is evidenced by the species data taken in the ion dump region, as seen in Fig. 6.

The efficiency of neutral beam heating is strongly dependent on beam penetration. Neutral beam energy must be optimally matched to the target or tokamak plasma density, which in turn presents a problem because the plasma varies by as much as an order of magnitude within a given pulse. It is, therefore, desirable to vary the beam energy. The extraction potential can be dynamically varied; however, the ion optics are strongly dependent on energy and the system efficiency is not preserved. Hydrogen and deuterium could be injected simultaneously into the source in a programmed manner that would allow for a wide range of penetration. However, this method would likely complicate the main thrust of the plasma studies and is therefore not desirable. A third option is possible. The species can be varied within a pulse (say from 50% atomic to 90% atomic), resulting in a variation in beam energy which is characteristic of the half energy for 50% atomic yield and full energy for 90% atomic yield. Exact species proportion is strongly dependent on energy and the type of hydrogen isotope used for heating (see Fig. 7).

Neutral beam injection efficiency is limited by the equilibrium fractions of the ions passing through the charge exchange neutralizers. Typical beam penetration requirements are beyond the region in which efficient neutral beams can be produced from positive ions — about 100 keV/nucleon. Energy recovery from the unneutralized portion of the beam leaving the neutralizer will alleviate the low efficiency associated with the high energy acceleration process. Figure 8 shows the typical neutral beam efficiency as a function of beam energy (lower curve). In an idealized situation in which there were no ions or neutral losses and the energy recovery scheme were considered perfect, the efficiency would be 100% (top curve) independent of energy. A more realistic evaluation (central curve) of energy recovery and focusing efficiencies yields a range of improved overall system efficiency that is worthy of investigation. This improvement in efficiency may allow the extrapolation of positive ion extraction for use in the heating of reactor-grade plasmas.

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3. R. C. Davis, O. B. Morgan, L. D. Stewart, and W. L. Stirling, Rev. Sci. Instrum. 43, 278 (1972).
4. J. H. Whealton, L. R. Grisham, C. C. Tsai, and W. L. Stirling, J. Appl. Phys., in press.
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ORNL-DWG 77-7406R

THE OAK RIDGE NATIONAL LABORATORY ION SOURCE DEVELOPMENT HAS BEEN PROCEEDING AS AN EXPONENTIAL FUNCTION OF THE EXTRACTED ENERGY.

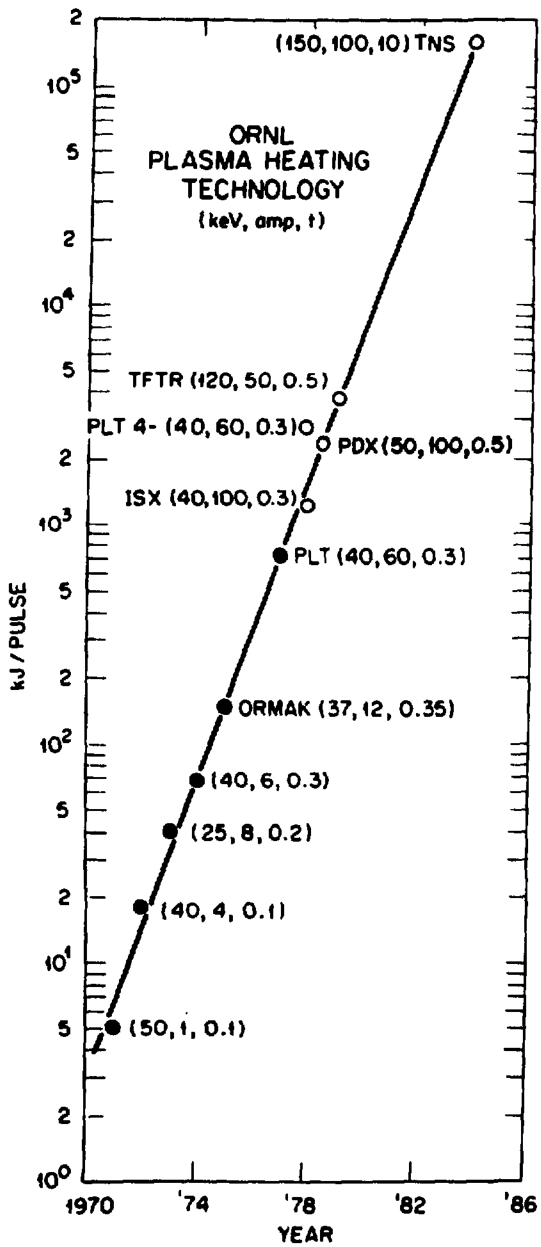
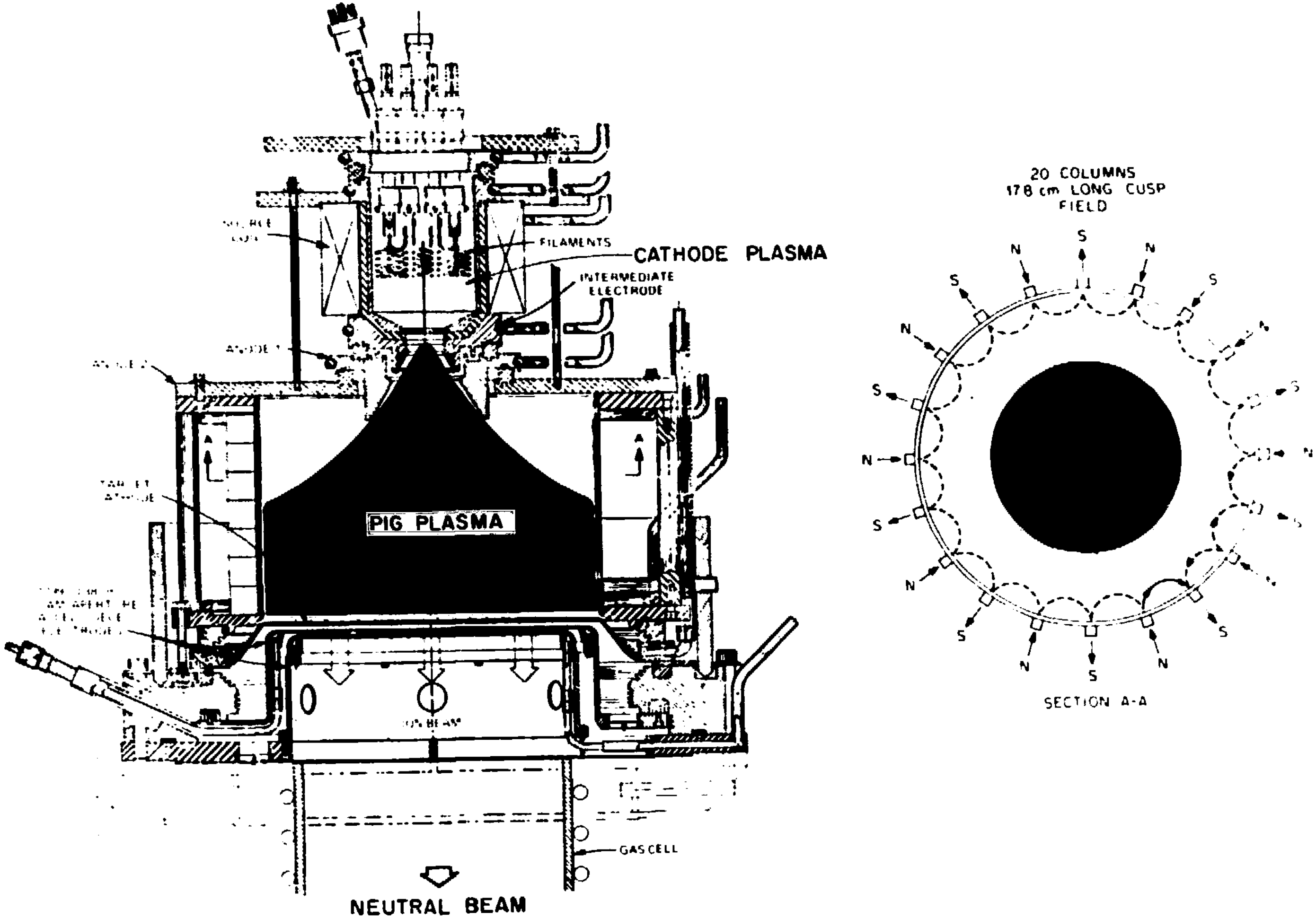
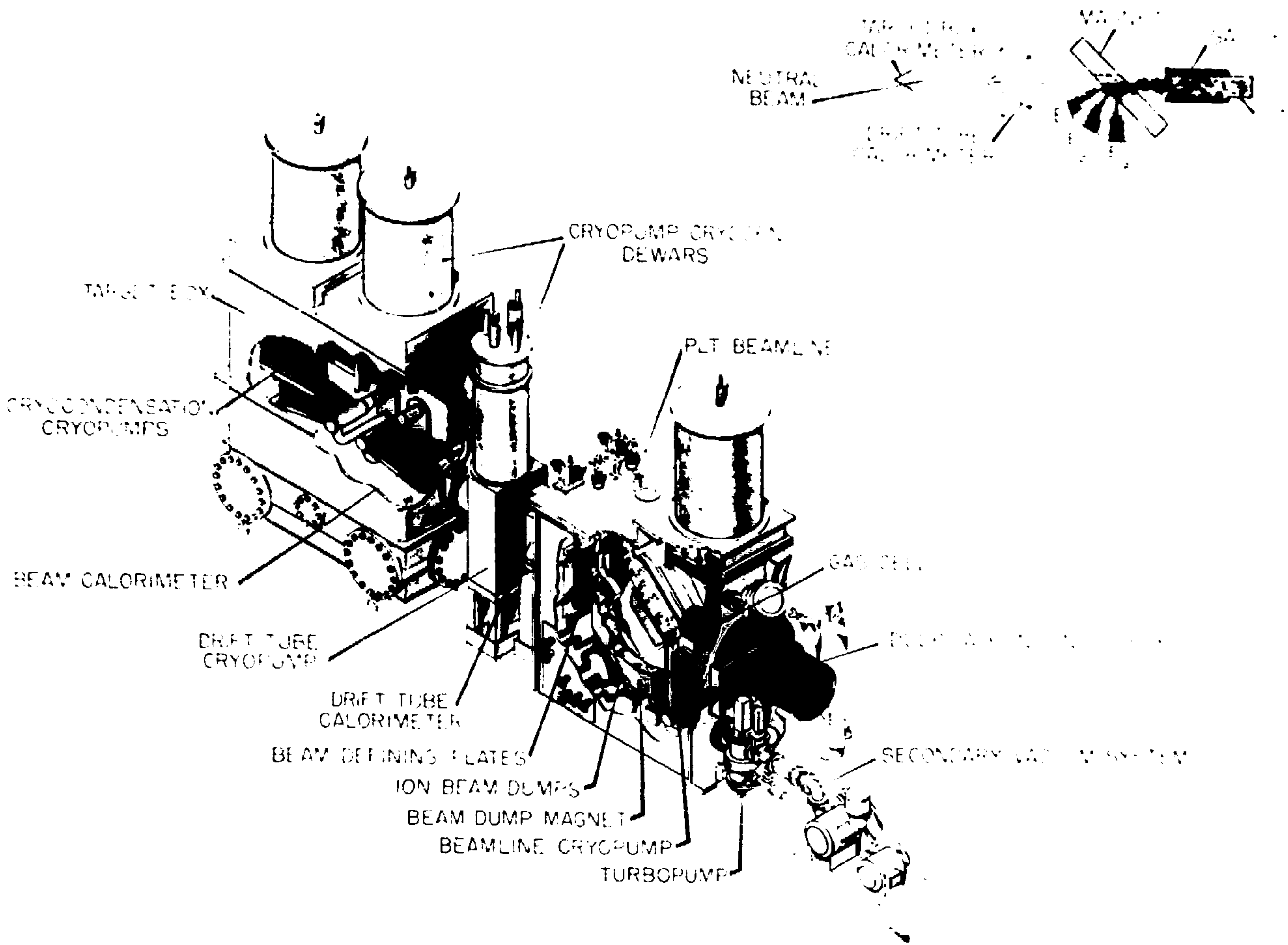


Fig. 1



PLT — ISX-B 22 cm Ion Source

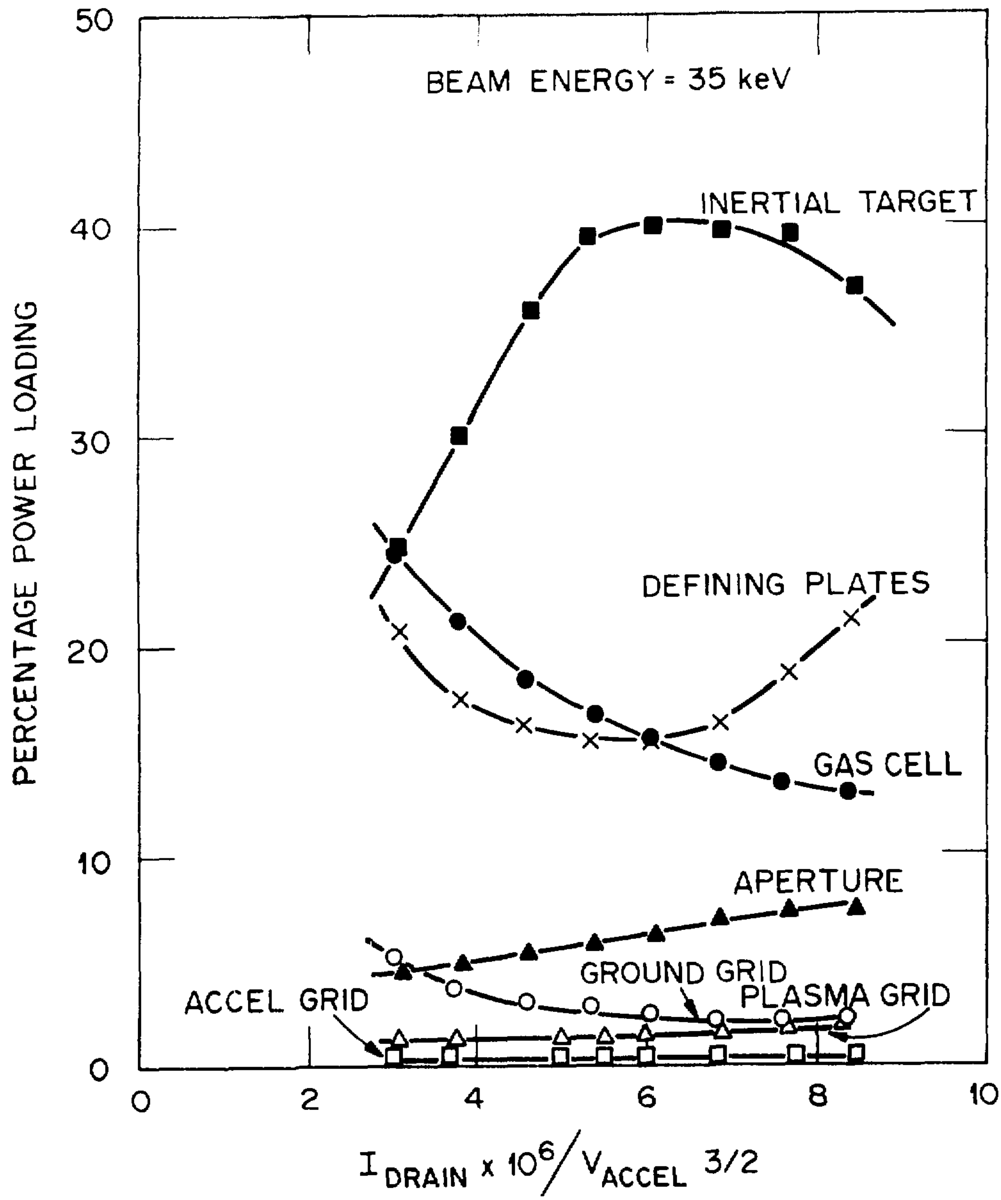
Fig. 2



ORNL/PLT BEAMLINE AND ORNL TEST FACILITY TARGET BOX

Fig. 3

ORNL/DWG/FED 78-147

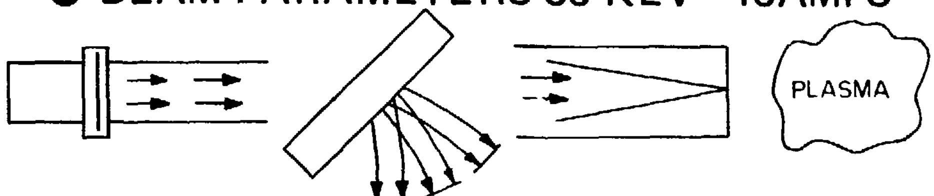


Typical Data From PLT Ion Source and Beam Line

Fig. 4

FROM MEASUREMENTS OF SPECIES IN THE ION DUMP REGION, THE SPECIES EMERGING FROM THE SOURCE AND THE NEUTRAL SPECIES RATIO CAN BE CALCULATED

- BEAM PARAMETERS 30 KEV–40AMPS



- ION SOURCE

  $E(H^+)$ – 85%

  $E(H_2^+)$ – 12%

  $E(H_3^+)$ – 3%

- ION DUMP

  $E(H^+)$ – 89%

  $E/2(H^+)$ – 10%

  $E/3(H^+)$ – 1%

- TOKAMAK PLASMA

  $E(H^o)$ – 84%

  $E/2(H^o)$ – 13%

  $E/3(H^o)$ – 3%

Fig. 5

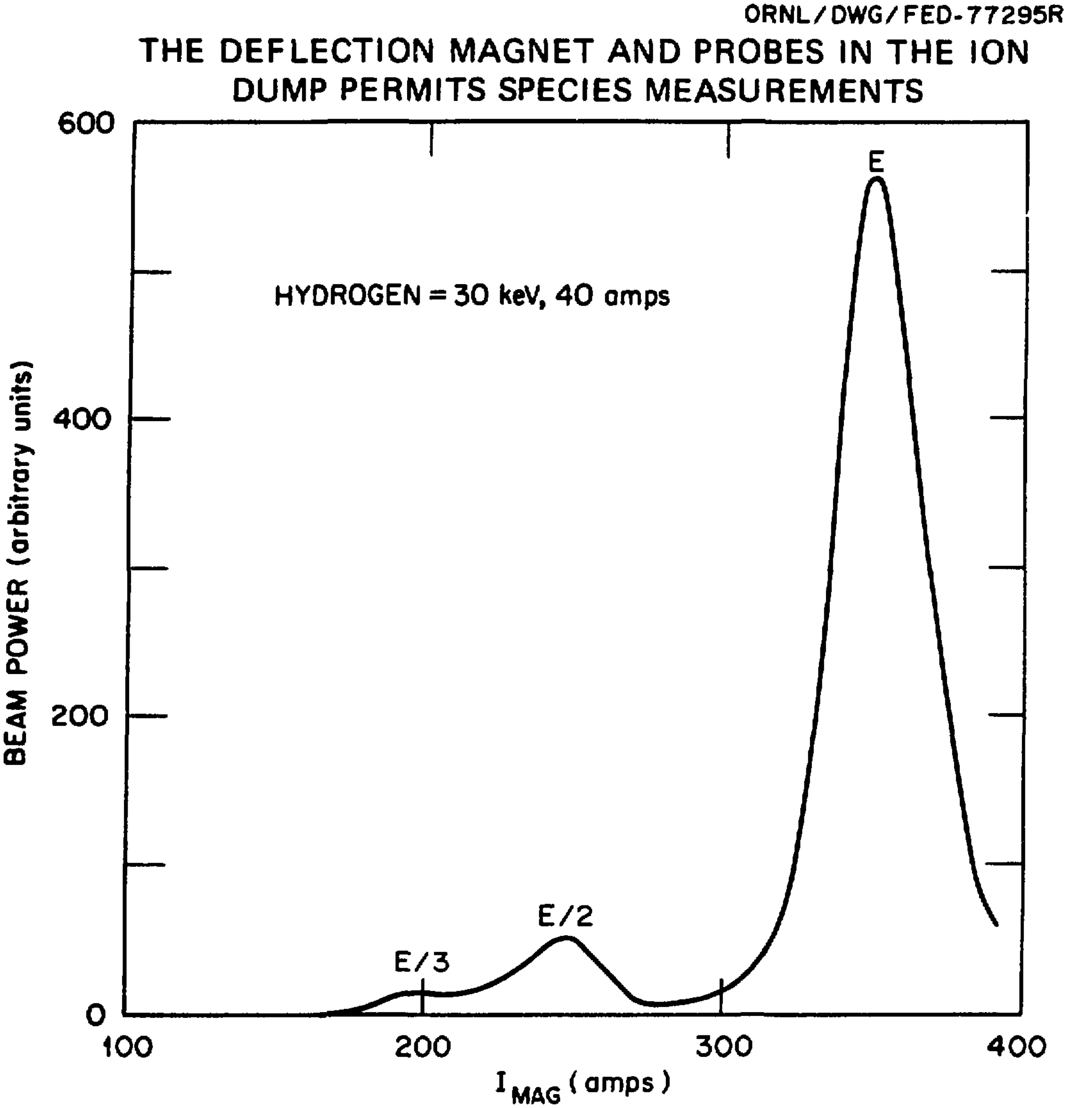

ORNL/DWG/FED-77295R
THE DEFLECTION MAGNET AND PROBES IN THE ION DUMP PERMITS SPECIES MEASUREMENTS
HYDROGEN = 30 keV, 40 amps
E
E/2
E/3
BEAM POWER (arbitrary units)
600
400
200
0
100
200
300
400
I MAG (amps)

Fig. 6

ORNL/DWG/FED-77906R

CONTROL OF SPECIES YIELD WILL PERMIT CONTROL OF BEAM DEPOSITION WITHIN TOKAMAK.

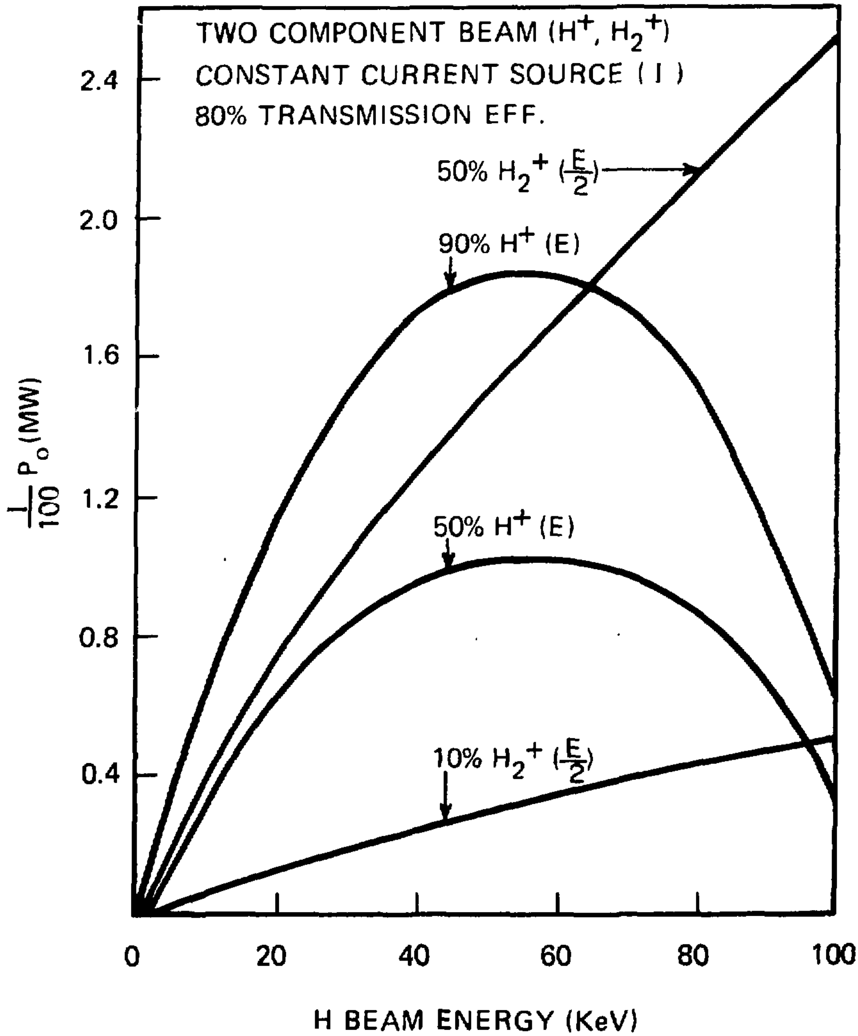


Fig. 7

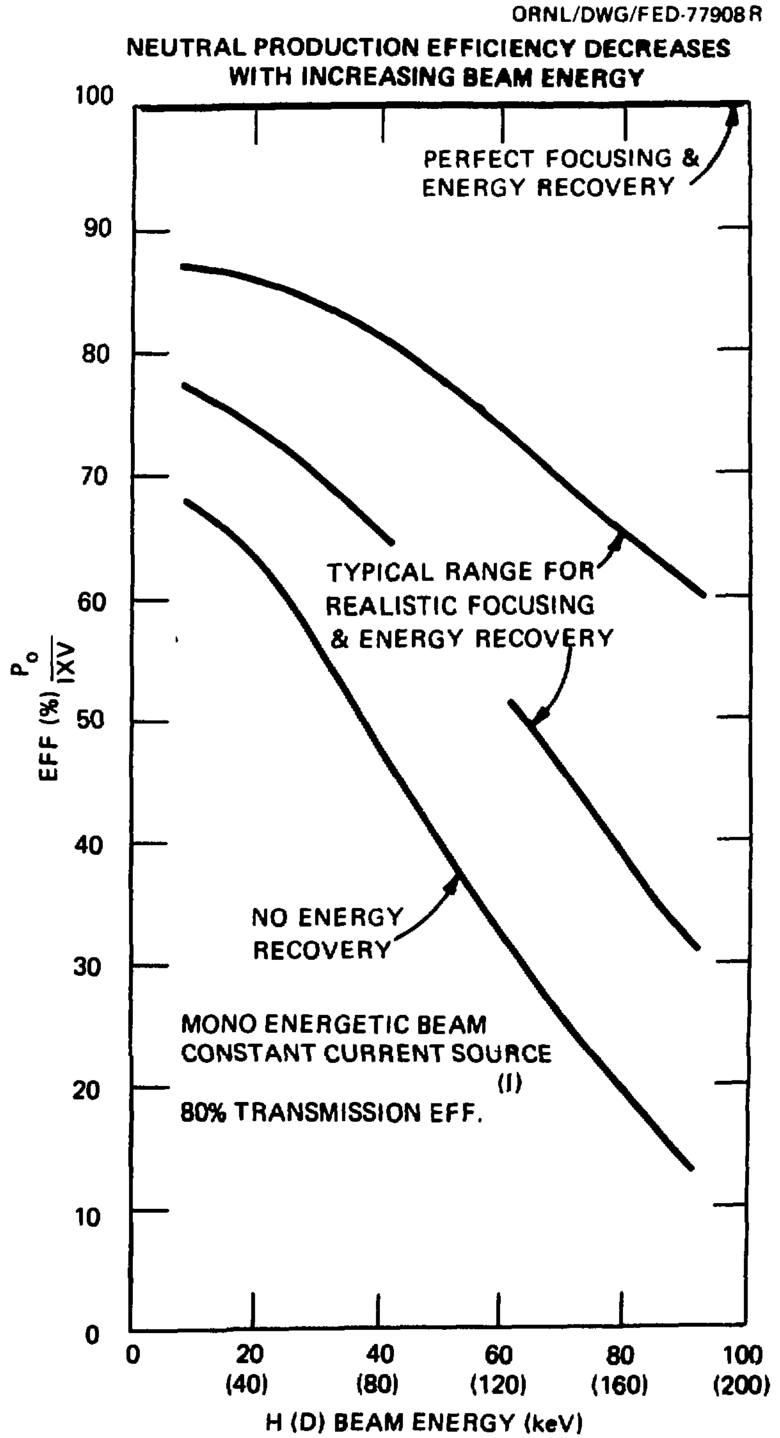

ORNL/DWG/FED-77908R
NEUTRAL PRODUCTION EFFICIENCY DECREASES WITH INCREASING BEAM ENERGY
PERFECT FOCUSING & ENERGY RECOVERY
TYPICAL RANGE FOR REALISTIC FOCUSING & ENERGY RECOVERY
NO ENERGY RECOVERY
MONO ENERGETIC BEAM
CONSTANT CURRENT SOURCE (I)
80% TRANSMISSION EFF.
EFF (%) $\frac{P_o}{I \times V}$
H (D) BEAM ENERGY (keV)
0
10
20
30
40
50
60
70
80
90
100
0
20 (40)
40 (80)
60 (120)
80 (160)
100 (200)

Fig. 8

UCID-3987

THE LBL/LLL POSITIVE-ION-BASED NEUTRAL BEAM DEVELOPMENT PROGRAM STATUS*

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Summary

The LBL/LLL neutral beam development program consists of two parts: a near-term (through ~ 1985?) program based on the acceleration and neutralization of positive ions, and a longer-term "efficient-beam" program. The emphasis in the present positive ion-based work is on 80- to 120- kV, 65- to 80-A, 0.5-S modules. We will go toward long-pulse (dc), and perhaps somewhat higher energy, operation during the next few years.

For efficient injection above 150- to 200- keV (D^o), the present LBL/LLL plan is to develop a negative-ion approach suitable for 200- to 400 kV injectors on confinement experiments in the 1985-90 period, although the timescale might be accelerated, if required. The negative-ion program is described elsewhere in the proceedings of this workshop.

I. Introduction

The Lawrence Laboratories Neutral-Beam Development Group is developing injection systems for mirror and tokamak confinement devices. The work proceeds along two lines: The first, based on positive-ion technology, is required for the near-term (at least through 1985) applications. The presently identified

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*Work done under the auspices of the United States Department of Energy.

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experiments, 2XIIB, TMX, MFTF, TFTR, and DIII, require injection at energies up to 120 keV, ion currents per module up to 80 A, and pulse lengths to 0.5 sec. This meeting should help to identify parameters for upgrades and future devices, if any, whose requirements can be met with positive-ion-based systems.

We have achieved 120-kV, 0.5-sec operation of a fractional-area (12 A) TFTR prototype source, and have tested a full-size TFTR source to 45 KV and 10 A for 200 ms (as of 12/1). A first model of a fractional area, unoptimized, 80 kV source has been operated with 80 kV, 14 A, 250 ms pulses. No basic problems with the source designs have shown up.

As a result, we are fairly confident about developing useful injectors for the near-term applications, including up-grades for longer pulses. However, it is important to emphasize the fact that much work remains to be done to increase efficiency and reliability, and decrease cost, as well as help develop an industrial capability. Better understanding of the physics is badly needed.

The second development effort is oriented toward longer-term applications requiring efficient neutral-beam systems at energies well above 120 keV. Most of these systems probably will require the production and acceleration of large currents of negative ions. Two of our goals are the demonstration of a 200-kV, 20-A (D^o), $\sim$dc system by 1981, and a 400 kV, 20-A (D^o), $\sim$dc system by 1983. These achievements will make it possible to have 200- to 400-kV injectors on confinement experiments in the 1984-90 period. The negative-ion based program is described elsewhere in the proceedings of this meeting.

II. The Positive-Ion System Status

Although positive-ion-based systems are not the main subject of this meeting, there are many elements and problems in common with negative-ion-based systems, so we will go into their present status in some detail to help the reader to evaluate

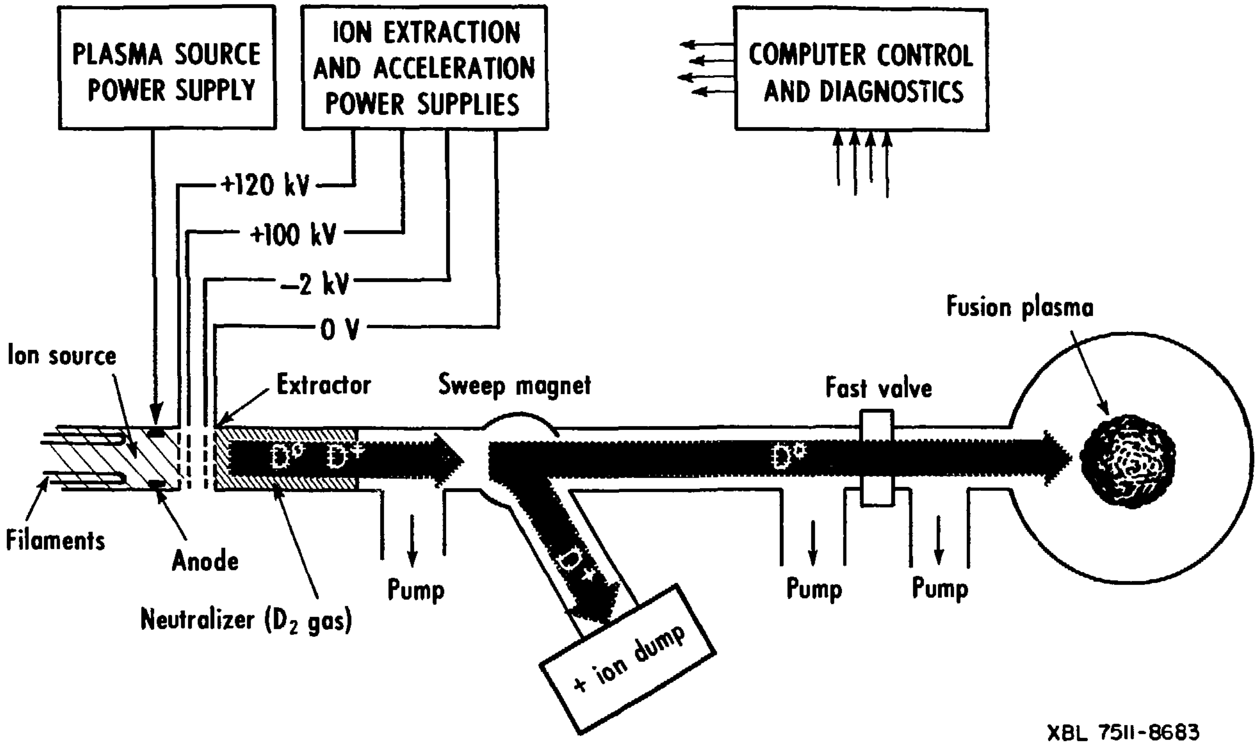

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Fig. 1. Schematic of a typical neutral-beam injection system.

the work yet to be done. (Six years of intensive effort have been required to reach the present point.) The principal components of these neutral-beam injection systems are shown schematically in Fig. 1.

The system operation is as follows: A deuterium plasma is created in the plasma generator by means of a high-current discharge. Ions from this plasma are accelerated in a carefully designed multi-electrode structure. The ions then pass through a neutralizer containing deuterium gas, and a fraction becomes neutralized by charge-exchange collisions. Remaining ions are removed from the beam by the sweep magnet; otherwise, the various reactor magnetic fields would bend the ions into surfaces near the entrance port, possibly releasing gas bursts or melting the surfaces. The considerable power in this ion beam at present must be handled by the ion-beam dump; in the future this power should be reduced by the incorporation of decel (energy recovery) systems.

The vacuum pumps distributed along the beam line remove most of the gas emerging from the neutralizer and the ion-beam dump, and must maintain the pressure between the sweep magnet and the entrance port at a sufficiently low value that very little of the neutral beam is re-ionized. Well-regulated power supplies are required to assure good beam optics. To minimize accelerator damage when a spark occurs, the power supplies must also be capable of rapid turn-off with a minimum of stored energy (e.g. in cable capacitance). Optical, mechanical, and electrical sensors determine the condition and performance of the neutral-beam system and permit the control system to adjust the power-supply voltages and to shut down the systems if a malfunction occurs.

Test Facilities

Before discussing beam-line elements further, we note that so far the pacing item in the neutral-beam program has been the development of suitable test facilities. Consequently, we have put the major part of our effort into this area for the past several years. Some of the parameters of our high-voltage facilities are given in Table I.

Facility	Electronics kV	A	S	Vacuum System V(ℓ)	Speed (ℓ/S)
III A	120	20	0.5	170,000	60,000 (D_2)
III B	120	80	0.05		
HVTS	80	85	30	20,000	650,000 (D_2)
	120	65	30		
	200	20	DC		

Table I. High-Voltage Test Facilities

The LBL 120-keV Neutral-Beam Test Facility[1,2] has two beam lines and associated power supplies: (A) a 150-kV, 20-A, 0.5-sec power-supply system[3,4] to test small-area injector modules (plasma source and accelerator), and (B) a 120-kV, 80-A, 50-msec power-supply system to test full-scale modules for short pulses. The beam-diagnostic system for these beam lines is described in Ref. 5.

Long-pulse testing of high-current ion sources will be completed on the High Voltage Test Stand (HVTS) at LLL. This facility is at present in a configuration suitable for testing positive ion sources in various voltage-current combinations. The initial application will be for TFTR 120-kV, 65-A, 0.5-sec modules. In a year or so, the HVTS will be modified to permit testing of a negative-ion system at 200 kV, 20 A, dc.

Plasma Source

A cross-section of the 120-kV, 10-x10-cm TFTR development module is shown in Fig. 2 The ions are produced in a high-current low-voltage discharge with no externally applied magnetic fields. The cathode consists of 84 (10-x10-cm) to 210 (10-x40-cm), 0.5-mm-diam, 11-cm-long tungsten filaments; the anode is made of one or more molybdenum plates, shown in the top of the figure. Details of this type of plasma generator can be found in Ref. 2.

These plasma sources operate reliably, and produce uniform, noise-free plasmas, in part because they do not have magnetic fields in the center. The atomic fraction is approximately proportional to the arc power. In the progression from 40 kV to 120 kV sources, the atomic fraction has dropped, partly because of the decrease in ion current density (see the next section). We have shown previously[6] that an extension of the source barrel (in the beam direction) can raise the atomic fraction from 65% to 88% and that the addition of multipole magnetic fields (to form a "MacKenzie bucket") can maintain the required plasma

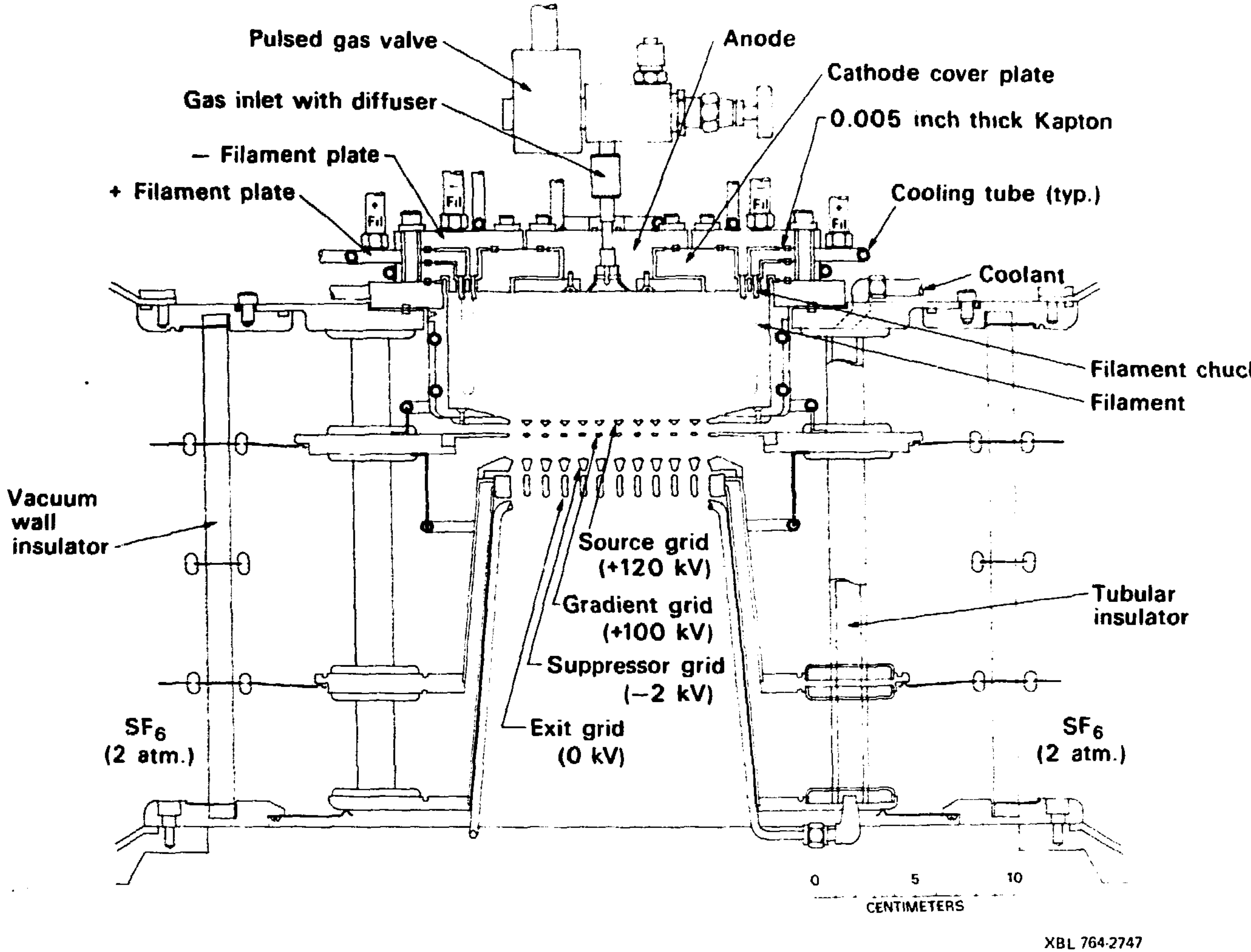


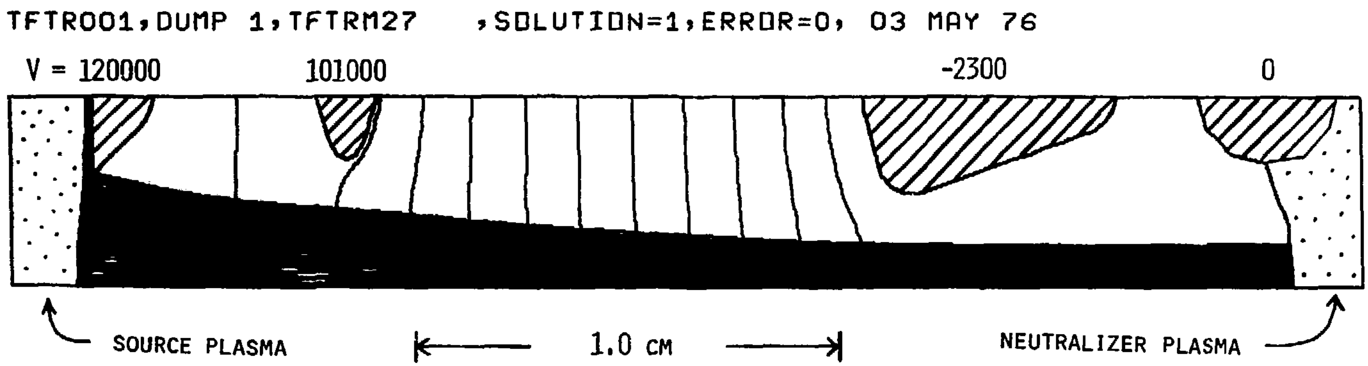
Fig. 2. Cross section of the 120-kV, 0.5-sec source module with a 10⁻ x 10- cm grid array.

uniformity. The ORNL neutral beam group has had good experience with these multipole arrays and the DITE group at Culham also reports good preliminary results with sources of the general LBL configuration, but with "MacKenzie buckets" magnets on the surface. There is no detailed understanding of the physics of any of these sources, but there are some apparent advantages: The atomic species mixture may be better, and we may be able to use fewer and thicker filaments or hollow cathodes (desirable for long-pulse and dc operation) as electron sources. Consequently, we have built a "MacKenzie bucket" plasma source to evaluate with our 10-x10-cm 80-kV and 120-kV accelerator structures.

Accelerator

A four-grid (three-gap) multiple-slot accelerator array shown in Fig. 2, (a cross section of half of a single slot of the TFTR array is shown in Fig. 3) is used. Ions are accelerated and electrostatically focused in the first two gaps; the third gap has a weak decelerating field to suppress down-stream electrons. The transparency of the array is 60%; the scale size was set by the desire to limit the maximum potential gradient to about 100 kV/cm (our estimate of the breakdown limit) and resulted in a design ion-current density of 0.31 A/cm^2 for a pure D^+ beam, or about 0.25 A/cm^2 for a beam with a realistic mixture of D^+, D_2^+, and D_3^+. The design shown in Fig. 3 was optimized, using the WOLF code,[7] by varying the shape of the first, beam-forming, electrode and the potential of the second, gradient-grid, electrode. The shapes of all electrodes except the first were chosen to minimize energy deposition in the structure by secondary particles created by ionization of the background gas or by secondary emission from grid surfaces.

Our 80 kV sources are of similar design, but the shapes, sizes, spacings, and curvatures of the electrodes are still being optimized.



ION CURRENT DENSITY = 0.31 A/CM^2 (D^+)

TRANSPARENCY = 0.6

BEAM DIVERGENCE ($\sqrt{2}\,\theta_{RMS}$) = 0.53 DEGREES

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Fig. 3. Calculated beam trajectories and equipotentials for a 120-kV accelerator.

An unfortunate consequence of the scaling laws, as we understand them, is that the current density, of a minimum divergence beam is inversely proportional to the square root of the beam energy; consequently j^+ has decreased from about 0.5 A-cm^{-2} (at the plasma) at 20-40 keV to about 0.25 A at 120 keV.

There are a number of important unanswered, or partially-answered questions that affect planning for the present and higher-energy designs: For example, we don't know the limits on energy storage (in the source, power supplies and cables) above which damage will occur. We have tried to determine the maximum stored energy allowed without adversely affecting operation of the module by adding capacitance to the source. Very preliminary results indicate that source performance is degraded, i.e., several lower-voltage conditioning pulses are required before operating at the original voltage, when more than 7 J are dissipated in a spark.

To date the power loading of the electrode structures by halo beams and secondary particles has not been measured; our first instrumented 10-x10-cm source will be finished in about a month. In the meantime, the computer program WOLF is used to adjust electrode shapes for minimum power loading.

The information on maximum voltages and electric fields in accelerator gaps of high-current devices is minimal. We are operating at gradients slightly above 100 kV/cm; this is about twice the gradient that conventional accelerator designers think prudent.

Neutralizer

The neutralizer is a closely-coupled structure constructed of iron to shield the beam from stray magnetic fields. For TFTR module testing, it has an internal cross section of 15 cm x 50 cm and is two meters long. D_2 gas emerging from the plasma generator through the grids produces a line density of $\sim 10^{16}$ molecules/cm^2 ($\sim$0.5 Pa-m) in this section.

Although present neutralizers are conceptually very simple, a fairly dense plasma is produced in them by the high-energy beam. We have some reason to suspect that beam-plasma interactions may contribute to the observed angular divergences. (This is one of the reasons that we can not assume that the angular divergence of a high-current beam will be the same as that of a low-current beam.)

As will be mentioned briefly later, there are reasons to have separate neutralizers for the individual beamlets rather than a single one for the total beam. The feasibility of this is being studied.

Sweep Magnet

Until in-line energy recovery systems are perfected, most injection systems (not MFTF) will use separate magnets to steer positive ions emerging from the neutralizers onto beam stops in the least-damaging way. To date these have been transmission magnets. However, J. Warren Stearns (LBL) pointed out that, in principle, reflection magnets are preferable for some applications. The DIII injectors will use magnets of this type.

Beam stop and calorimeter

The beams are stopped by copper-plate calorimeters, instrumented with arrays of thermistors. The beam power and divergence are determined from temperature measurements after the beam pulse. The plates are water cooled (at the backs) between shots. It is estimated that this technique is suitable for pulse lengths up to about 3 seconds.

Vacuum system

Suitable pressures are maintained in the present systems by combinations of expansion, diffusion pumps, gettering, and cryocondensation pumps.

We assume that most future installations will use cryocondensation pumps. Some concern has been expressed in the past that radiation might remove condensed gases from panels, in which case cryosorption pumping might be preferable. We have an experimental program to look at this.

Diagnostics and controls

Because of the very high power in present-day positive-ion-based systems, it is difficult to diagnose their performance; and it is catastrophic if poor performance or malfunctions are not diagnosed and acted on promply. We use, in conjunction with hard-wired and/or computer-controlled circuits, many thermal, electrical, and optical systems.[5,8] The computer-based diagnostic and control system is described in Ref. 9.

High-Power Electronics

More effort and money has gone into the design, construction, and debugging of the power supplies, switches, modulators, and their controls, than into any other element of the R and D program. This careful, innovative development work[3,4] has been the key to successful operation of high-current sources. Principles of working systems are described in a separate paper at this meeting.

Operation

A typical beam-pulse sequence is as follows: The water to the calorimeter is turned off and the diagnostic computer receives a signal to record the pre-shot thermistor temperatures on the calorimeter. Approximately 5 sec later the filament-power supply is turned on; approximately 2 sec later the filaments have reached their emission temperature and gas is pulsed into the plasma chamber. Within 20 msec the are-power supply is turned on. When the discharge has stabilized the accelerator voltage is applied to the grids by firing the series switch.[3,4]

The potential for the gradient grid (see Fig. 2) is obtained from the accelerator power supply by means of a resistive voltage divider; the suppressor supply is slaved to the accelerator supply in that it is gated on by a signal from the high-voltage divider. The rise time of the potentials applied to the three grids is approximately 30 sec. When any of a number of fault sensors is activated (drop in high voltage below a preset threshold, loss of voltage across the first gap, excessive suppressor current, excessive gradient-grid current, etc.) the series switch is "opened" by firing the shunt switch.[3,4] This removes all high voltage from the grids; the discharge, however, is not turned off. After a present "interrupt" time, typically 21 msec, the series switch is again fired and the sequence is repeated. At the end of the 0.5-sec pulse all power supplies are turned off. Approximately 5 sec later the final temperatures of the calorimeter thermistor array are recorded by the computer, and the calorimeter water is turned on to cool the plates.

A saturable-core reactor[3,4] dissipates much of the energy stored in the stray capacitance of the power supplies and cables; the stored energy delivered to the grids by a spark is approximately 2 J (the energy stored in the capacitance of the module).

Operation of the module by interrupting only the grid voltages and leaving the discharge on permits rapid re-starts. It does, however, cause a problem when the potentials are applied to the grids: Without any potential difference between the grids, ions and electrons from the discharge fill the grid region; when the potentials are applied, these ions and electrons must be swept out of the gaps (possibly emitting secondary electrons as they strike the grids) resulting in large currents from the power supply. If the power supply is not capable of providing the extra current, this can load down the supply and prevent the required potentials from being applied to the grids -- which in turn gives rise to large currents, and so on. We have found that we can minimize this effect by "step-starting" the discharge: An RC circuit, with a time-constant of about 20 μsec, in series with a thyristor, is connected in parallel with the discharge; when the high-voltage series switch is fired, this thyristor is also fired to shunt the discharge current until the capacitor is charged. Because of the rapid deionization time (sub msec) of the discharge, this puts a corresponding dip in the plasma density and thus decreases the ion and electron density in the grid region, minimizing the current surge when the voltages are applied to the grids. The recovery time of the plasma density in the discharge is comparable to the RC time of the circuit.

As a general observation, we note that the electronics and controls are becoming increasingly complex. Only in this way are we able to learn about the interactions between sources and the supplies and controls, and achieve successful operating conditions. At the present time we can not write specifications for an electronic system that will work with a new type of ion source without some modification. However, we expect to learn how to simplify and reduce the costs of future systems.

Deuterium operation

Operation with deuterium has been limited because of the large neutron fluxes produced by d-d reactions between the energetic deuterium ions of the beam and deuterium atoms buried in the copper calorimeter. In our experiments the reaction rate built up gradually as the copper was loaded with deuterium from successive beam pulses. After approximately one-hundred 120-keV beam pulses the neutron-production rate reached an asymptotic value -- approximately 10^{11} neutrons/sec during a beam pulse.

III. Near-Term Development Program

Since most of our present activities are oriented toward applications at 120 kV and below, we won't go into plans and schedules in any detail here. Suffice it to say that there is a tremendous amount of work to do to improve the performance of components and systems of the kind just described. In addition we will work toward longer pulses (30 seconds to dc) and somewhat higher energies for applications in the early 1980 s. Transfer of technology to industry is an important element of the program.

One item that deserves special mention is recovery of energy from un-neutralized ions: By recovering the energy of the un-neutralized positive ion beam, it should be possible to extend the range of the positive-ion approach to neutral beam production to 150 or possibly 200 keV (D). This approach has the advantage of capitalizing on the relatively well-understood technology of positive ion beam production and transport.

Several efforts are underway throughout the world to develop energy recovery units with clear apertures large enough to operate with beams from large area ($\sim$500 cm^2) sources; the plans and preliminary measurements made on our 120 kV III A beam line by the LLL direct recovery group will be described in another paper at this meeting.

Another approach, offering additional advantages, also is being considered. In this second approach, a conceptual design for which is shown in Fig. 4, each <u>beamlet</u> from the ion source passes through a very short neutralizer section containing a low temperature, high density neutralizing gas, then enters a direct recovery section scaled to each individual beamlet. Some of the additional advantages of this scheme are a) it is very compact--conventional beamline designs could be shortened 2 to 3 meters, b) since the direct recovery section is of the same scale as the accelerator section, the ion beam space charge can

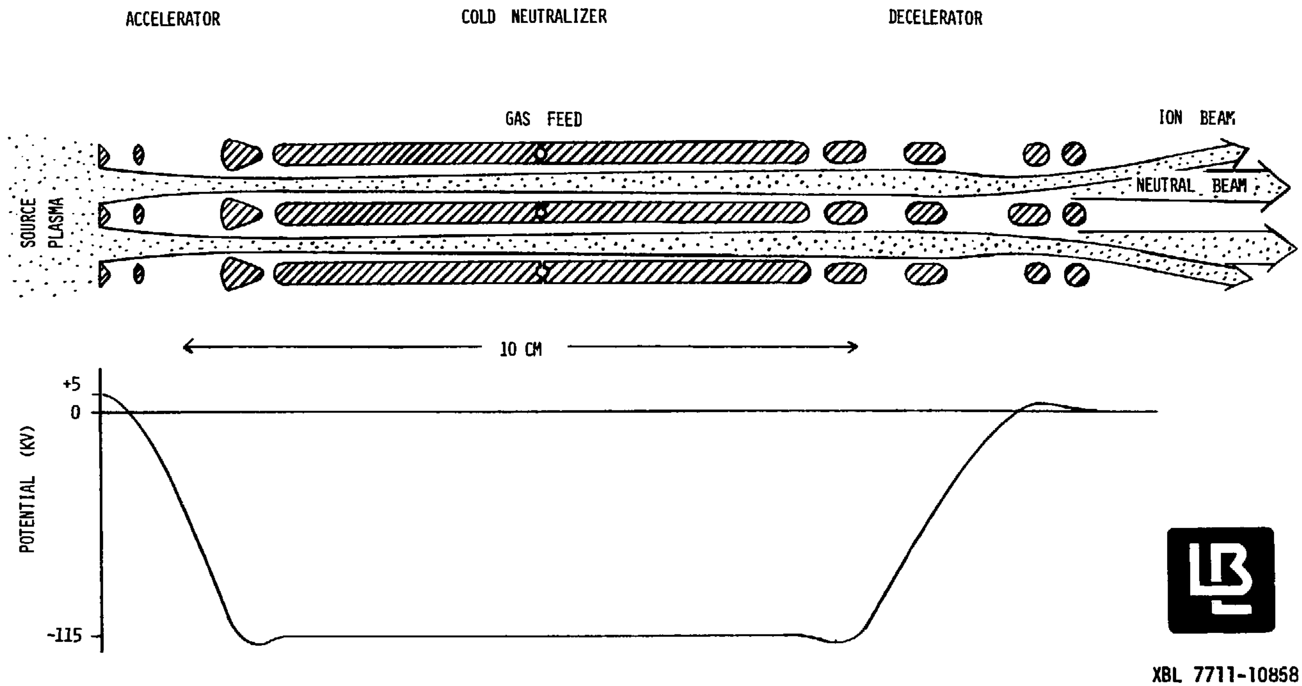

Fig. 4 Conceptual design of an efficient, close-coupled neutral beam source based on positive ions, showing the grid structures and potential distribution through the system.

be "handled" without difficult, c) full-energy ions are not stopped in the direct convertor, resulting in a smaller gas load in that region, and d) (highly speculative) some advantages in beam optics may result from the short neutralizer. Experimental tests are required to determine the feasibility of developing systems based on this approach of a close-coupled accelerator neutralizer decelerator.

IV. Potential of Positive-Ion-Based Beams

The potential for positive-ion-based neutral beams for heating and fueling fusion plasmas is very great if designers can keep the injection energies below about 200 keV for deuterium ($\sim$100 keV H^0; $\sim$300 to 400 keV He, Fig. 5), and energy recovery systems are perfected (Fig. 6). (From almost all aspects of beam-line design, helium is an easier element to work with than hydrogen, needing only the development of suitable cryopumps. However, there has been little user interest so far.)

At energies above $\sim$200 keV (D^0) the as yet undeveloped negative-ion approach is much better, probably essential for many applications. If efficiency and cost were not a primary consideration, then high-power neutral beams could be obtained from molecular ions ; however, it would seem more desireable to accelerate the negative ion program.

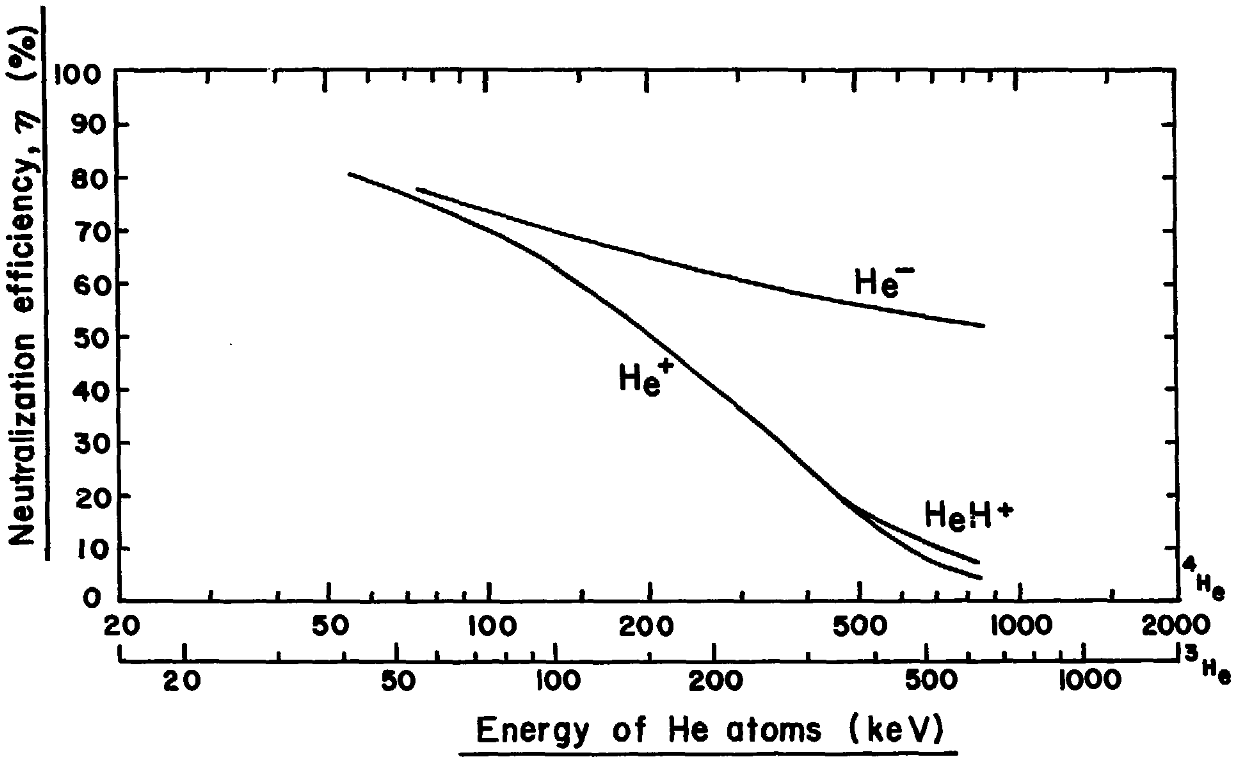

Fig. 5 Efficiency for converting power in He^+, HeH^+ ion beams to neutral helium power.

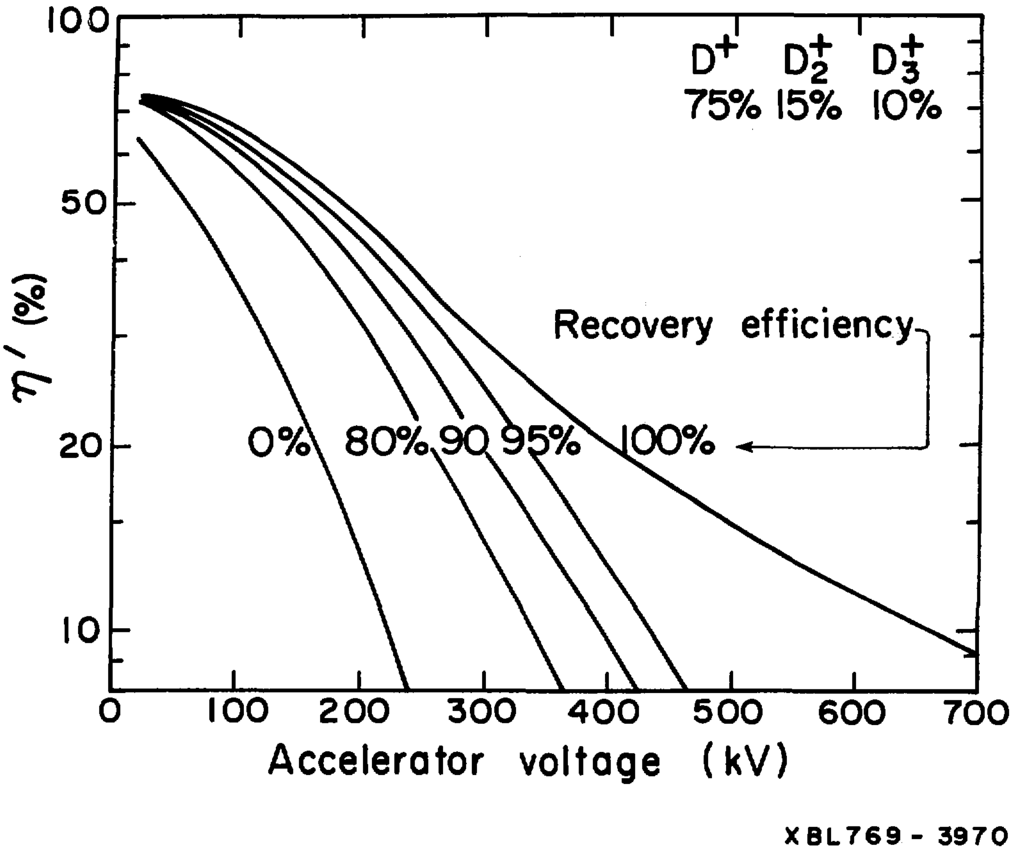

Fig. 6 Net efficiency of converting power in a deuterium ion beam to full-energy neutrals in a system with energy recovery. The assumed ion species mixture is shown at the top.

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NEUTRAL BEAM BASED ON POSITIVE IONS WITH DIRECT ENERGY CONVERSION

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NEUTRAL BEAMS BASED ON POSITIVE IONS WITH DIRECT ENERGY CONVERSION*

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ABSTRACT

Positive ions can make efficient neutral beams when direct energy conversion is incorporated at energies up to 150 keV for D°, 225 keV for T° and 300 keV for $^{3}He°$. Above these energies the efficiency is low (< 50%) and falling rapidly, requiring other means for making neutral beams such as negative ions. The virtues of $^{3}He°$ beams as a heater are discussed. The role of direct conversion is discussed and the various conversion concepts and the experimental data base are reviewed. The development problems facing direct conversion are: space charge handling, secondary and primary electron suppression, and the fractional energy ions. The next step in the development of efficient neutral beams based on positive ions is argued to be a developmental beam which integrates an advanced ion source with a neutralizer, cryopump, direct converter, heat removal system, and power conditioning system.

* Work performed under the auspices of the U.S. Department of Energy under contract No. W-7405-Eng-48.

1. Components of a Neutral Injection System based on Positive Ions with Direct Energy Conversion

a. Plasma Source. A source of plasma is required from which positive ions such as D^+ may be extracted with the required characteristics of beam density, uniformity, and stability.

b. Extraction System. The ions are extracted from the plasma by focussed electric fields applied by high-voltage grids. A substantial quantity of electrical power is invested in the production of the fast ion beam.

c. Neutralizing Cell. Normally this is a duct filled with gas of sufficient thickness such that the fast beam attains a state of equilibrium. In an equilibrium neutralizing cell, the rate of fast atom production by charge exchange collisions with gas molecules is equal to the rate of fast atom loss by re-ionization. The maximum neutralization efficiency is therefore determined by the ratio of charge exchange and ionization cross-sections. The efficiency becomes low at D^+ energies above 100 keV because of the small charge exchange cross-section at high speeds.

d. Direct Energy Converter. The converter improves the overall efficiency and also disposes of the residual charged beam by deceleration of the charged beam and collection of the ions at a potential close to the potential of the plasma source.

e. Gas Pumping System. High-speed pumps are required to remove the gas flowing from the neutralizer and produced at the beam targets. The pressure must be reduced to the 10^{-5} Torr range to avoid beam loss by re-ionization and to avoid secondary products in the Direct Converter.

These components may be seen in Fig. 1. This figure shows a long, narrow beam cross-section to facilitate the suppression of electrons. A pumping space of about 0.5-meter between the neutralizer and the converter is required for the pumping system and also to reduce the density of gas streaming from the neutralizer collisionlessly into the converter. Cryogenic panels are probably the only available technique to attain the high-pumping speed required.

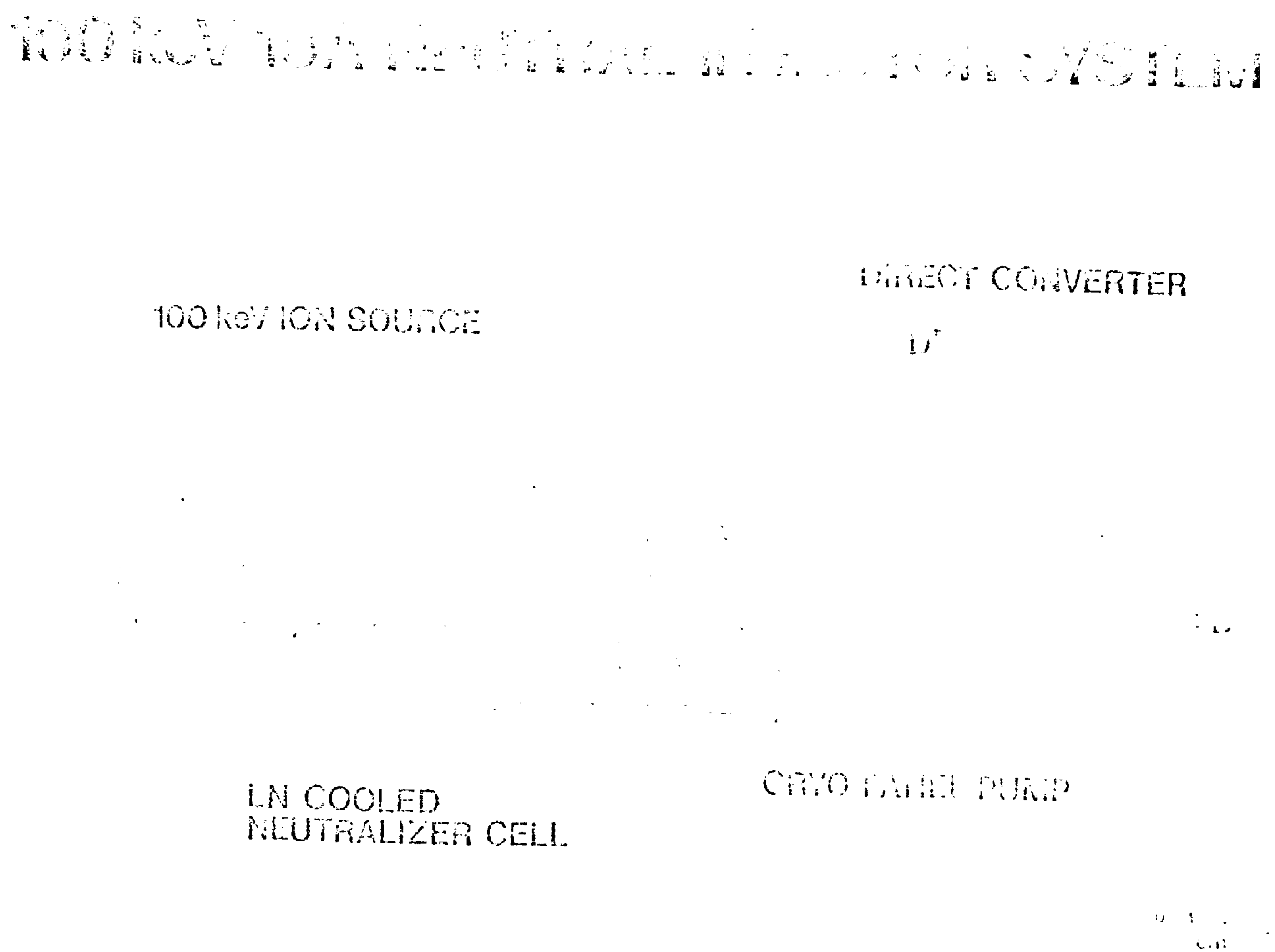

100 keV ION SOURCE
DIRECT CONVERTER
LN COOLED
NEUTRALIZER CELL

Figure 1.

2. The Role of Beam Direct Conversion in Large Experiments and Fusion Reactors

a. Disposal of the high-power charged beam may be possible by direct conversion under conditions not possible by other techniques. Designs indicate that beam power densities of 10 to 20 kW/cm^2 may be handled by a direct converter although these densities exceed the heat-transfer limitations for steady-state beam targets. This is because most of the beam energy is removed by deceleration before the ions strike the collector and because the charged beam is spread over a large area by electrostatic blow-up. This possibility is interesting, aside from other advantages, because the converter occupies less volume than the large sweep magnets and beam dumps otherwise required to dispose of the charged beam. Our layouts indicate more compact assemblies and more ion sources in a single beam line than the alternatives.

b. A substantial saving in the capital cost of injector power supplies will be possible in large experiments by using the converter to supplement the accel power supplies. This saving may amount to several million dollars for a large experiment.

c. The reduction in power requirement will significantly reduce the cost of electricity and also will affect the feasibility of large experiments limited by power availability.

3. Comparison of D^+ and D^- efficiency on a common set of assumptions

Figure 2 indicates the overall efficiencies of injection systems based on D^+ and on D^- with and without beam direct conversion.

The efficiency estimates for D^+ sources assume that a 0.75:0.15 : 0.1 mixture of D^+ : D_2^+ : D_3^+ is extracted from the arc source. 5000 eV are expended for every ion that is extracted. We assume that 5% of the extracted ions (5% of each of the three types) experience charge exchange in the acceleration space; the contribution of the resulting reduced-energy neutrals to the neutral beam power is neglected. We chose the neutralizer thickness to achieve a $D°(W)$ current equal to 95% of the infinite neutralizer case. For a structure of this thickness the currents of the $D^+(W)$, $D_2^+(W)$ and $D_3^+(W)$ ions are computed; if a direct energy converter is present, 80% of their power is recovered. The power carried by the fractional energy ions is not recovered; it is discarded as heat. The energy carried by the currents of $D°(W)$, $D_2^\circ(W)$, $D°(W/2)$, $D_2^\circ(2W/3)$, and $D°(W/3)$ neutral atoms is calculated to determine a neutral beam power. The efficiency is defined by the expression

$$\eta = \frac{\text{Power carried by full, half, 2/3 and 1/3 energy neutrals}}{\text{Net power in}}$$

where Net power in = source current x (Accel Voltage + 5000)

- power recovered in the direct converter.

The efficiencies of negative ion neutral beam injectors are calculated assuming that positive ions are created with an energy expenditure of 5000 ev/ion. These ions are first accelerated to 10 keV; 5% of the beam is assumed lost by charge-exchange during acceleration. The contribution of the charge-exchange-produced low energy neutrals to the final neutral beam power is neglected. The 10 keV ions are then slowed to 1750 eV and sent through a cesium cell in which 20% of the injected beam is converted to negative ions. The negative ions are then accelerated up to the indicated voltage, V_a. During this process 5% are lost due to charge exchange and again the contribution of the resulting neutrals to the neutral beam output power is neglected.

The energetic negative ions enter a plasma cell in which 80% of the ions are neutralized. If direct conversion is present, 80% of the energy carried by the remaining 20% of the negative ions is recovered. Therefore the efficiency is computed using the following "Net power in":

$$I_s \times (5000 + 10^4 \times .05 + .95 \times V_a \times .05 + .95^2 \times .2 \times (1-R)\ V_a)$$

$$+ \text{ Neutral beam power}$$

where R is 0.8 if direct conversion is used and zero otherwise; I_s is the ion source current, and eV_a is the acceleration energy.

The He injector efficiency calculations are based upon a model which is identical in form to that used for the positive ion deuterium injector. The ion source expends 5000 eV per ion extracted; 5% of the beam is lost because of charge exchange during acceleration; and the neutralizer thickness is adjusted to allow 95% of the infinite-thickness neutralization. If a direct energy converter is incorporated, 80% of the energy borne by the unneutralized fraction is directly recovered.

The efficiency η can be expressed as

$$\eta = \frac{\text{Neutral beam power}}{\text{Net power input to source}}$$

$$= \frac{(.95)^2 F_\infty V_a}{5000 + 0.05\ V_a + (1-R)\ .95\ (1-.95F_\infty)\ V_a + (.95)^2\ F_\infty\ V_a}$$

where $F_\infty = \dfrac{\sigma_{10}(W)}{\sigma_{10}(W) + \sigma_{01}(W)}$, the infinite thickness neutralization

and R = 0.8 if direct energy conversion is assumed and zero otherwise

(Note that the ion energy, W, is equal to eV_a.)

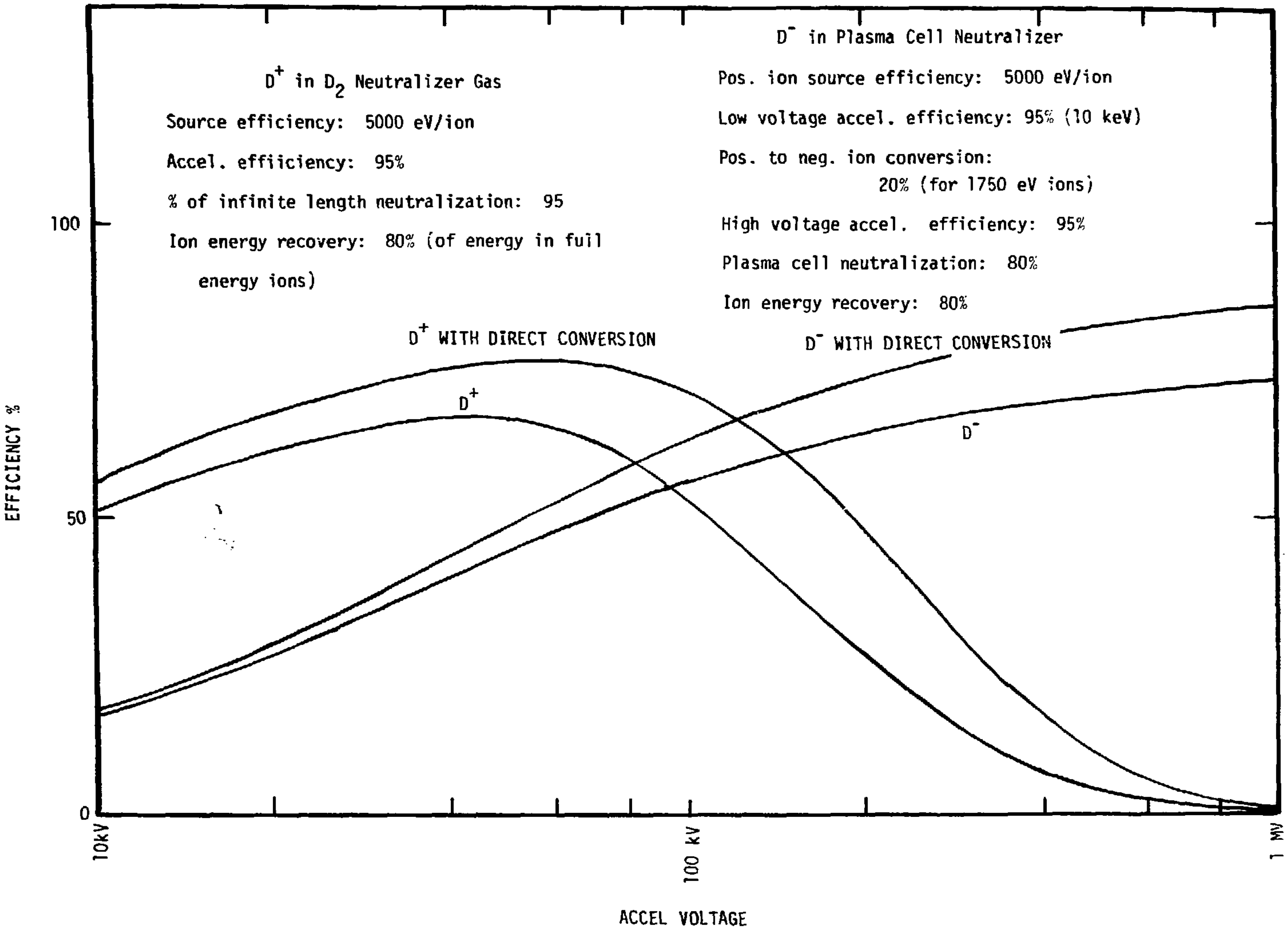

D^+ in D_2 Neutralizer Gas
Source efficiency: 5000 eV/ion
Accel. effiiciency: 95%
% of infinite length neutralization: 95
Ion energy recovery: 80% (of energy in full energy ions)
D^- in Plasma Cell Neutralizer
Pos. ion source efficiency: 5000 eV/ion
Low voltage accel. efficiency: 95% (10 keV)
Pos. to neg. ion conversion: 20% (for 1750 eV ions)
High voltage accel. efficiency: 95%
Plasma cell neutralization: 80%
Ion energy recovery: 80%
D^+ WITH DIRECT CONVERSION
D^+
D^- WITH DIRECT CONVERSION
D^-
EFFICIENCY %
100
50
0
10kV
100 kV
1 MV
ACCEL VOLTAGE

Figure 2.

4. Neutral Injection Systems Operating on Helium

Several advantages may be obtained by operating neutral injection systems on helium rather than on hydrogen isotopes. Some of the arguments for and against the use of energetic helium atoms for heating toroidal systems have been presented by Ernie Thompson in his Letter in Nuclear Fusion 15, 347 (1975). In the list below, Ernie's arguments are identified by asterisks. The new development of cryogenic pumping for helium, by T. Batzer, has removed the most important liability that existed at the time of Ernie's Letter. In addition to Ernie's arguments, we will include some additional thinking on this subject.

(1) Arguments in favor of injection of fast helium atoms:

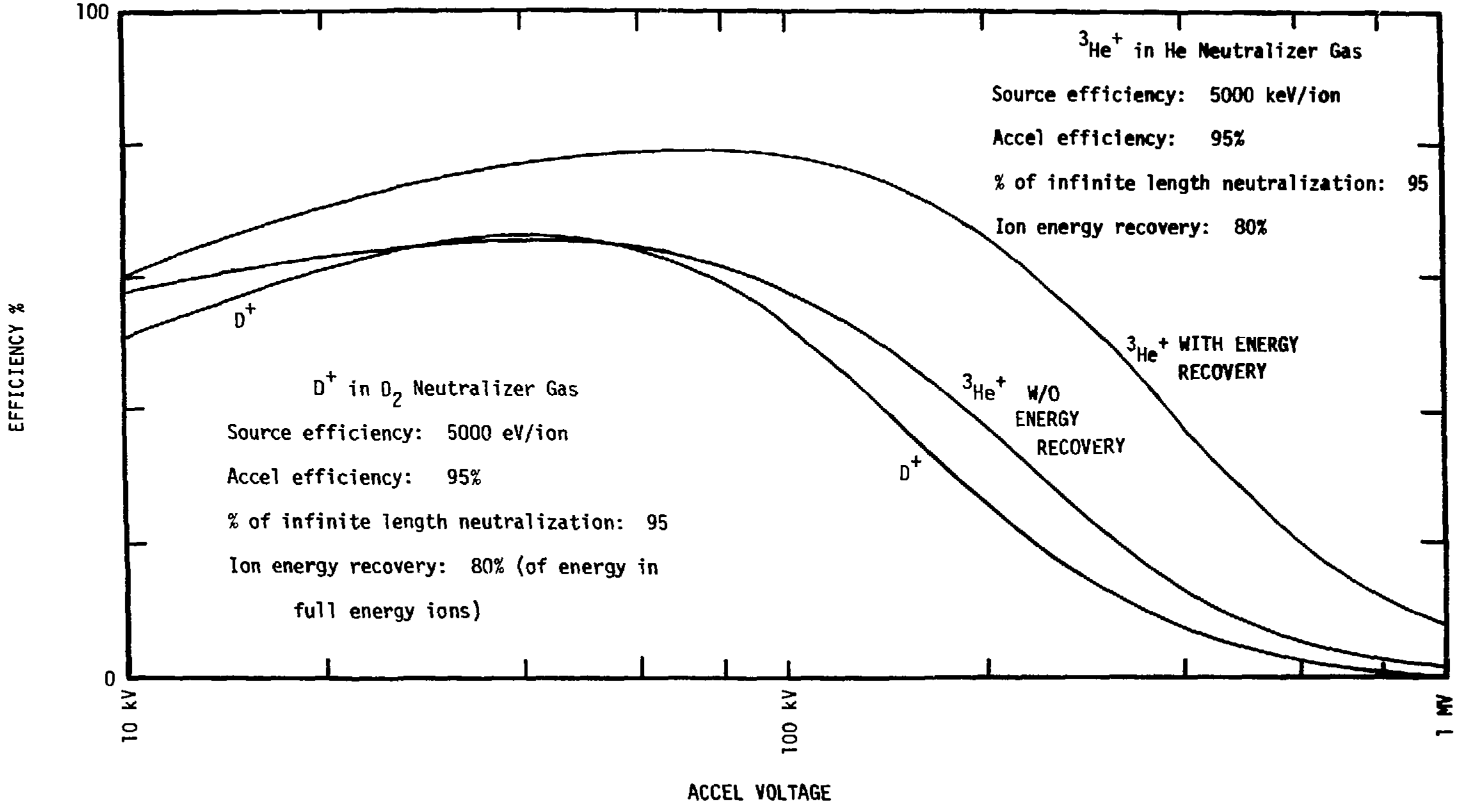
*a. Deeper penetration of He° in comparison with H° and D° at equal energies up to 60-100 keV.

*b. Almost constant trapping cross-section of He° in the energy range from 10 to 500 keV. Therefore the injection energy can be selected for other reasons.

*c. More efficient production of fast neutrals because of the larger electron capture cross-section for He^{+}. This improvement is shown by Fig. 3, showing the overall efficiency for the understated assumption for D and for ^{3}He injectors.

d. A helium ion source normally produces only one ion species, He^{+}, in comparison with the three species H^{+}, H_2^{+}, H_3^{+} produced by sources of hydrogenic ions. This avoids the production of half-energy or third-energy neutrals derived from negative ions, and avoids several disadvantages:

(1) Optimum heating of toroidal plasma requires an injection energy above a certain value. Injection of half-energy or third-energy particles tends to heat electrons (undesirably) rather than ions.



100
0
EFFICIENCY %
10 kV
100 kV
1 MV
ACCEL VOLTAGE
$^3He^+$ in He Neutralizer Gas
Source efficiency: 5000 keV/ion
Accel efficiency: 95%
% of infinite length neutralization: 95
Ion energy recovery: 80%
$^3He^+$ WITH ENERGY RECOVERY
$^3He^+$ W/O ENERGY RECOVERY
D^+
D^+
D^+ in D_2 Neutralizer Gas
Source efficiency: 5000 eV/ion
Accel efficiency: 95%
% of infinite length neutralization: 95
Ion energy recovery: 80% (of energy in full energy ions)

Figure 3.

(2) The half-energy or third-energy injection enhances the charge-exchange at the outer portion of the plasma, causing wall bombardment, gas loading, and other undesirable effects.

(3) Penetration of a mixed-species beam is non-uniform.

(4) Beam Direct Conversion is more complicated and less efficient if a mixture of energies is present in the charged beam emerging from the neutralizer.

e. Injection of He° rather than D° avoids the need for neutron shielding in plasma experiments and in neutral-beam test stands.

f. An He injection system involves a smaller gas load than an equivalent H or D system because of the higher neutralization efficiency.

g. The design of certain types of D-^{3}He reactors would become possible if ^{3}He neutral injectors are available.

h. Field reversal in steady-state mirror devices probably will require the injection of at least two ion components with different charge numbers and different speeds. This is required as shown by Ohkawa so the electrons dragged along with the circulating ions do not completely neutralize the circulating current. The logical species to inject for this reason are D^+ and He^{++}.

(2) The arguments against the injection of fast He atoms are:

*a. The impurity buildup of trapped He^{++} ions is tolerable only up to a certain level.

*b. A new type of gas pumping is required for helium gas. (This requirement now appears to be satisfied.)

5. Evolution of direct conversion concepts.

The in-line concept for beam direct conversion evolved from the biased Faraday cup, but with two important changes. First, the back is left open to allow the neutrals to pass on through. And second, the suppression of electrons is much more difficult because of the cold plasma generated in the neutralizer. As the in-line concept evolved, the changes that were made were mainly in the electron suppressor. The first in-line direct converter used a magnetic field to suppress electrons as discussed below. The second in-line converter used immersed grids (Fig. 4). The ions diverge due to their own space charge and are caught on the "Pierce-shaped" decelerator electrode. With round grid wires immersed in the beam, the product of total beam-power density times pulse length must be kept below about 300 j/cm^2 for 1-mm diameter wires (e.g. $1kW/cm^2$ for 0.3 sec). By using grid wires with aligned rectangular cross sections, the limit can be increased to about $3kj/cm^2$. Ribbon grids were tested on a direct converter at LLL in 1970. A 1-kW beam of 20 keV ions gave $200W/cm^2$ in steady state. The efficiency was 70%. This concept was dropped because of the low limit on power density.

A scheme for reducing the power loading on the grid is being developed at Fontenay-aux-Roses by Fumelli and co-workers. Figure 5 shows the grid arrangement. Only the ion component of the beam passes through the grid, and the power density is further reduced by the large grid surface area. This is being tested at 35 kV on a large, 14-cm-diameter beam with a goal of 60 kV, 35A total (~7A ions) for several seconds.

At LLL we are developing a system which uses no intercepting grids. Electrons are suppressed by a negative electrode in the form of a solid band around the beam (Fig. 6). This system has been tested in steady state at up

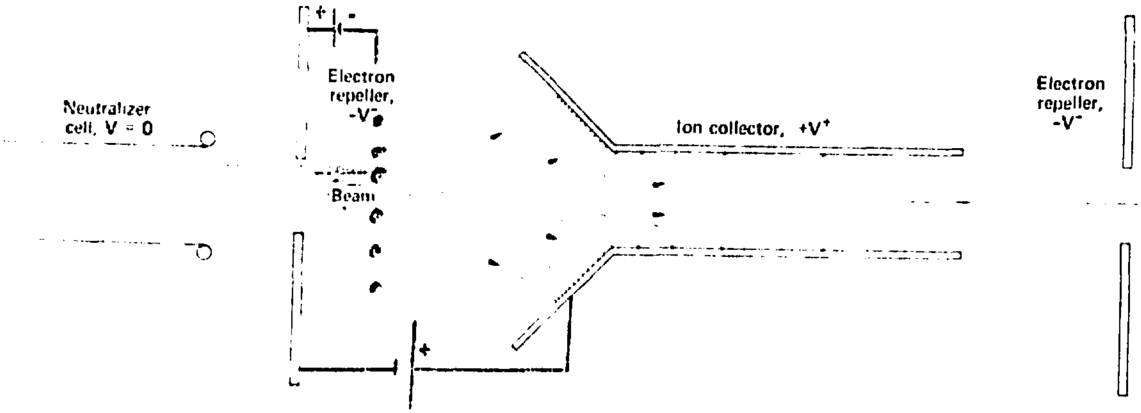


Figure 4. Beam Direct Converter with Immersed Grid

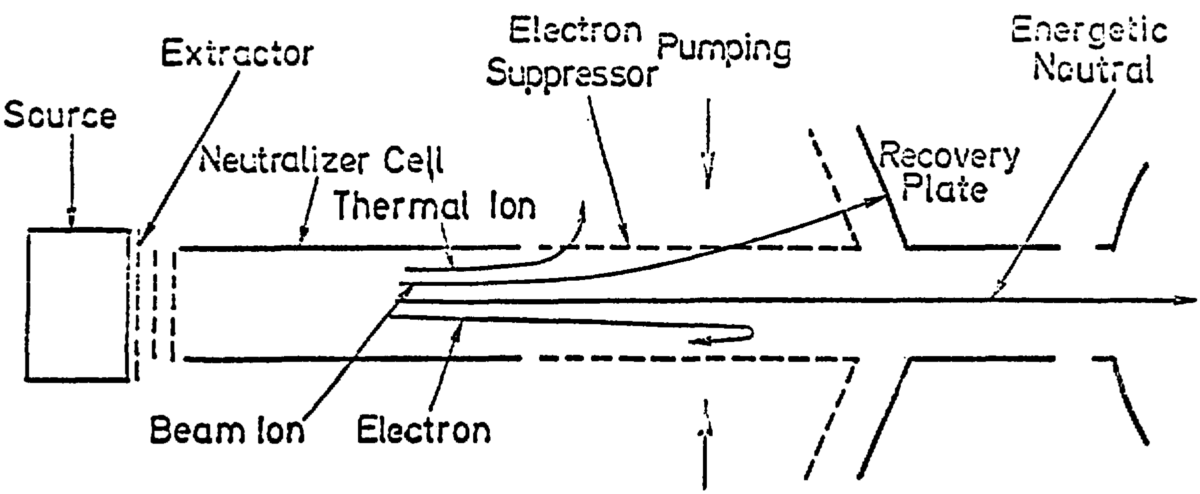


Figure 5. Beam Direct Converter with Partially Immersed Grid

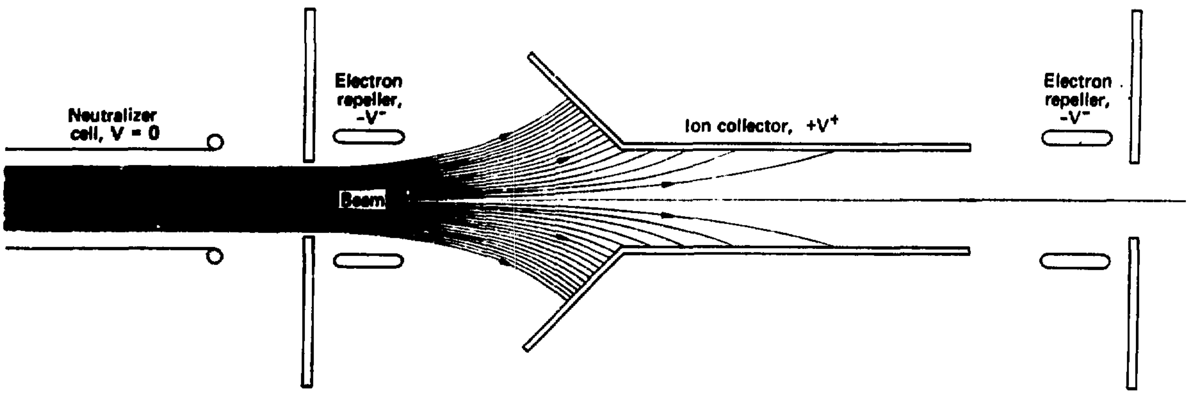


Figure 6. Beam Direct Converter with Non-Immersed Grid

to 2 kW and 15 keV, with a net recovery efficiency of 70 $\pm$ 2%. Tests are in progress on LBL's 120 keV, 20A test stand. The direct converter should be capable of steady state operation at this power level, but cannot be pushed much further. The limit is due to the high negative potential required at the electrode. The high-negative potential is required partly to overcome the effect of the neighboring, positive, collector electrode, and partly to penetrate the beam. At LBL, -20 kV is required. The part required to penetrate the beam scales as nd^2 (ion density times beam thickness squared), and becomes impractical for dense beams much thicker than the 9cm-thick LBL beam. On TFTR the 120 kV beams will be slightly thicker and four times as dense. On MFTF the 80 keV beams will probably be only slightly thicker and also about four times as dense as in the present LBL tests. This scheme for electron suppression can only be used on these larger beams if the beams are divided into thinner ones.

One way to avoid this nd^2 limit might be to use a magnetic field to suppress the electrons. Figure 7 shows a configuration that was tested in 1970 at LLL using a 2 keV beam in steady state. Because of the aperture, electrons had to cross magnetic field lines to reach the ion collector. These tests were not successful and the experiment was dropped.

Figure 8 shows a similar scheme now being tested at LLL on the 15 kV test stand. These new tests resulted from a suggestion by O. B. Morgan of ORNL*. Electrons are observed to reach the ion collector in spite of the magnetic field, and present efforts are directed at understanding the electron currents.

* G. Schilling, et al. "ORNL/TFTR Neutral Beam Injection Systems", Rept. ORNL/CF-76/400 (1976)

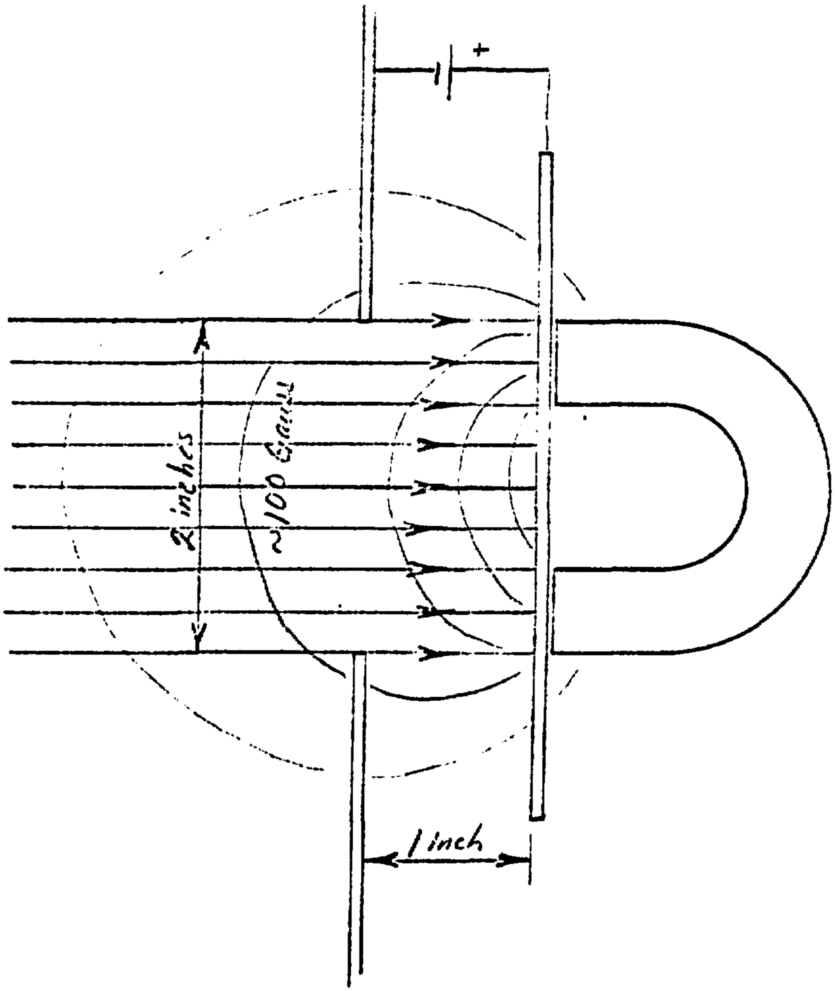


Figure 7. Magnetic electron suppressor with a beam direct converter

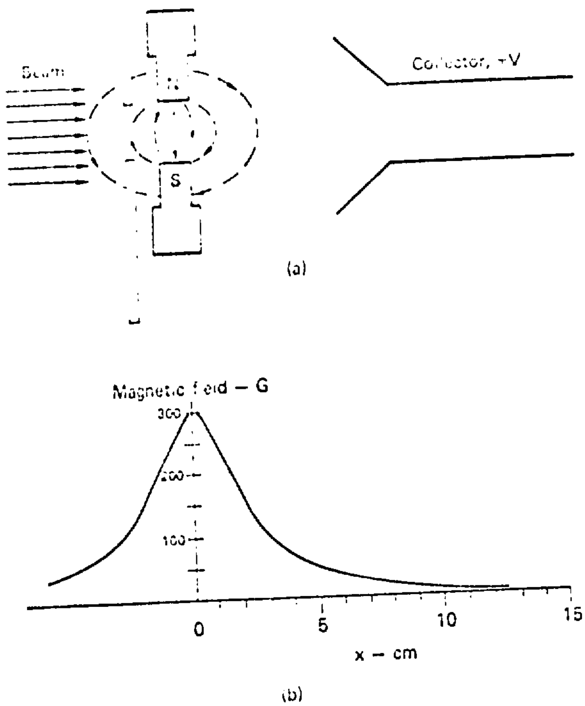


Figure 8. (a) The beam direct converter with a perpendicular field to suppress the electrons. (b) Field strength along the axis.

6. Grounded Neutralizer, or Grounded Plasma Source

Figures 9 and 10 show two possible circuits for recirculating the recovered power back to the injector. In both cases a low-voltage supply is needed to maintain a potential difference between the ion source and the collector. In both circuits several options are available in the choice of where to define the ground potential.

It is conventional in most systems to ground the neutralizer at point A and to operate the ion source at positive high voltage. However, another option (preferred at Fontenay-aux-Roses) is to ground the collector (Point B) and to operate the neutralizer at negative high voltage. In principle, the direct converter and the ion source operate equally well in either ground condition although there are some practical differences:

a. If the neutralizer is operated at negative-high voltage (ground condition B) it is necessary to prevent the electrons produced within the neutralizer from escaping. This requirement for electron suppression is fulfilled by the beam-direct converter. Therefore the converter is almost mandatory for condition B but is optional for condition A.

b. Ion source electronics will be simplified under ground condition B, since the arc and filament circuits are operated near ground potential. Voltage-holding conditions will be different since the energy stored by the capacitance of the neutralizer is not the same as the energy stored in the plasma source and isolation transformers.

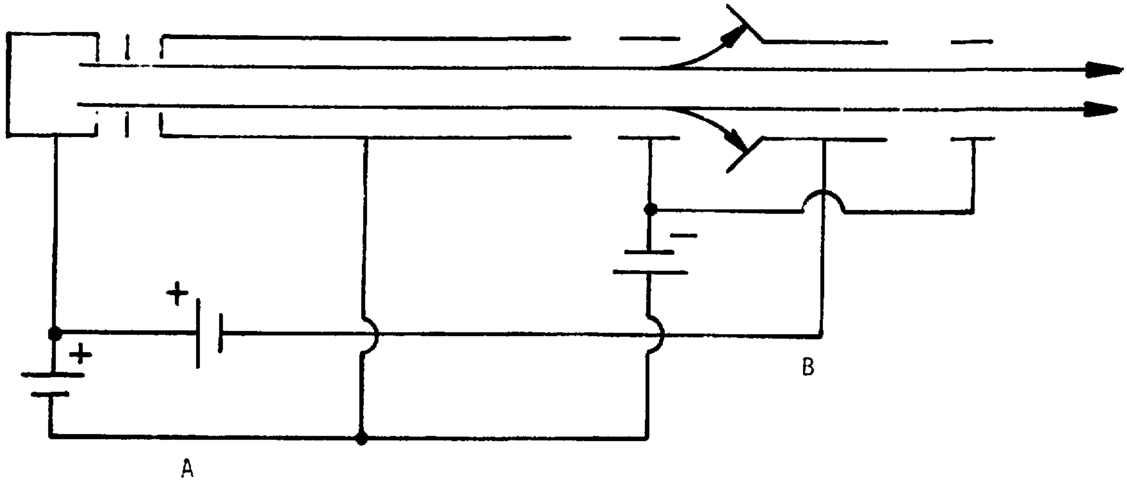


Figure 9. A circuit for re-circulating the power recovered by a Beam Converter to supplement the accel power supply.

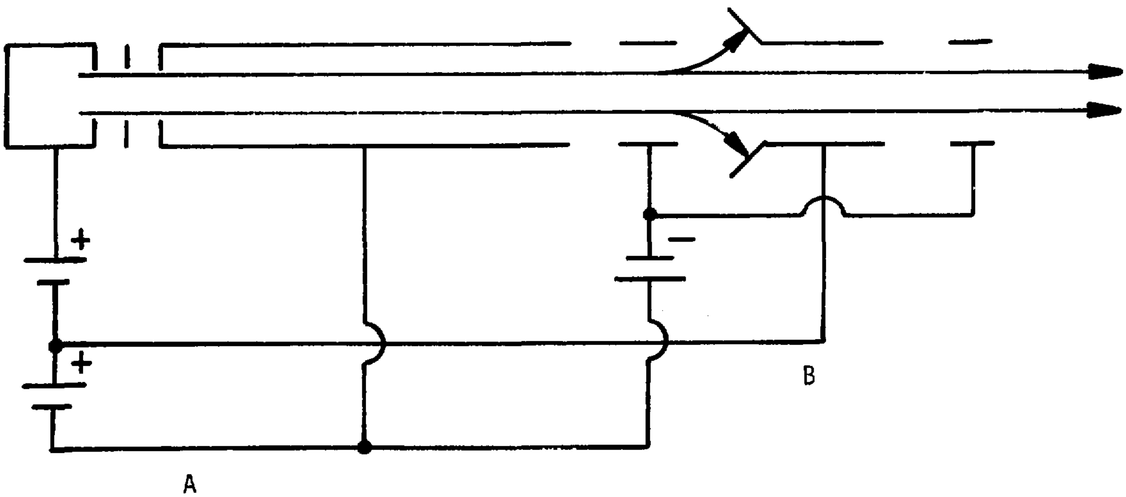


Figure 10. Another version of a circuit for recirculation of the recovered power. The only difference between the two circuits is the connection of the power supply required to maintain the difference between the source potential and the collector potential.

7. Effects of Fractional Energy Ions in Direct Converters

All existing neutral beams of hydrogen or deuterium consist of a mixture of full-energy and fractional-energy ions and neutrals. The fractional-energy ions cause serious problems in the direct converter because they are unable to reach the positively biased ion collector. Some of them are accelerated into the suppressor electrode where they not only generate heat, but also release about 4 secondary electrons per ion. These electrons are accelerated onto the ion collector where they generate heat and cancel an equal amount of collected ion current. This problem is equally serious whether the source or the neutralizer is grounded and remains serious for all types of electrostatic electron suppressors. The solution to this problem is to develop ion sources which produce fewer molecular ions and therefore fewer fractional-energy particles. MacKenzie picket-fence magnets and hot walls are being tested for this purpose at ORNL, and elsewhere. We plan to run with a He beam to eliminate the fractional-energy ions.

8. Development Problems Affecting High Efficiency

In the above discussion several development problems were mentioned that affect the direct converter and its efficiency. One is the development of sources to maximize the full-energy component in the beam. Another is the design of direct converters with intermediate voltage collectors to collect the fractional-energy ions. Such a design will probably have a lower nd^2 limit than present designs, and this compromise should be evaluated. The lower nd^2 would certainly make the problem of electron suppression easier. A third development problem is that of magnetic electron suppression. The combination of magnetic suppression with intermediate electrodes might be best and will be analyzed and tested.

9. Development of an integrated system for production of an efficient long-pulse neutral beam based on positive ions.

Figure 11 shows an integrated module for neutral beam production with high efficiency using positive ions with direct conversion. Such an integrated module will be the logical next step after each of the components are tested separately. The principal elements are the following:

a. An ion source producing a single specie at full energy -- either D^+ or He^+.

b. A neutralizer and pumping system using regenerative cryopumps or cryotraps.

c. A direct energy converter with heat removal for long-pulse collection of high-intensity beams.

d. A power conditioning system with fault protection and with a feedback circuit to recycle the recovered power to the accel circuit.

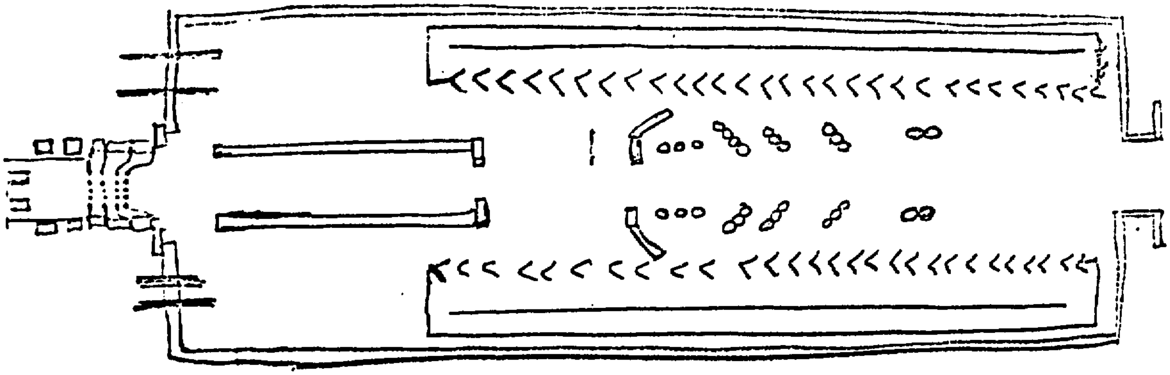


Figure 11.

120 keV D°/He° INJECTOR

FEATURES

- LONG PULSE
- HIGH D^+ FRACTION SOURCE
- DIRECT CONVERSION COUPLED TO
- SOURCE POWER SUPPLY
- REGENERATIVE CRYO PUMP
- NO BENDING MAGNET

10. Direct Energy Recovery using Negative Ions.

At beam energies above approximately 200 keV it is clear that negative ions (D^-) must be used to attain a reasonably good neutralization efficiency. Beam direct conversion will still be required to improve the overall efficiency of these systems. The most important change required for negative ion systems is caused by the fact that the residual charged beam emerging from the neutralizing cell may consist of either positive ions (D^+), negative ions (D^-), or a mixture of both depending on the thickness of the neutralizer cell. Also consideration must be given to the fact that the polarity of the high voltage power fed back to the negative-ion accelerator must be negative. A simplification results from the fact that only one negative ion species D^- exists in large quantities.

Table I summarizes the performance of negative ion injection systems classified according to the species of the residual charged beam. If the fast neutrals are produced by electron detachment in a gas cell or a plasma cell at a density optimized for the neutral output, the residual charged beam will consist of a mixture of D^- and D^+, since the number of electrons stripped by collisions may be 0, 1, or 2. This option involves an unnecessary complication for direct conversion because it will be necessary to separate the D^- and D^+ somehow and to collect them at two different potentials.

The second possibility illustrates how almost all of the D^- can be eliminated by increasing the cell thickness slightly beyond the optimum value for D° production. A very few D° will be lost by stripping because the stripping cross-section for the second electron is much smaller than the stripping cross-section for the first electron. Consequently direct recovery of the D^+ ions will be possible by the techniques already described. However, the electrical power recovered must be transformed from positive high voltage to negative high voltage to feed back to the accelerator.

The third possibility, shown in Table I, is the stripping of negative ions by photons rather than by gas or plasma collisions. No D^+ will be produced by photon-stripping. The output beam will consist only of D° and D^-, depending upon the thickness of the optical cavity. Direct recovery of the D^- energy will be at negative high voltage, which can be fed back to the accelerator without transformation. However, the problems of electron suppression will be different because of the negative high voltage of the D^- collector. Our studies indicate that photon-stripping is economical only for large systems in which hundreds of amperes of D^- ions are passed through a single optical cavity.

11. Conclusions

Direct energy conversion will play a useful role in improving the efficiency of neutral beams, in disposing of the residual charged beam, and in overall economy. These considerations are valid for neutral beams derived from positive ions such as D^+ or He^+ and also from negative ions such as D^-. Helium neutral beams are especially interesting because of direct conversion and because of other advantages. Several types of direct conversion systems have been developed or designed. Further development is required to optimize the performance and to integrate the direct converters into injector modules.

DIRECT CONVERSION OF RESIDUAL BEAMS DERIVED FROM NEGATIVE IONS
THREE VERSIONS, DEFINED BY THREE TYPES OF RESIDUAL IONS

Residual Ion Species (No half-energy ions derived from negative ions.)	D^+ and D^-	D^+ only	D^- only
References, Fink, Hamilton, & Barr	UCRL-52173	UCRL-79643	UCRL-52173
Neutralization Cell	optimized gas or plasma cell	over-dense gas or plasma cell	under-dense optical cavity
Advantages	Most efficient without beam direct conversion. (if beam current is less than several hundred amperes)	Most efficient with beam direct conversion (if beam current is less than several hundred amperes)	Most efficient if beam current is sufficiently high
Polarity of recovered power (if neutralizer is grounded)	Both positive and negative high voltage	Positive high voltage only.	Negative high voltage only.
Polarity of high voltage required for accelerator	Negative d.c.	Negative d.c.	Negative d.c.
Fraction $(F^+ + F^-)$ of total accel power into direct converter.			
gas neutralizer	34%	40%	5 to 40%
plasma neutralizer	14%	18%	
Fraction (F^+) of total accel power requiring transformation by inverter-rectifiers (with plasma cell, conditions of UCRL-79643)	8.4%	18%	0
Sources of secondary electrons requiring suppression			
D^+ collectors at positive potential no suppression required.			
D^- collectors at negative potential suppression required?	yes	no	yes
Electrons from neutralizer suppression required?	maybe	yes	no

Table 1.

BNL 23583

NEGATIVE ION SYSTEMS BASED ON DIRECT EXTRACTION SOURCES*

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November 1977

For deuteron energies of 150 keV and above, neutral beam systems based on negative ions are being considered as an alternative to the existing positive ion based systems. The main advantage of negative ion systems is a neutralization efficiency that may approach 100%, depending on the stripping target, and that does not depend very much on the energy of the negative ion. In contrast, the neutralization efficiency of positive ions decreases with energy. According to the method negative ions are produced, two approaches are presently being pursued, one using a direct extraction source of negative ions, and the other, conversion of low energy positive ions in an alkali vapor cell or jet. Figure 1 shows schematically components of the two systems; some of them (e.g., ion source, alkali vapor cell) require separate development efforts, while the others will have common problems (transport of beams, acceleration, neutralization). Each component of the system will be considered, with more emphasis on those specific for negative ion systems based on direct extraction sources. Fundamental processes will be discussed, if necessary, in relation to the development of the system components, state of the art reviewed, areas of further work suggested and, where possible, ultimate potential estimated.

a) Ion Source

Several types of discharges have been used as sources of negative hydrogen ions[1-3]: magnetrons, Penning discharges, duoplasmatrons, and most

*Work performed under the auspices of the U.S. Dept. of Energy, Washington, D.C.

recently, the modified duopigatron. Although all of them may operate with hydrogen gas as the only medium, H^- (D^-) yields, normalized to the discharge current or power, are very low and not usable in neutral beam lines. Addition of a small amount of cesium vapors into the discharge increases the normalized yield by one to two orders of magnitude. It is generally accepted that in the former mode of operation volume processes play the dominant role in the production of negative ions, while in the latter case, surface conversion of charged and neutral particles on cesium-covered surfaces is the factor responsible for a much higher yield. Calculations[4,5] of the surface conversion efficiency indicate that, depending on the energy of incident particles and surface conditions, values up to 50% may be expected for hydrogen. This seems to be confirmed by the measurements of the ratio of the emitted H^- current density to the discharge current density,[2] implying effective values of the secondary H^- emission coefficient (including neutral incident particles) between a few and up to 80%, depending basically on the degree of cesium coverage. Further theoretical and experimental work is necessary, especially related to the dependence of the secondary H^- ion emission coefficient on incident particle energy and species (isotope effects) and on surface conditions.

Magnetron source (Fig. 2) was the first high current source to be developed using the surface production of H^- ions and it still remains the most intense and efficient among them. Table I shows parameters of existing magnetron sources, at BNL,[1] Novosibirsk,[2] and Fermilab[6] laboratories. The comparison of source parameters points out to a difference in objectives: the BNL source compared to other sources operates with longest pulses, H^- current densities of 0.5 to 1 A/cm^2 and cathode power densities around 1.5 kW/cm^2. From the point of view of a long pulse or steady state operation of the source, the removal of the heat from the cathode may be the limiting factor for the achievable H^- current density. Recent studies[7] done at BNL have shown that it should be possible to keep the cathode surface temperature around 500°C (normal operating regime) under a load of 1 to 2 kW/cm^2 by using pressurized water cooling. Parameters of the BNL source, as given in Table I, may therefore serve as a basis for scaling up a magnetron source to higher currents and longer pulses. A possible isotope effect has to be studied as well.

Penning discharges (Fig. 3) have also been investigated at several laboratories, BNL,[1] LASL,[8] Novosibirsk,[2] and Efremov[9] (Leningrad, USSR); parameters are given in Table II. Achieved H^- currents are generally lower than from magnetron sources, due to smaller dimensions of presently operating sources. At BNL a pronounced isotope effect was found: D^- yields and power efficiencies were considerably lower. However, this has apparently not been observed by other investigators. All these sources have been designed for relatively short pulse lengths (high cathode power densities), so that operating parameters are far from those for a steady state regime. It seems that a Penning source with a separate emitting electode opposite the extraction slits[2] may improve the performance of the source by a factor of 2. If this is so, these sources would have characteristics comparable to those of magnetron sources and might be scaled up to much larger sizes.

Duoplasmatrons have the disadvantage of an extraction along the magnetic field lines, resulting in a high electron component of the beam and, consequently, a low power efficiency. Modified duopigatrons will have the same problem; plans are to use perpendicular magnetic fields in the extraction aperture to remove the electrons, but this program is still at a very early stage of investigation.

A common drawback of all the described sources is the lack of efficient cooling during the discharge pulse (some of the sources have a cooling system to remove the heat between the pulses). The larger BNL Mark IV model,[1] that may operate in both magnetron and Penning modes, has a limited cooling capability. In addition to this feature, which should assure operation at pulse lengths up to 50 ms, the method of cesium injection has also been changed. Instead of placing the cesium compound into a cavity inside the cathode a separately heated container will be used, resulting in a better control of the cesium diffusion rate. As shown in Table III, H^- currents up to 1 A are expected in the Penning mode and up to 2 A in the magnetron mode. The source is presently being tested and normal operation of the discharge in the hydrogen mode achieved with pulses 50 ms long. To our knowledge, there are no other sources of either magnetron or Penning type designed for long pulse or steady state operation. A 10 cm diameter ring magnetron source[10] is being studied at Efremov Institute, but only preliminary results are available at discharge currents up to 300 A in pulses of 0.2 ms. Similarly, their 30 cm long Penning source has no provision for the cathode cooling during the discharge pulse.

There are several areas where further studies may lead to an improvement of source performance. The power efficiency of a source may be defined as the ratio of the extracted H^- or D^- current to the input discharge power. Present sources operate at values between 10 and 50 mA/kW (Tables I and II), although theoretical considerations[2] predict an order of magnitude higher values (up to 0.5 A/kW). Power efficiency of a source is a function of the average energy required to produce an incident particle (positive ion, fast neutral atom) to bombard the H^- emitting surface, surface conversion efficiency, transport efficiency from the surface to the extraction slits and efficiency in beam formation. When estimated separately, these processes predict a high overall efficiency, with the conclusion that present source designs have still a lot of room for improvement (e.g., optimized discharge parameters, development of low work function surfaces).

A very important characteristic of H^- sources is their gas efficiency, defined as the ratio of the number of extracted negative ions to the total number of particles (ions, neutrals) leaving the source. Losses of H^- ions during beam formation and transport and vacuum requirements in the whole beam line will depend on this parameter. Present sources, as indicated in Tables I and II, have gas efficiencies ranging from about 1 to 3% (BNL sources) to about 5% or more (USSR); the latter value would correspond to 3.7 Torr ℓ/As. All these values are quite uncertain because it is difficult to measure the gas flow during the pulse--there are indications from Novosibirsk measurements, that the gas flow may even decrease during the pulse and this may explain a part of the discrepancy above, the rest being due to a lower density of H^- current in the extraction aperture of BNL sources (it has been observed that the gas efficiency increases with H^- current density). It is expected that by separating the discharge region from the H^- production region,[2] one could improve the gas efficiency to 15% for the Penning source with an independent H^- emitter. In a magnetron source a similar improvement may be possible by using a gap wider in the back of the source to facilitate the breakdown and maintaining of the discharge at a lower pressure (Fig. 4).

While the extracted and accelerated beam from an H^- source has a relatively small heavy negative ion component (< 10%), and even this may be attenuated by the source magnetic field, electrons diffusing out of the plasma have to be removed from the beam before they can reach an electrode on a potential higher than the cathode potential. Perpendicular magnetic fields are used for that purpose in all direct H^- sources; in some of them the source field extends far enough (Penning, magnetron), in others additional fields are necessary, as mentioned before for the duoplasmatron and modified duopigatron.

Assuming now that a power efficiency of 50 mA/kW is achievable in a large source, with a cathode power load of 1.3 kW/cm^2 to be removed by pressurized water cooling, parameters of a steady state source are estimated and given in Table III for an extracted H^- current density of 0.25 A/cm^2 in the Penning and 0.5 A/cm^2 in the magnetron mode of operation. While there is a certain freedom in choosing the shape of the magnetron cathode, in a Penning source the scaling up has to be essentially lengthwise and a 5 A source will probably have cathodes 20 cm long. Information gained from experiments with Mark IV model may change the assumptions above as well as parameters of a scaled-up source.

b) Extraction and Transport to the Accelerating Column

The purpose of this component of the system is to extract negative ions from the source, remove accompanying flux of electrons, serve for initial beam formation (often under space charge limiting conditions) and transport the ions into the accelerating system with losses as small as possible. It is also desirable that some of the gas diffusing out of the source be pumped from this region so as to decrease the gas load further downstream. Two systems are being considered presently: close coupling between the source and the accelerating system and separating the accelerating system from the extraction by using a bending magnetic field.

The advantages of a close coupled system are a compact structure producing the high energy beam, fast acceleration of ions to an intermediate or final energy thus avoiding beam losses by collisional detachment and, possibly, a better beam quality. The disadvantages are also related to the closeness of the accelerator column: difficulties in achieving high voltage gradients in the presence of a relatively high density background gas streaming out of the source, possible contamination of the column elements (electrodes, insulators) by cesium coming from the source, and the necessity to operate at relatively high beam current densities. Although the experience with close coupled positive ion systems would speak in favor of this method, in the case of negative ions there is a number of factors to be considered as well, as e.g. presence of the magnetic field, removal and dumping of electrons, loss of H^- ions in collisions with neutral molecules, production of secondary ions and electrons in the extraction gap and on the extractor, etc. The system presently being studied at BNL is of this type.[11] Calculations of particle trajectories, taking into account only H^- ions and their space charge (i.e., neglecting the presence of the magnetic field, diffusion and drift of electrons, secondary particles) have shown that it should be possible to extract and immediately accelerate H^- beams of 0.5 to 1 A/cm^2 density. Preliminary results[11] show that H^- currents of about 1 A, at an initial current density of about 0.8 A/cm^2, can be extracted and accelerated; about 15 cm downstream the density was about 25 mA/cm^2. Further studies should yield more information on the beam optics, beam losses, electrode loads and possibly lead to an improved electrode design.

The second approach has been applied in several laboratories, Novosibirsk,[12] LASL,[8] and Fermilab;[6] their results are shown in Table IV. All the systems have in common an extracted H^- current density of several A/cm^2 at around 20 kV extraction voltage; the output (after the bending magnet) H^- current density is 20 to 100 mA/cm^2, corresponding to a transport efficiency of 50 to 90%. In spite of the focusing effect of the magnetic field in the direction perpendicular to the extraction slits ($n \approx 1$), space charge forces would not allow the observed transport efficiencies without a high degree of neutralization by positive ions. It seems to be very important that the source operates in a low noise mode;[12] explanation could be in a poor neutralization of a fluctuating beam. Advantages of this approach would be an operation of the source with much higher extracted current densities (better gas efficiency; this may change the requirements for a scaled-up source), an efficient removal of electrons and heavy ion impurities, differential pumping of the space for beam transport (lower downstream gas load), absence of cesium contamination of the beam line, and lower beam densities in the accelerator. One might even envisage the extraction from the rear part of the magnetron source,(Fig. 5) doubling in this way the power efficiency of the source, but for one side of the source the field would have to be reversed fast, which adds to the complexity of the system. Disadvantages are related to the new element in the system, the bending magnet, which for an order or so higher beam currents becomes bulky; also, in order to utilize the space more efficiently larger source units may be required, operating as well with higher power densities, although, in principle, simple stacking of sources and magnet pole tips in the direction perpendicular to the slits is possible. Presently there is no low energy, high current transport system of this type in operation and further studies are necessary.

c) Acceleration to the Final Energy, Transport to the Neutralizer

Acceleration of negative ions to the final energy is a system component where an overlapping exists with other methods for production of high energy neutral beams (systems based on positive ions or on negative ions produced by double electron capture). A distinction between the two negative ion based systems may be mentioned at this point: the double electron capture system may require some means to separate negative ions from low-energy neutrals originating in the charge change cell.[14] As mentioned in the previous paragraph, acceleration may be close coupled or separated from the first, extraction stage.

In the former group, acceleration of multiampere, low divergence, positive ion beams to energies around 100 keV has been studied extensively and the LBL, 120 kV system is in operation. Higher energies, up to 500 kV across a single gap and up to 750 kV across two gaps, have been achieved with proton currents up to 1 A, but pulses were much shorter, 1 ms or less (applications in pre-injectors of high energy accelerators). The only negative ion accelerator for energies above 100 kV and currents of around 1 A has been constructed at BNL and is being studied presently, with pulses 10 ms long.[11] In the second group, a system has been tested for separate, post-acceleration of H^- ions to an energy of 700 keV[6] (Table IV), at a beam current density of 20 mA/cm^2.

In principle, parameters (current density, energy, geometry) of any single-gap, close coupled structure for acceleration of either positive or negative ions are determined by the space charge effects in the beam and the voltage hold-off between the electrodes. Studies done for single-gap (they should be valid for multiple gap Pierce systems as well) have shown,[13] that the maximum current density is given by:

$$J_{max} = k \cdot E_{max}^2 \, V_o^{-1/2} \quad ,$$

where E is the maximum (or safe) field strength and V_o the final beam energy. The highest electric field will exist at the high energy end because as the particle energy increases, focusing forces of the electric field decrease somewhat faster than the defocusing forces of the space charge do. For $V_o \approx 500$ kV and for D^+ and D^- ions, J_{max} is less than 0.1 A/cm^2. Secondary effects (stripping for negative ions, ionization in collisions with molecules of the residual gas for both, positive and negative ions) will be different for different ion species and will have to be taken into account in more detailed studies.

This limitation of a single-gap or Pierce close coupled structure points out to the system with separated extraction and acceleration as possibly more advantageous for negative ion current densities of about 0.1 A/cm^2 and voltages higher than a few hundred kV. In such a post-acceleration system additional magnetic focusing could be included before acceleration, so that a good beam optics should be achievable with multiple gaps operating at field strengths lower than required for a Pierce structure.

The role of the next element (Fig. 1) is essentially the same in both negative ion based systems: to transport the beam from the accelerator to the neutralizer. Requirements may be summarized as follows: losses as small as possible, no deterioration of beam properties (some additional magnetic focusing may be envisaged), separation of positive ions before they enter the accelerating column, and differential pumping of gas diffusing out of the accelerating column. It is expected that the beam will be space charge neutralized.

d) <u>Neutralizer</u>

Three types of targets have been considered for neutralization of an H^- beam: gas or metal vapors, plasmas and lasers (photodetachment). Their characteristics are summarized in Table V.

Gas or metal vapors have lowest neutralization efficiencies,[15] but are simple, reliable and have low power requirements. Depending on the target, values of neutralization efficiency at the optimum target thickness (cm^{-2}) range from about 40% for N_2 and CO_2, to about 50% for He and Ar and to about 60% for H_2, Cs and Mg. The choice of the target is a compromise between the efficiency of neutralization on one hand and the required target thickness and ease of pumping on the other.[15] In plasma targets[16] neutralization efficiencies of about 85% are expected; this has been experimentally confirmed[17] for ions of energies between 0.5 and 1 MeV in Li and Mg plasmas. Although the required plasma target thickness is usually lower than for gas or metal jet targets, it is still high enough (for energies of several hundred keV of the order of 10^{15} cm^{-2}) to represent a challenge if large cross section dc beams have to be neutralized. Laser beams, using photodetachment, may in principle approach 100% neutralization, but due to very high power requirements[13] they would be economical for very large systems only.

The power efficiency of the whole system depends directly on neutralization efficiency. Charged components (H^+, H^-) represent a net power loss. Schemes have been proposed to recover a part of the energy carried by charged particles. They include the use of waste heat from the beam dump for operation of diffusion pumps, conversion into electricity or direct recovery.[13] The problem with the last scheme is the presence of two species, positive and negative ions in roughly equal proportion, if an optimum target thickness is chosen (except in the laser system, where only negative ions would remain). It has been proposed[13] to use an over-dense target, sacrificing a small part of the neutralization efficiency (e.g., from 65 to 60% for a Cs target or from 86 to 82% for a cesium plasma target) but achieving a very high percentage of positive ions in the charged component of the beam. This scheme has, however, a disadvantage, because another high voltage power supply of the opposite polarity is necessary to decelerate the ions and recover their energy. In a negative ion system based on direct sources, ion sources will eventually be at the ground potential so as to be more accessible for maintenance, while the neutralizer will be on the high voltage terminal. It may be more advantageous to use an under-dense neutralizer, first because for the same neutralization efficiency as achieved with an over-dense target one would need a target thickness two to three times lower, and second, a large part of the charged component after the neutralizer would be in the form of negative ions. To recover the energy from negative ions at the ground potential would require equipment simpler and less expensive.

The power efficiency of the neutralizer would be further improved by separating positive ions at the high potential and by using heat obtained from dumping them to operate diffusion pumps.[13] Table V shows beam compositions for both, under-dense and over-dense, targets, at 90% of the optimum efficiency;[13] the data are for D^- beams of 1.2 MeV energy, in Cs vapor or Cs plasma targets.

System Performance

It is difficult to estimate the performance of a neutral beam system based on direct extraction sources because of a number of uncertainties. If the system power efficiency is defined as the neutral beam power divided by the total net input power (total input power minus recovered power), Fig. 6 shows the power flow for several cases, assuming a 10 A source and 500 kV accelerating voltage. Two values of the source efficiency were assumed, 0.05 A/kW (present sources) and 0.1 A/kW (improvement by a factor of 2), a current load on the extractor equal to the extracted negative ion current (present sources), two values of the beam transport efficiency through the bending magnet (50% and 80%), a power loss in the accelerating system equal to 0.5% of the beam power (a similar value has been assumed in Ref. 13 for a double electron capture system), a plasma neutralizer with a 75% efficiency and, if used, direct energy recovery system with 70% efficiency. It follows from the diagram that the most important elements are the neutralizer and energy recovery system. An improvement by a factor of two in the source efficiency affects very little the total efficiency, more important is the beam transfer efficiency through the bending magnet. The neutral beam power is 3 MW in cases A, B and D (6 A of D^o) and 1.87 MW in the case C (3.75 A of D^o).

The second important system parameter is the gas efficiency. The range of quoted values for different ion sources is even wider than for the power efficiency. If, for an argument sake, a value of 5% is assumed, a 10 A source would have a gas flow of 37 Torr ℓ/s. If a separate acceleration is used, one may estimate that less than 10 Torr ℓ/s would flow through the accelerating column, the rest being pumped away from the back of the transfer system. In cases A, B and D this would correspond to an overall gas efficiency of 11%, lower in the case C.

TABLE I MAGNETRONS

Parameters \ Source		Novosibirsk H_2		Fermilab 15 Hz H_2	BNL H_2
H^- (D^-) current	A	0.9	1	$\approx$0.1	0.9 (0.6)*
H^- (D^-) current density	A/cm^2	2.9	3.3	$\approx$1	0.7 (0.45)
Pulse length	ms	1	1	0.09	10 (20)
Discharge current	A	450	150	150	260 (180)
Cathode current density	A/cm^2	110	50	15	20 (14)
Discharge voltage	V	100	120	175-200	120
Total discharge power	kW	45	18	16-30	30 (22)
Cathode power density	kW/cm^2	7.5	4	1.8-2	1.5
Power efficiency	mA/kW	20	56	3.0-4.0	30 (20)
Gas efficiency	%	5 - 6		---	2-3

*Parameters for 20 ms pulses are given in parenthesis.

TABLE II PENNING SOURCES

Parameters	Source	Novosibirsk H_2 emitter No	Novosibirsk H_2 emitter Yes	Efremov H_2	LASL H_2	BNL H_2	BNL D_2
H^- (D^-) current	A	0.15	0.2	0.48	0.11	0.44	0.2
H^- (D^-) current density	A/cm^2	3	5.4	1.2	2.2	0.44	0.2
Pulse length	ms	0.2		1.2	0.7	3	6
Discharge current	A	180	450(80*)	260	60	65	40
Cathode current density	A/cm^2	300	(90)	[100]*	[100]	33	20
Discharge voltage	V	100	100(100)	150	80	220	400
Total discharge power	kW	18	45(8)	39	4.8	14.3	16
Cathode power density	kW/cm^2	20	(9)	[10]	[4.8]	4.8	5.3
Power efficiency	mA/kW	17	3.8	12	23	30	12
Gas efficiency	%	5	[15]	[5]	0.8	1.1	

*Values in (): emitter parameters.
Values in []: expected or not confirmed.

TABLE III

Parameters	Source	BNL cooled cathode source, design values Penning	Magnetron	5 A scaled-up Penning	10 A scaled-up Magnetron
H^- (D^-) current	A	0.9	1.8	5	10
H^- (D^-) current density	A/cm^2	0.5	0.5	0.25	0.5
Pulse length	ms	25-50	25-50	dc	dc
Discharge current	A	150	500	1000	2000
Cathode current density	A/cm^2	20	20	20	20
Discharge voltage	V	200	120	100	100
Total discharge power	kW	30	60	100	200
Cathode power density	kW/cm^2	2.7	1.6	1.3	1.3
Power efficiency	mA/kW	30	30	50	50
Gas efficiency	%			10	10

TABLE IV

Parameters \ Source		Novosibirsk Penning	LASL Penning	Fermilab magnetron
Extracted H^- current	mA	100-150	≈120	100
Extracted H^- current density	A/cm^2	2-3	≈2	≈1
Extraction voltage	kV	20	18	20-25
Output H^- current	mA	100	110	50
Output H^- current density	mA/cm^2	125(10*)	>15	20
Transport efficiency	%	80-90	90	50
Post-acceleration voltage	kV	--	--	700
Field index		1	0.85	1

*Noisy discharge.

TABLE V

Target	Gas or vapor jet	Plasma	Laser
Efficiency	40 - 60%	80 - 85%	up to 100%
Power requirements	low	medium	high
Charged components at optimum	$\approx$ 1/2 H^-, 1/2 H^+	$\approx$ 1/2 H^-, 1/2 H^+	H^-
Cs target, optimum density, cm^{-2}	1.6×10^{16} 20% D^+, 15% D^-	2×10^{15} 8.4 D^+, 5.9 D^-	
Over-dense target, 90% efficiency	2.6×10^{16} 36.6 D^+, 4.4 D^-	4.5×10^{15} 23% D^+, 0.2% D^-	
Under-dense target, 90% efficiency	0.8×10^{16} 3% D^+, 38% D^-	1.2×10^{15} 5% D^+, 18% D^-	

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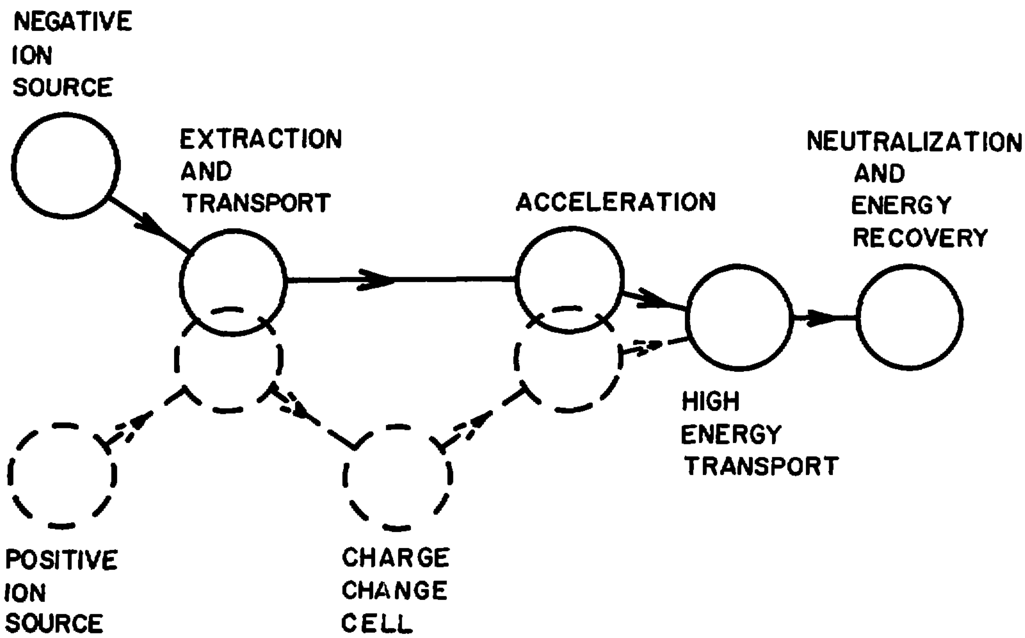

Fig. 1 Negative ion based neutral beam systems

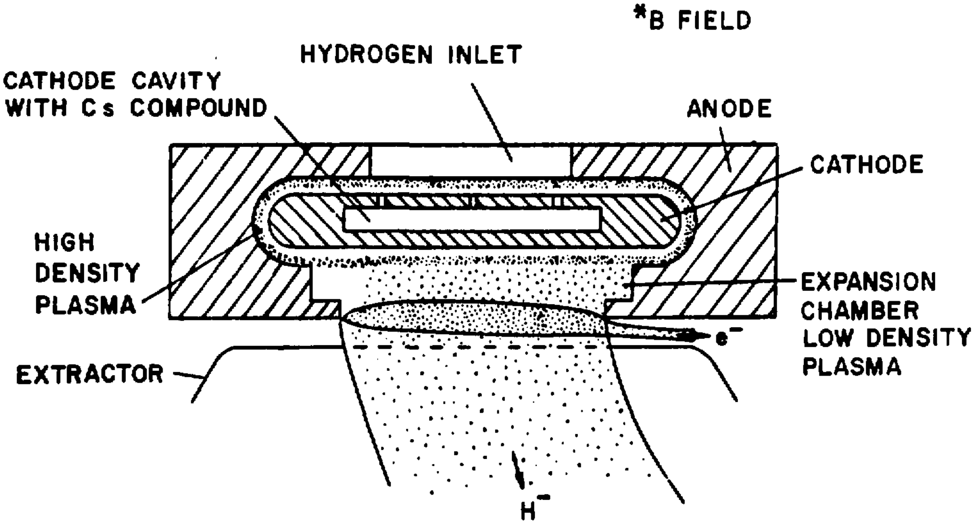

Fig. 2 Magnetron source

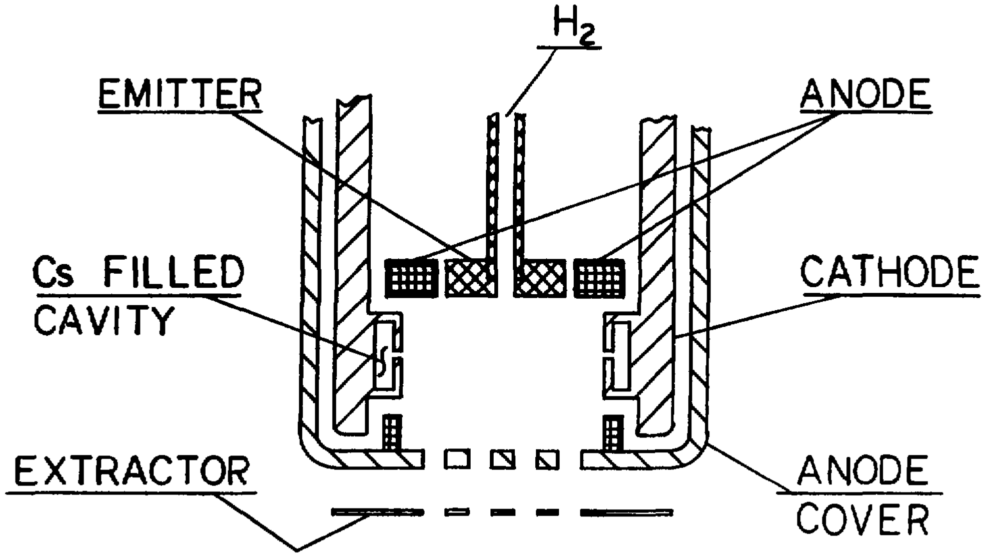

Fig. 3 Penning source

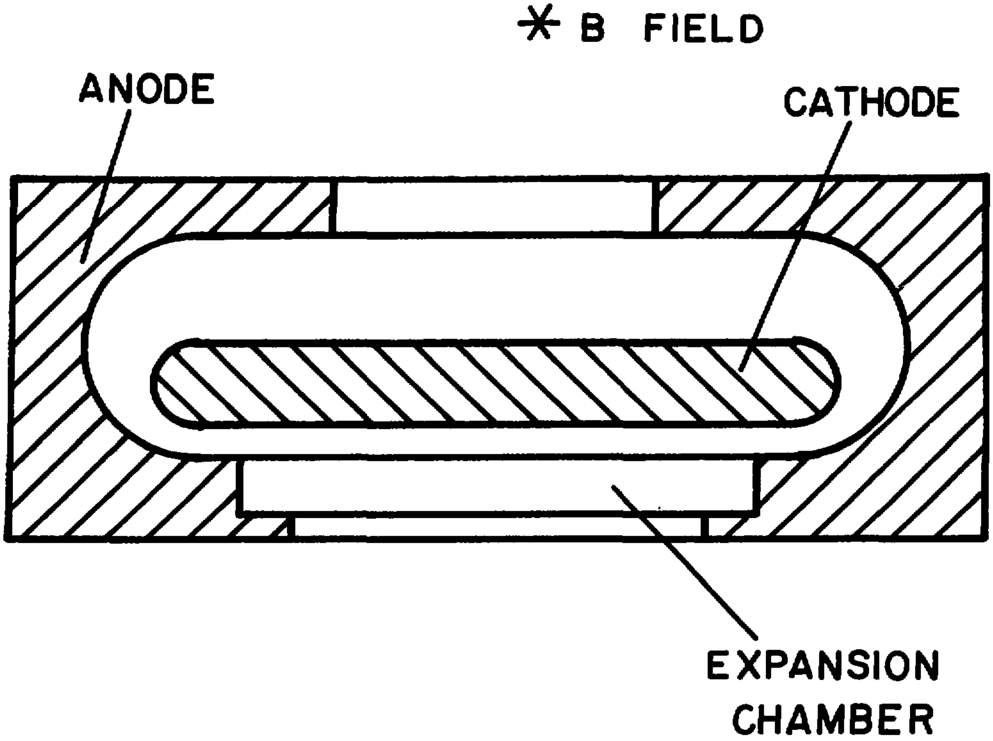

Fig. 4 Asymmetric magnetron

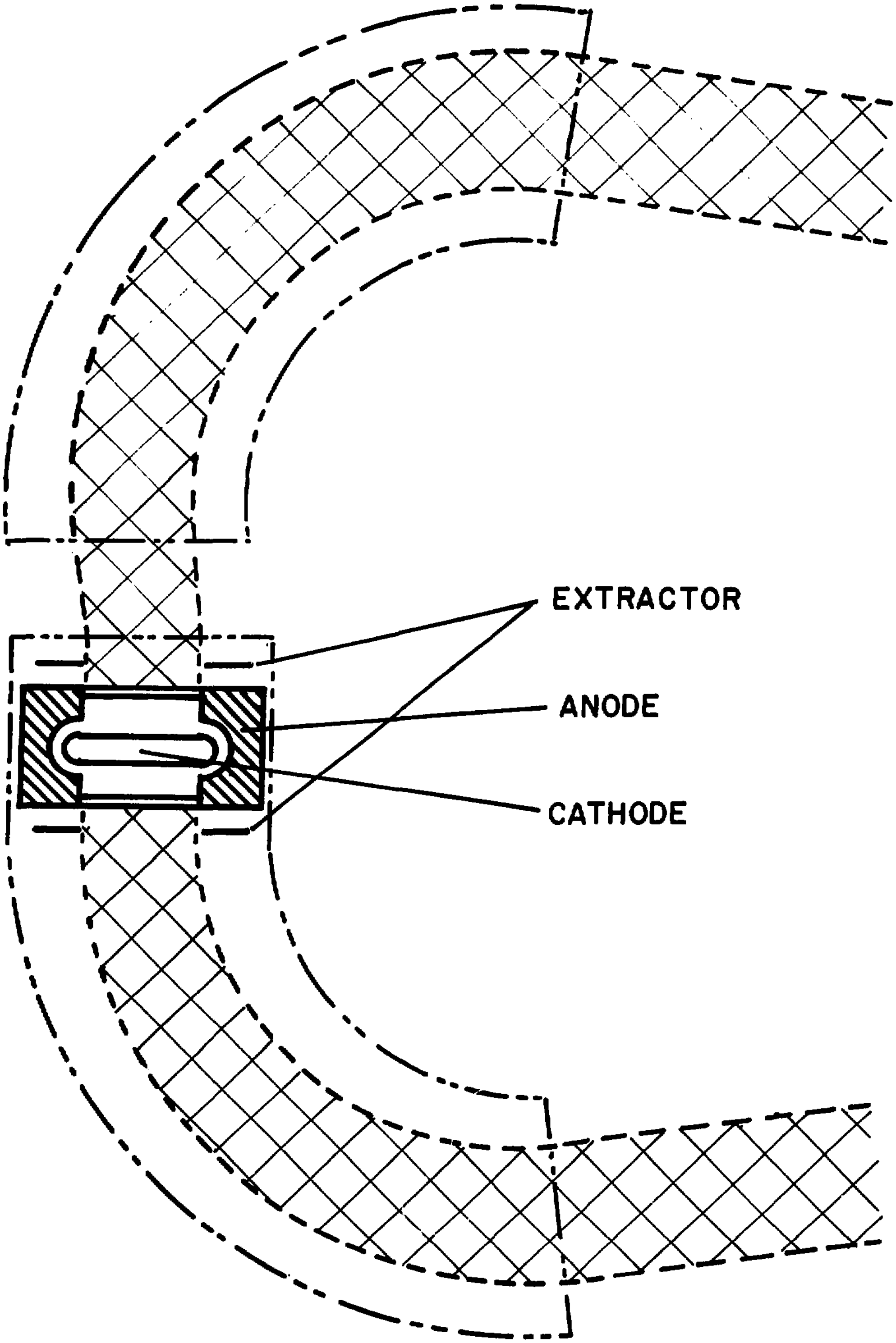

Fig. 5 Two-side extraction

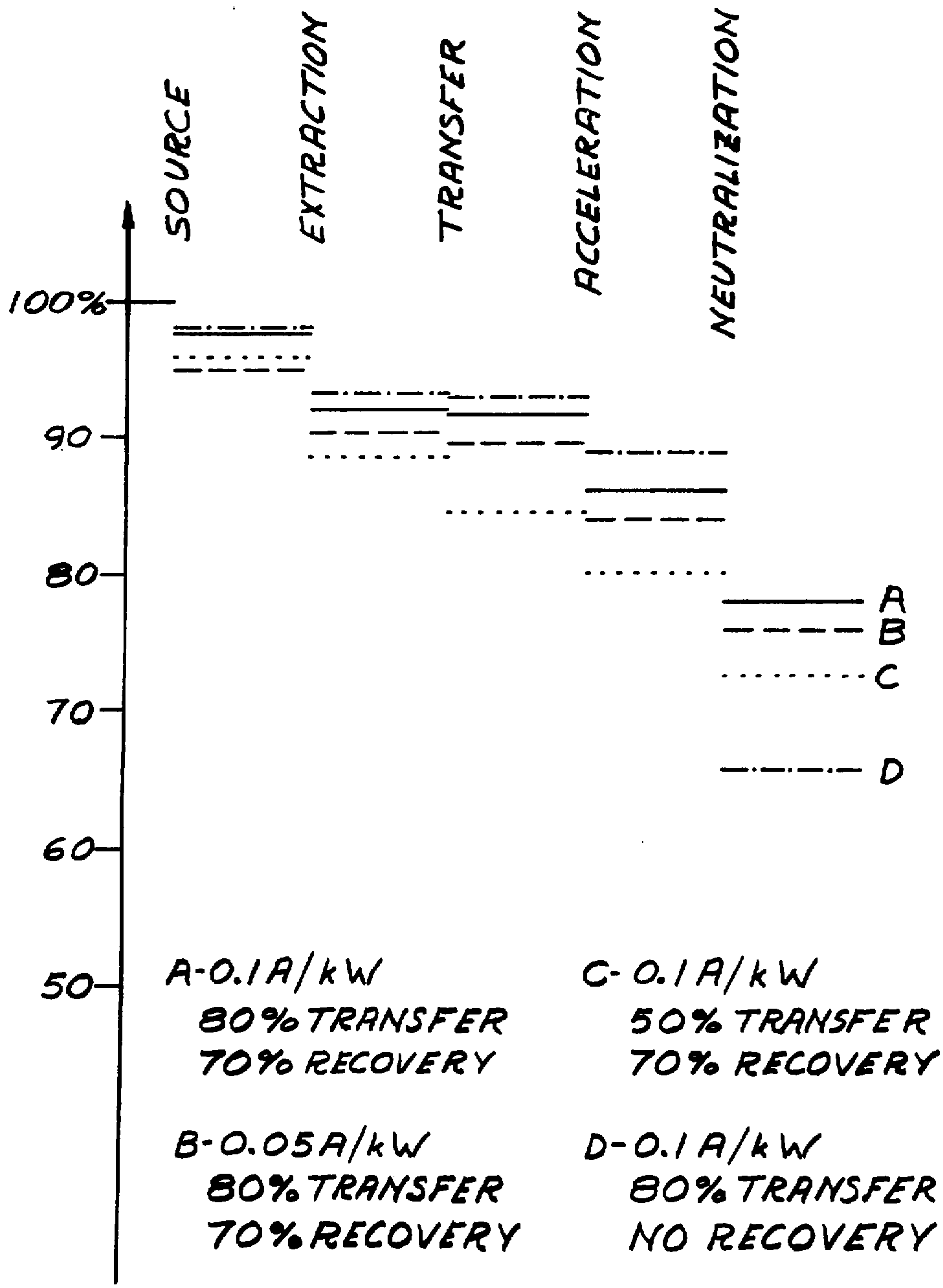

Fig. 6 Power flow diagram

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HIGH ENERGY NEGATIVE DEUTERIUM BEAMS USING DOUBLE CHARGE-EXCHANGE OR SURFACE PRODUCTION

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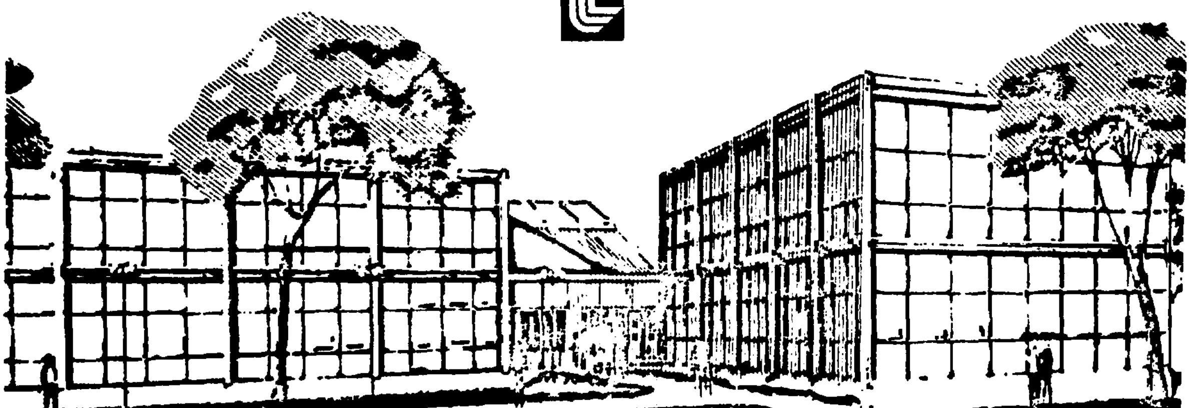

High Energy Negative Deuterium Beams Using Double Charge-Exchange or Surface Production

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ABSTRACT

Experimental and theoretical research on production of negative ion beams is described. Results from a double charge-exchange experiment include 10 ms pulses of 100 ma of D^- accelerated to 60 kV. Equilibrium fractions of D^- in several metal vapors are presented. Mechanisms and measurements of D^- on surfaces are described, and a scheme is shown for producing high current, high energy beams originating on surfaces.

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I. Introduction

This paper describes the Lawrence Laboratories' experimental and theoretical research on the production and acceleration of negative ion beams. The goal of this work is to provide the information required for the design and construction of large, high voltage neutral beam systems within the near future. The present results are encouraging for such systems, although considerable work remains before beams at hundreds of keV will be available for fusion experiments.

The major negative ion experiment uses double charge exchange in cesium vapor. This is described in Section II. Atomic and surface physics research is described in Section III; this work was performed by researchers, listed there, other than the above authors. Its purpose is to support the larger developmental effort and to provide the information and guidance required in the development of advanced negative ion systems. Finally, in Section IV we describe one proposed advanced system, based on the surface production of negative ions.

II. High Voltage Beams Using Double Charge-Exchange

The double charge-exchange method of producing negative deuterium beams in principle can be scaled to arbitrarily high currents. Also, as discussed below, the geometry of a double charge-exchange system permits control of gas and electrons. Because of these and other considerations, it was chosen as our primary approach.[1]

Osher et al.[2] have previously produced beams of 50 mA of D^- using double charge-exchange in cesium. In the present experiments,[3,4] conditions for stable propagation of a negative ion beam have been studied and found, and a beam of 0.1 amp has been accelerated to 60 keV. The experimental arrangement, illustrated in Fig. 1, consists of a positive ion source, a charge exchange cell, a drift space, and a 100 keV accelerator structure. A stripping cell is not employed in the present experiment. Each of the component parts of the

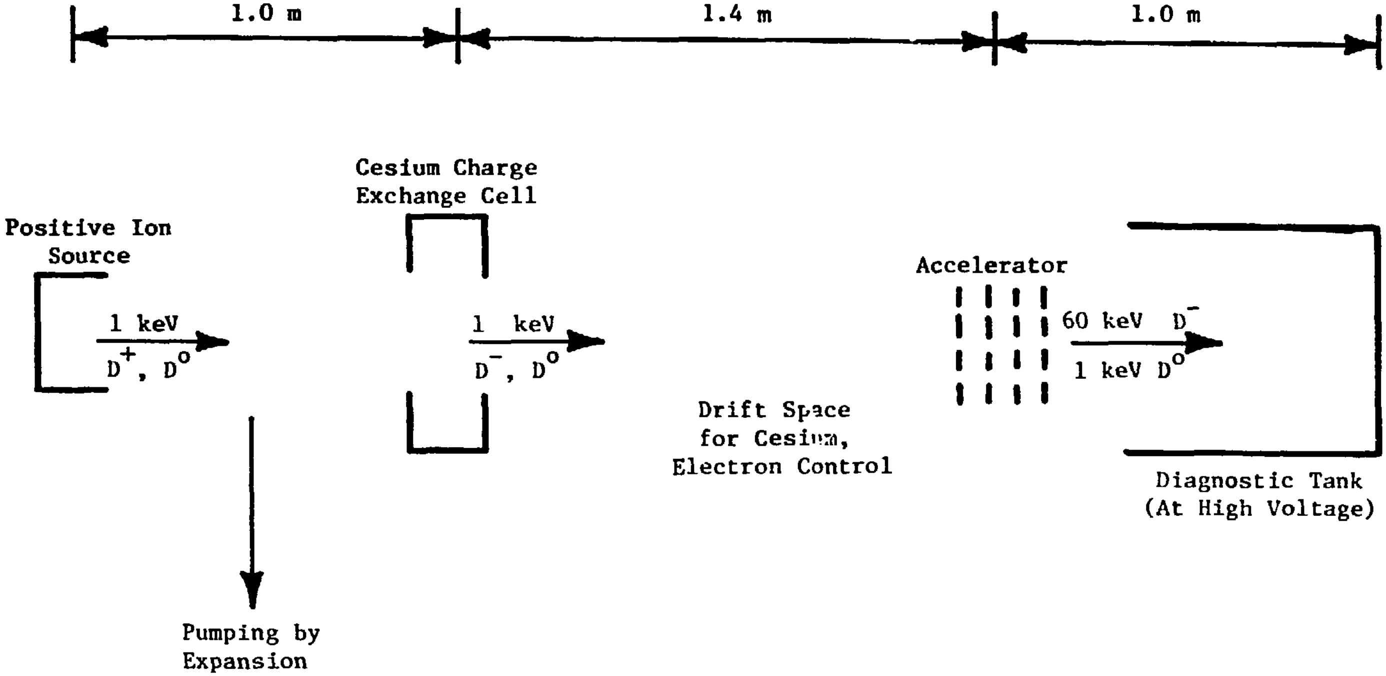


Fig. 1. Schematic of the D^- (double charge-exchange) Experiment.

beamline is discussed, beginning with the positive ion source and the cesium charge exchange cell. The formation of plasma and its flow is described; it is shown that a drift space can be employed to stabilize the beam and to control the electron concentration at the entrance to the high voltage accelerator structure. The design of the accelerator is given in the last paragraph of this section, and data on the accelerated beam is presented.

The Positive Ion Source

The double charge exchange method of producing a negative ion beam requires a well directed flow of positive ions into the charge exchange target. The source of positive ions employed in the experiments described here is a "10-amp" source developed at LBL.[5] A strong accel-decel grid array[6] is employed for the ion extraction and beamlet focusing; the grid is 60% transparent, with 10 amps extracted over a 7 cm x 7 cm area. Due to breakdown limitations in the grids, the RMS angle of the flow divergence has been 8° in the tests reported here. The extracted ions are substantially neutralized by charge exchange on the background gas flowing from the source. Pumping of the excess gas is achieved by expansion into the volume of the source vacuum chamber, with a resulting limit upon the pulse duration of 10-25 ms.

The grids have been redesigned to improve the divergence. Preliminary tests indicate that space charge in the neutralizer near the grids is important, so electron emitters are being installed to reduce the problem. We are also testing a very different source, the MPD arc, described later in the paper.

The Charge Exchange Cell

The primary function of the charge exchange cell is to convert an incident positive ion beam to a negative ion beam. The charge exchange medium employed for this purpose, cesium, can adversely affect parts of the beamline such as the accelerators, and is an undesirable impurity in the target plasma. The cesium, therefore, must be contained within the confines of the charge exchange cell. An effective way of minimizing the loss of cesium along the beamline is to employ a directed flow of cesium transverse to the beamline. Upon crossing the region of charge exchange, the cesium vapor is pumped by cooled surfaces. The randomly directed component of the cesium flow is restrained from flowing along the beamline by a set of cooled baffles.

The charge exchange cell employed in this experiment is shown in Fig. 2. The flow of cesium is pulsed in order to reduce the quantity of the cesium flowing in the cell. The cesium is vaporized in the oven and passes through a transfer tube to a pulsed valve that controls the flow of vapor to the plenum and a slit nozzle orifice. The valve is open for $\sim$1 sec. The cesium vapor forms a jet through which the beam passes and is charge changed. The flow is directed onto a liquid nitrogen cooled surface, the hopper, and is collected. A set of cooled baffles along the beamline aid in the collection of the vapor. The cesium is periodically recycled into the oven. Operation of this system has been highly reliable, with many recyclings before additional cesium is required.

The effect upon beam intensity of the angular scattering during charge exchange collisions was investigated. Calculations[7] show that the scattering is small at the line density (2×10^{15} cm^{-2}) of cesium that is required to reach the equilibrium fraction (24% at 1 keV)[8] from an incident D^0, D^+ beam. A measurement of the D^- production efficiency from the ion beam employed in this experiment yielded a value of 20%. This value, when corrected for the contribution of molecular ions in the beam, corresponds well to the measured equilibrium fraction of 24% for D^0, $D^+ \rightarrow D^-$, D^0 conversion, and thus confirms that angular scattering does not reduce the beam intensity.

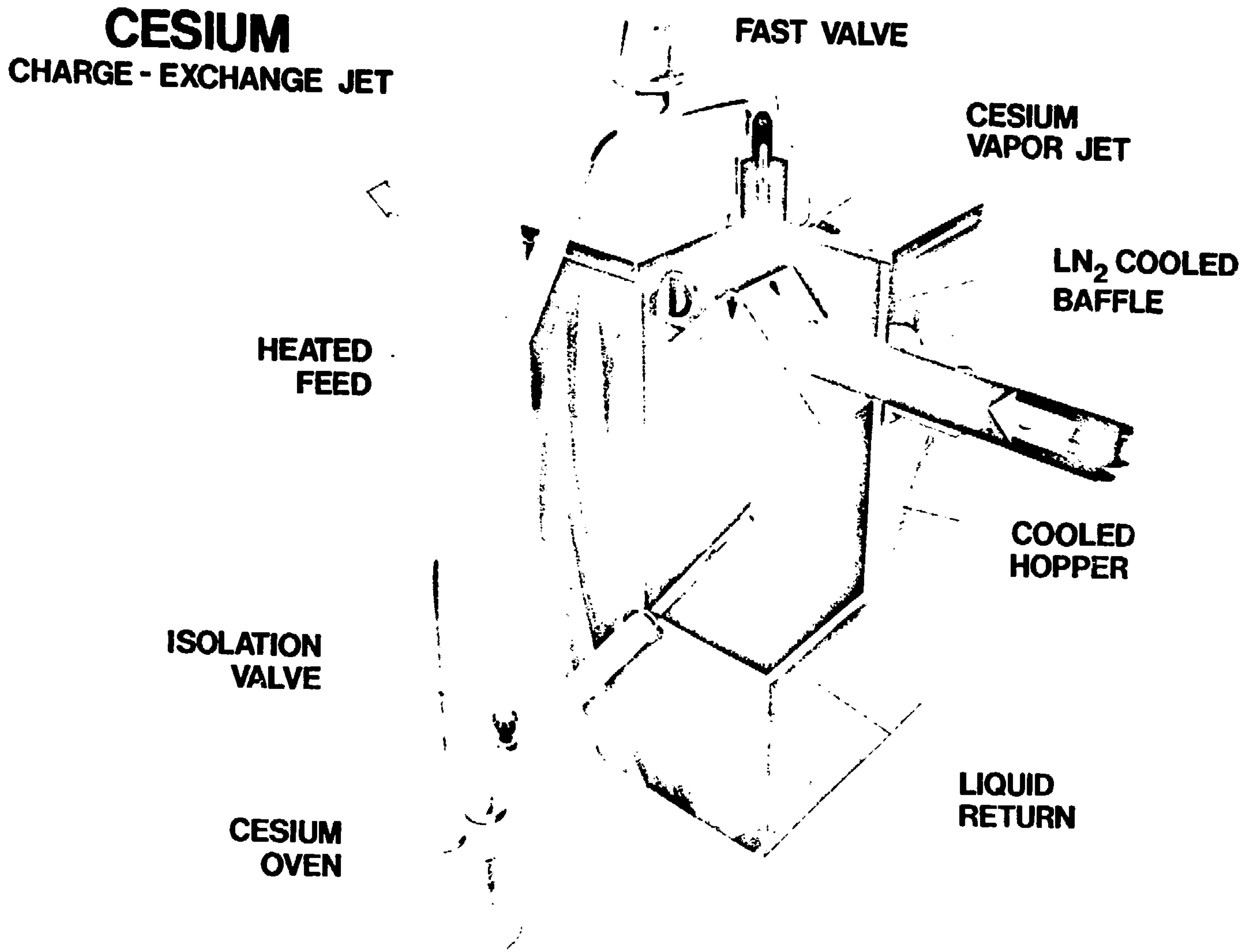
Plasma Formation

The ion beam generates a plasma as it passes from the positive ion source through the charge exchange cell to the high voltage accelerator.[9] The plasma, which flows along and away from the beam, serves to neutralize the space charge within the ion beam. The presence of electrons within the beam and near the aperture of the accelerator will cause a current drain due to the acceleration of electrons along with the ions. Interaction of the beam with the plasma may cause unstable propagation. Thus, the self-generated plasma has a strong effect upon beam acceleration and propagation.

The background gas density required to neutralize the space charge within a beam can be estimated. We model the beam in cylindrical geometry, with a beam radius R_b, velocity v_b, and density n_b. The continuity equation for the radial outflow of ions is written

$$\frac{1}{r} \frac{d}{dr} (rnv) = S \quad ,$$

Fig. 2.



CESIUM
CHARGE - EXCHANGE JET
FAST VALVE
CESIUM
VAPOR JET
LN_2 COOLED
BAFFLE
HEATED
FEED
COOLED
HOPPER
ISOLATION
VALVE
LIQUID
RETURN
CESIUM
OVEN

where S is the rate of creation of the background positive ions. In this experiment, the dominant source of ions is ionization by a 10 keV component of the beam.[10] This component, about 5% of the extracted beam, is formed by charge exchange in the extraction grids, and is highly directed. Due to its low divergence, the density of the 10 keV beam component, which does not charge exchange in the cesium cell, is estimated to be equal to the negative ion beam density. The ionization cross section $\sigma \approx 10^{-16}$ cm^2.

Thus $S = \sigma v_b n_b n_g$, where n_g is the background deuterium gas density. Other sources of positive ions, such as ionization by electrons, by the 1 keV beam, charge exchange processes, and effects in cesium, are estimated to be weaker.

In order to estimate a gas density at which the positive ion density roughly equals the beam density, we estimate $v = \sqrt{kT_e/M_i}$ at the beam edge $r = R_b$; then the gas density for which $n = n_b$ is $n_c = (2/R_b\ \sigma)\ (\sqrt{kT_e/M_i V_b^2})$. For $R_b = 5$ cm and $T_e \approx 1$ ev, $n_c = 2 \times 10^{13}$ cm^{-3}, i.e., a gas pressure of 6×10^{-4} Torr. At this gas pressure, therefore, the beam and electron densities are nearly equal.

The experiment operates at an order of magnitude below this pressure, so that stripping of the D^- ($\sigma = 10^{-15}$ cm^2) reduces the D^- beam intensity by no more than 20%. It is found experimentally that the beam propagation is not stable below a pressure of 4×10^{-5} Torr.

The generation and two-dimensional flow of plasma along and across the beam was investigated computationally.[11,12] Of particular interest was the flow of plasma from the cesium cell, where it is relatively dense, toward the accelerator where the presence of electrons is not desirable. The beam is again assumed to be rigid, and the geometry is axially symmetric. A radial beam density profile $n_b(r)$ and an axial distribution of background gas $n_g(z)$ is employed. The results of a typical computation are shown in Fig. 3. It is noteworthy that the electron density is localized near the region of the greatest ionization rate, e.g., the electron density falls off sharply away from the cesium cell. Thus, these results show that the accelerator structure may be placed within a few beam diameters of the cesium cell without adverse effects due to the formation of plasma in the cesium cell.

Infinite medium analysis of the stability criteria indicate that the negative ion beam should propagate stably for electron velocities greater than the beam velocity, as in our experiment. A computer code has been employed to

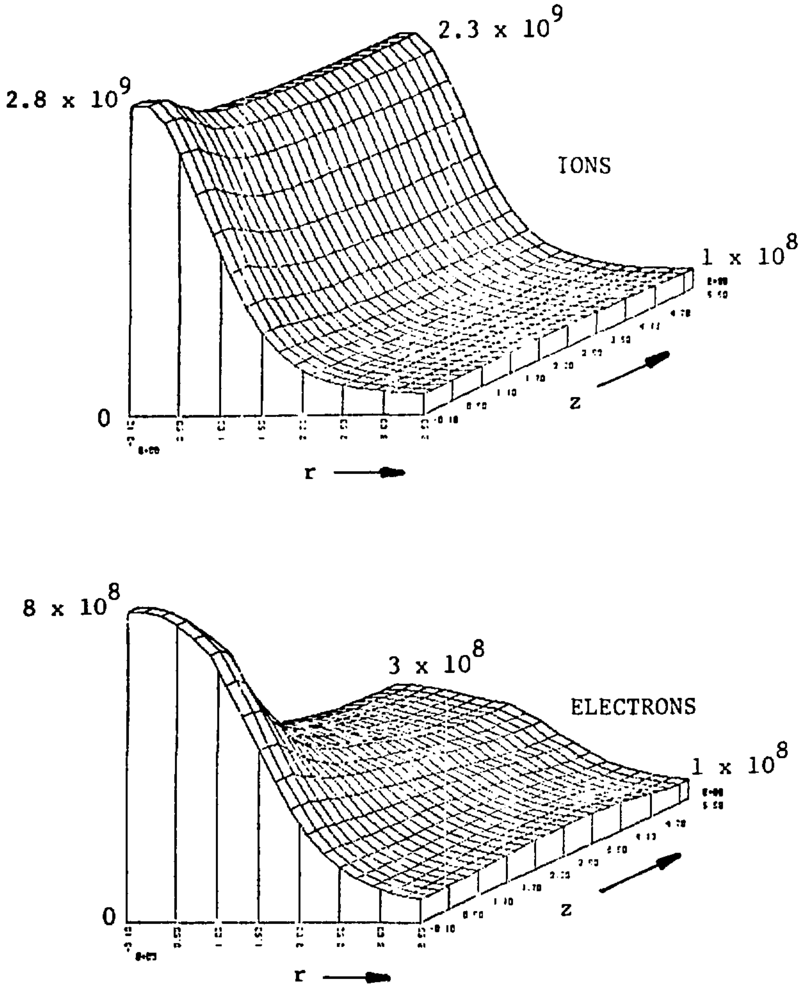


Fig. 3. Plasma densities in two-dimensional flow; $n_b = 2 \times 10^9 \exp(-r^2)$,

$n_g = 0.26\, n_{cr}\, (1 + 3e^{-z})$.

study the instabilities associated with the non-uniform beam. Preliminary results[13,14] indicate instability at low pressures, resulting from coupling to a radial acoustic wave. Detailed comparison with the experimental observations is not yet possible.

D^- Acceleration

Approximately 100 mA of D^- is injected into the accelerator shown in Fig. 4. The trajectories have been calculated by the Wolf code. For these results, the beam has been assumed to have zero temperature.

The detailed design takes into account the actual particle trajectories, of course, but the electrodes have specific purposes. The first (entrance) electrode defines the beam size. The third (focusing) electrode is used to concentrate the beam to optimize the beam optics in the main accelerating gap. To the first approximation, the resulting space charge in this gap satisfies the Child-Langmuir law. The second (control) electrode is biased to minimize overfocusing of the particles at the beam edge; in essence, it plays the same role as the Pierce angle in standard charged particle accelerators. The last pair of electrodes can be biased to prevent backstreaming ions, although in the present experiments they were set to the same voltage. The shapes of electrodes near the walls prevent a direct line of sight between the beam and insulators.

Initial measurements of the accelerated beam indicate that the electron current is significantly less than the negative ion current. These measurements were made using a weak magnetic field at the exit of the accelerator, and confirm the measurements made in the 1 keV beam upstream from the accelerator.

Faraday cup measurements 1 meter from the accelerator, Fig. 5, yield a beam divergence of $\pm$ 2.5° both along and across the aperture, which compare with $\pm$ 1.6° calculated by the Wolf code.

Neutrons produced in a titanium disk saturated with deuterium give a qualitative confirmation that a high energy deuteron beam is produced.

Although these results are preliminary in nature, they are very encouraging for the accelerator techniques used in the experiment. Work will continue to compare experimental and theoretical results in this area.

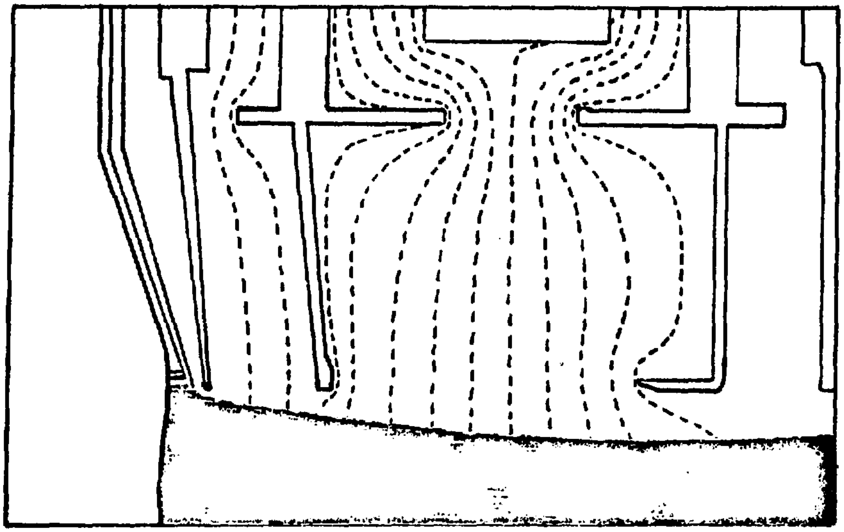


Fig. 4. High Voltage Accelerator. $j = 3mA/cm^2$. Electrode potentials are (from the left) 0 kV, 2.5 kV, 10 kV, 35 kV, 60 kV, and 60 kV.

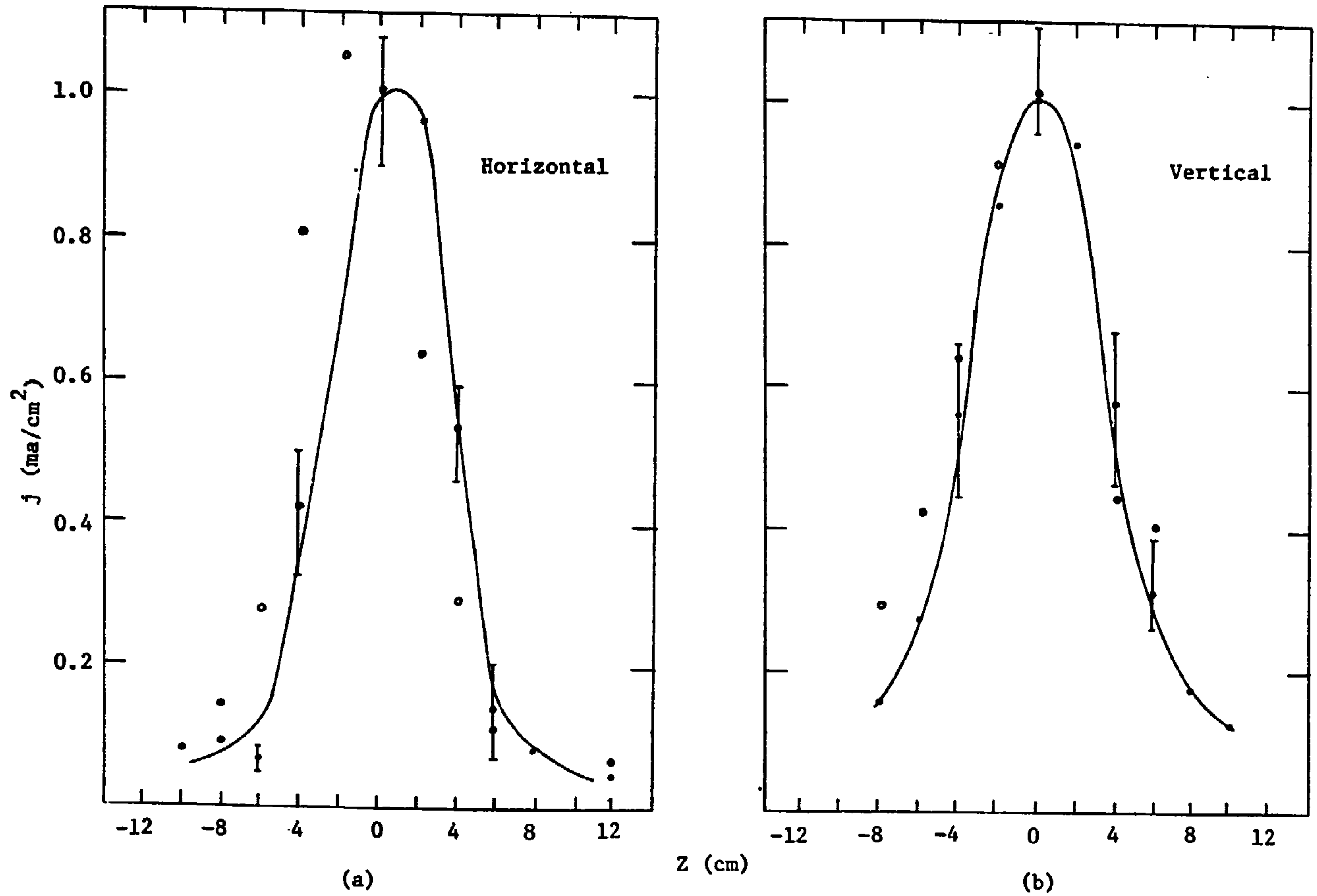


Fig. 5. Beam profiles at end of diagnostic tank, as measured by a Faraday cup. Solid dots: D^- current. Open dots: D^- plus electron current.

Advanced Double Charge-Exchange Techniques

To extend the results described above, we are developing new methods of producing intense positive ion beams and of producing a highly directional cesium jet. The goals of these developments are the production of D^- current densities of 50-100 mA/cm^2, and the confinement of the cesium to a small section in the beamline. The latter goal will permit a much more compact beamline to be constructed.

The fraction of a proton or deuteron beam which can be converted to negative ions exceeds 30% at 600 eV, and is independent of energy at least as low as 300 eV (c.f. Section III and Ref. 8). In this energy range, the cross sections for conversion from neutral atoms to negative ions and vice versa decrease somewhat, but their ratio is constant. Extrapolation of the results is uncertain, of course, but it is quite likely that large conversion efficiencies exist to considerably lower energy.

In the energy range of hundreds of eV and below, intense plasma beams can be produced with a highly directed flow. Such beams contain equal densities of electrons and ions, and are thus not subject to space charge limitations. These could then be used to produce quite intense negative ion beams.

We are working with one such source, the MPD arc,[15,16] shown in Fig. 6. MPD arcs are extremely intense, typically operating at 50 volts and 500 amperes. The injected gas is consequently highly ionized; heating and compression near the cathode tip, followed by expansion as the plasma is ejected, produces a highly directional flow at energies up to at least 50 eV. These arcs were originally produced for use as space thrusters, and are surprisingly sturdy and reliable. Testing of the MPD arc for negative ion production has just begun, and no results are available at the time this is written.

Further development of the cesium jet is important for successful development of intense beams. The present jet, described above, is formed from a simple slit geometry. Measurements of the resulting cesium density distribution[3] show that it can be modeled as coming from a diffuse source on a line several slit widths below the actual slit. The resulting distribution, $(\cos\theta)/r^2$, drops as distance cubed for large distances along the beamline. This rate of fall is satisfactory for an experiment, but requires improvement in an actual neutral beam system.

Metal vapor jets have been developed in the USSR[17] using Laval nozzles as illustrated in Fig. 7a. More than 50% of the flowing material is

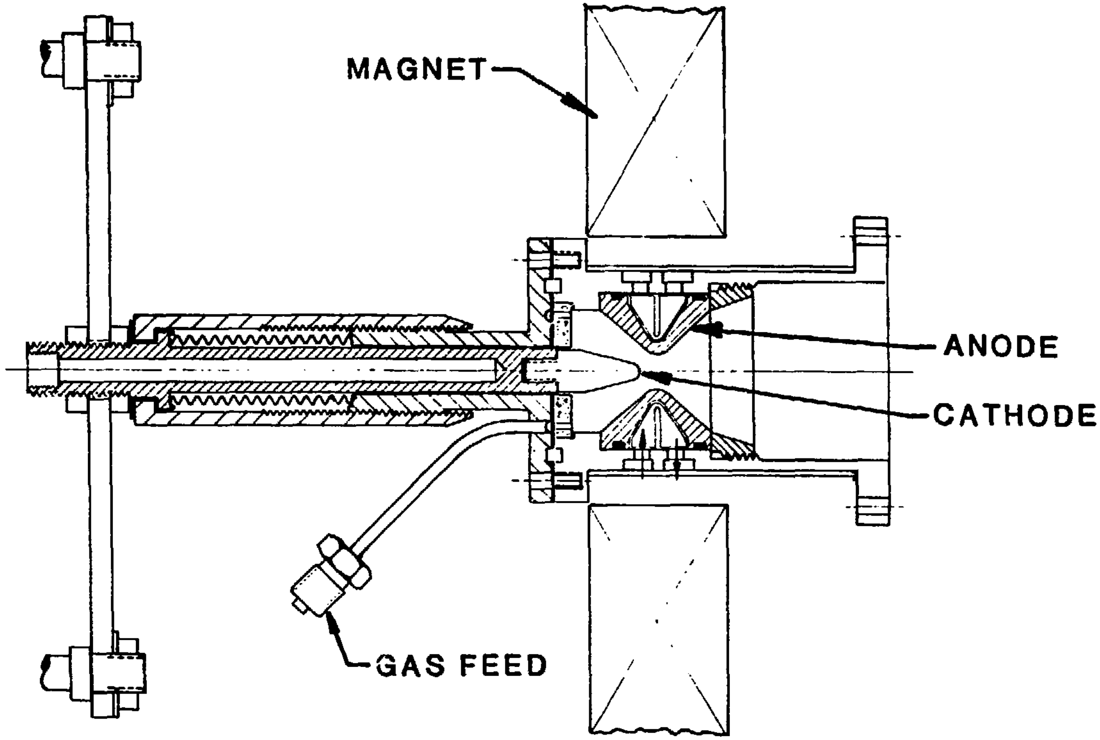


Fig. 6. MPD (Magnetoplasmadynamic) arc. Current is drawn from the center cathode to the coaxial anode.

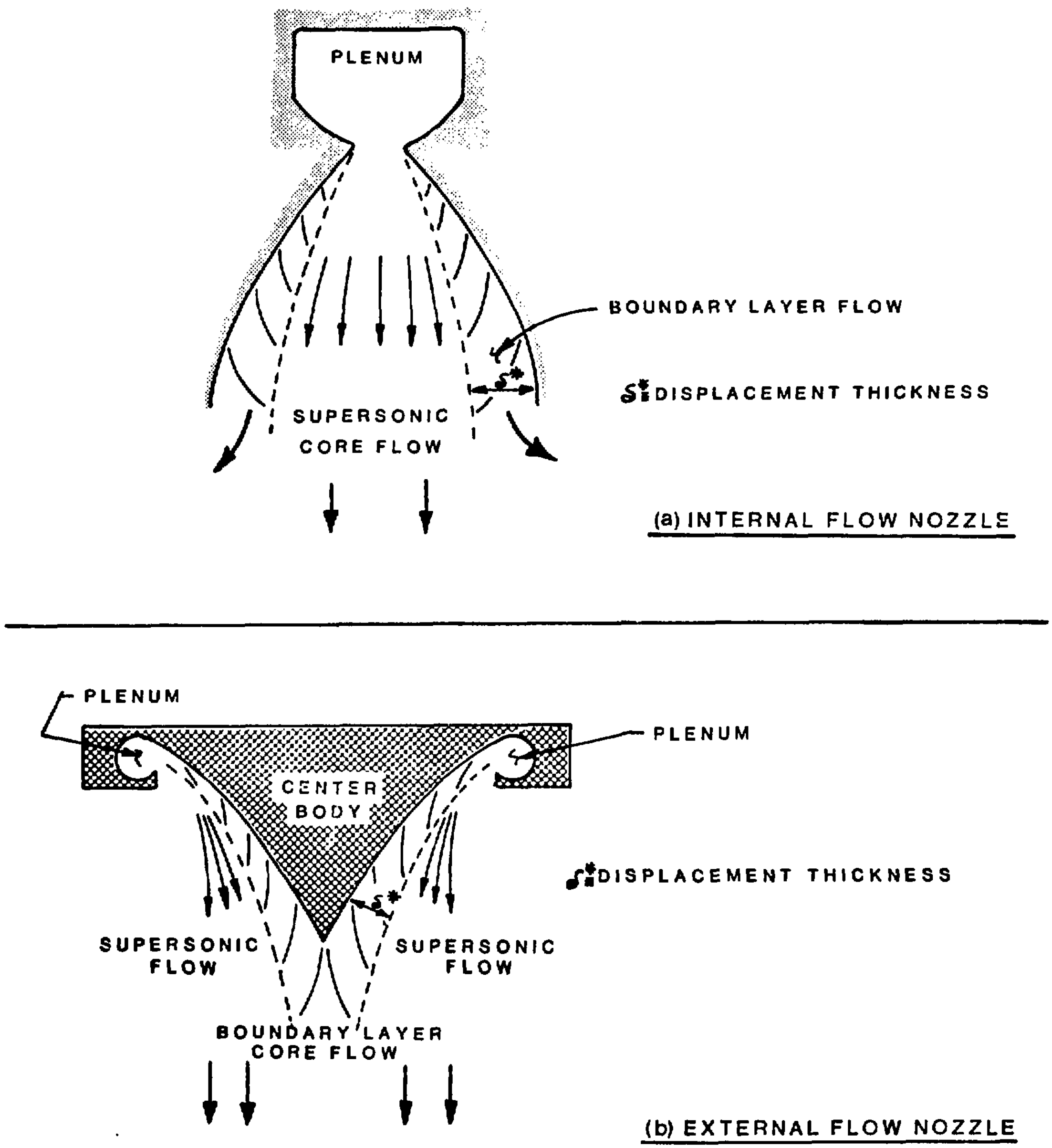


Fig. 7 Nozzles for metal vapor jets.

in a highly directional supersonic jet. A substantial amount of material is in boundary layers, however, and is emitted over a wide angle. Skimmers are used to trap this material, but as the flow is not in the free molecular regime, collisional effects destroy the uniformity and directivity of the jet.

This problem can be alleviated by a jet of the design shown in Fig. 7b. Material is emitted from two slits. Calculations show that boundary layers develop along the "center body" between the two slits. These boundary layers merge at the end of the center body, and are confined by momentum exchange with the supersonic flow. Small boundary layers also form on the outer edges of the slits, of course, but because the amount of material in these boundary layers is small, they rapidly become collisionless and can be trapped by cold surfaces and skimmers. Detailed calculations and design of such a jet are underway, with initial testing planned to be in the present cesium cell.

Charge-Exchange in Sodium

Vapors other than cesium may also be used for the double charge-exchange production of D^-; Schlachter[18] has summarized the cross sections and equilibrium fractions. Sodium has been used by Semashko and co-workers[19] to produce greater than 1 ampere of H^-. Millisecond beam pulses have been accelerated to 40 kV.

Table 1 compares sodium and cesium. Cesium has one major advantage: the conversion efficiency is very high at low energies. As a result, the beam power and gas efficiencies can be considerably higher than with other substances. Furthermore, the low energy requirements may permit the use of plasma accelerators, as discussed above, with resultant high current densities and greater simplicity.

Sodium, on the other hand, does have several advantages.[21] The initial positive ions will have better beam optics at the higher energy, and thus can be easily transported. Furthermore, molecular ions can be dissociated without angular scattering broadening the beam significantly. At the low energies required in cesium, the large vapor line densities required for dissociation will appreciably broaden the beam,[7] so that the use of molecular ions will be difficult.[22] Also, because of the lower atomic number of sodium, considerably more leakage of sodium than cesium can be permitted into the fusion experiment, and the lower sodium vapor pressure at the melting point makes it easier to control. For both cases, however, extremely good control of the metal is required.

Table 1 - Comparison of Cesium and Sodium Charge-Exchange Targets

	Cesium	Sodium
Maximum D^- Fraction	0.35 at 0.3-0.5 keV	0.12 at 2-8 keV
Vapor Pressure at liquid point	2×10^{-6} Torr at 302° K	0.9×10^{-8} Torr at 371° K
Atomic Number	55	11
Permitted n(Cs)/n(ion) in Tokamak[20] (TFTR, Q = 10)	1.5×10^{-4}	2.5×10^{-2}

III. Atomic and Surface Physics*

D^- Formation by Charge-Exchange

Measurements of the equilibrium fraction of negative ions in deuteron beams have been made in several metal vapors.[8,18] Results for cesium are shown in Fig. 8. As emphasized previously, the equilibrium fraction has not started to decrease at 300 eV, and significant generation of D^- may be possible to quite low energies. Measurements have also been made in rubidium, magnesium, and strontium.

D^- Formation on Surfaces

The generation of intense negative hydrogen ions in cesiated magnetron and other plasma sources is believed to result from surface effects.[23,24] The formation of H^- on partially cesiated surfaces has been analyzed,[25,26] and the results support this conclusion.

The coefficient for formation of negative ions is calculated as $\kappa = R_n P_- f$. The backscattering particle reflection coefficient, R_n, is

* The work described in this section was done by A. S. Schlachter, K. R. Stalder, J. W. Stearns, W. G. Graham, and P. J. Schneider at LBL, and by J. R. Hiskes, A. Karo, and M. A. Gardner at LLL.

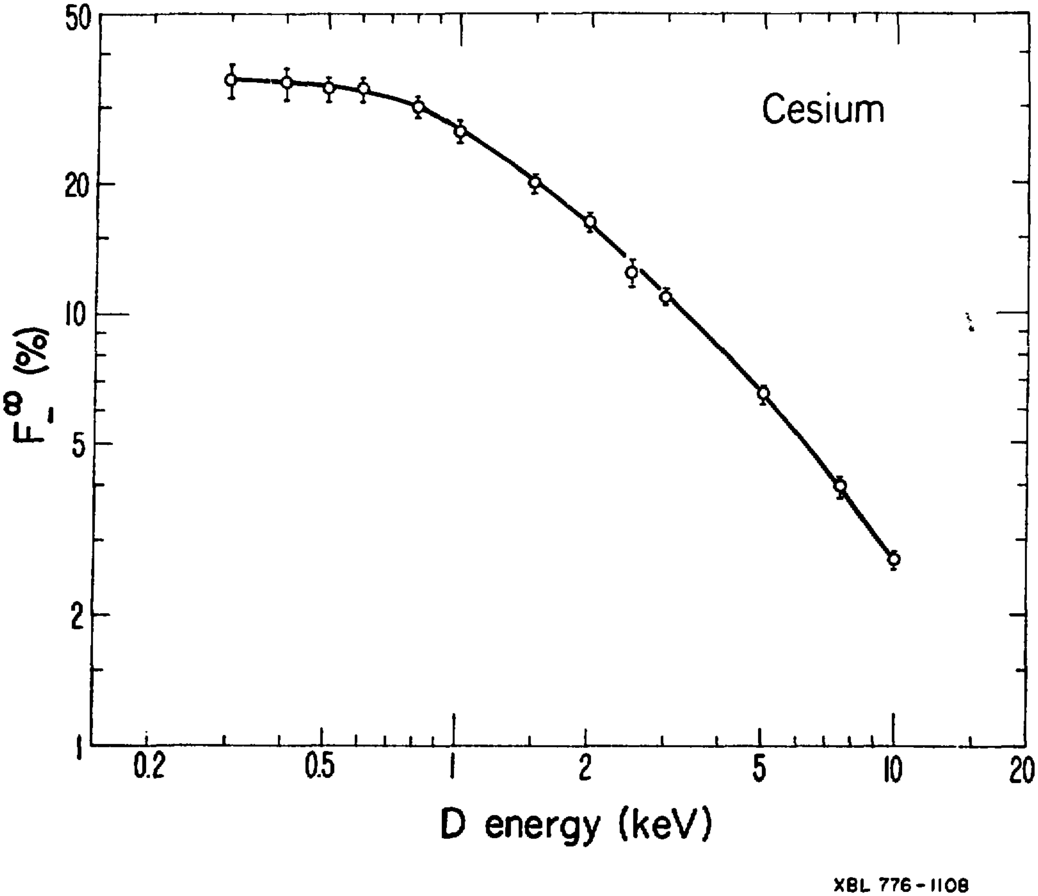


Fig. 8. Equilibrium fraction of D^- in cesium (from Schlachter[18]).

estimated at 60% to 80% from published measurements. The probability for negative ion formation, P_-, is calculated to vary from 55% for a 100 eV H^- ion to > 99% at 1 eV. The survival fraction, f, is calculated to be close to unity in the range >4 eV for a partially cesiated surface. The resulting coefficient κ is thus 30% to 50% for energies 10-100 eV; no reflection coefficients are available below 10 eV. For thick alkali metal coatings, f, and thus κ, are considerably smaller.

Measurements have been made[27,28] of the total backscattered D^- yields from thick surfaces of Cs, Rb, K, Na, and Li, bombarded with D_2^+ and D_3^+ in the energy range 0.05 to 3.5 keV/nucleon. As seen in Fig. 9, the D^- yield was found to be a maximum (as high as 12% per incident deuteron for Cs) between 150 and 300 eV/nucleon.

Some preliminary measurements for partial coverage of the target with alkali metals have also been made. Fig. 10 shows the negative ion yield as a function of Na evaporation time onto a Cu substrate. The maximum yield is believed to correspond to a coverage where the reflection is predominantly from the Cu substrate, while the work function is that of Na. Therefore, partial coverage substantially increases the negative ion yield.

This is also shown in Fig. 9 from the measurements by Dudnikov,[29] using a magnetron ion source. At 130 eV, the H^- yield from a surface with an "optimum" cesium coverage (one where the work function is minimum) is twice that for a thick cesium surface. The decrease at higher energies is believed to be due to decreasing cesium coverage of the surface.

An experiment is also underway to investigate H^- production by the reflection of very low energy H^0 incident on treated metal surfaces. Surfaces with partial Cs coverage are bombarded with H atoms produced in a furnace. The average energy of the H atoms is about 0.25 eV, but it is estimated that 10%

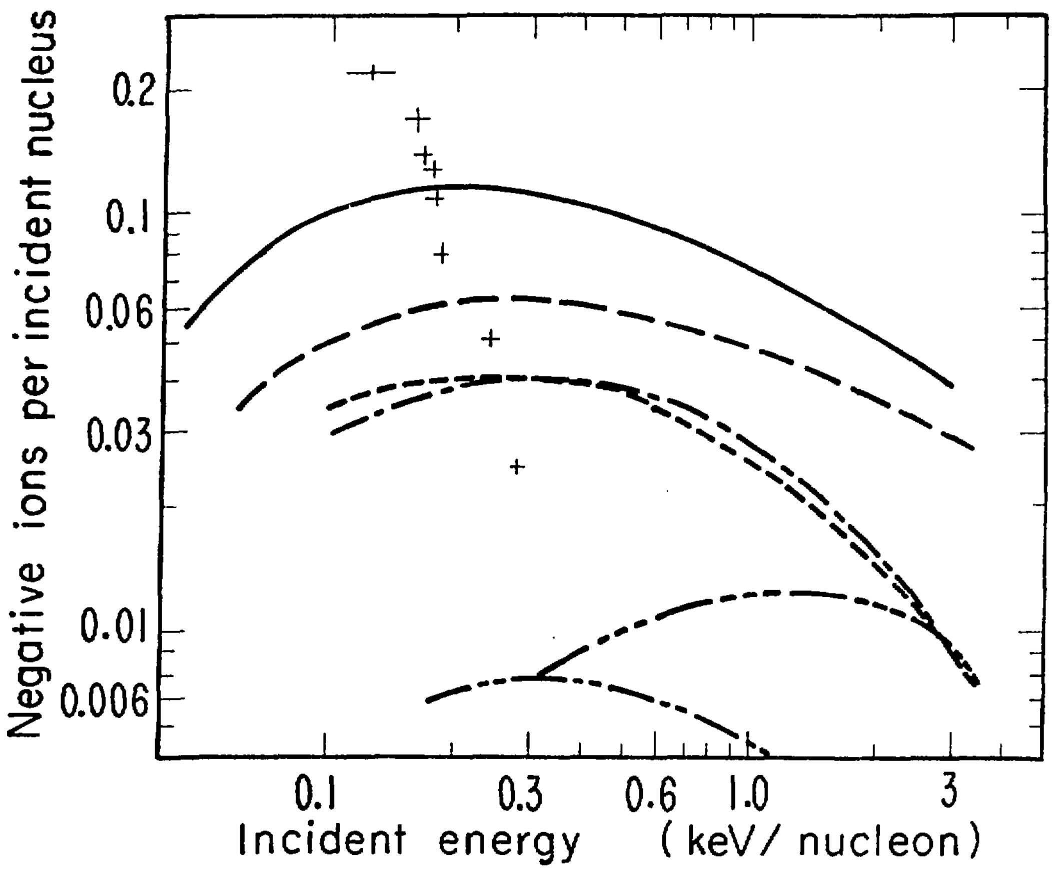


Fig. 9. Total H^- yi..ds from various surfaces. D_2^+ and D_3^+ incident; —— Cs surface; — — Rb surface; ---- K surface; — - — Na surface; — -- — Li surface; — --- — Mo surface, +, total H^- yields from H^+ incident on a Mo surface with partial Cs coverage (Ref. 27).

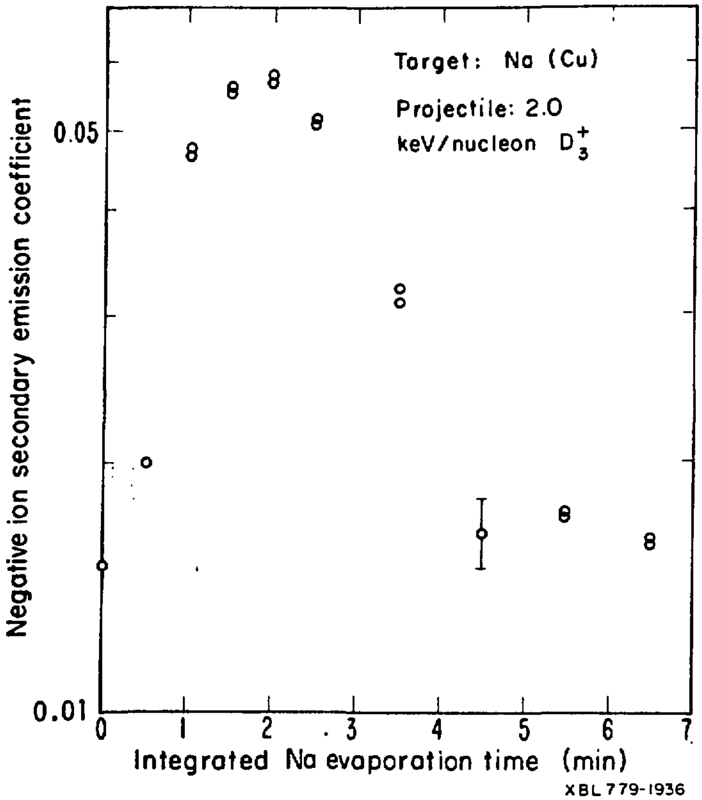


Fig. 10. Negative ion secondary emission coefficient for 2.0-keV/nucleon D_3^+ vs Na-evaporation time for Na deposited on a Cu substrate.

of the H atoms have an energy in excess of 0.75 eV. Since large fluxes are available, H^- production from particles with energies as high as 2 eV can be determined.

IV. Proposed Neutral Beam System Using Surface Production of D^-

It follows from the results in magnetron sources,[23] supported by the above calculations[25,26] and experiments,[27,28] that it may be possible to produce large currents of D^- on surfaces. As discussed, the conversion from D^0 to D^- may be 30-50%, even at 10 eV or less. Further supporting the production of negative hydrogen on partially cesiated surfaces, are preliminary measurements in thermionic diodes at Rasor Associates.[30] Negative hydrogen is observed using mass spectroscopic measurements, even at zero applied voltage.

To take advantage of this production, the system shown in Fig. 11 has been proposed.[31,32,10] Neutral atoms are produced far from the conversion surface, so that excess gas can be pumped before it enters the accelerating structure. An initial ion beam is produced using an MPD arc similar to that described earlier in this paper. The beam is guided by a magnetic field and converted to neutrals by recombination with electrons or by charge-exchange in vapor. The D^- beam is then accelerated by a grid structure similar to that used in positive ion sources with total currents of 50 A or more.

This system would have several major advantages. In particular, (a) excess gas is pumped in a region where it is not harmful; (b) the acceleration structure is simple and compact; and (c) the system can be scaled to any required current, with d.c. operation.

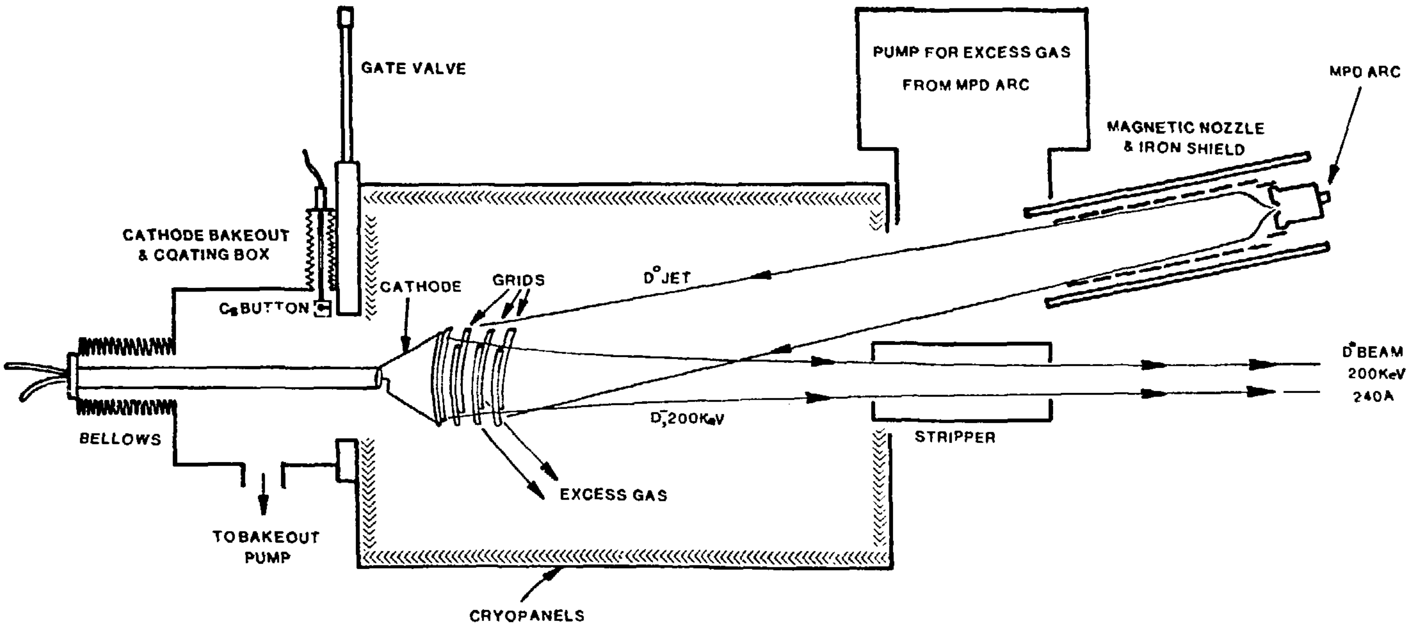


Fig. 11. Proposed neutral beam system based on surface source of negative deuterium ions.

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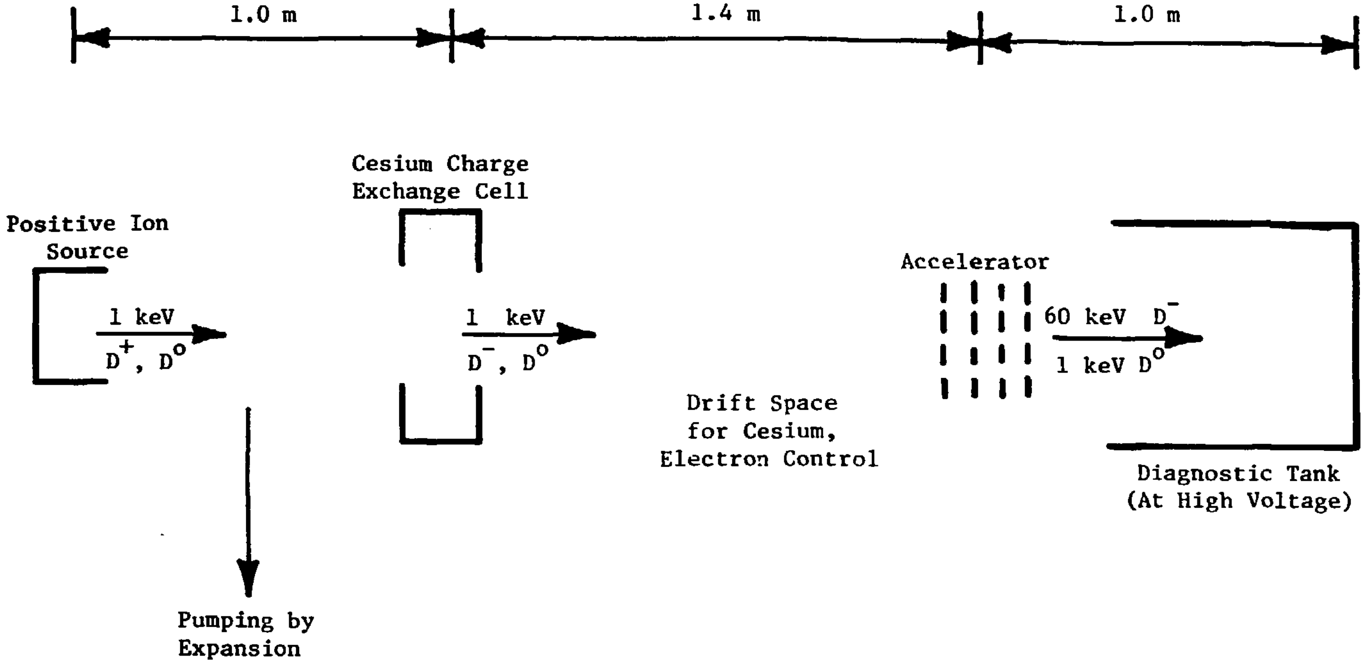
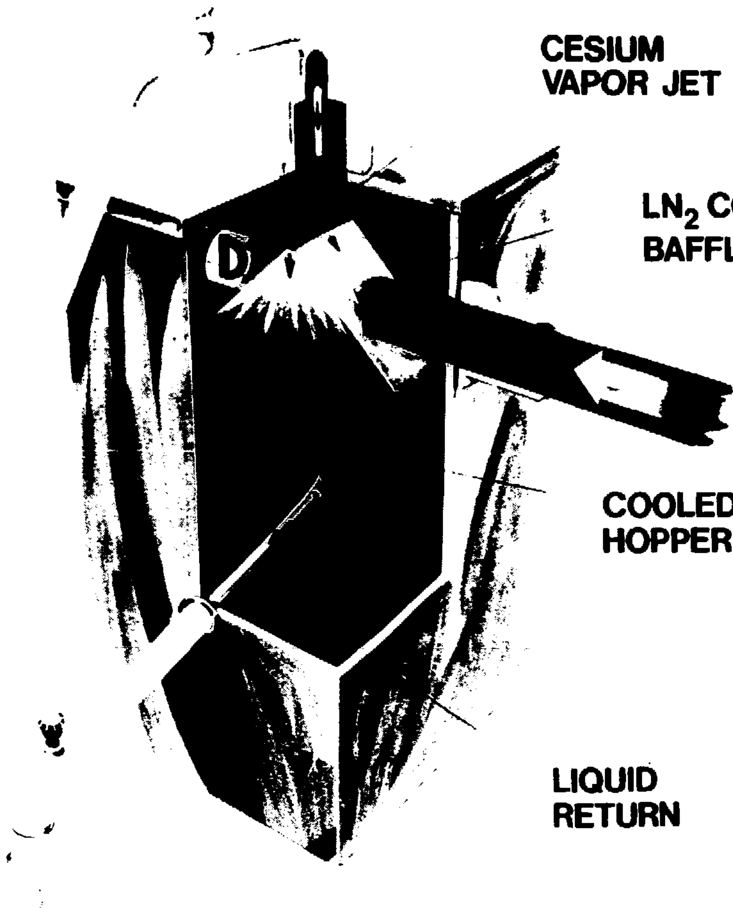
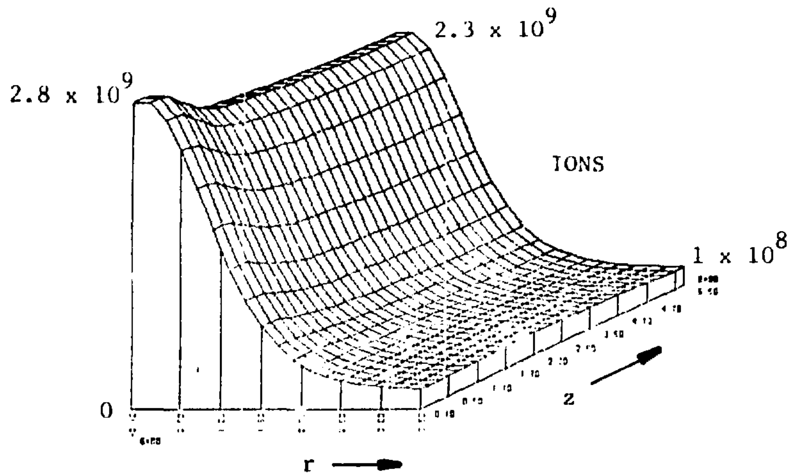
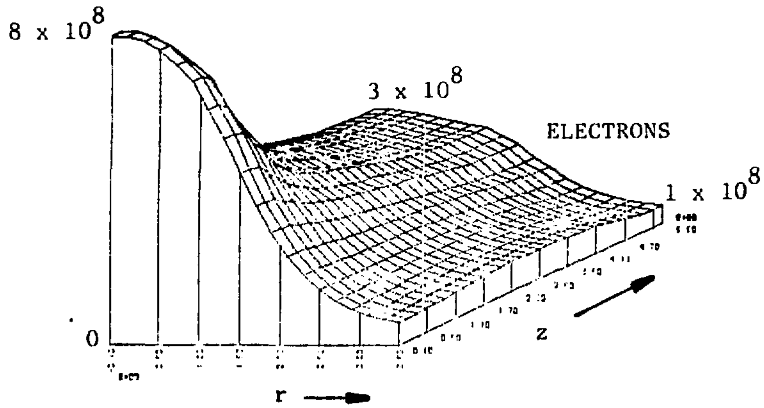


Fig. 1. Schematic of the D^- (double charge-exchange) Experiment.

Fig. 2.


CESIUM
CHARGE - EXCHANGE JET
FAST VALVE
CESIUM VAPOR JET
LN_2 COOLED BAFFLE
HEATED FEED
COOLED HOPPER
ISOLATION VALVE
LIQUID RETURN
CESIUM OVEN





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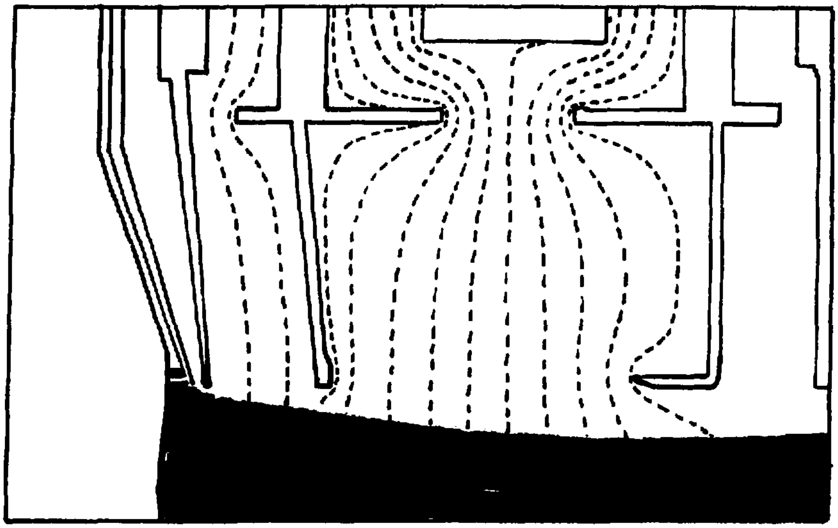

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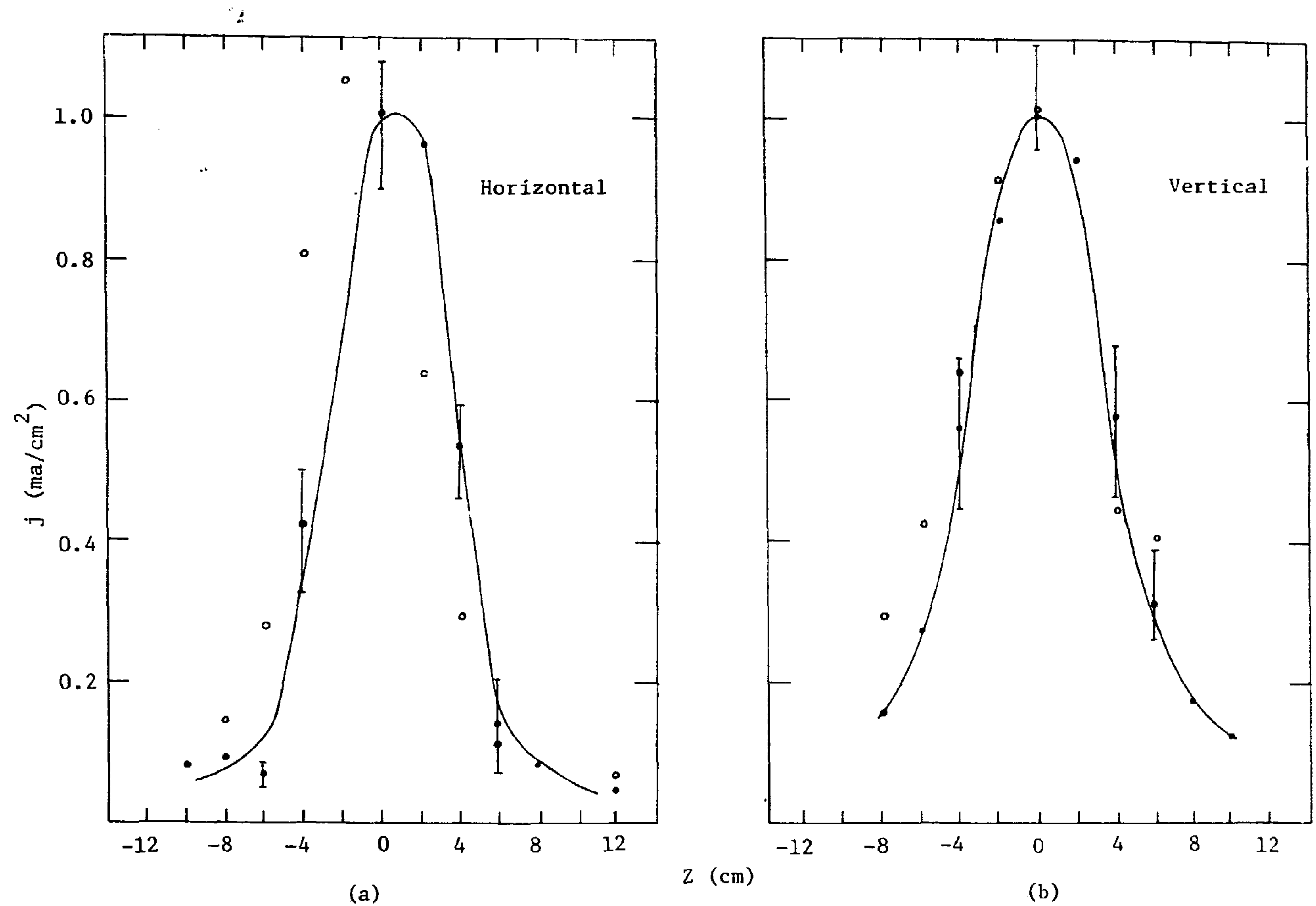

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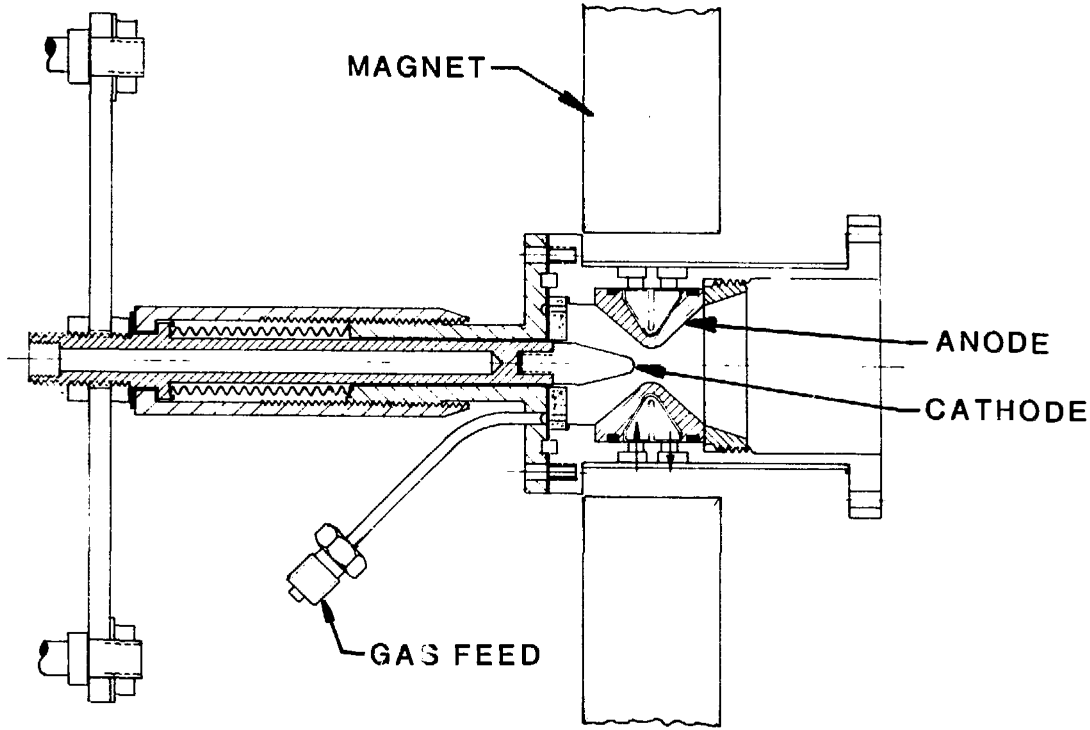

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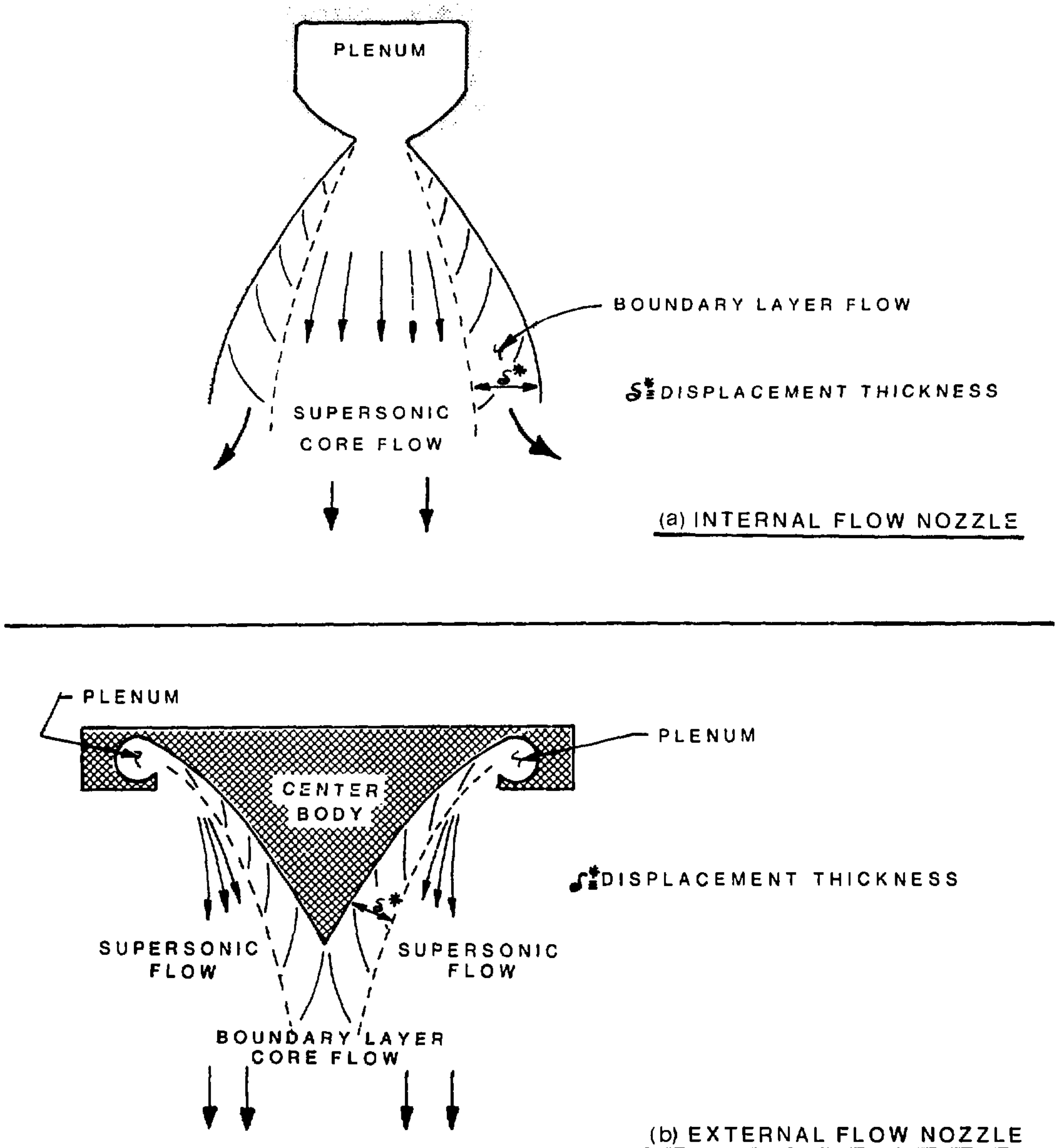

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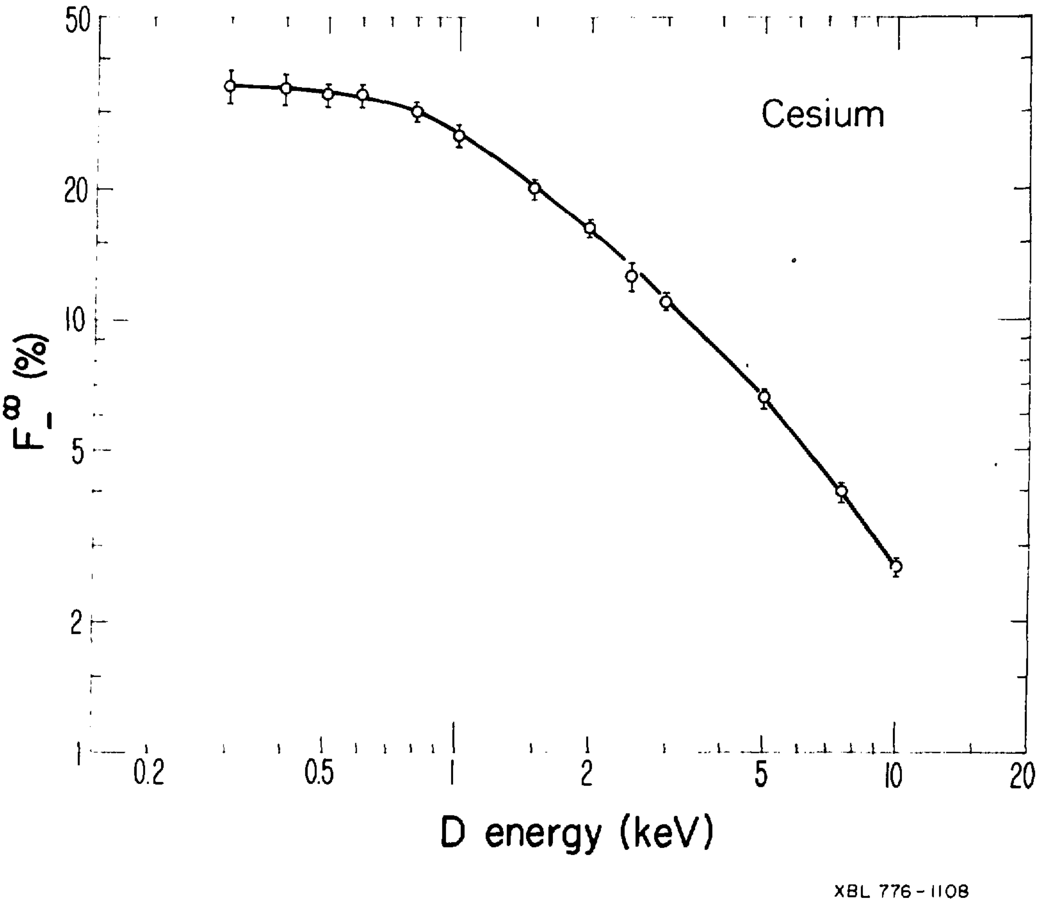

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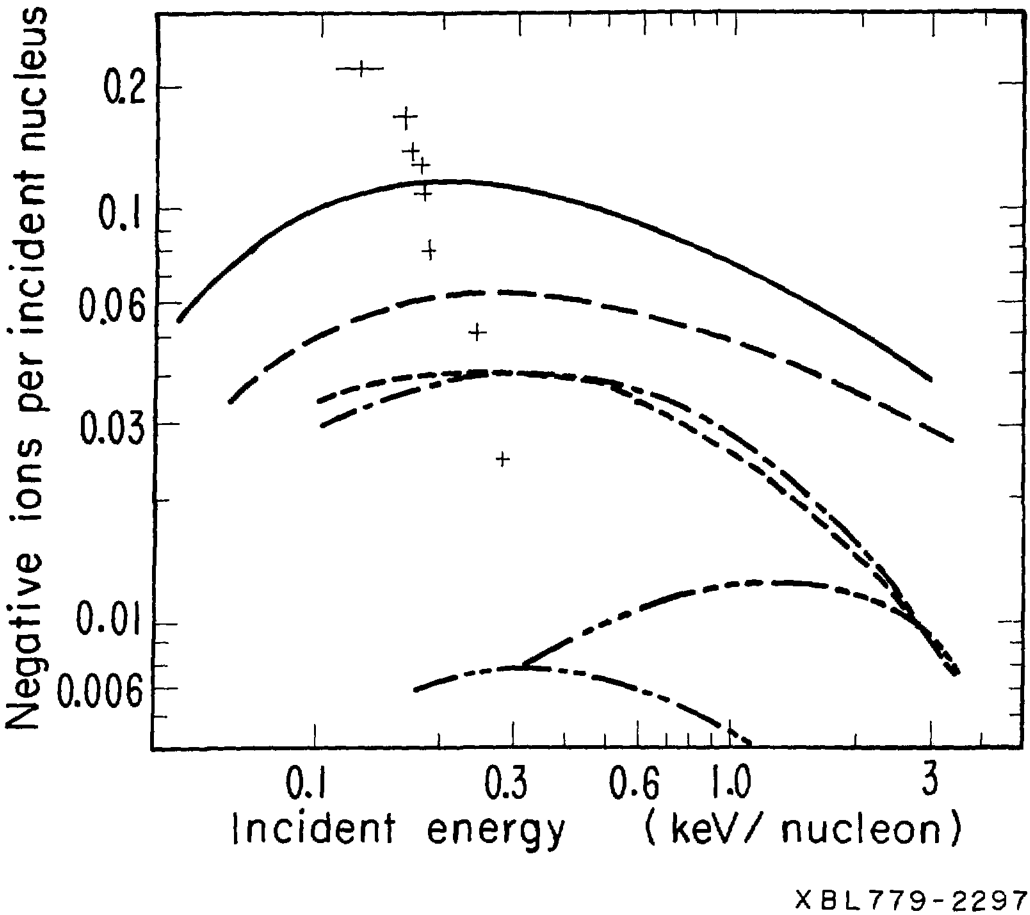

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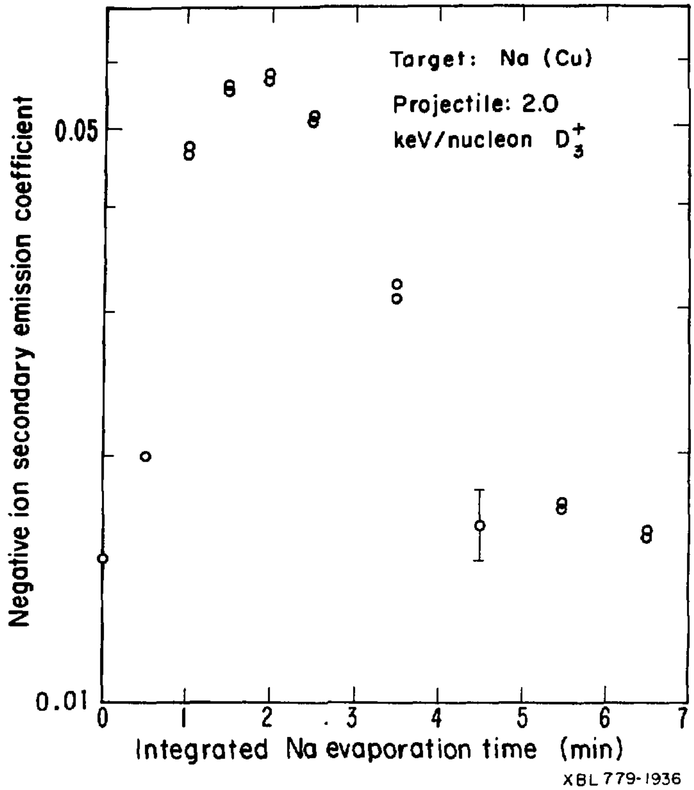

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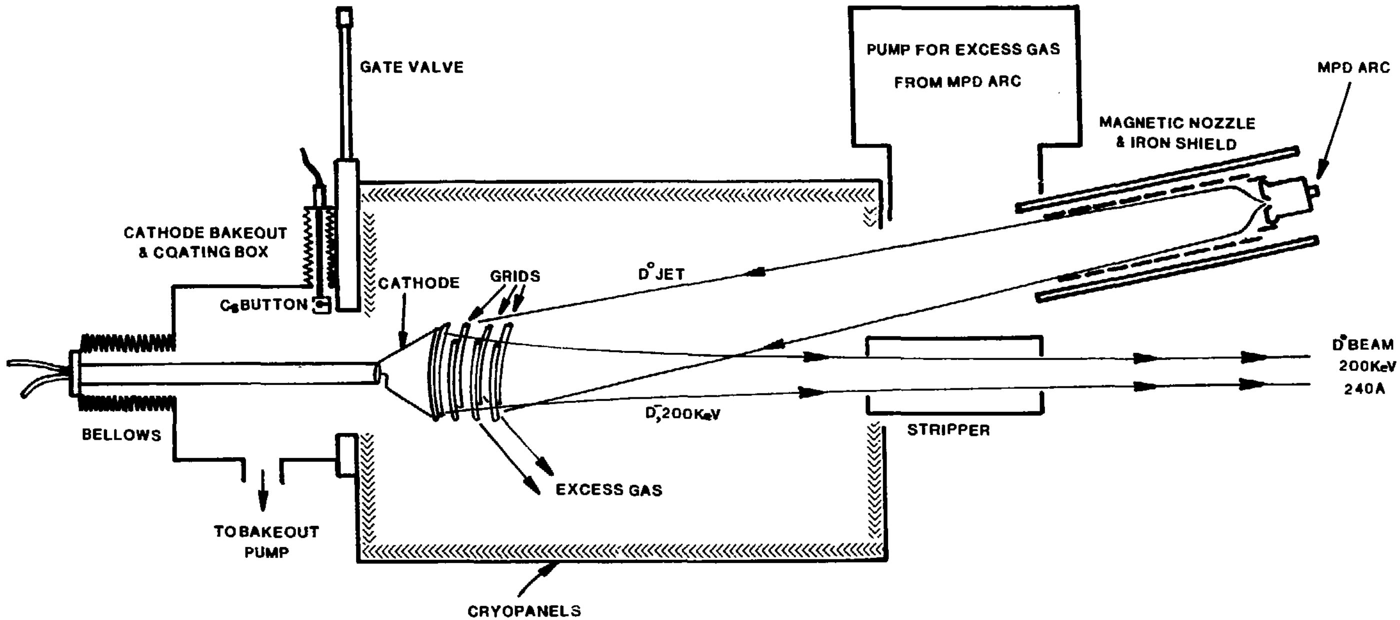

Fig. 11. Proposed neutral beam system based on surface source of negative deuterium ions.

Oral presentation given at Workshop on Plasma Heating Development Requirements, Washington, D.C., 12/5-7.

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BNL HIGH ENERGY NEUTRAL BEAM DEVELOPMENT PLANS *

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Upton, New York 11973

November 1977

ABSTRACT

After having achieved a well defined, one ampere, 120 keV negative ion beam from a magnetron surface plasma source, the BNL Neutral Beam Development Group has entered its second phase development program. This program is aimed at demonstrating the feasibility of a multi megawatt, economical, high energy neutral injector, based upon direct extraction negative ion sources and close coupled or separated accelerator structures. This document details the BNL neutral beam research and development program in the years ahead.

INTRODUCTION

The negative ion beam program at BNL started in 1973 with the development of negative ion sources such as the hollow discharge duoplasmatron and magnetron sources with applications for high energy accelerators. The early success and the promising aspects of the sources to generate very high negative ion beam currents, triggered in 1975 a vigorous BNL neutral beam development program for the next generation of fusion devices based upon intense direct extraction H^- sources. This program added a new dimension to the already existing neutral beam development programs which were based on double electron capture and the well-developed positive ion sources.

In recent years surface plasma (SP) sources, magnetrons and Penning sources, have been developed into intense negative ion sources. Beam currents in excess of one ampere in beam pulses more than 10 msec have been achieved. A 150 kV

*Work performed under the auspices of the U.S. Department of Energy.

accelerator has been constructed and one ampere negative ion beams were recently accelerated to 120 keV. After having obtained these results, the BNL Group entered its second phase development program. In the years ahead, we will concentrate our research to develop quasi-steady state, multiampere negative ion sources, accelerate those beams in close coupled and separated accelerator structures to energies up to 250 keV and neutralize negative ion beams in gas-jets.

SOURCE DEVELOPMENT

Presently there are two types of sources under investigation at Brookhaven: the magnetron and the Penning sources. In either source the discharge is established in hydrogen or deuterium gas with a small admixture of cesium vapor. In both sources negative ions are produced in very high densities on or near the cathodes or anode surfaces. They can be extracted with densities up to as large as 3 A/cm^2 for short beam pulses or they can be extracted at lower densities for longer beam pulses, limited by the heat removal from the cathode and anode. Parameters of present BNL sources are very close to parameters that should be achievable with a cooled source operating in a quasi-steady regime (pulse duration > 1 s). The first objective of our research and development is to design a multiampere H^- (D^-) source capable of yielding 1 A/cm^2 current density at the emission aperture, and operating in a quasi-steady state. This research includes the development of cooling techniques to remove heat (1 to 3 kW/cm^2) from the cathode, studies to measure the effect of the source discharge on gas consumption and other fundamental parameters in plasma sources. This research is essential to optimize the operation of reliable plasma surface sources and to improve the gas and power efficiencies.

Recently we have constructed a versatile 7.5 cm plasma source with forced cathode cooling and independent cesium control. It may operate in two modes, either in the magnetron or Penning mode. With this third generation of SP sources we may expect the extraction of multiampere negative ion beams and long beam pulses. During this fiscal year (1978) we aim to extract at least 2 A, 25 ms negative ion beams at energies around 20 keV. In the next three to four years, the output of the SP sources will be gradually increased to about 5 to 10 A and the pulse length up to about one second, following as closely as possible the goals and milestones as summarized in the first chart.

CHART I

Negative Ion SP Source Development

	Project \ Fiscal Year	FY'78	FY'79	FY'80	FY'81	FY'82
1	Test and extract 2 A, 25 ms beams	———				
2	Design and construct a multiampere, 100 ms source with cooled electrodes	———				
3	Test and extract multiampere 100 ms beams		———	———		
4	Design and construct a multiampere 1 s source		———	———		
5	Test and extract multiampere 1 s beams				———	———
6	Design and construct 10 A, dc sources				———	———

HIGH ENERGY NEUTRAL BEAM FORMATION

In the present BNL scheme the accelerator consists of a 1 A magnetron source, closely coupled to the accelerator. In this arrangement (see Fig. 1), the ions are extracted from the source and accelerated directly across a single accelerating gap to an energy of 120 kV. The beam is then transported to the gas neutralizer

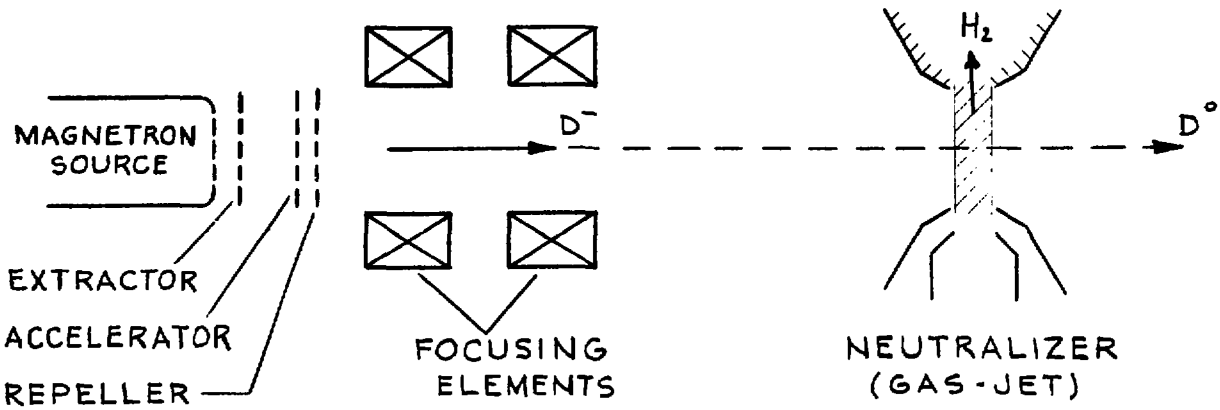


Fig. 1 Close-coupled acceleration

by space charge neutralization and (if necessary) external magnetic focusing. The advantage of this system is its basic simplicity. During FY 1978 this system will be further investigated with and without quadrupoles. In a later stage a CO_2 neutralizer jet will be installed in order to test its operation in the presence of high gradient columns.

An apparent problem of close-coupled acceleration is holding high voltage gradients in high background pressures and eliminate cesium contamination in the accelerator. An alternative is separating the source from the accelerator by a focusing bending magnet. Figure 2 shows a neutral injector based upon this principle. The SP source is mounted on a 120^{o} vertical focusing bending magnet. In the plane perpendicular to the emission slit the dense ion beam initially diverges and the current density decreases until space charge neutralization and focusing by the bending magnet compensate for space charge and the initial

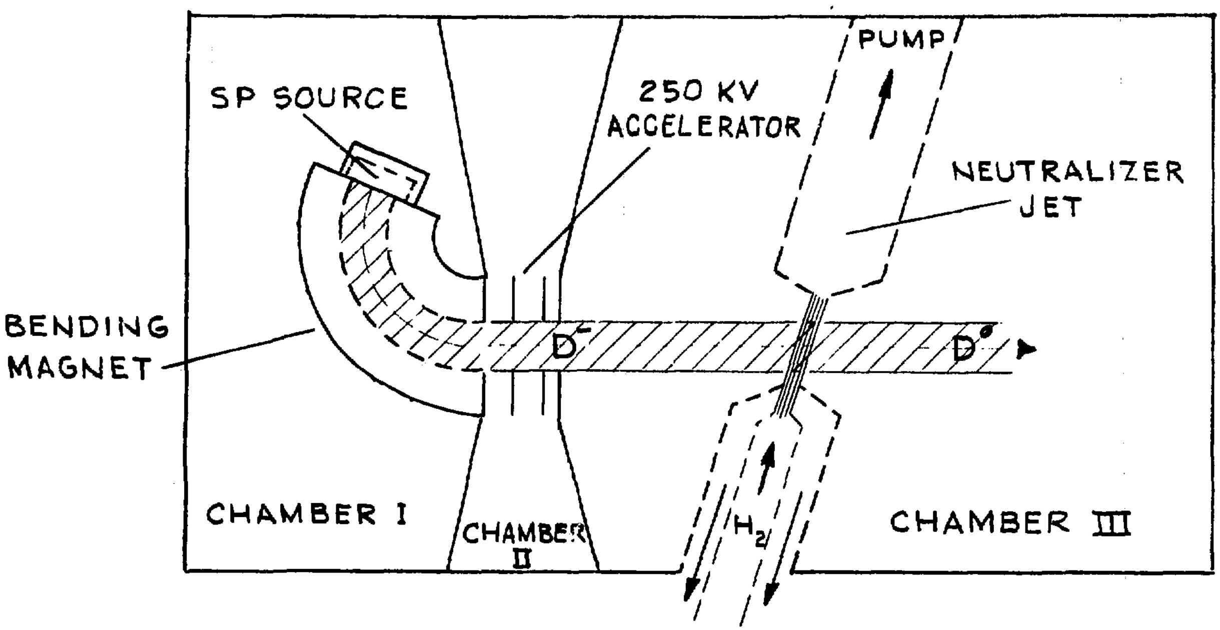


Fig. 2 Separated accelerator neutral beam injector

divergence (see Fig. 3a). In the plane parallel to the slits, there is no significant beam blow-up (Fig. 3b). This method of beam transport to the

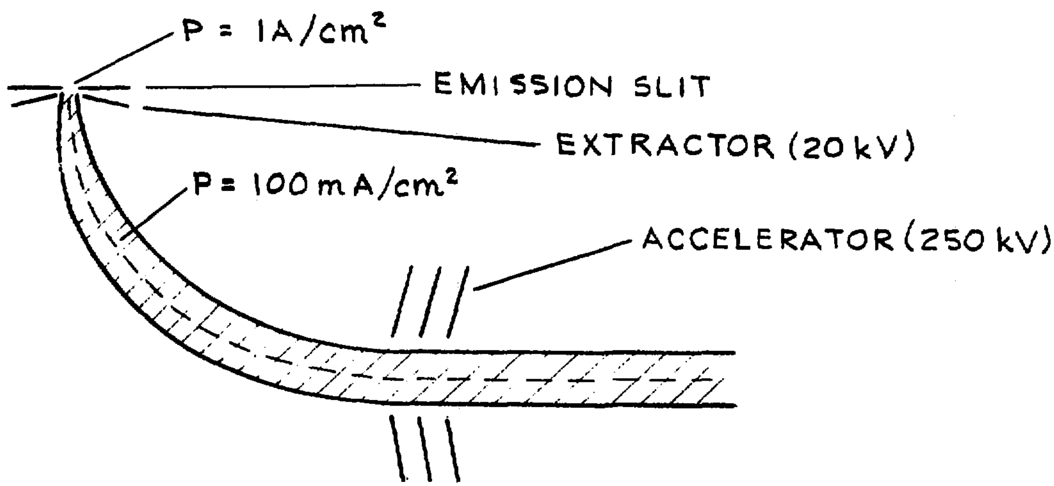


Fig. 3a Beam in the plane ⊥ to emission slit

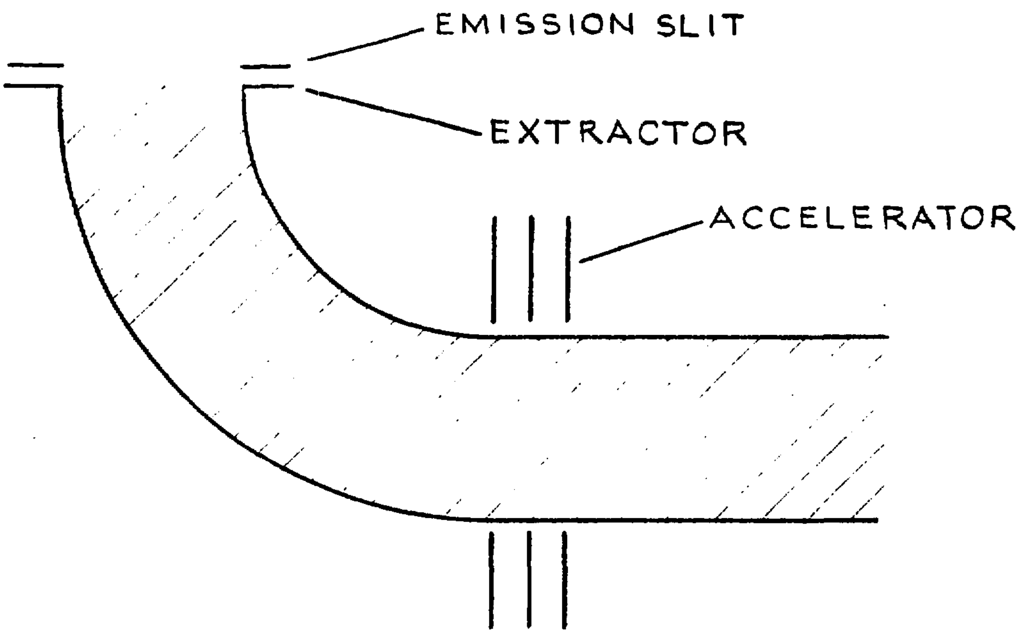


Fig. 3b Beam in the plane || to emission slit

accelerator has been tested successfully in the accelerator laboratories of FNAL, LASL and Novosibirsk for low current (0.1 A), short pulsed (100 μsec), 15 to 25 kV H^- beams. In the suggested scheme for neutral beam injectors (Fig. 2), the ribbon beam injected into the electrostatic accelerator is slightly focused in both planes and the neutralizer jet is located close enough to the accelerator as to avoid additional external focusing means. Each of the three components of this injector (source, accelerator and neutralizer) is located in a separate pumping chamber.

The advantages of this scheme of separating source and accelerator are summarized below.

a. Independent control of the vacuum around the source, the accelerator and the neutralizer.

b. Exclude cesium contamination of the accelerating column.

c. Possibility to apply high current density SP sources.

d. Operate the accelerating column with relatively low current density ($100\ mA/cm^2$) ribbon beams.

e. Possibility to focus extracted beam into the accelerator column and neutralizer jet.

f. Separation of species.

g. No beam losses in the accelerator.

h. No external focusing elements required between accelerator and neutralizer.

In our development program of high current, neutral injectors, we will initially study the focusing properties of a 1 A beam from a single slit magnetron source in à bending magnet, that can be either a U-shaped magnet with pole face windings (Fig. 4a) or a hyperbolic-shaped bending magnet without correction coils (Fig. 4b). The beam will then be injected into an upgraded, high gradient accelerator (250 kV).

FIG. 4a

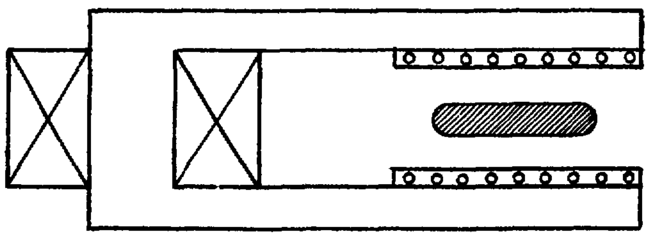
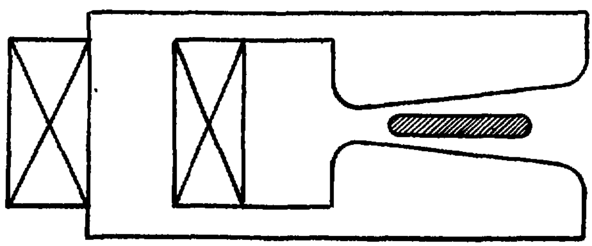


FIG. 4b



The milestones for the negative ion beam formation and neutralization program are summarized in the next chart.

CHART II

Negative Ion Beam Formation and Neutralization

	Project \ Fiscal Year	FY'78	FY'79	FY'80	FY'81	FY'82
1	Test close coupled acceleration and transport of 1 A, 120 kV, 10 ms ion beams	—				
2	Measure gas-flow characteristics of a CO_2 gas jet neutralizer (effusive jets)	—				
3	Upgrade 150 kV, 1 A, 10 ms test stand to 250 kV, 10 A, 10 ms (or 250 kV, 1 A 100 ms)	—	—	—		
4	Neutralization of 1 A, 120 kV, 10 ms negative ion beams with CO_2 gas jet		—			
5	Construct and test bending magnet for a separated acceleration system		—			
6	Acceleration of negative ion beams in a separated acceleration system			—	—	
7	H_2 gas jet tests in close coupled accelerator systems			—		
8	Neutralization of negative ion beams in separated acceleration system and H_2 gas jet				—	—

In conclusion: it appears feasible to assume that, with adequate cooling of the cathode and independent cesium control a reliable multiampere, quasi steady-state negative ion surface plasma source can be realized within three to four years. Although high current negative ion beams have been accelerated in close coupled acceleration, the new development of separating source and accelerator by a bending magnet promises to become practical for high current, high energy neutral injectors. Feasibility studies for high beam currents will take some three to four years depending on the financial support.

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PROGRAM PLAN

LLL/LBL EFFICIENT NEUTRAL BEAM DEVELOPMENT

E. B. Hooper, Jr. and R. V. Pyle

Summary

This material describes our Lawrence Laboratory's negative-ion-based "efficient" neutral beam development plan. Other present closely related programs are not described here, but their developmental work is implicity included, in particular the APP-sponsored programs on theoretical (LLL) and experimental (LBL) negative-ion physics, and on ion-source research (LBL), the D & T (LLL) Direct Recovery and Reactor Design Studies, and the various confinement systems interfaces. Some of this work is contained in other parts of these proceedings.

Aside from the physics and engineering unknowns in scheduling what is still basically a research program, there is no well-defined users' schedule. Our informal poll of possible users shows potential near-term applications of 200- to 400-kV injectors in 1984-85. The program described here deals with the 200-kV application in 1984 and 400-kV injectors on an experiment about five years later. However, we believe that such a program requires funding at a much higher level then is indicated by our present DMFE FY 1978 and FY 1979 guidelines. This workshop should be very useful in helping to formulate realistic plans.

I. Introduction

Present high-current neutral beam systems accelerate positive ions and convert them to neutrals via electron capture and/or dissociation in a gas or vapor. The efficiencies for converting the power in positive ion beams to neutral power are shown in Fig. 1. These curves show rapid efficiency drops, with increasing velocity, for the production of hydrogen atoms with energies above about 30 keV and deuterium atoms above about 60 keV.

Figure 2 shows, as an example, the power flow in a 120-kV TFTR injector with a plausible mixture of atomic and molecular ions; only 30% of the energy goes into the full-energy neutral component. At the higher energies envisioned for future experiments and reactors, say 200 keV and above, it is difficult to imagine practical neutral beam systems for heating and fueling that are based on positive ions, even with systems incorporating electrostatic energy recovery. (For some specialized applications, neutral-beam systems based on positive molecular ions or helium ions may be useful. Recirculation of positive ions through a charge exchange cell, or through several charge exchange cells, also has been proposed as a way to increase efficiency.[1] However, no detailed studies have been carried out.)

From Fig. 1 we see that negative-ion based neutral-beam systems can yield high efficiencies at arbitrarily high energies. Our 1974 document[1] "Neutral Beam Development and Technology: Program Plan and Major Project Proposal" discussed some of the different approaches to high efficiency. It also explains our choice of negative deuterium-

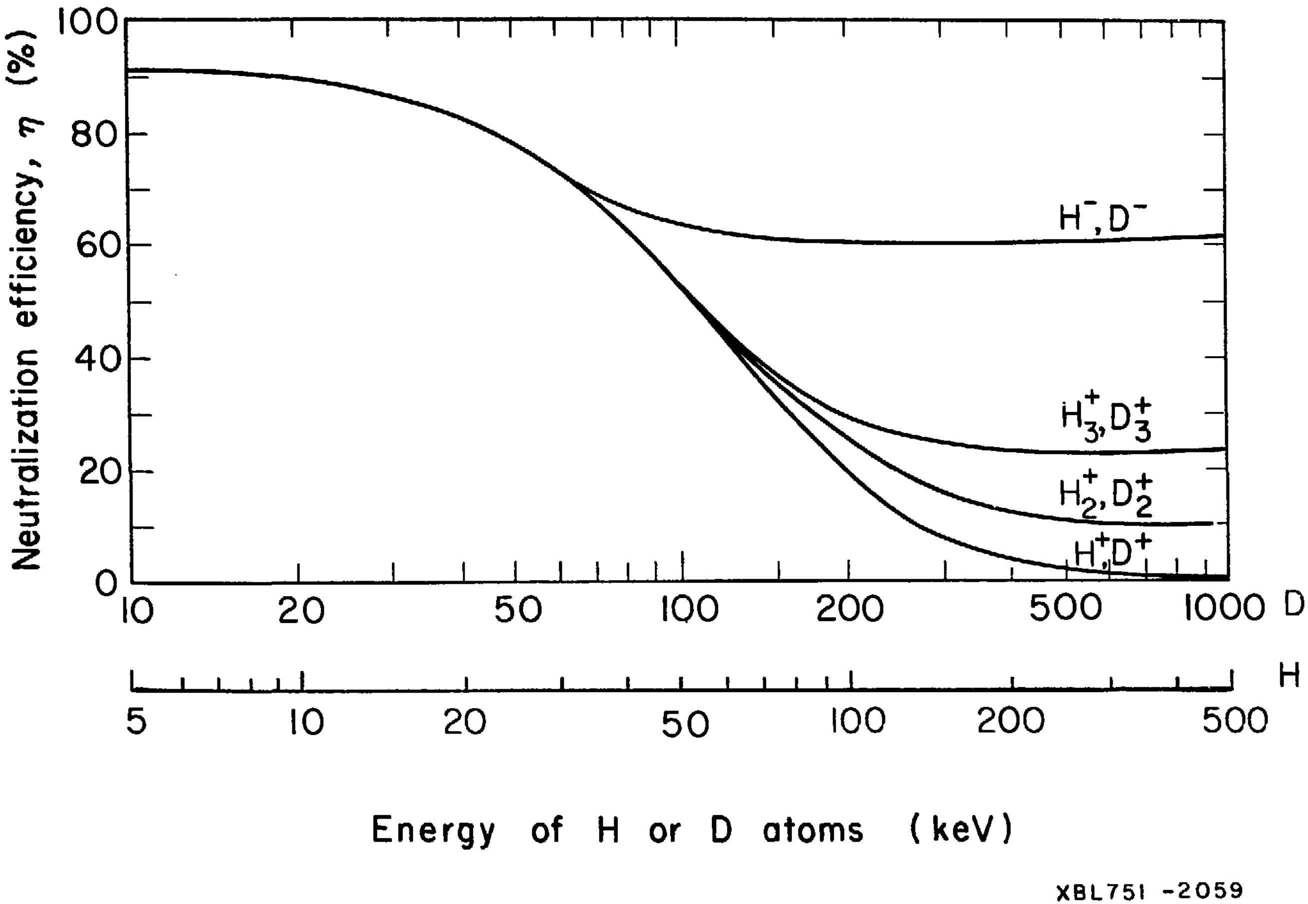


Figure 1. Efficiency of converting ion power to neutral power.

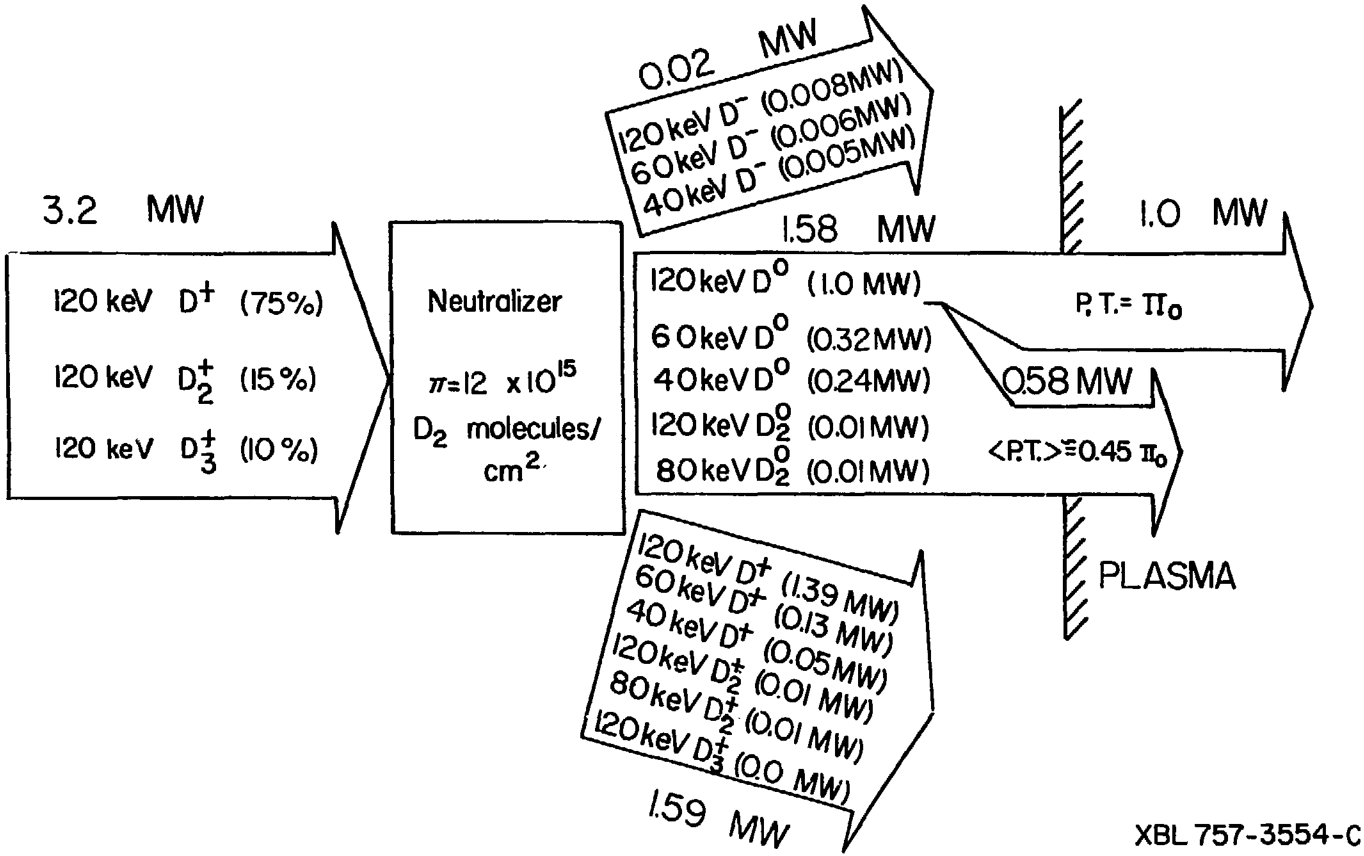


Figure 2. Power flow in a TFTR injector (under development)

ion production by double charge exchange in Cs vapor, followed by dc acceleration to 200 keV and (60%) conversion to 5 A (equivalent) of neutrals in a gas cell, as the first development goal.

In the three years following LLL Prop-115, alternative negative ion scenarios have been tried or proposed at the Lawrence and other Laboratories. The direct-extraction work at Novosibirsk and Brookhaven has been carried farthest, and has some very attractive features. The use of Na vapor as a double-charge-exchange medium at the Kurchatov laboratory also is worthy of consideration. However, we don't know of any reason, at this time, to substantially alter the program laid out in 1974. We estimate that a well-founded decision as to the best way to proceed to useful injectors can not be made for perhaps two years.

Parameters and start-up dates on upgraded and subsequent confinement experiments have not been determined by users. However, for planning purposes we have obtained informal guidance from members of the various confinement laboratories, and from ERDA. The least ambitious program is shown in Table I. Some more recent (and more ambitious) estimates have 300- to 400-kV injectors on a confinement experiment by 1985. On a longer time scale, probably in the 1990's, injectors in the 700 keV to MeV, or higher, energy range may be required. Serious conceptual design studies of such injectors should be started soon for planning purposes.

LLL/LBL D & T NEGATIVE ION GOALS (BASED ON INFORMAL POLL 6/77)

1977	200 kV, 5A, DC CONCEPTUAL DESIGN COMPLETED
1978	BUILD 200 kV, 5A, DC SYSTEM START CONCEPTUAL DESIGNS OF 200 kV, 400 kV, 800 kV, 1200 kV SYSTEMS
1979	COMPLETE CONVERSION FOR 5A SYSTEM START TESTING 200 kV, 5A SYSTEM ~JULY 1979
1980	DEMONSTRATE 200 kV, 5A, LONG PULSE (~DC) SYSTEM
1981	DEMONSTRATE 200 kV, 20A, ≥ 30 SEC.
1982	CONVERT TO 400 kV
1984	200 kV, 20A, ≥ 30 SEC ON EXPT.
1985	FREEZE DESIGN OF 400 kV, 25A, ≥ 30 SEC INJECTORS
1988	FREEZE DESIGN OF ≥ 700 kV INJECTOR
1988-90	OPERATE 400 kV INJECTORS ON CONFINEMENT EXPT.
199?	OPERATE ≥ 700 kV INJECTOR ON CONFINEMENT EXPERIMENT

Table I. Goals based on informal poll 6/77

The building blocks of a negative-ion program are:

- Research on negative ion production, destruction, transport, and neutralization.
- Development of the required technology, including electronics, pumping and diagnostics.
- Design, construction, and operation of test facilities.
- Liaison with users.
- Coordination with ERDA and with other R & D groups.
- Transfer of technology to industry.

The Lawrence Laboratory's program includes all of these items.

A generalized neutral-beam system based on negative ions is shown schematically in Fig. 3. The negative ions may be produced in a plasma, on surfaces, or by double-electron capture in gases and vapors. The first negative-ion-beam production in reflex or diode discharges was by Ehler's et al. at LBL. The extracted current densities were in general less than 0.1 A/cm^2; however, they could operate continuously, as was needed for use with cyclotrons. The Dimov group at Novosibirsk added Cs to small pulsed sources (also intended for accelerator applications), hoping to enhance negative ion production in the plasma. They found instead that the negative ions are produced at the surface of the cathode. Subsequent development of this type of discharge at Novosibirsk and at Brookhaven has produced promising sources operating in the 1 A/cm^2 regime for several milliseconds. If the pulse length and gas efficiency can be improved, this approach may yield the simplest and least costly system for future applications.

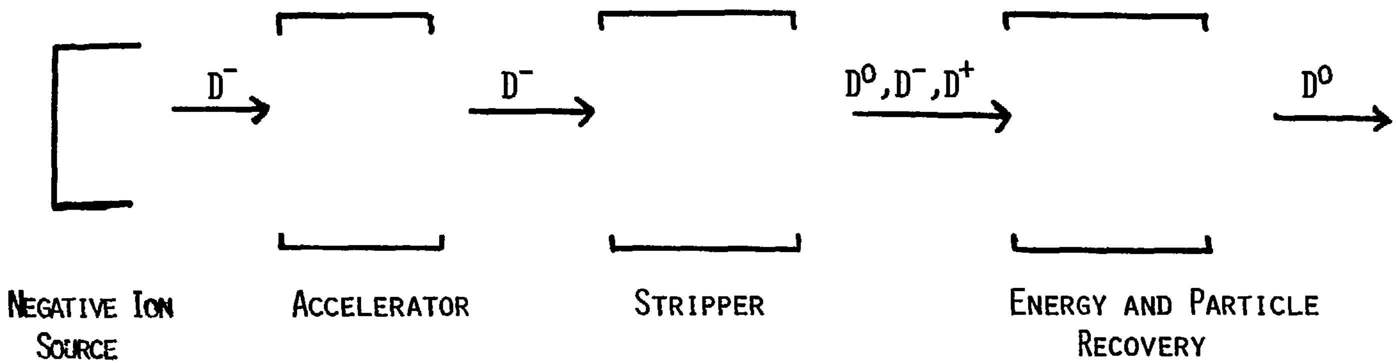


FIG. 3 GENERALIZED NEUTRAL BEAM SYSTEM BASED ON NEGATIVE IONS.

The production of negative hydrogen ions by double charge exchange in cesium or other vapors appears to have a present advantage of being scalable to arbitrarily large currents and long pulse lengths. Because of this, we use this technique as the basis of our near-term development plan. In our current experiments we start with a 1-keV D^+ beam, of which somewhat more than 1 A passes through a cesium charge-exchange cell. The measured conversion efficiency is about 20%, in reasonable agreement with the value of 24% obtained in an atomic collision experiment. The plan is to determine the optimum D^+ energy and current density, considering beam transport, Cs vapor transport, and optical problems.

The accelerator is a single-aperture, five-electrode 100 KV structure designed with the "Wolf" computer code. The first two stages of acceleration are used as focus electrodes to optimize beam optics. The experiments in progress will provide the data for the design of the 200-kV, 10-A (D^-), dc structure.

Possible problem areas include voltage breakdown in the presence of intense beams (and perhaps cesium vapor), and electrode-damage by the stored electrostatic energy if a spark occurs. The problem of keeping the accelerated electron current to a small fraction of the negative-ion current is critical in all negative-ion schemes. Calculations indicate that the equilibrium electron current in a propagating negative-ion beam can be made arbitrarily small. Possible limitations due to beam stability are being studied both experimentally and theoretically. If necessary, electrostatic grids or weak magnetic fields will be used to further reduce the stable equilibrium electric current.

The design of a stripping cell for the high-energy beam has received relatively little attention. Three possibilities have been considered: A vapor cell can yield neutrals with greater than 60% efficiency. The design of such a cell is very similar to the charge-exchange jet presently used in the production of negative ions. For higher efficiencies a highly-ionized plasma cell can be used, and (for maximum efficiency) a conceptual design has been made of a photon stripper using near infra-red lasers.

In any of the stripping schemes some fraction of the final beam is ionized. To maximize power efficiency the energy of the ionized part of the beam must be recovered. Perhaps as important is to control the trajectories of these ions so that they do not end on surfaces which would be damaged. Techniques for recovering these particles are being developed in the LLL "Direct Conversion" program, but their application to intense negative ion beams will require considerable additional work by the neutral beam development groups.

The vacuum system, power supplies, and controls are not shown in Fig. 3, but they are expensive and critical items. Although much of the required technology is being developed in the positive-ion-based program, improvements in reliability, simplicity, and cost are required. Operation at voltages above 200-400 kV may require substantial new technology.

II. Test Facilities

We have several 20-40 kV test stands, and four beam lines for 100- to 200- kV development work (Table II). The Bldg. 442 (LLL) installation is used for the basic double-charge-exchange experiments. The HVTS (LLL)

Facility	Electronics kV	A	S	Vacuum System V(ℓ)	Speed (ℓ/S)
442	100	1	0.01	6,000	15,000 (D_2)
III A	120	20	0.5		
				170,000	60,000 (D_2)
III B	120	80	0.05		
HVTS	80	85	30	20,000	650,000 (D_2)
	120	65	30		
	200	20	DC		

Table II. Test facilities for V $\geq$ 100 kV.

initially will be used for positive-ion source and component tests, and will be modified so that 200-kV negative-ion tests can start in about two years.

The beam lines at LBL, labeled IIIA and IIIB, at present are used in the positive-ion development program, but may be used for negative-ion work if the occasion arises.

We propose that the 400-kV goal will be handled by upgrading the HVTS, i.e. by adding a second 200-kV supply with the opposite polarity. The accelerator will operate from -200-kV to +200-kV, with the stripper and energy recovery unit providing the isolation to ground.

An alternative, or supplementary, supply at lower voltage, $\sim$340-kV, (-170-kV to +170-kV can be obtained by using the two 170-kV, 17A, dc supplies at LBL that will be shared with the TFTR prototype program for the next few years. According to the present plan, it would be necessary to construct a new vacuum chamber, and perhaps new radiation shielding, if deuterium operation is required.

We are unable to comment in detail on the design, schedule, and problems of 700-kV to >1-MV test facilities until a serious conceptual design study has been carried out, although rough schedule and cost estimates have been made. It is quite likely that the size and cost will be such that a separate prototype or full-scale test facility will not be built; instead, construction and testing may be done only at the site of, and as part of, a confinement experiment. However, we include the possibility of building a 700-kV to 1200-kV test facility, pending resolution of this topic.

III. Schedule

In Figure 4 we show a plan for efficient beam development based on Table II, but delayed somewhat to conform with actual FY 78 funding and proposed FY 79 funding. Funding as needed is assumed for FY 80 and beyond.

As can be seen, the lead time required for design and construction of sources to go on experiments makes the development schedule much tighter than one might expect.

An important point to note for the overall beam program is that the HVTS becomes almost totally devoted to the negative-ion program in about two years.

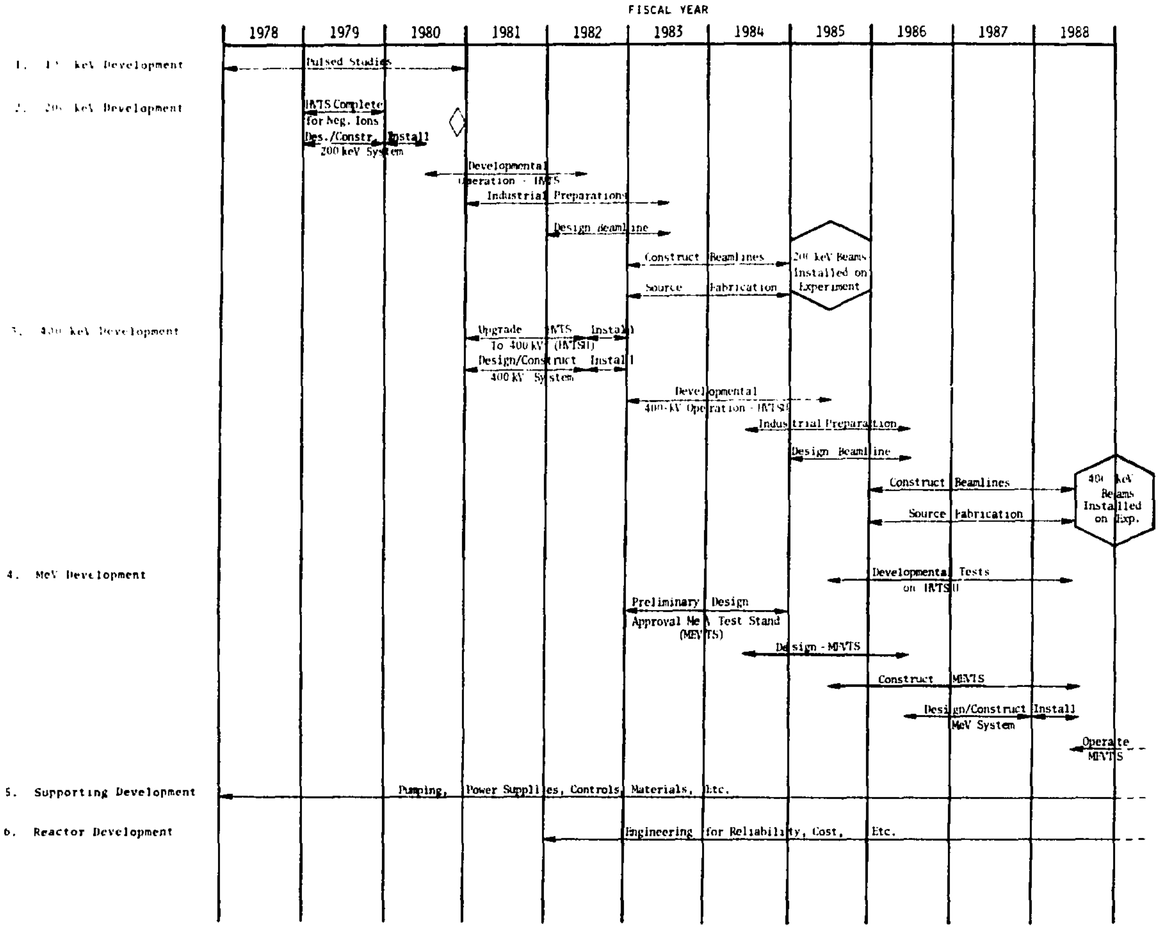
IV. Manpower

Manpower estimates for subsequent years are available, but are not included here. We note that a large amount of neutral-beam-trained engineering manpower in principle becomes available in a year or two as the design and construction phases of the HVTS and DIII and TFTR prototype beamlines change to the de-bug and operation phase.

V. Funding

Because of the large programmatic and R and D uncertainties, we do not include our estimate of the cost of carrying out the development program shown in Fig. 4.

EFFICIENT BEAM DEVELOPMENT - PLAN B



Decision point as to whether to stick with the double-electron-capture approach or switch to an alternate negative-ion production scheme.

Figure 4. Sample development plan

Reference

1. "Neutral Beam Development and Technology: Program Plan and Major Project Proposal," LLL Prop-115 (September 1974).

ALKALI DEUTERIDE NEGATIVE ION SOURCE

DEVELOPMENT PLAN

Prepared for

DoE, Division of Magnetic Fusion Energy

by

TRW Defense and Space Systems Group

Redondo Beach, California 90278

December 6, 1977

Introduction.

TRW is developing a negative deuterium source for high energy neutral beam lines. The scheme employed is based on direct plasma production from solid sodium deuteride by impacting its surface with a short intense pulse of energy. Figure 1. shows a schematic view of an embodiment of this source currently under investigation. NaD (or NaH) is suspended in the vacuum chamber and its surface is irradiated by a laser pulse guided into the vacuum chamber by steering optics. Surface layers of sodium deuteride are evaporated and ionized, producing a multispecies plasma which expands into the vacuum and fills a confining system which is directly coupled to the accelerator extraction system. Measurements of the quantity of negative ions produced per laser pulse and the efficiency of their production indicate that this type of source is attractive for the development of high energy neutral beam systems.

In this report we discuss a three phase program for the development of such a system. In the first phase, concluded in May, 1977, the laser initiated source was characterized. In phase two, scheduled for completion in September, 1978, negative ion confinement and extraction are investigated using laser energy deposition as a baseline method to produce D^- . In addition other energy deposition schemes are studied in order to define a baseline energetic beam source system. The third phase is devoted to producing an integrated baseline system and scaling it up in current and energy to meet magnetic confinement system requirements.

Characterization of Single Laser Pulse Source.

In Figure 2. the parameters characterizing H^- and D^- sources are summarized. Positive and negative ion abundances were determined with a quadrupole mass analyzer. Electron temperature was measured with rapidly swept Langmuir probes. Negative (H^-) ion temperature was obtained using the mass analyzer and focusing grids to provide electrostatic energy resolution. The evolution of the electron density near the target surface, the total number of ions produced and the efficiency of the process for H^- and D^- generation was obtained by analysis of double pulse optical holograms of the target plasma region. Details of the source characterization experiments can be found in references (1), (2).

A favorable property of the laser produced plasma as far as the negative ion extraction is concerned is relatively low electron and ion temperature ($T_e \sim 0.4$, $T_i \sim 0.04$eV) and high D^- drift velocity. Other important favorable features are the efficiency of the negative ion production and a very low neutral background pressure.

Planned Continuation:

A plan for the second phase of the program, which is currently underway, is shown in Figure 3. We have begun to test a variety of single pulse extraction and electron elimination methods in the apparatus shown in Figure 1. Installation of a 600Hz repetition rate CO_2 laser is planned for January, 1978. Since this rate is insufficient to produce a constant density incident on the extraction grids (required for optimum extractor performance) we are currently designing a multidipole surface confinement chamber ("MacKenzie Bucket") to smooth the flow. To increase the duty cycle from the expected 6% (the plasma evolution time is 100 microseconds), as well as to reduce density temporal fluctuations if the confinement is ineffective, a rep.-rate approaching 10kHz would be required. Although no such laser is currently commercially available a supplier has estimated that it could be developed within two years at a relatively modest cost. A potentially more desirable approach, particularly for the high powers required for a high current source, is to utilize a more readily developable high repetition rate energy deposition method. One possibility, an electron beam pulse, has been studied at low intensity and shown to be relatively inefficient ($\sim$ 40 keV/H^- vs. $\sim$ 1.4 keV/H^- for the laser). We currently plan to investigate its efficiency at higher intensity, as well as the merits of other alternate methods such as laser initiated and ablative arcs. At the end of phase 2 a selection of a baseline system consisting of an energy deposition method and a confinement-extraction configuration will be made.

A nominal FY'82 objective of developing a 50A, 200 keV, CW beam has been selected for phase 3. The first milestone is a 50 keV, 1A, 50 msec. beam line to be developed on a TRW test stand in FY'79. The beam line would be scaled to 200 keV on the Lawrence Livermore Laboratories test stand while a higher performance (e.g., 5A and 100 msec.) 50 keV beam line was developed at TRW. This process would continue until the objective, or some limitation, were reached. A depiction of beam line performance progression, for certain funding and "goodness of physics" assumptions, is given in Figure 4.

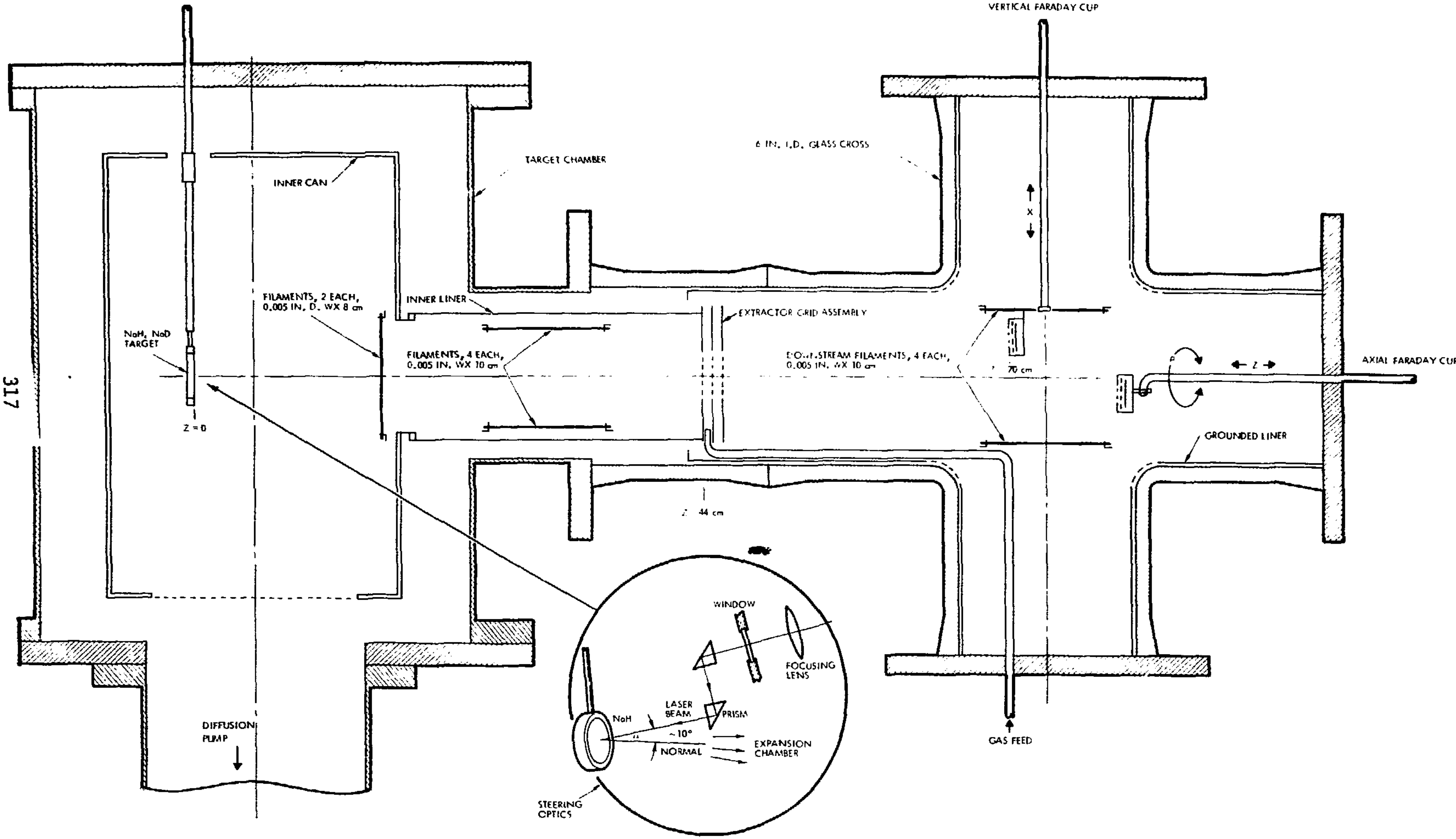


FIGURE 1. SCHEMATIC VIEW OF D⁻ SOURCE.

FIGURE 2. H^-, D^- SOURCE PARAMETERS (ENERGY DEPOSITED BY A LASER PULSE)

Abundance Measurements In Low Density Region ($N_e \sim 10^{10} - 10^{12}$ cm^{-3})	$H^-/H^+ \sim 0.2$, Relative abundance is independent of incident Laser Energy	$D^-/D^+ \sim 0.2$,	Very small N_a^+ Content
Temperature Measurements In Low Density Region	$T_e \sim 0.4$ eV,	$T_{H^-} \sim 0.04$ eV,	Plasma Drift Velocity $\sim 3.10^6$ cm sec $^{-1}$
Particle Production Rate 20% in H^-	3.10^{18} cm^{-2}/pulse, $E_{Laser} = 1.3$ Joule		
Total Number of Particles Produced per Pulse	$\sim 3 \times 10^{16}$, Slight Dependance on the laser energy		
Energy Spent Per H^- or D^- Ion	1.4 KeV/ H^- ion		

Ref: (1) Vanek, Gekelman, Wong, Proceedings of First International Conference on the Production and Neutralization of Negative Hydrogen Ions and Beams, Brookhaven September 1977.

(2) Gekelman, Vanek, Wong, Journal of Applied Physics, to be published.

Figure 3. D^- Beam Development Program Plan FY 78

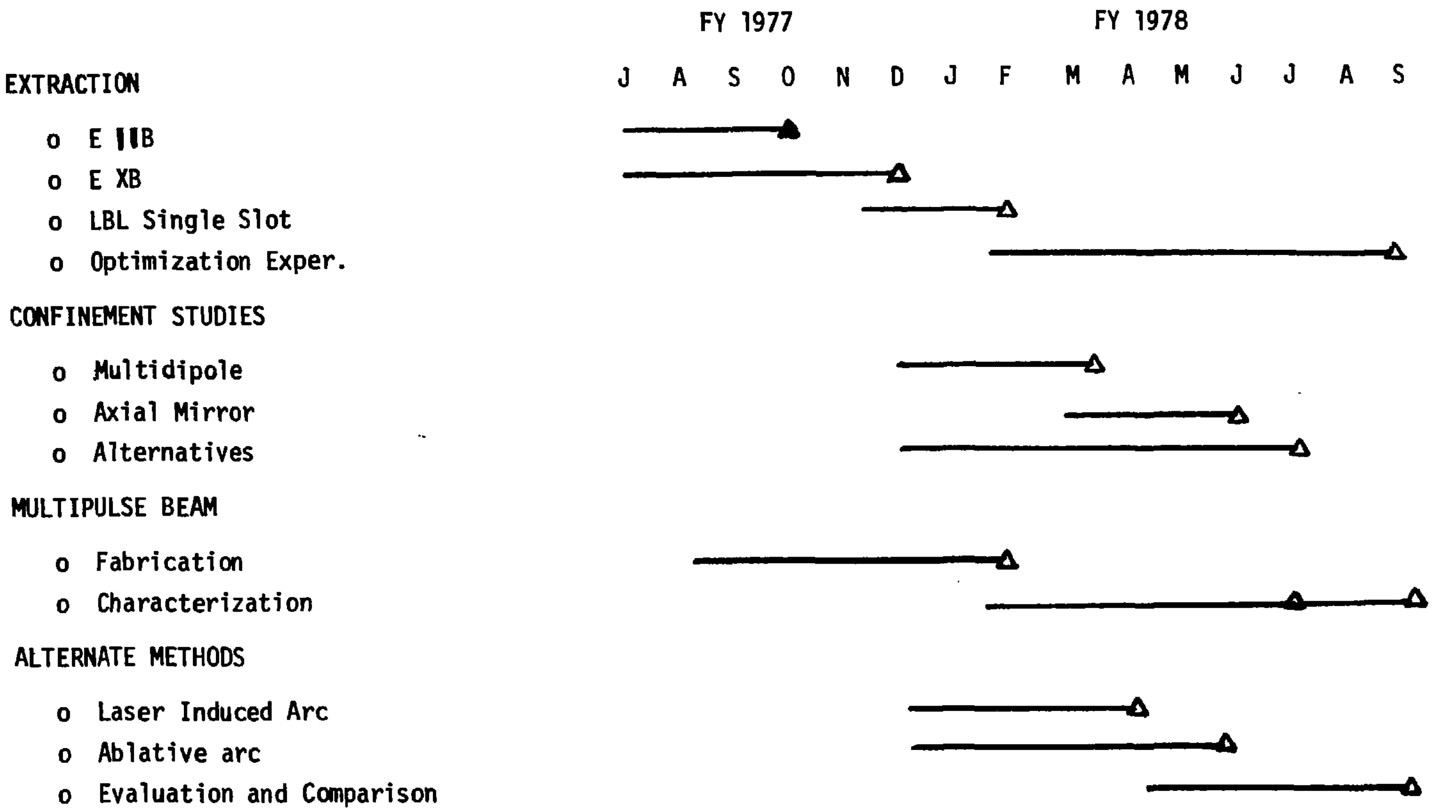
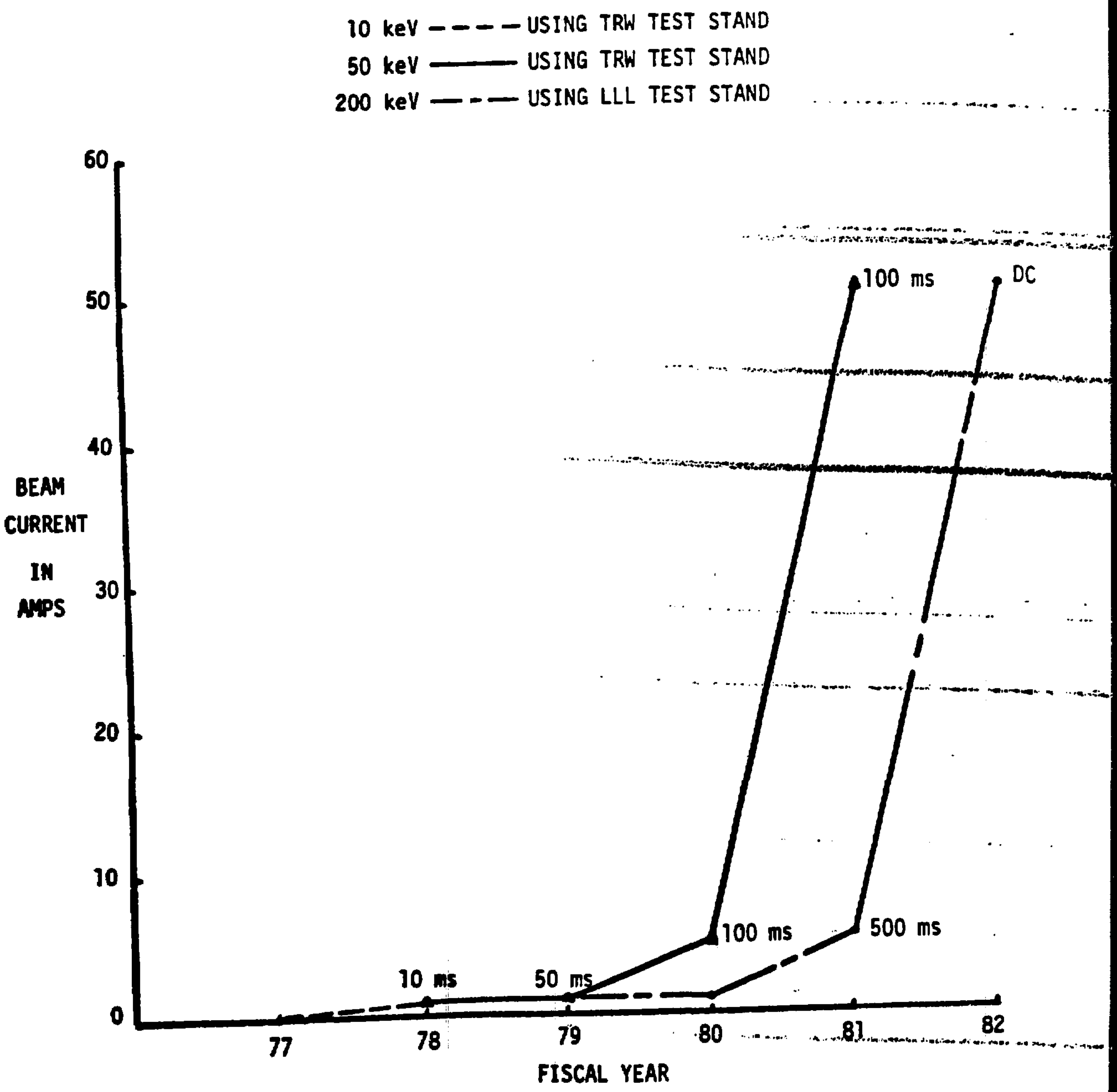


Figure 4: TECHNICAL GOALS



The ORNL Negative Ion Program*

W. L. Stirling

The ORNL Negative Ion Program is interested in developing small-scale or bench type experiments which are inexpensive and are designed to test principles involved in negative ion plasma generator development. As shown in Fig. 1, there are four basic ideas at which we are now looking. Based on our experimental findings, we will select one or two of these ideas for further development into a negative ion source. These four ideas are tabulated in Fig. 2: (1) H^- production from H^+ ions in a modified duoPIGatron; (2) double charge exchange production of H^- ions utilizing the double sheath of a duoPIGatron; (3) surface generation of negative ions from thermally dissociated hydrogen gas; (4) calutron or Penning discharge.

To understand the first idea, the production of H^- from H^+ ions in a modified duoPIGatron, we can look at Fig. 3, where there is a schematic of the modified duoPIGatron. This source consists of a MacKenzie type bucket in the region labeled PIG plasma; the bucket consists of a line cusp magnetic confinement system illustrated in section A-A on the right. The electron feed comes from the cathode plasma, which is at the top of the figure; this feed system is a development of the old duoplasmatron source. Electrons are accelerated from the cathode plasma and become the ionizing agent in the PIG or anode plasma at the bottom. An important region in the source is that between the cathode and the PIG plasma in which an electrostatic double sheath exists. (The second idea, above, is based on the existence of this double sheath.)

*Research sponsored by the Department of Energy under contract with Union Carbide Corporation.

Turning now to the PIG or anode plasma in the modified duoPIGatron, consider the region right above the target cathode or plasma electrode. This anode plasma has a density of 2-3 x 10^{12} cm^3. In this region, directly above the grids, we place a system of chevrons or vanes on which the positive ions impinge. These vanes are covered with cesium; therefore, the positive ions are converted to negative ions on these surfaces. The negative ions proceed on towards the grids, where extraction will take place. Figure 4 shows a simplified layout of this principle. There are two grids shown with a voltage supply V_E connected; this is the extraction voltage. At the top we see a region labeled plasma. In between the grids and this plasma is the cesiated surface or collection of vanes on which the positive ions impinge. The cesium oven is not labeled in the diagram but is in the vicinity of the arrows where the H^+ ions pass.

Figure 5 shows a better view of the vanes on which the cesium is deposited. Figure 6 shows another view and depicts how the magnetic field is set up in the direction which is parallel to the vanes. This direction is perpendicular to the flow of the plasma through the vanes. The magnetic field is used as a means of retarding the electron flow from the plasma into the grid region from which we extract the negative ions. In addition to the magnetic field for retarding the electron flow we also have a bias voltage labeled V_c in Fig. 4. This is a means of electrostatically retarding the electron flow from the cesiated surface to the grids.

A great deal of data has been taken with this ion source or plasma generator. Figure 7 shows two curves; one is a variation of extraction current with extraction voltage, and the other is a variation of the Faraday cup current with extraction voltage. Consider the extraction current curve. For a negative extraction voltage a small ion current flows in the system, and for a positive extraction voltage we see a much larger negative current (either negative ions or electrons, yet to be determined) flowing in the system. The ratio of positive to negative current seen by the Faraday cup is not in agreement with the ratio determined from the extraction current signal. We feel that this is explainable by the very high destruction rate of the H^- ions, which become either neutrals or positive ions at this very low energy. If they become positive ions, they reduce the negative signal and contribute to the positive ion current. Calculations show that the reduction in the H^- signal could be a factor of 30 to 40; therefore, the overall conclusion from these results is encouraging.

Figure 8 shows the variation of the extraction current and the Faraday cup current signals as a function of the biasing potential of the convertor. Even though the magnetic field in the convertor region should stop the electrons coming out of the plasma above, it is possible that secondary electrons will be emitted from the convertor surfaces themselves as the positive ions impinge upon these surface. There is no way the magnetic field could stop the secondary electrons that are produced at the bottom of the convertor or surface, and therefore, we use the biasing potential to retard them. Simultaneously, the negative ions are also slowed down, and thus both the Faraday cup current and the extraction current decrease as the biasing potential of the convertor is increased positively. The Faraday cup current signal is not shown in this figure, but the ratio of the Faraday cup current to the extraction current is shown. This ratio is constant as the biasing potential is increased above 10-15 V. We interpret this as meaning that the electrons proceeding from the convertor surface are all turned around above a biasing potential of 10-15 V, and the signal then is all (or most likely all) H^- ions.

Using this ratio as a normalizing factor, we can go back and correct the extractor current. We see a curve of I_E', which is the derived H^- current or the actual H^- ion current extracted. The difference between curves I_E and I_E' indicates the presence of electrons in the ion beam whenever the bias potential is lower than 10-15 V. Hence, it is possible to suppress electrons in the extracted beam by a combination of electrostatic and magnetic methods. Thus, the data from both Figs. 7 and 8 indicate the presence of negative ions. We have not as yet determined what the ratio of electrons to negative ions is in the extracted beam.

Figure 9 gives the performance of the modified duoPIGatron for H^+ ions and the estimate for H^- ions. The positive ion plasma density should be the same in the two cases. In fact, when operating as a negative ion source, we can increase the density above the positive ion density perveance limit. The density increase is permissible due to the loss conversion efficiency of positive ions to negative ions. We expect the plasma uniformity to remain the same for both positive and negative ions. The arc should typically run at the same values of voltage and current; the beam current density will certainly be different for H^+ and H^- ions depending upon the conversion efficiency of the positive ions to negative ions on the cesium surfaces. We have run as high as 400 mA/cm^2 of positive ions. We plot two values for the negative ions depending upon the conversion efficiency; these two values are for 10% or 30% conversion efficiency. The beam currents are then reduced correspondingly from 70 A of positive ions to 7 A (perhaps 21 A) of H^- ions.

These efficiency values of 10% or 30% affect the overall arc efficiency and gas efficiency of the system. We think that the pulse length can be maintained at the same value (up to 500 msec). These estimates are for a PLT type source which is a 22-cm-diam grid system. All of our work at this point has been with a 10-cm-diam source, which should be easily scalable to the larger sources when the final parameters of the convertor system are worked out.

The second idea is similar to the double charge exchange method of producing H^- ions. As mentioned earlier, there is an electrostatic double sheath existing between the cathode and anode plasma. Electrons are emitted from the cathode plasma and pass into the anode plasma region. To maintain the stability of this electrostatic sheath, the Bohm criterion states that there is a certain fraction of ions going back into the cathode plasma. Looking at Fig. 10, we can relate the cesium plasma in this figure to the cathode plasma existing in the modified duoPIGatron from which electrons proceed into the hydrogen plasma. The hydrogen plasma then becomes an anode or PIG plasma of the modified duoPIGatron. Thus, low energy positive ions come from the hydrogen plasma and proceed into the cesium plasma, where they are converted to negative ions. The voltage of the positive hydrogen ions can be adjusted by the gas pressure existing in this device. We run typically in the modified duoPIGatron up to 80-90 V drop across the double sheath. If desired we can also add the vanes or convertor as shown in the bottom of this figure to make sure that the positive ions do indeed have an opportunity to be coverted into negative ions by striking the cesiated surface of the convertor. Some of the expected parameters are also listed in Fig. 10. This idea has not yet been incorporated into an active experiment.

The third idea, utilizing surface generation of negative ions from thermally dissociated hydrogen gas, can be examined in Fig. 11. At the top of the figure are two small circles which represent tubes in which the deuterium or hydrogen gas is fed and thermally dissociated. Small holes, not shown, permit this thermally dissociated gas to proceed towards the cesium coated tungsten tubes which are the larger circles at the top of the figure. These H or D atoms then strike the cesiated tungsten tubes and some fraction are converted to negative ions. The negative ions are extracted by the same system of tubes, which form an extraction electrode system. Beneath the tubes is an electron shield to reduce electron drain. A final acceleration potential is applied to the negative ions coming through the electron shield. Figure 12 gives some of the pertinent data of this system and the expected performance. The energy of the hydrogen atoms must be above 0.75 eV, according to theoretical calculations, in order to produce a negative ion. Since the hydrogen or deuterium atoms will thermalize to the temperature of the tube, which is about 0.2 eV, only 10% of the tail of the atom distribution will have sufficient energy (i.e., energy greater than 0.75 eV) when striking the cesiated tungsten surface. Other factors, such as the overall gas efficiency of 1%, are also listed. The atomic flux on the cesiated tungsten surface is estimated to be between 1-10 mA/cm^2 for the initial experiments in which two 1-mm holes are employed. This experiment has been assembled and installed in a vacuum chamber and work is now proceeding.

The fourth idea is one that employs a Penning discharge, which exists in a calutron ion source. An exploded view of the source as we have modified it is shown in Fig. 13. A heated filament emits electrons parallel to the magnetic field across the rectangular box or arc chamber. Gas is injected into the filament region and/or arc chamber. The electrons are reflected at the other end of the chamber and oscillate back and forth in a typical Penning discharge fashion. Positive ions created in this discharge can then be accelerated into the cesiated surface. Cesium is released from the oven at the back of the device. Negative ions are accelerated through the grids to a collector not shown in this figure. As part of the mounting bracket we have an electron dump, which is necessary for draining off electrons originating in either the arc discharge or accelerating gap. Figure 14 shows the electrical connections of the components of the discharge. The filament is heated by the filament supply V_F, the arc voltage supply V_A is used to supply the arc power necessary to run the discharge, and V_E is a supply which is used to attract positive ions into the cesium coated molybdenum surface. Extraction of the negative ions is accomplished by the supply of V_B. In this system, we have decoupled the filament heating requirement from the arc voltage requirement necessary to run the discharge. Since we have the supply V_E to accelerate positive ions into the cesiated surface, we can vary this bombardment energy independently of the arc voltage and thus determine the optimum value for negative ion production. The cesium itself is emitted from the cesium oven at a rate determined by the oven temperature, and it too is independent of the arc parameters or the voltage used to accelerate the positive ions into the cesiated surface. By dividing the discharge into two vertical ribbons, it is possible to extract the negative ions produced on the cesiated surface without having them pass through the main portion of the discharge itself; therefore, we can minimize the destruction of negative ions in the discharge. This source has been assembled and voltage tested. The arc discharge has been run at low power. The source is now ready for full-scale testing.

In summary, our goals are to take these small-scale experiments and examine them in sufficient detail to determine which ones are appropriate for enlarging into a full-scale negative ion source. We feel we should complete this preliminary examination by the end of this fiscal year.

ORNL NEGATIVE ION PROGRAM

TO INVESTIGATE METHODS OF NEGATIVE ION PRODUCTION WITH SMALL SCALE EXPERIMENTS.

HAVE FOUR IDEAS FROM WHICH ONE OR TWO WILL BE CHOSEN FOR DEVELOPMENT OF A NEGATIVE ION SOURCE.

Figure 1

NEGATIVE ION SOURCE DEVELOPMENT

1) H^- PRODUCTION FROM H^+ IONS IN MODIFIED DUOPIGATRON.

2) DOUBLE CHARGE EXCHANGE PRODUCTION OF H^- UTILIZING DOUBLE SHEATH OF DUOPIGATRON.

3) SURFACE GENERERATION OF NEGATIVE IONS FROM THERMALLY DISSOCIATED HYDROGEN GAS.

4) CALUTRON OR PENNING DISCHARGE.

Figure 2

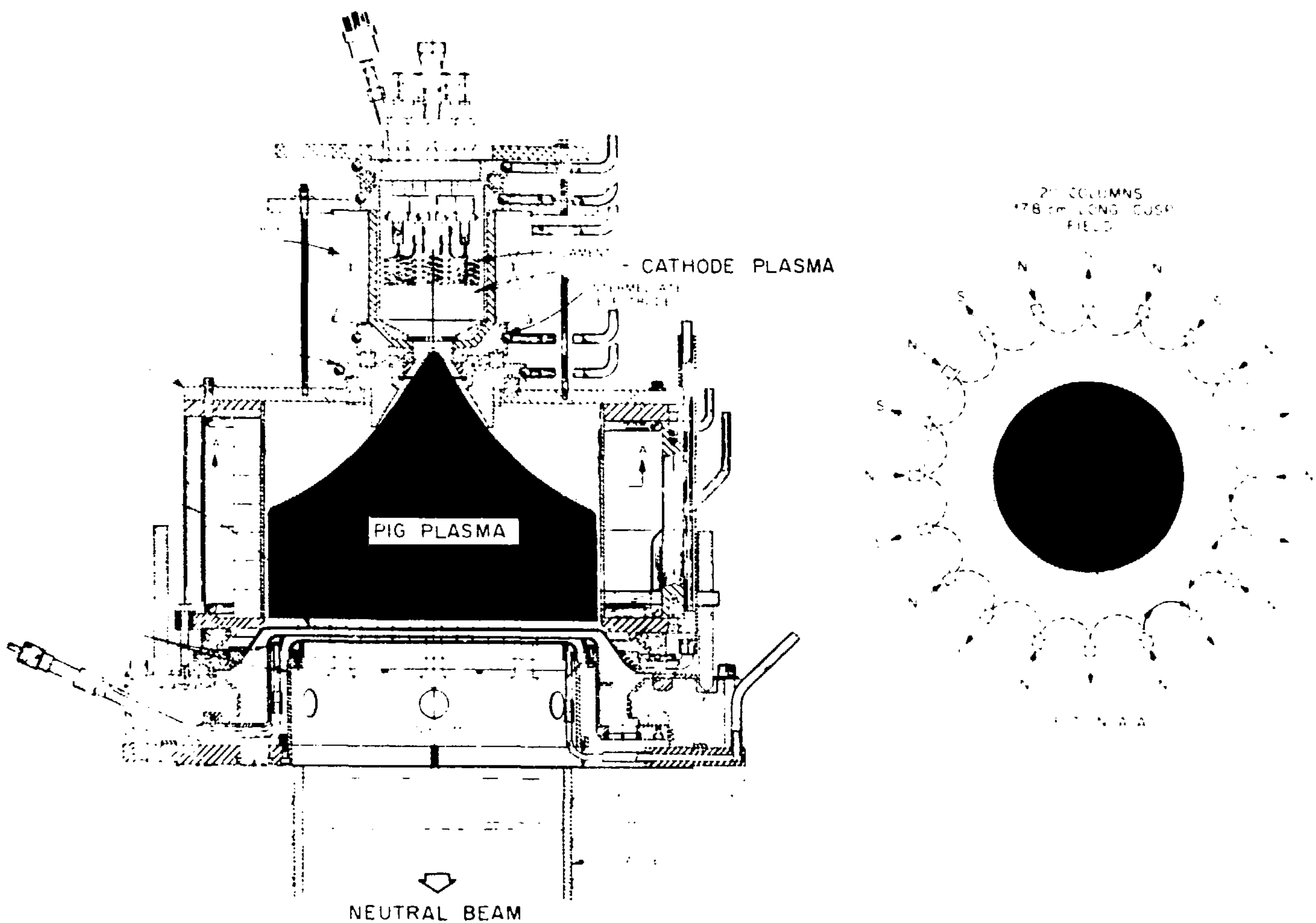


Figure 3

ORNL/DWG/FED-771004

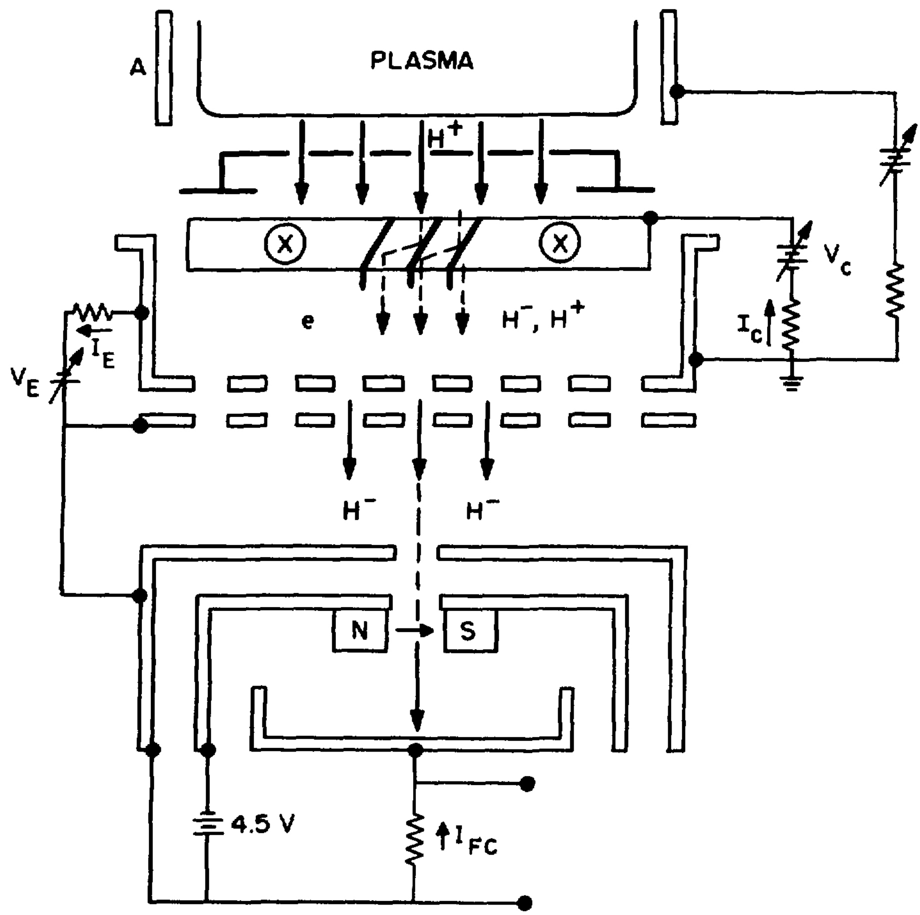


Figure 4

ORNL/DWG/FED-77569R

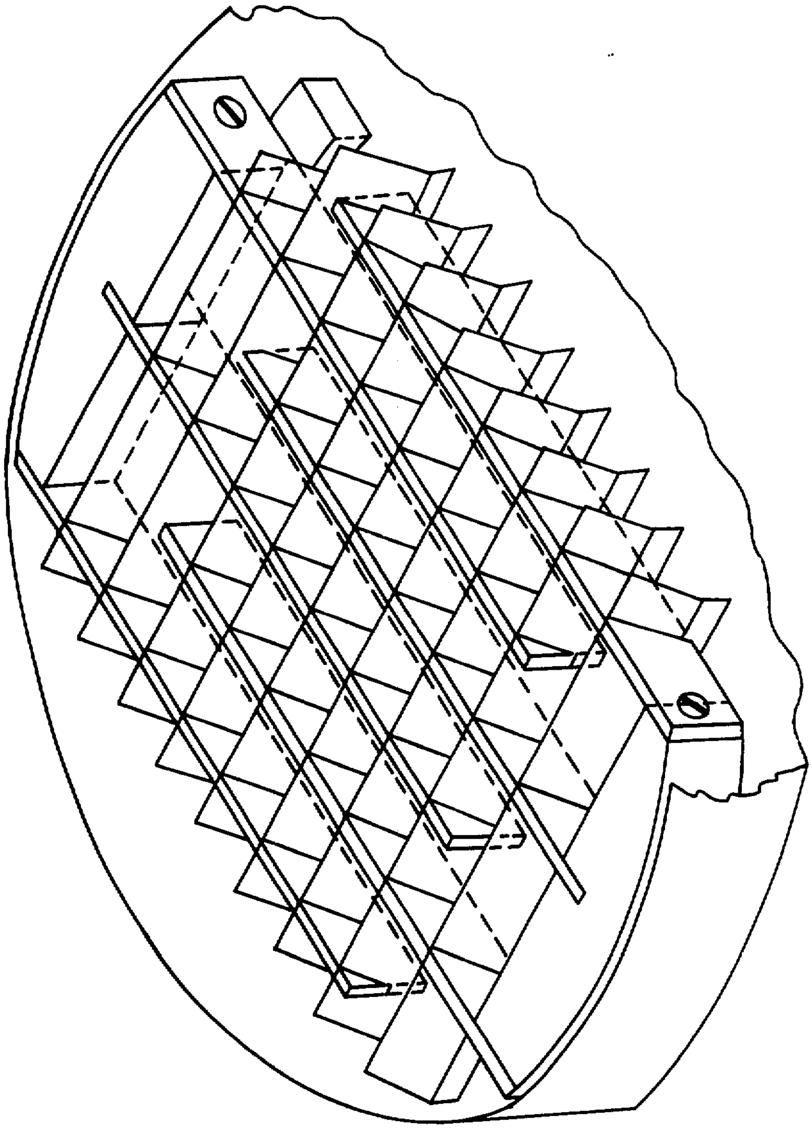


Figure 5

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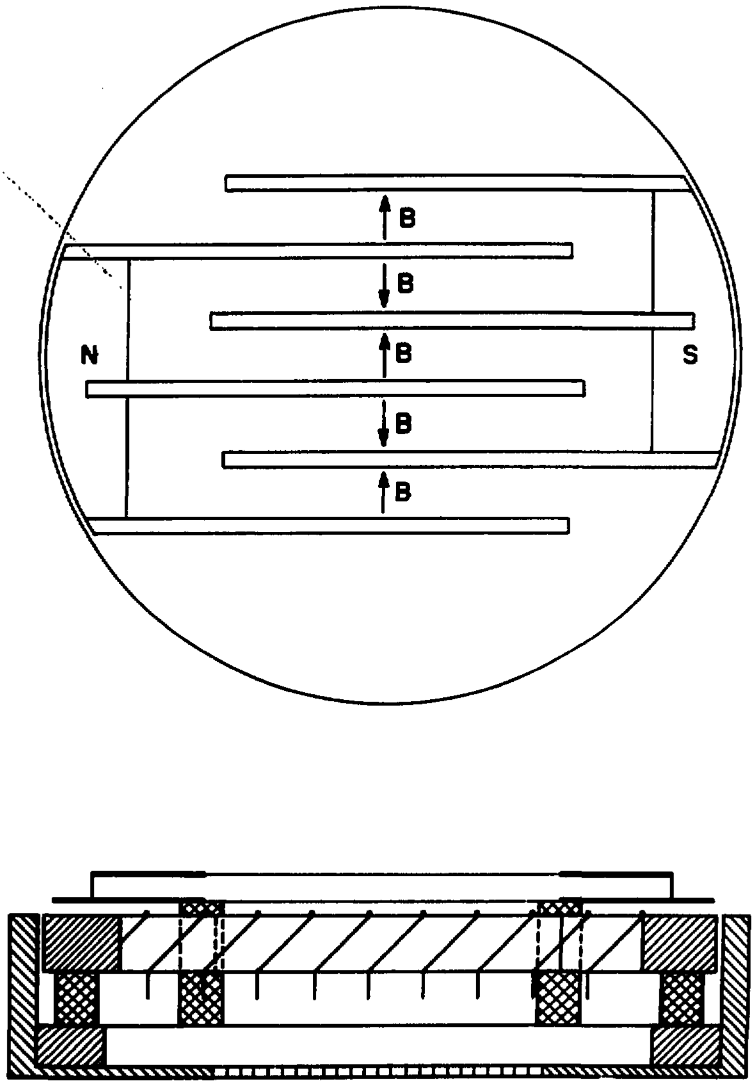
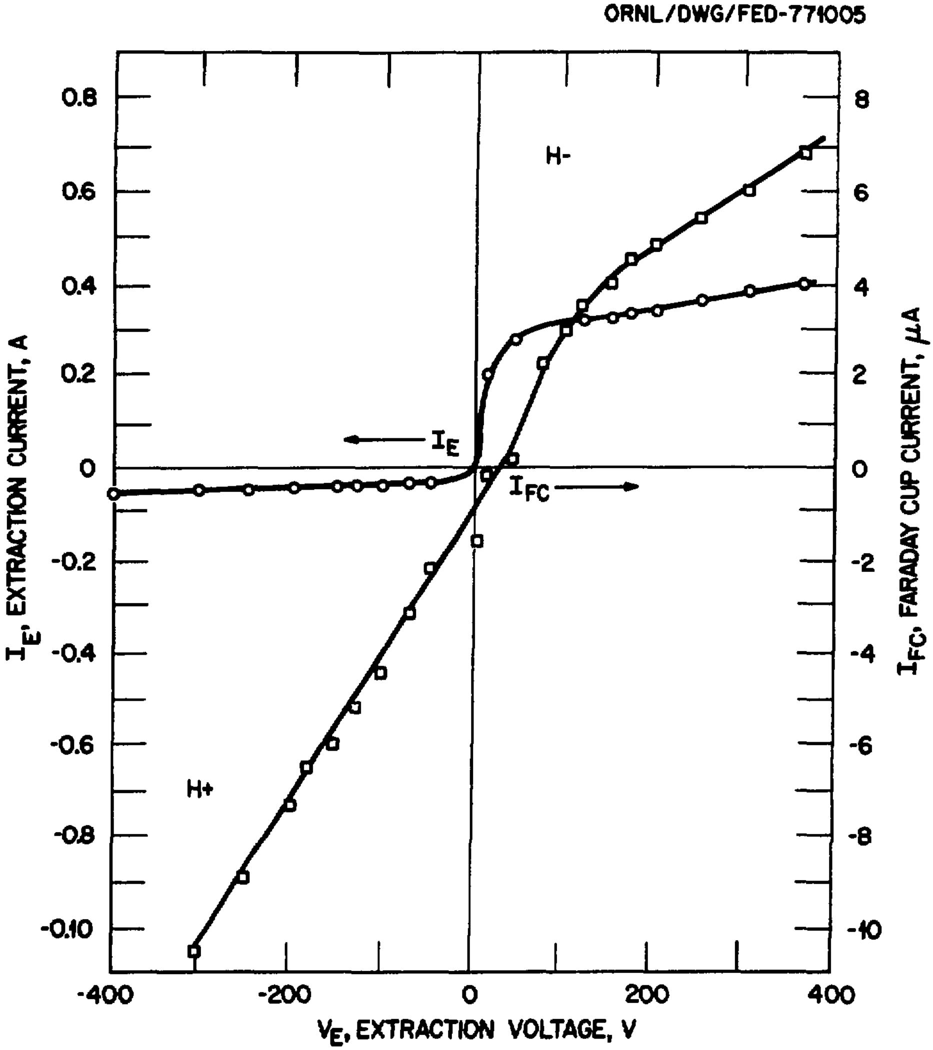


Figure 6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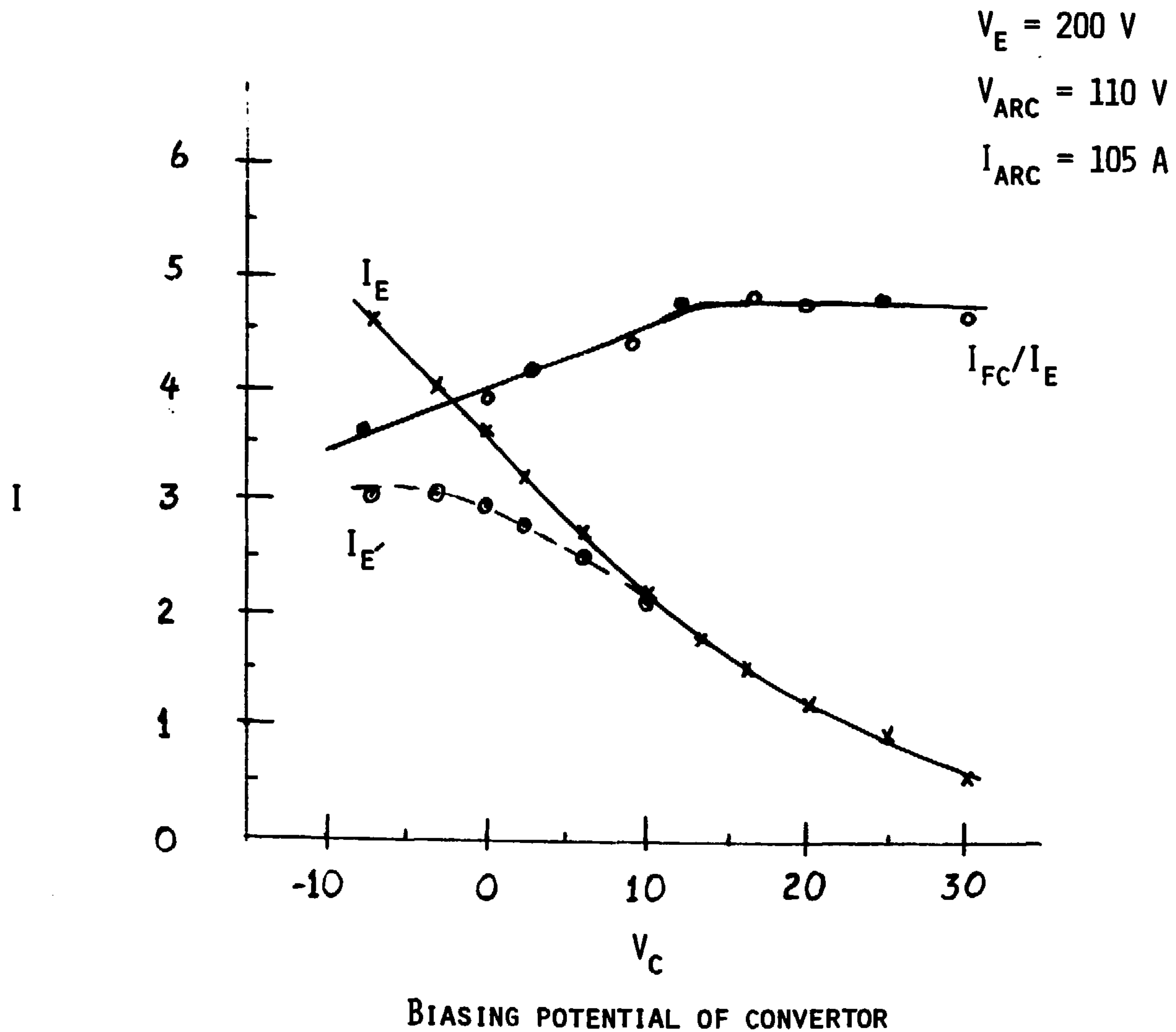
0.8
0.6
0.4
0.2
0
-0.2
-0.4
-0.6
-0.8
-0.10
8
6
4
2
0
-2
-4
-6
-8
-10
H-
H+
I_E
I_{FC}
I_E, EXTRACTION CURRENT, A
I_{FC}, FARADAY CUP CURRENT, μA
-400
-200
0
200
400
V_E, EXTRACTION VOLTAGE, V

Figure 7



DERIVED H^- CURRENT, $I_{E'}$

Figure 8

PERFORMANCE OF MODIFIED DUOPIGATRON

	H^+	H^- (ESTIMATED)	
PLASMA DENSITY, cm^{-3} (POSITIVE ION)	$\sim 2 \times 10^{12}$	2×10^{12}	
PLASMA UNIFORMITY,%	±5	±5	
ARC VOLTAGE, V	<150	<150	
ARC CURRENT, A	<900	<900	
		(10%)	(30%)
BEAM CURRENT DENSITY, A/cm^2	0.4	0.04	0.12
BEAM CURRENT, A	70	7	21
ARC EFFICIENCY, A/kW	1	0.1	0.3
GAS EFFICIENCY,%	≥50	≥5	≥15
PULSE LENGTH, ms	500	≤500	≤500

Figure 9

ORNL/DWG/FED-77572R

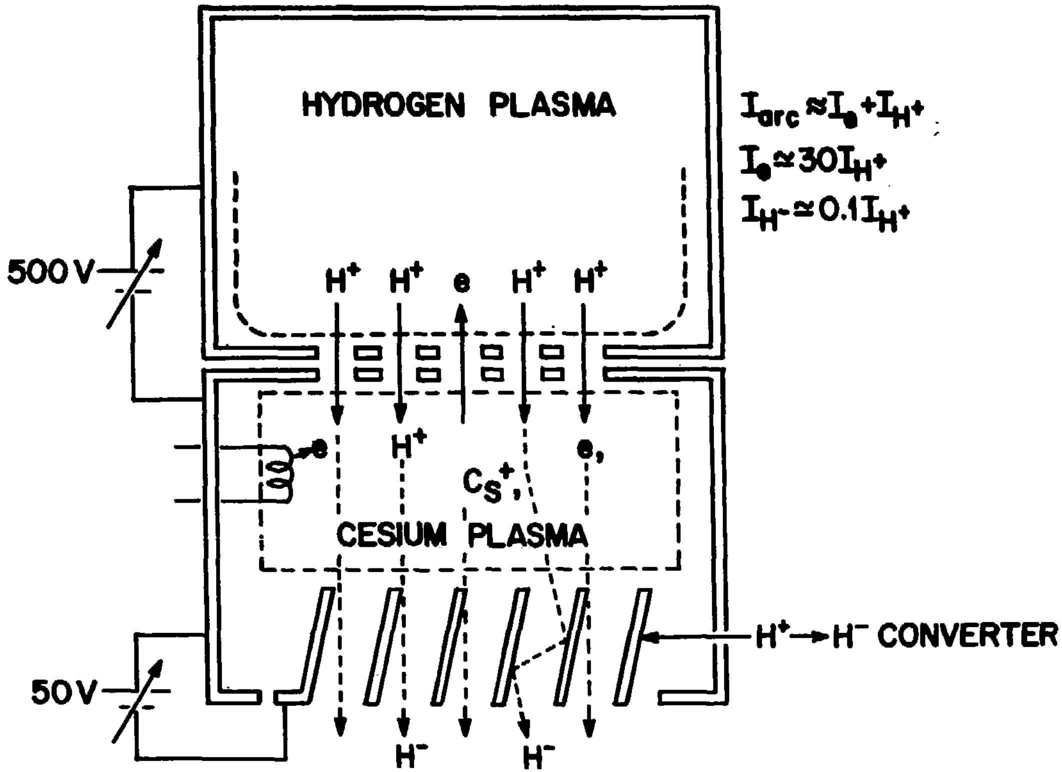
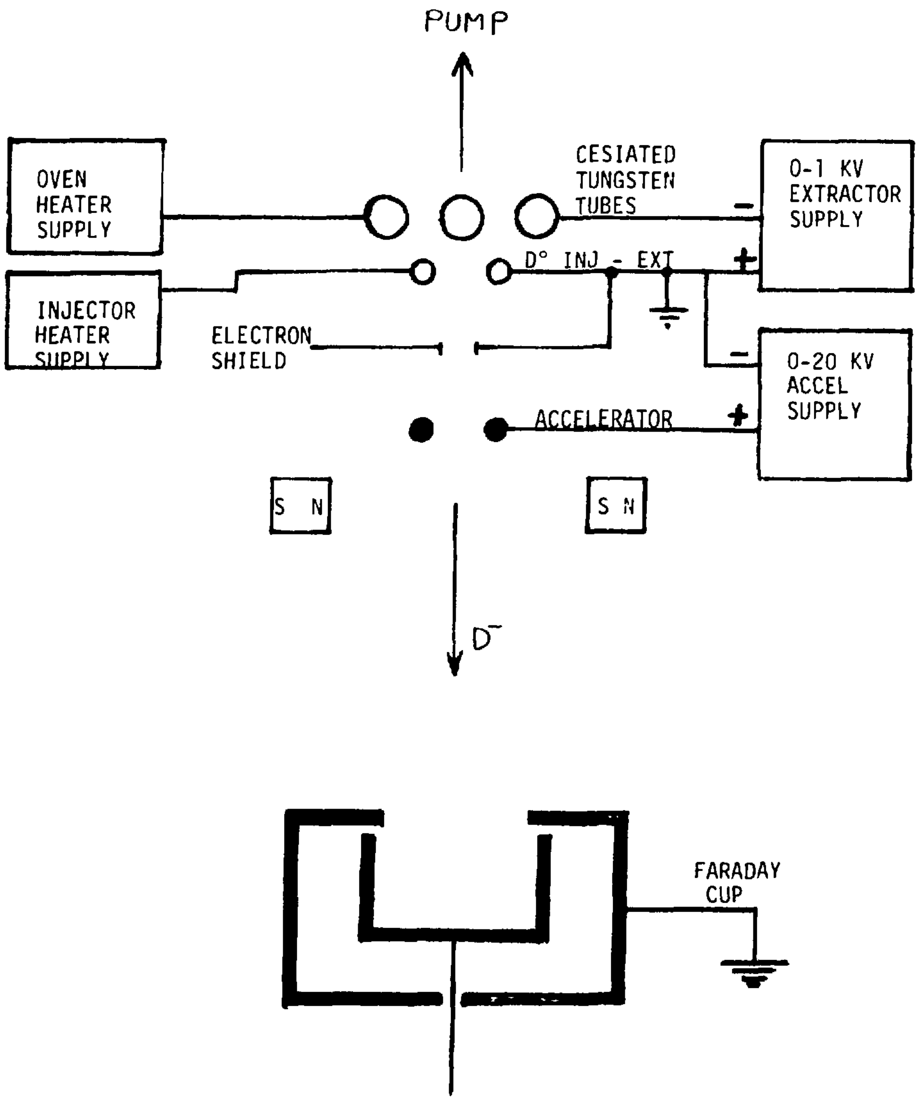


Figure 10



SCHEMATIC OF SURFACE-PRODUCED NEGATIVE ION SOURCE EXPERIMENT

Figure 11

NOZZLE/EXTRACTOR

FILLING PRESSURE ----------------0.1-1.0 TORR

TEMPERATURE ---------------- ≈2400 K

DIAMETER OF PINHOLE -------------0.1 CM

FLOW REGIME -----------------MOLECULAR

(COS θ DISTRIBUTION)

EXPECTED PERFORMANCE

ATOMIC FLUX ON W/C_S SURFACE (ϕ_S)

$\phi_S(E > 0.75\ \mathrm{eV}) = G \cdot \Gamma_O \cdot F_{TAIL} \cdot F_{DISS} \approx \underline{1\text{-}10\ \mathrm{mA/cm^2}}$

G: GEOMETRIC FACTOR (≈0.2)

Γ_O: TOTAL NUMBER FLUX PER HOLE (VARIED)

F_{TAIL}: FRACTION OF PARTICLES IN $E \approx 0.75$ (≈0.1)

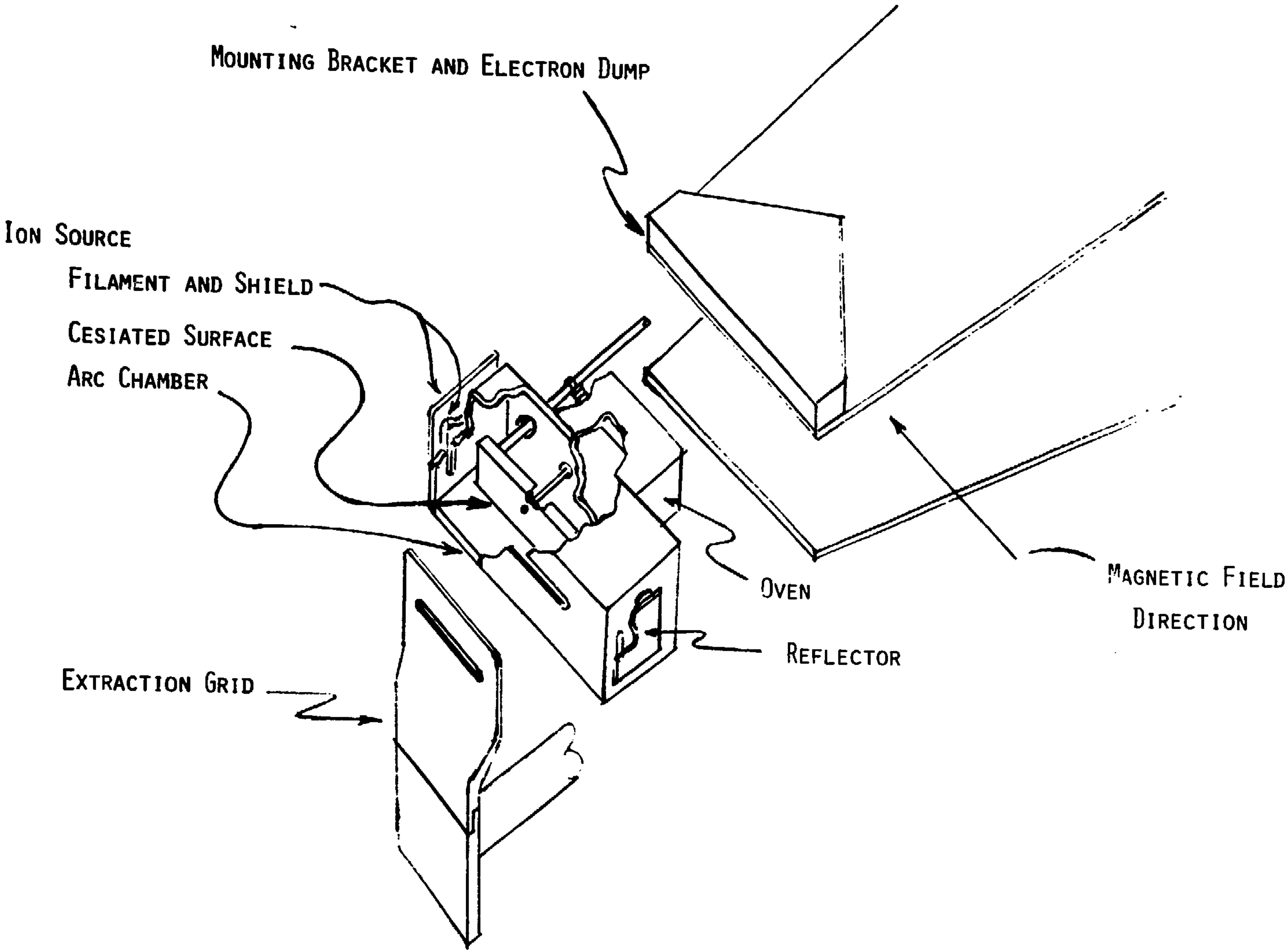
F_{DISS}: EQUIL FRACTION OF DISSOCIATED H (≈0.8)

THEORETICAL EFFICIENCY -----------40-50%

OVERALL GAS EFFICIENCY -----------1%

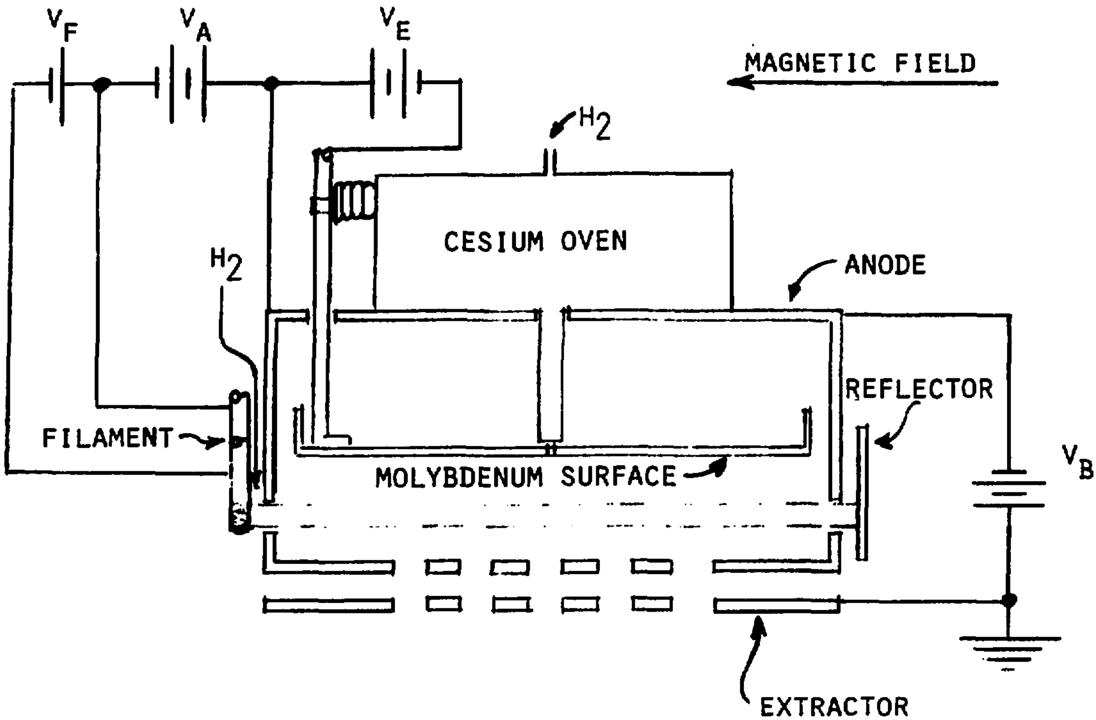
(I_{H^-}/I_{GAS})

Figure 12



Calutron Negative Ion Source

Figure 13



CALUTRON NEGATIVE ION SOURCE

Figure 14

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NEGATIVE-ION-BASED NEUTRAL BEAMS AND UPGRADED FUSION DEVICES (TFTR, MFTF)

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NEGATIVE-ION-BASED NEUTRAL BEAMS AND UPGRADED FUSION DEVICES (TFTR, MFTF)*

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Abstract

The development of negative-ion-based (H^-, D^-) neutral beams for TFTR and MFTF is discussed. Problems with existing equipment include providing beam current and current density, voltage holding, and gas pumping. The problem of stripping the D^- is described; it is concluded that photodetachment would be desirable to prevent formation of high-energy D^+. Estimates of the parameters associated with photodetachment show it may be feasible although a detailed study will be required.

*Work performed under the auspices of the U.S. Energy Research and Development Administration under contract No. W-7405-Eng-48.

†On loan from Westinghouse Electric Corporation.

Introduction

The first uses for neutral beams based on negative ions will probably be for upgrades of devices now under construction — TFTR, MFTF, and possibly TMX or DIII. If this is so, there are many implications for the beam system,[1,2] many of which are independent of the detailed techniques used to generate the beams. In this paper we begin to identify these areas and discuss their consequences for the development program.

General Comments

We start with three general comments. First, any upgrade will reflect the need for improvement in some area, presumably beam energy. As the energy is increased, the power requirements probably will not decrease, so that injected currents will remain high. Pulse lengths will undoubtedly be several seconds at least, and could be dc in MFTF if the power can be handled. The point is that either a few large beams or many small ones will be needed, but in any event, considerable current is required for long times to drive a fusion experiment into the thermonuclear regime.

A second point is that we will wish to use existing equipment to the extent possible to keep cost and construction time to a minimum. Thus, for TFTR, we can achieve 300 KeV by operating two power supplies "back to back" in a tandem mode.[1] We will consider the consequences, both good and bad, below. Note, however, that we will clearly wish to use the full power supply current capability, 65 A in TFTR. For MFTF and DIII, two available supplies will yield 80 A at 160 kV; if higher voltage is necessary, additional supplies can ride on top of these. To take advantage of these existing supplies, we need single sources with 65 or 80 A at high voltage, or we need to learn how to run several smaller sources off the tandem supply.

The existing mechanical equipment — vacuum box, cryopumping, etc. — also should be used. This probably precludes the use of tritium. As the use of tandem supplies requires that both ends of the source system be at high voltage, voltage holding will have to be done within the vacuum chamber. This will be discussed later.

A third comment is that beam power efficiency, per se, is not the primary consideration in these experiments, as it would be in power-producing reactors. Getting the injected current at high energy with a low capital expenditure is more important than the power efficiency, and trade-offs are possible that would not be acceptable in a reactor. For example, because high-voltage power is more expensive than low-voltage power, power at low voltage may be used e.g., in a stripping cell or elsewhere, to increase beam current despite decreases in total power efficiency. Alternatively, it is not necessary to reach the ultimate in stripping efficiency if other considerations are more important.

Voltage and System Design

The use of two power supplies with the center grounded means that the voltage will be related to the system components as shown in Fig. 1. The use of excursions of $\pm V_b/2$ also appears to be the only way that development can proceed above 200 kV in existing test stands. In the HVTS, for example, voltage hold-off problems due to limited space will make voltages above 400 kV (-200 kV to +200 kV) very difficult to handle.

Operating "double ended" has advantages in addition to voltage holding:

(1) Energy stored in capacitance to ground is reduced by a factor of four.

(2) Transformers supplying low-voltage power to high-voltage regions need isolation only to $V_b/2$.

A primary disadvantage is that both ends of the system are at high voltage, so that a high-voltage region will be closer to the experimental plasma than required in a single-ended system. The system design will be complicated by this feature. Also, simultaneous control of tandem high-voltage supplies requires greater care than control of one supply.

The electrical implications of the use of tandem supplies is discussed in the paper by Baker and Hopkins at this conference.[3]

Mechanical Arrangement — TFTR

To take a "first cut" at the mechanical system (Fig. 2), we laid a block diagram of system components over a drawing of the TFTR beam line.[4] This is not a design, of course, but rather is prepared for purposes of estimating parameters. Detailed numbers must come from a careful design.

In the diagram a 1-metre-long section has been left for the negative-ion source. For present purposes we will not discuss the nature of this source.

After passing through the rest of the system, the beam must enter the tokamak through a port about 80-cm high by 46-cm wide. At 65 A this corresponds

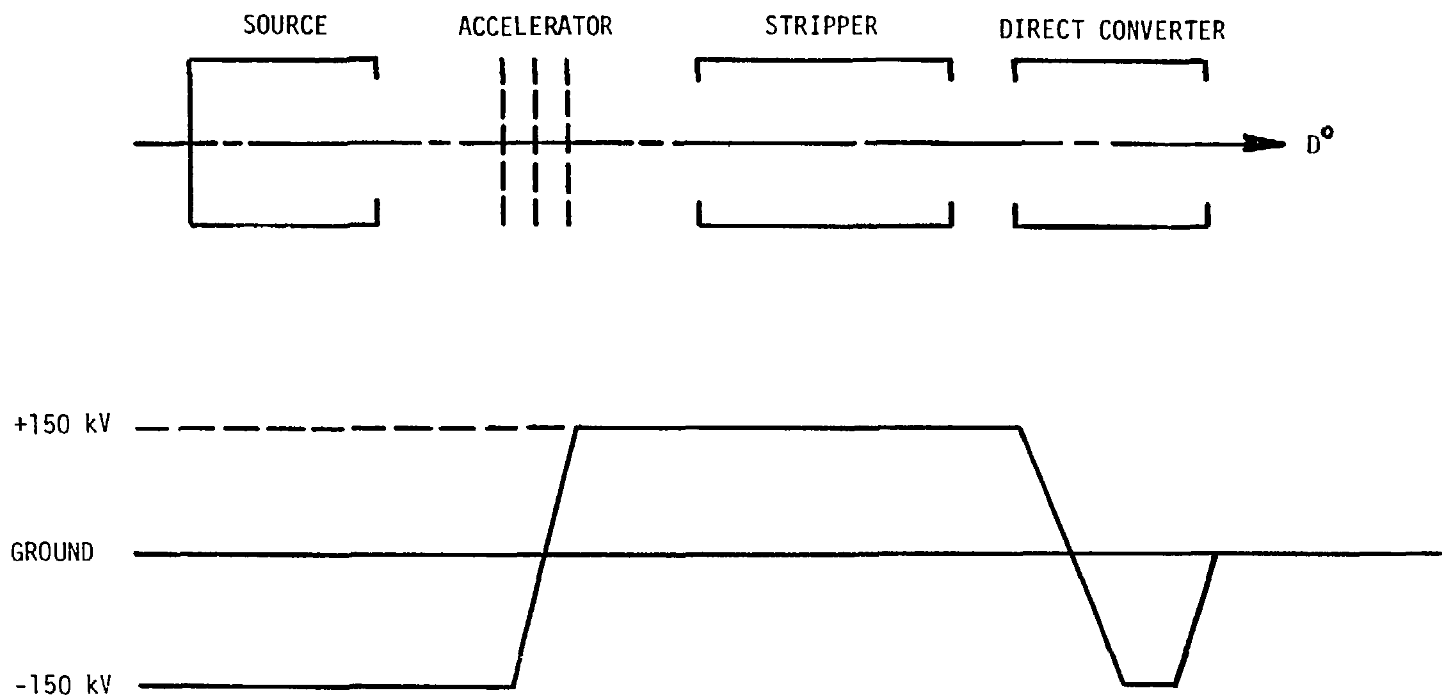
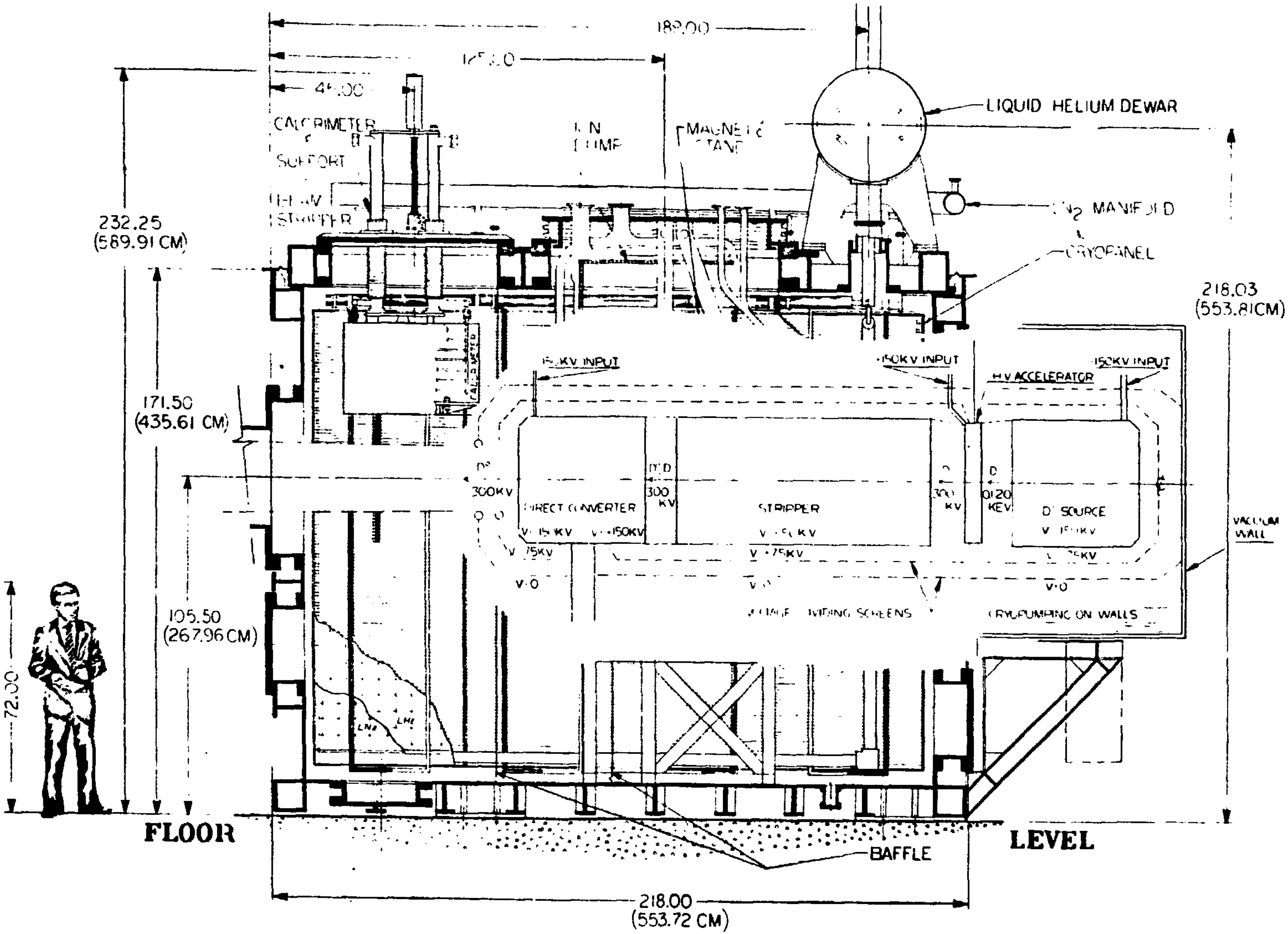


Fig. 1 Voltage variation along D^- Beamline. Minor variations in potential may occur in the source or direct converter. This potential profile assumes negligible D^+ enters the direct converter.



NOTE. DIMENSIONS IN INCHES

FIG 2 TFTR BEAMLINE WITH OVERLAY OF NEGATIVE ION SYSTEM (SCHEMATIC) to a current density $j = 18\ mA/cm^2$. In the TFTR beamlines operating in the positive-ion mode, beam particles originate from an area 40-cm high, with 140-cm width between the extreme trajectories, corresponding to a *net* current density at the source of $j = 12\ mA/cm^2$; the three beams cross to enter the tokamak. For present purposes, then, we assume that the negative- ion current density at the source must be at least 15 mA/cm^2; as we will see from other considerations, higher current densities are desirable.

This current density is several times higher than achieved to date in double charge-exchange sources, so improvements are required if these sources are to be used. For direct extraction sources, current densities of 1.5 to 2 A/cm^2 have been achieved in the extraction slits. However at currents per source of 5 to 10 A, 6 to 13 sources will be required. The ratio of the area of each source to its slit area will thus have to be 50 to 100 or less, including source structure, bending magnets, etc., for successful application.

Considerable excess gas is generated in present negative-ion sources. This gas must be pumped in the source region. A D^- current of 65 A corresponds to 5.5 torr-litres/s of D_2; at 10% gas efficiency, for example, the gas load is 55 torr-litres/s. If this is cryopumped at 10^{-4} T, a cryoarea of $6 \times 10^4\ cm^2$ (250 cm × 250 cm) will be required. The box drawn (in Fig. 2) about the source region has a wall area of about $1.3 \times 10^5\ cm^2$, so a pressure of 5×10^{-5} T is the best that can be achieved in it. There is clearly a need both to increase the gas efficiency and to pump as much gas as possible at higher pressures near the source. The problem could also be alleviated by opening up the back of the vacuum box in a way to take advantage of the cryopanels within the present box. Finally, baffling will be required to keep the pressure in the accelerator well below 10^{-3} torr to minimize heating and sputtering damage to the acceleration electrodes.

As discussed in the presentation on double charge-exchange,[5] it appears possible to construct a jet with extremely low losses of cesium. Thus, if the D^- ions are formed in double charge-exchange, cesium contamination should not be a problem in normal operation. Careful design and planning of interlocks, etc., will be required to ensure that the probability of an accident is very small.

The approximate length of the high-voltage accelerator follows from the Child-Langmuir law for D^- as

$$d = 3.48 \times 10^{-2} V^{3/4} j^{-1/2} \quad .$$

Thus, for V = 300 kV and j = 20 mA/cm^2, d = 16 cm. Even at 0.1 A/cm^2, d = 8 cm and the average field is 38 kV/cm, well below vacuum-breakdown limits.

The accelerator clearly will require multiple apertures. If many sources at low current are used, each source can feed a single circular aperture or slot with little current loss. If the current per source is large, on the other hand, it will need to feed an array of openings, or slots and negative ions will be lost on the structure. In either case the current density into the accelerator openings will need to be larger, perhaps by a factor of two or more, than the simple calculation would allow.

In the layout shown in Fig. 2 the stripper has been assumed to have a length of 2 metres. This would be required if the stripper used photodetachment; a gas or plasma stripper might be shorter. This issue is discussed later.

If photodetachment is used, the exiting beam contains only D^0 and D^-. This beam can be handled in a direct converter similar to those designed for positive ions, assumed here to require 1 metre of length.

To take advantage of the TFTR beam vacuum box, and because the power supply will be operated in a tandem mode, we will have to isolate the voltage by vacuum at the direct-converter end of the system. In the schematic both ends are in vacuum. Screens (two shown) are used to divide the voltage and to give a well defined ground surface. Fields of 50 kV/cm are feasible, but on such a large structure 5 kV/cm is more conservative, requiring 30 cm to the ground screen.

Care must also be taken to keep far away from gas breakdown. We are operating on the low-pressure side of the Paschen curve for deuterium.[6] For $V = 3 \times 10^5$ volts, breakdown occurs at pd = 0.1 torr-cm. Thus, for d = 30 cm, the pressure must be below 3×10^{-3} torr, which is well satisfied in the above design.

The capacitance-to-ground of the illustrated system is roughly 6×10^{-10} F, so that the stored energy (at ±150 kV to ground) is less than 6 J. This is about the acceptable limit in present LBL positive-ion sources and 4 to 8 times less than that acceptable in vacuum breakdown. The energy available in the event of a spark is not sufficient to damage surfaces, and thus cannot cause sustained breakdown. Additional electrostatic shields can further reduce the stored energy available for initiating sparks.

The TFTR magnetic field is 100 to 200 gauss in the beamline. This field *is large enough to affect the energetic-ion motion*; it will have to be shielded from the rather large volume of the negative-ion system.

Mechanical Arrangement — MFTF

As shown in Figs. 3a through 3c, the beam layout for MFTF is very different from that for TFTR. A negative-ion system, probably similar to the schematic sketched on the TFTR beamline, can be inserted into the source array by removing several sources. The detailed design will be complicated by the likely requirement that as many 80-kV sources be retained as possible. The final layout of the 80-kV array is not available, so that no attempt has yet been made to consider specific modifications for negative ions.

Because of the requirement to keep as many 80-kV sources as possible on MFTF, there will be a greater premium on current density and compact design than in TFTR. This consideration, already affecting the design of positive-ion sources, may make the beam system development somewhat more difficult for MFTF.

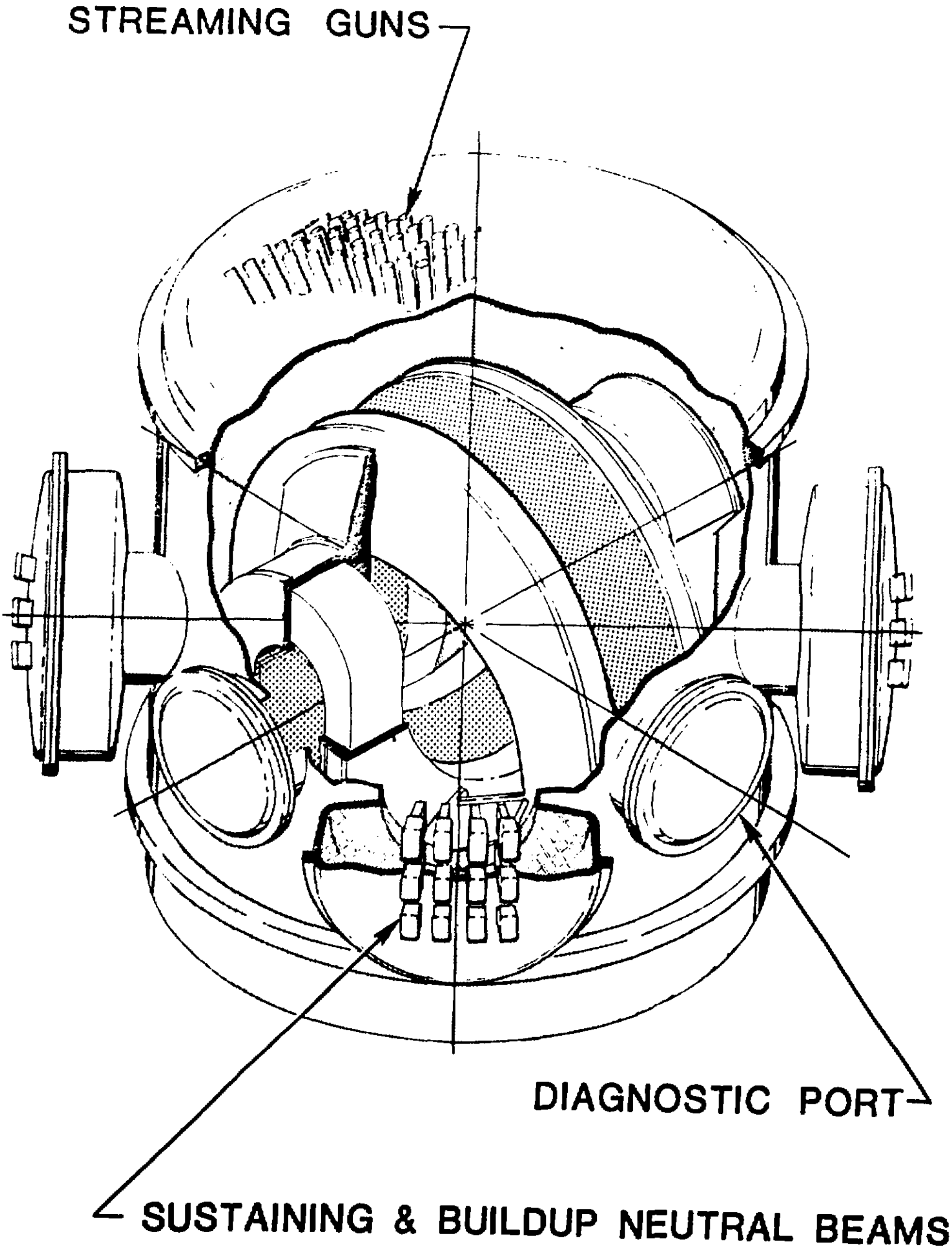
A system design for MFTF will have one significant advantage. Because of the large ports into the plasma, and because of the large dimensions of the MFTF beam array, the geometry of the negative-ion source is more flexible than in TFTR. If, for example, it proves desirable to work with a beam much longer in one dimension than the other, this will be possible in MFTF but probably not in TFTR.

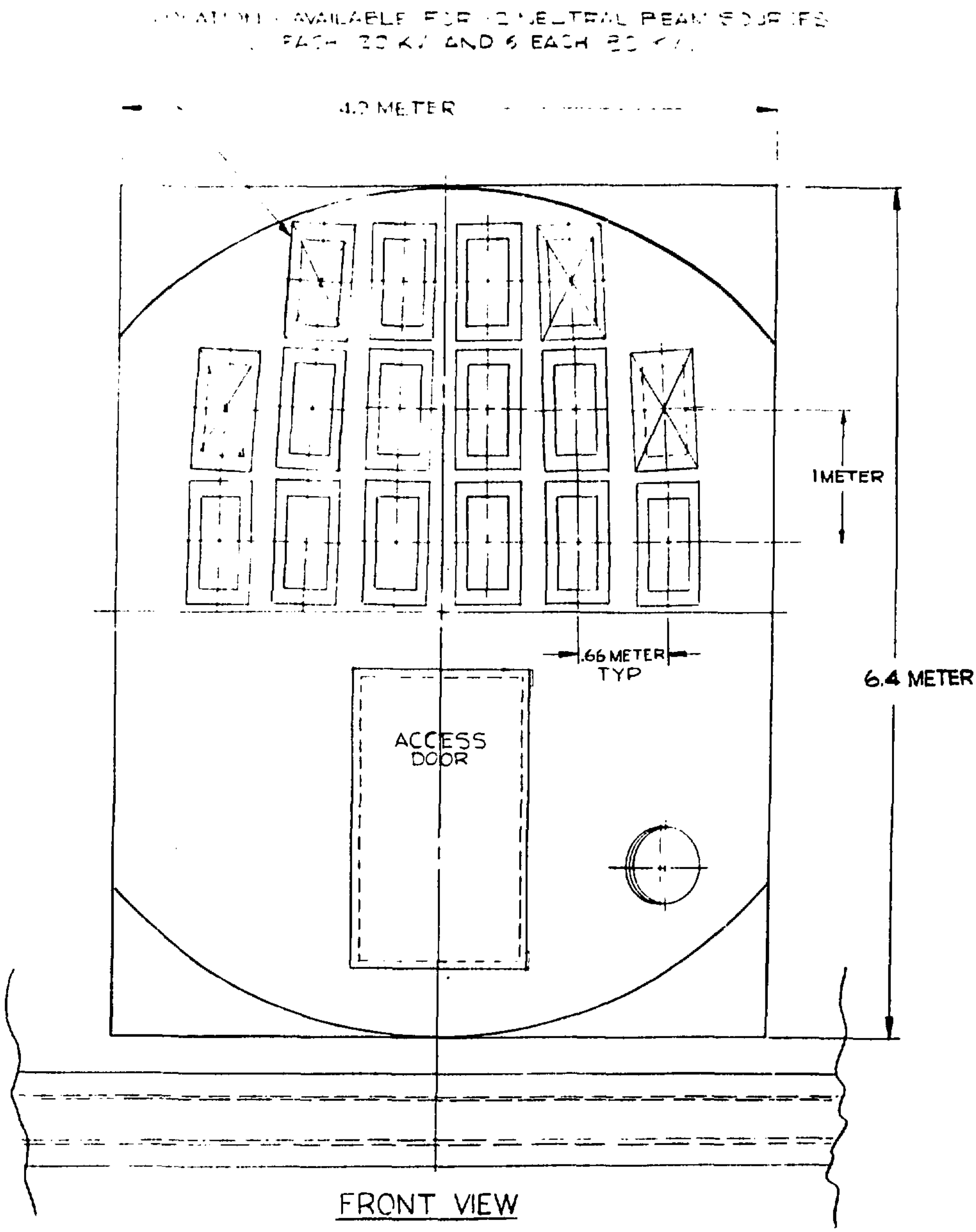
If hydrogen beams are required instead of deuterium, the helium cryopumps will have to be subcooled to obtain a base pressure $<<10^{-6}$ torr.

Except for the above caveats, general considerations for MFTF are the same as those for TFTR. At this stage, therefore, the development of negative-ion-based systems is not very sensitive to the application.

FIG 3a

MFTF

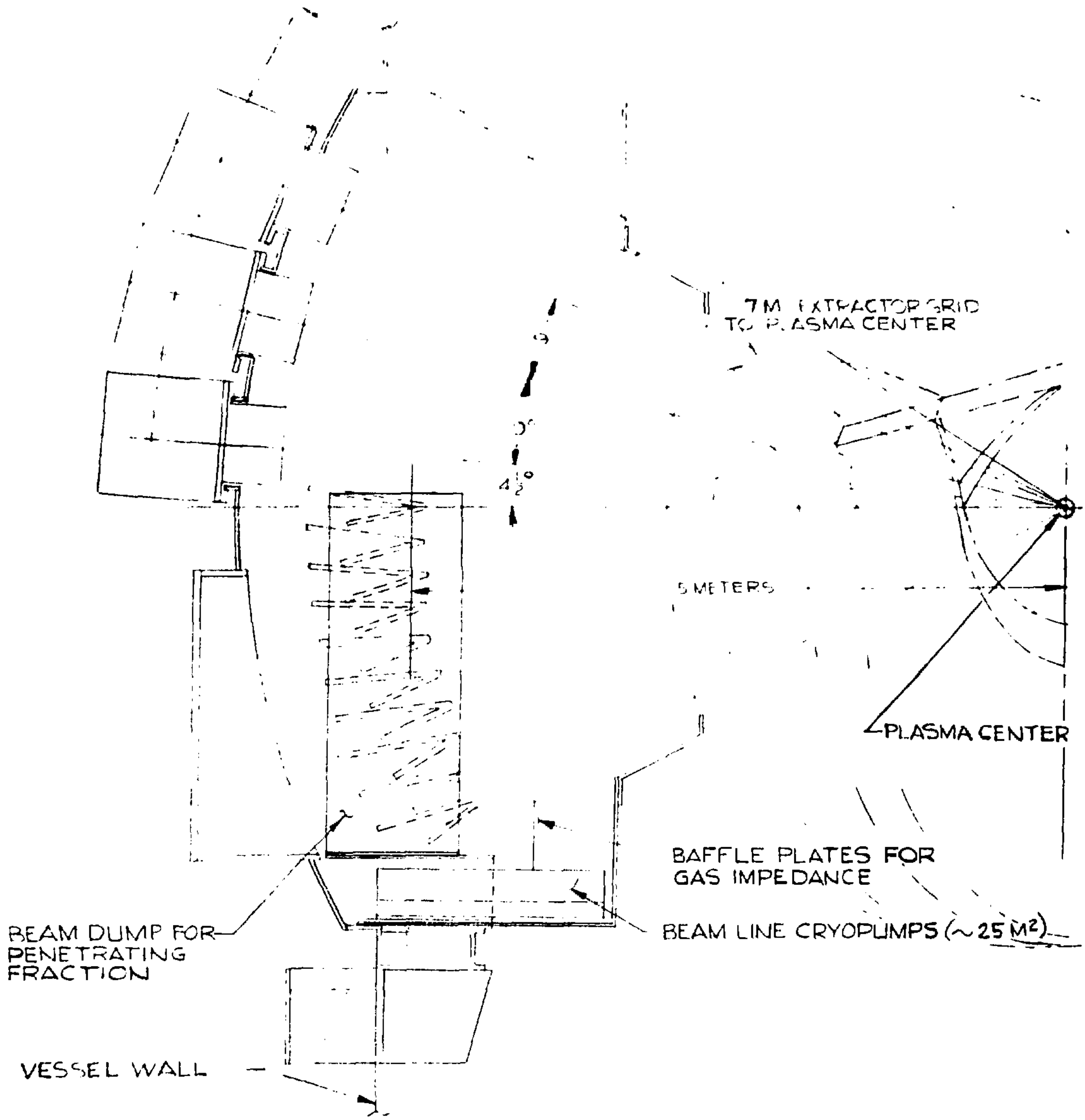

STREAMING GUNS
DIAGNOSTIC PORT
SUSTAINING & BUILDUP NEUTRAL BEAMS



M.F.T.F NEUTRAL BEAM INJECTOR

TYPICAL OF FOUR LOCATIONS ON VESSEL

Fig. 3b



CROSS SECTION

M.F.T.F. NEUTRAL BEAM INJECTOR

TYPICAL OF FOUR LOCATIONS ON VESSEL

Fig. 3c

Stripping, Beam Dumps, and Direct Conversion

Three different types of stripping cells have been proposed: gas or vapor, plasma, and photon. Each has advantages and disadvantages as listed in Table 1.

Table 1. Possible stripping media.

Stripping medium	Advantages	Disadvantages
Gas or vapor	Simplicity.	Maximum conversion 60% to 65%.
		Energetic D^+ formed by ionization.
	Low power requirements.	Gas or vapor used near fusion experiment.
Plasma	Conversion efficiency = 80%.	High ionization required.
		High power requirements.
		Energetic D^+ formed by ionization.
		Gas or vapor used near fusion experiment.
Photon	No D^+ formed.	High (advanced) technology.
	Conversion efficiency arbitrarily high.	
	No additional gas or vapor used.	High power requirements.

In both gas (vapor) and plasma strippers, half or more of the final ion beam is positively charged. To decelerate these ions requires a potential equal but opposite to that used in accelerating. Thus one could separate the positive and negative beams (e.g., by a magnetic field) and decelerate the negative beam back to −150 kV. Deceleration of the positive beam would require +450 kV. In both cases the power could, in principle, be coupled back into the power supply.

The voltage requirements to decelerate the positive ions are not very palatable; it would probably be better to dump these ions on a surface. This means, however, that a positive-ion current of 10 to 40 A will be dumped at 300 keV if the dump is at high voltage or at 450 keV if it is at ground. In both cases a large water flow for cooling is required, heat transfer for long pulses is necessary, and sputtering and other effects must be controlled.

Positive ions are not formed in a photon stripper (with photon energy = 1.5 eV) except through collisions with background gas. In this case a direct converter can be used with a design very much like those being developed for positive ions. As the payoff for simple handling of the residual-ion beam is so great, we will consider a photon stripper later.

The gas stripper, of course, has the major advantage of simplicity. The gas could have low Z to minimize problems of leakage into the experiments. A highly directional jet, such as that under development for charge-exchange sources,[5] or a capillary array could be used.[7]

A plasma stripper can obtain higher efficiency than gas: 80% rather than 50%. Considerable power is required, however. If an arc[8,9] is drawn across the ion beam, a lower limit to the current is that carried by the ions into the cathode, $en(T_e/M_i)^{1/2}LW$, with n the plasma density, T_e the electron temperature in eV, M_i the ion mass, L the plasma length along the beam, and W the width perpendicular to the current. A line density $nL = 2 \times 10^{15}\ \mathrm{cm}^{-2}$ is required. In lithium with $T_e = 2$ eV, for example, the ion current is $I = 240 \cdot W$ amperes. At a discharge voltage of 20 V (for example) and W = 40 cm, the corresponding power is 200 kW. Electron considerations will probably increase this by a factor of 2 to 5. The power requirements, while not large compared to beam power, are high enough to require some care.

The cross section for photodetachment of hydrogen and deuterium is $4 \times 10^{-17}\ \mathrm{cm}^2$ at 1.5 eV.[9,10] Suppose we require α stripping lengths for beam particles in the stripper; note that at α = 1, 63% of the beam is converted to neutrals, which is as good as in a gas cell. At α = 2, 86% is converted to neutrals, better than can be achieved in a plasma.

The light intensity required, I_ℓ, is

$$I_\ell = e\alpha v_b \varepsilon_p/\sigma L = 1.9 \times 10^5 \alpha\sqrt{E_b}/L\ \mathrm{watts/cm}^2 ,$$

with v_b the beam velocity, E_p = 1.5 eV, and E_b the beam energy in keV. Mirror reflectivities of 99.8% are available in large mirrors (18 in. × 18 in.), so the power input required at E_b = 300 kV is

$$P_\ell = 2 \times 10^{-3}\ I_\ell LW = 6.6 \times 10^3 \alpha W\ \mathrm{watts} .$$

Thus, at $\alpha = 1$ and $W = 40$ cm, the light power required is 260 kW, comparable to that required for the plasma stripper. Gallium arsenide lasers operate at 0.85-μm wavelength with efficiencies of 30% to 40%, so the total power input is 650 to 870 kW for $\alpha = 1$, or 1300 to 1770 kW for $\alpha = 2$. These power requirements, although greater than for a plasma, are still reasonable when compared to the high voltage power of 20 MW. The powers and power densities required are at or beyond the state of the art for lasers, however, so that detailed study will be required to determine feasibility of an operating system.

To keep the power density absorbed in the laser-cavity mirrors to a reasonable level, the cavity should be quite long. At L = 200 cm, as illustrated in Fig. 2, the absorbed power is $P_\ell/LW = 33\alpha$ watts/cm^2, which could be handled quite straightforwardly.

Because of the considerable advantage arising from the absence of D^+ in the final beam, it will be important to consider development of a photon stripper in some detail. Problems requiring attention include laser cooling, laser coupling to the optical cavity, and the cavity shape and characteristics for long confinement of the light.

It is noteworthy that any stripper will require less power and/or material flow if the beam width, W, can be reduced. Considerations such as the above will thus benefit from high current densities and from "slab" geometry in which the beam cross section is much longer in one direction than the other.

Conclusions

The above discussion assumes that the first applications of negative-ion-based neutral beams will be on TFTR or MFTF. These machines are already essentially designed, and the developer must consider the resulting constraints. To meet the expected deadlines, preliminary design must begin soon, to act as guidance to both developers and users of these beams.

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ENGINEERING PROBLEMS OF FUTURE NEUTRAL BEAM INJECTORS

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ENGINEERING PROBLEMS OF FUTURE NEUTRAL-BEAM INJECTORS*

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ABSTRACT

Because there is no limit to the energy or power that can be delivered by a neutral-beam injector, its use will be restricted by either its cost, size, or reliability. Studies show that these factors can be improved by the injector design, and several examples, taken from mirror reactor studies, are given.

* Work performed under the auspices of the U.S. Department of Energy under contract No. W-7405-Eng-48.

† On loan from Westinghouse Electric Corporation

INTRODUCTION

Neutral beams have been so successful in plasma heating experiments that their role in future experimental fusion reactors is almost assured. However, major advances in the technology will be needed before neutral beams can be used in a power-producing fusion reactor.

Although there is no fundamental limit to the power or energy that can be delivered by a neutral-beam injector, several constraints must be met; that is, the specified current of neutrals must be delivered at the desired energy via an injector of reasonable cost, acceptable size, and adequate reliability.

High-power, high-energy injectors will be very costly and very large. The injector cost is estimated to be $320 divided by the injector power efficiency per kW of neutral beam. As a consequence, to meet the cost objectives of a fusion reactor [$1000 to $2000/kW(e)], the efficiency of the injector must be better than 70%.

To form an operating ensemble, the injector must be reasonably compact and physically compatible with the reactor layout. For instance, the aperture through which the beam leaves the injector and enters the reactor must not be too large. Furthermore, the presence of the injector must not interfere with other reactor components.

Finally, the injector must be sufficiently reliable to sustain at least 6 months of continuous operation. More frequent interruptions would be intolerable. To achieve this reliability, only the most conservative designs can be considered and only the most ideal materials used.

CONCEPTUAL STUDIES

During the past few years, we have studied several conceptual designs of different mirror reactors. The associated neutral-beam injector designs represent our attempts to resolve some of the problems previously mentioned.

The injectors for the Fusion-Fission Mirror Hybrid[1] uses positive ions as a source of neutrals, a liquid nitrogen-cooled neutralizer cell, and energy-recovery electrodes to enhance the injector efficiency. The injector for the Reference Mirror Reactor[2] employs a negative-ion beam formed by double charge exchange in cesium vapor and a photodetachment cell. Finally, the injector for the Tandem Mirror Reactor[3] runs on negative ions with a plasma-stripping cell. A thermal beam dump is used to recover the energy remaining with the unstripped negative ions.

These studies of neutral-beam injector designs have proven useful not only in the reactor studies but also as a guide to establish neutral-beam development objectives. In the following, some features of these designs are described.

GENERAL DISCUSSION

A neutral beam (Fig. 1) is formed from either positive or negative ions that have been extracted from a source, then accelerated, and finally focused at or near the reactor plasma. While traveling from the ion source to the reactor, the beam passes through a neutralizer cell in which positive ions become neutralized by picking up an additional electron or negative ions are neutralized by being stripped of their extra electrons. Subsequently, the beam passes through an energy-recovery unit in which some fraction of the energy remaining with the un-neutralized portion of the beam is recovered.

Conservative source design entails a low-extraction current density, possibly 100 mA/cm^2. This assures a long operating life by limiting sputtering and holding down the grid dissipation. As a result, the emitting source-area of a high-current injector will be quite large. However, the aperture through which the beam enters the reactor blanket can still be small if the entire beam is aimed at a common focus in the plane of the aperture. Thus, as shown in Fig. 2, an injector will assume the approximate shape of a pyramid, with the ion sources at the base and the beam aperture at the apex.

INJECTOR DESIGNS

Several source details are shown in Fig. 3. Using the Lawrence Berkeley Laboratory/Lawrence Livermore Laboratory (LBL/LLL) source as a reference, the preferred ion source design is found to be long and relatively narrow. This configuration makes it easier to pump the gas that escapes from the source out of the beam line. It also makes it easier to cool the grids, forms a favorable configuration for neutralizing the beam, and makes the design of the energy-recovery electrodes less critical. In general, smaller sources are advantageous because they are easier to make, easier to align, and less difficult to install. Furthermore, they require less expensive equipment for testing. Also, sources with smaller grids store less electrical energy and are not as vulnerable to damage from arcing.

The ion source shown in cross section in Fig. 3 has a hollow cathode that is proposed as a substitute for the tungsten hairpin filaments now in use. Studies[4] indicate that this type of cathode should be capable of thousands of hours of continuous operation.

The anode mounting detail in Fig. 4 shows how small alumina buttons can be used for low-voltage insulation between adjacent components of the ion source. This mounting can be effective in regions of intense neutron bombardment: a small percentage change in the insulator dimensions will not seriously alter the spacing between the electrodes and, because of the low voltage and the small contact area, degradation of the insulation will not cause a significant increase in leakage current. More massive, high-voltage insulators must be shielded from the neutron radiation and gamma flux originating in the reactor.

Figure 5 shows a version of the LBL/LLL ion source[5] that was designed for the Mirror Fusion Test Facility. The arc source is supported by two triaxial feedthroughs, each of which is housed within its own cylindrical insulator column. All of the services (i.e., power, gas, and water) are brought through the back of the source. The high voltage (80 kV) is sustained over the outer surfaces of the insulator and across the low-pressure gaps between the arc chamber, the 60-kV corona shield, and the grounded frame. The structure is compact, making it possible to house the source within the grounded vacuum wall (Fig. 6). In this way, many individual sources can be stacked, one upon the other, to form a relatively small injector assembly.

Figure 7 shows an injector made of many individual beam lines. The sources, at high voltage, are mounted within grounded electrostatic shields. This makes it possible for each source to be turned on or off without disturbing its neighbor. As a result an assembly of many beam lines (Fig. 8) can provide reliability through redundancy. In the event of an arc in any beam line, vacuum switches open the circuit, crowbars short out the residual voltage, and special arc snubbers dissipate the residual stored energy. Thus, the injector continues to operate with one beam line turned off, while the faulted beam line remains unharmed, ready to be re-activated in a few seconds.

It is well known that the fraction of neutrals available from a positive-ion beam passing through a neutralizer of optimum design falls off with increasing beam energy.[6] Therefore, it is necessary to use negative ions (Fig. 9) to obtain efficient neutral beams at energies greater than 150 keV. Negative-ion stripping in a gas cell is 62% effective. In a plasma, the stripping efficiency is 82% while in a photodetachment cell it reaches 95%. Thus, negative ions are a desirable source of high-energy neutrals.

Unfortunately, negative-ion sources are still in development. Figure 10 shows a conceptual design of a negative-ion beam line that uses an LBL/LLL positive-ion source, operating at about 2 kV to form negative ions via double charge exchange in a cesium vapor cell.

The cesium vapor also acts as a pressure barrier. This allows the neutral gas, escaping from the positive-ion source, to be pumped away at roughly 2×10^{-3} Torr, whereas the pressure on the other side of the vapor cell is maintained at 10^{-4} Torr by many cryopanel pumps.

The 20% of the incident beam that becomes negative in the cesium cell is accelerated to high voltage. Meanwhile, the balance of the positive-ion beam becomes neutral and is collected at the low-energy neutral target.

Photodetachment can be used to form a very efficient injector of high-current beams. Many negative-ion beam lines can be operated in parallel so that they pass through a large common, stripping cell (as shown in Fig. 11). The high-voltage insulators, not shown in Fig. 11, are mounted in a shielded region above and below the beam line. To enhance the efficiency, energy-recovery electrodes are also included in the system.

A 1.2-MeV injector has also been considered (Fig. 12). As before, a double-charge-exchange source of negative ions is used. The source is mounted in a pump duct that is supported at high voltage by insulators above and below the beam line (Fig. 13). To minimize the high-voltage insulating problems, the source is maintained at -600 kV and the stripping cell at +600 kV. Perforated electrostatic shields at -400 and -200 kV help maintain the high-voltage standoff between the ion source and the grounded injector walls. Similar shields at +200 and +400 kV surround the stripping cell.

To keep to a minimum the loss of negative ions via charge exchange with the background gas in the beam acceleration region, it is essential that the background gas pressure be low. Thus, the outer walls at the injector are covered with cryopanel pumps.

The ions are stripped using a cesium-plasma cell that is maintained by surface ionization on hot-tungsten plates. To maintain the space-charge neutrality of the cesium plasma, the tungsten plates also emit electrons. If we assume that 82% of the negative ions get stripped, the balance of the beam (consisting predominantly of positive ions) is collected in a beam dump at ground potential.

CONCLUSION

In this paper, we have briefly described some conceptual neutral-beam injectors that were designed in various mirror fusion reactor studies. From these studies and from some of the more detailed analysis that went into their preparation, it is possible to draw some specific conclusions about neutral-beam development.

For instance, we conclude that negative ions are essential to the formation of very high-energy beams. The power efficiency of the source is not critical because it does not have a significant impact upon the overall efficiency of the injector.

Although a compact source of ions would be advantageous, this cannot be achieved. Considerations of long life, grid heating, sputtering, and reliable high-voltage stand-off between the acceleration grids require low ion current densities, albeit large emitting areas for the ion source. Those procedures that might be used to enhance the extracted current density make the source less reliable and unfavorably affect the beam optics.

Even though gas efficiency is often considered an important source parameter, the critical factor is the density of the residual background gas in the region of the extraction grids. Charge exchange between the ions in the beam and the residual gas molecules can cause the loss of a significant fraction of the ion beam and also result in excessive grid loading. The most effective method to reduce the density of the background gas is to minimize the operating pressure of the ion source. Because several different types of negative-ion sources are now under development there is no point in discussing this problem further. However, it is important to note that the removal of excess gas is more economical at higher pressures, near the ion sources, rather than further down the beam line. In all of the conceptual injector designs, the gas coming out of the ion sources has been drawn back into a pump duct behind the ion sources.

In a continously operated system, grid cooling appears to be the most serious problem of ion source design. Solid grid rods can not transfer the heat load; as a result, direct cooling of hollow rods is necessary. To do this, the grid rods must be large in diameter and subject to puncture as a result of arcing or sputtering. To minimize this problem the grid structure must also be large, the grid transparency reduced, and the allowable beam current density limited. All these are undesirable, and the compromises needed to form a workable system must be carefully evaluated.

Another factor of importance in grid design concerns the prospect of metal flaking as a result of bombardment by neutrons and alpha particles. Although the problem appears to be far less severe at the ion source than at the first wall, the prospect of this flaking causing high-voltage arcs is very serious. Fortunately, there is hope that the flaking will not be as bad as it once appeared: there are techniques that can mitigate the effect.

High-voltage insulation in an injector needs considerable study. After 100 years of research, we are still unable to accurately specify the minimum spacing required to hold off high voltage in vacuum. Small spacing at low pressure works, but the factors that assure reliability over large areas (to provide compactness) are not known. To overcome this, we propose a redundant design in which arcing components can be turned off before serious damage occurs. The actual details of such procedures have yet to be worked out.

As for stripping negative ions, a gas stripper can be ruled out. At the price of adding gas to the system the 62% is not that good, particularly when there is a prospect of getting 82% efficiency from a plasma cell. We have considered a cesium plasma cell, recognizing that the cesium containment is a problem. But despite that, plasma stripping is not too desirable either, because at optimum stripping the unneutralized fraction of the beam is composed of almost equal parts of positive and negative ions. It is difficult to obtain energy recovery from such a beam composition in a compact beam line. If the plasma cell were made over-dense, the stripping efficiency would drop to 80%, and the remainder of the beam would become almost 100% positive ions. However, the recovery of the energy of the positive ions in a negative-ion beam line creates undesirable voltage-holding problems. Of course, a thermal beam dump can be used, but its efficiency is not very good. Thus, we turn to photodetachment. It is expected that laser technology can be brought along to meet our needs.

In this discussion, we have not mentioned either continuous cryopanel pumps or the power supply requirements. Obviously, much effort is also needed in these fields. To make reliable, high-energy, high-power, neutral-beam injectors for operating fusion reactors, much effort, time, and money are required. However, there are no insurmountable obstacles.

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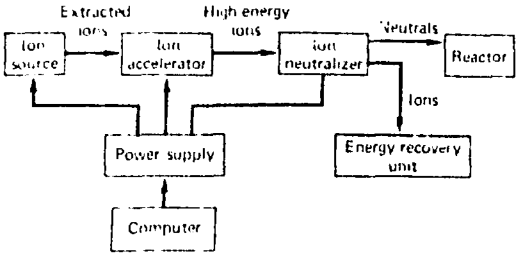


Fig. 1. Formation of neutral beams.

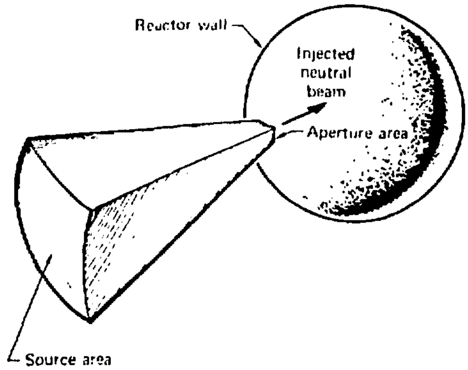


Fig. 2. Injector configuration.

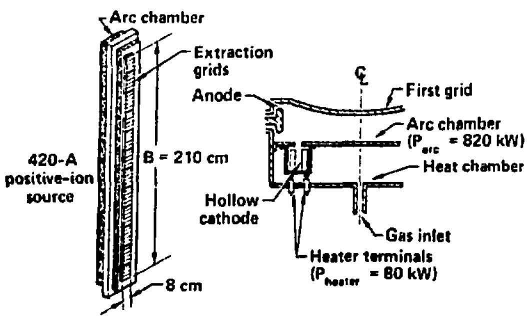


Fig. 3. Positive-ion source. Inset shows design of hollow cathode.

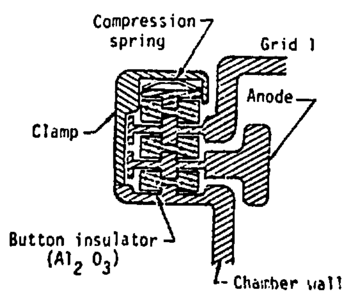


Fig. 4. Anode Mounting Detail.

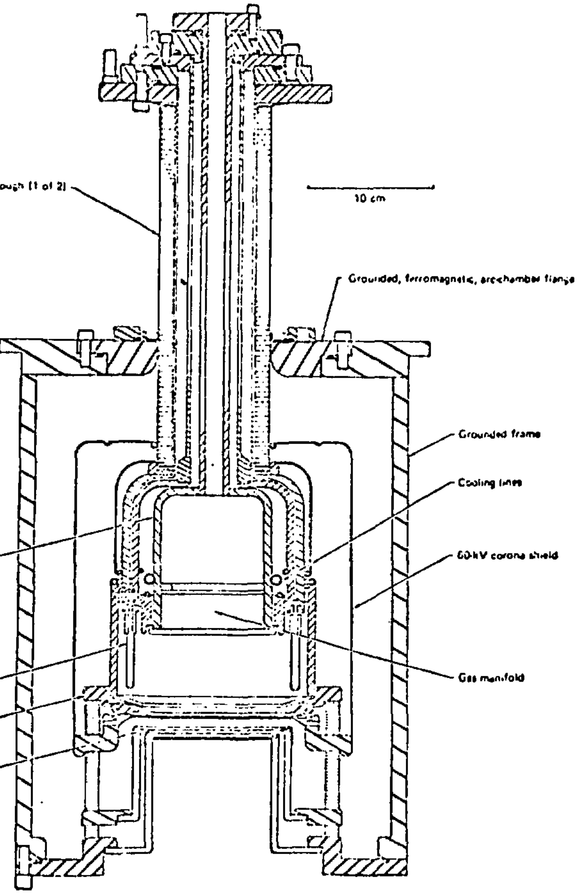


Fig. 5. A compact, 80-keV neutral-beam module.

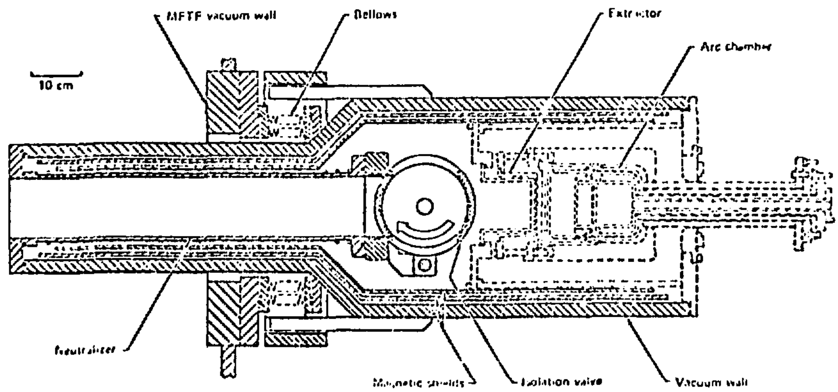


Fig. 6. The outer magnetic shield forms the vacuum wall around the arc chamber, extractor, and cylindrical isolation valve. The arc chamber is mounted off the grounded back plate by two triaxial feedthroughs. The extractor is supported by a grounded frame from an annular backplate. The isolation valve is mounted on the neutralizer tube, which is cantilevered from the exit end.

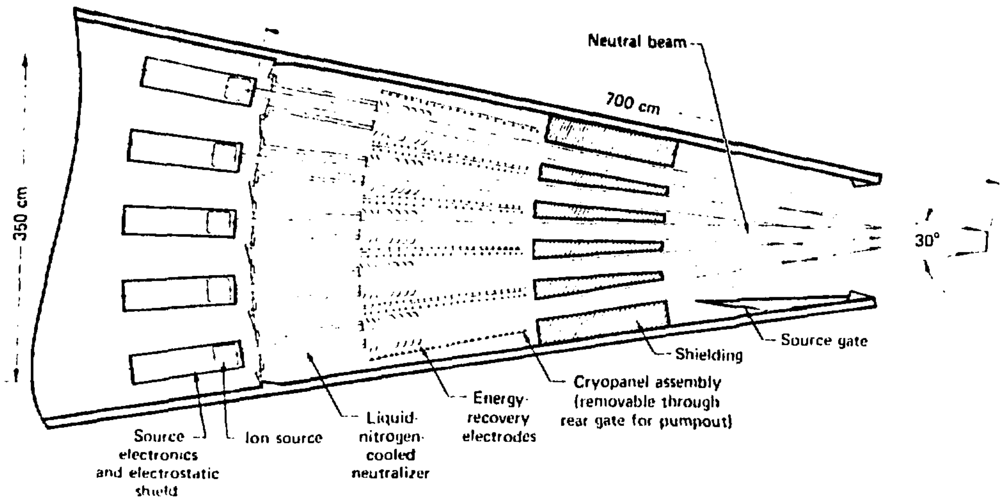


Fig. 7. A 225-MW neutral-beam injector delivering a mixture of 100-keV deuterium atoms and 150-keV tritium atoms.

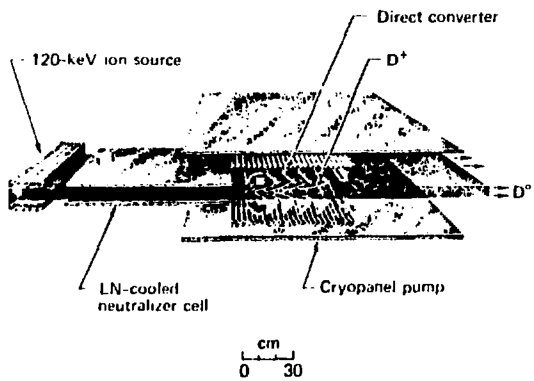


Fig. 8. 100 keV neutral-beam injector beam line.

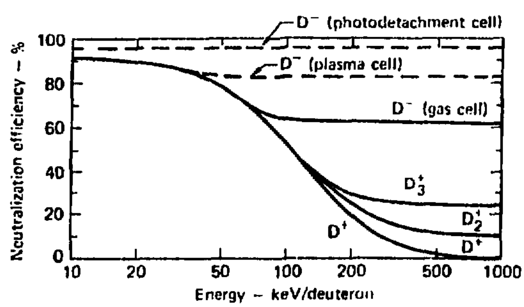


Fig. 9. Optimum neutralization efficiency of a deuterium-gas cell as a function of the energy of an incident deuterium atom.

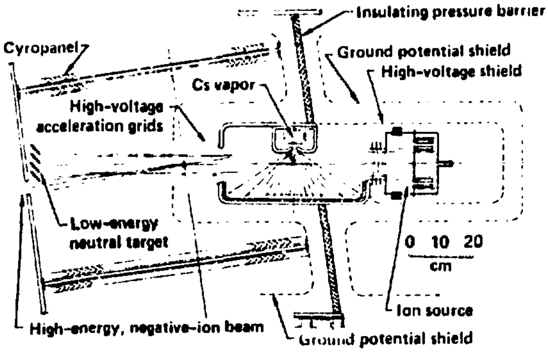


Fig. 10. Negative-ion injection module delivering 84 A of 150-keV D^- ions.

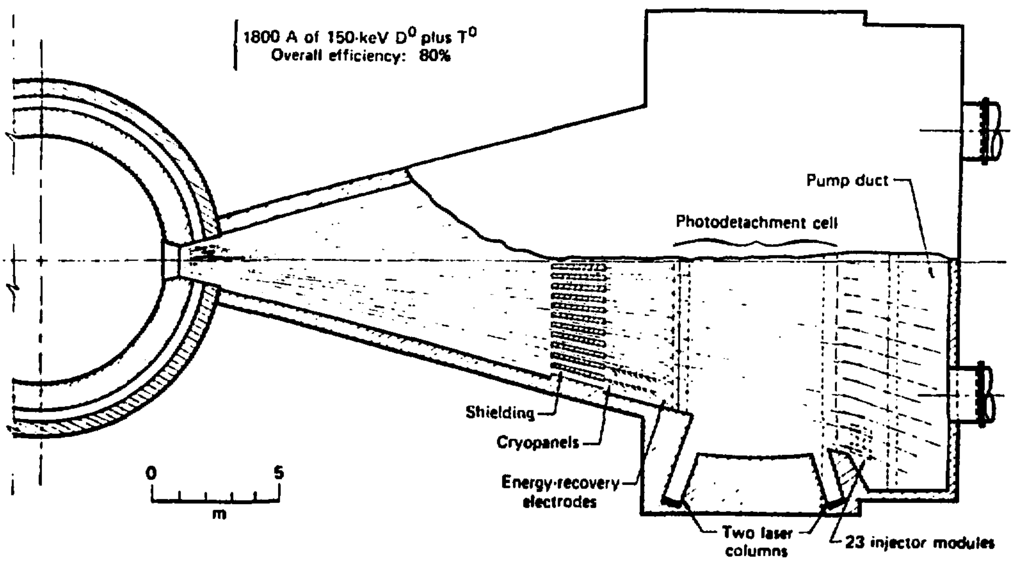


Fig. 11. Neutral-beam injector delivering 1800 A of 150-keV deuterium and tritium atoms.

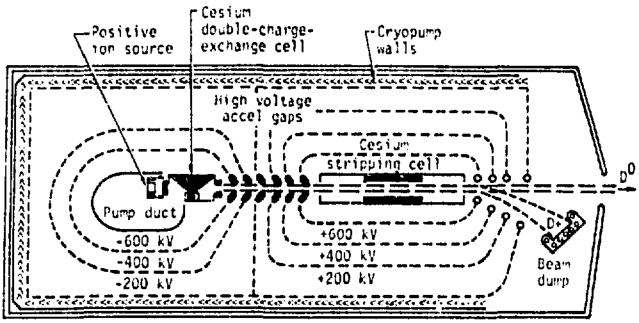


Fig. 12. A 1.2 MeV neutral-beam injector.

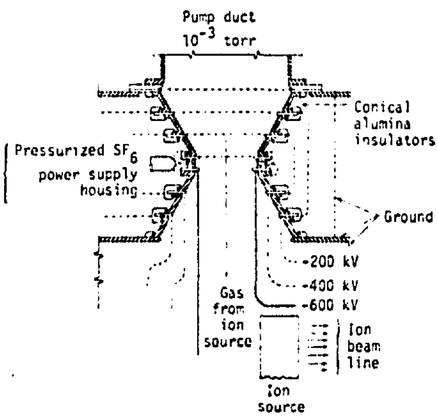
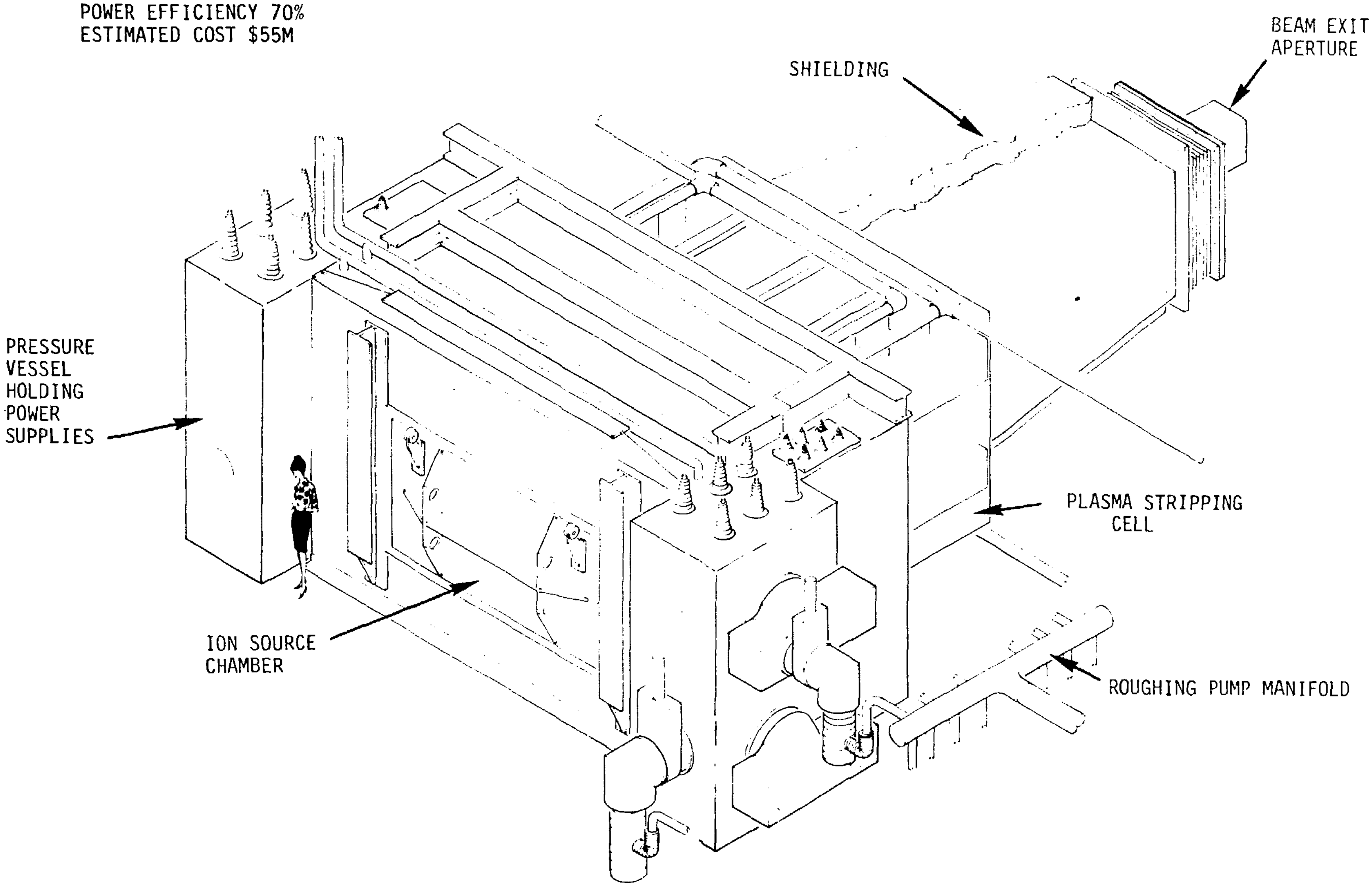


Fig. 13. Pumping through the high-voltage insulators.

CONCEPTUAL DRAWING OF A
120 MW, 1.2 MeV D^0 INJECTOR
POWER EFFICIENCY 70%
ESTIMATED COST \$55M



PRESENT AND FUTURE TECHNOLOGY OF HIGH VOLTAGE SYSTEMS FOR NEUTRAL BEAM INJECTORS

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INTRODUCTION

The intent of this paper is to present:

1. a brief review of existing neutral beam (NB) power supply technology for operating up to $\sim$200 kV, 65 A;
2. possibilities for using existing systems for next-generation NB sources, and associated problems;
3. a summary of the features of present systems which contribute to a high degree of complexity and/or cost;
4. a plea and proposal for minimizing cost and complexity of future systems operating up to $\sim$300 kV;
5. a few comments pertaining to special problems associated with operating in the 300 to 1000 kV range; and
6. a listing of some specific task areas which we believe should receive early R and D effort.

*Work done under the auspices of the United States Department of Energy.

EXISTING NB POWER SUPPLY TECHNOLOGY

General System Description

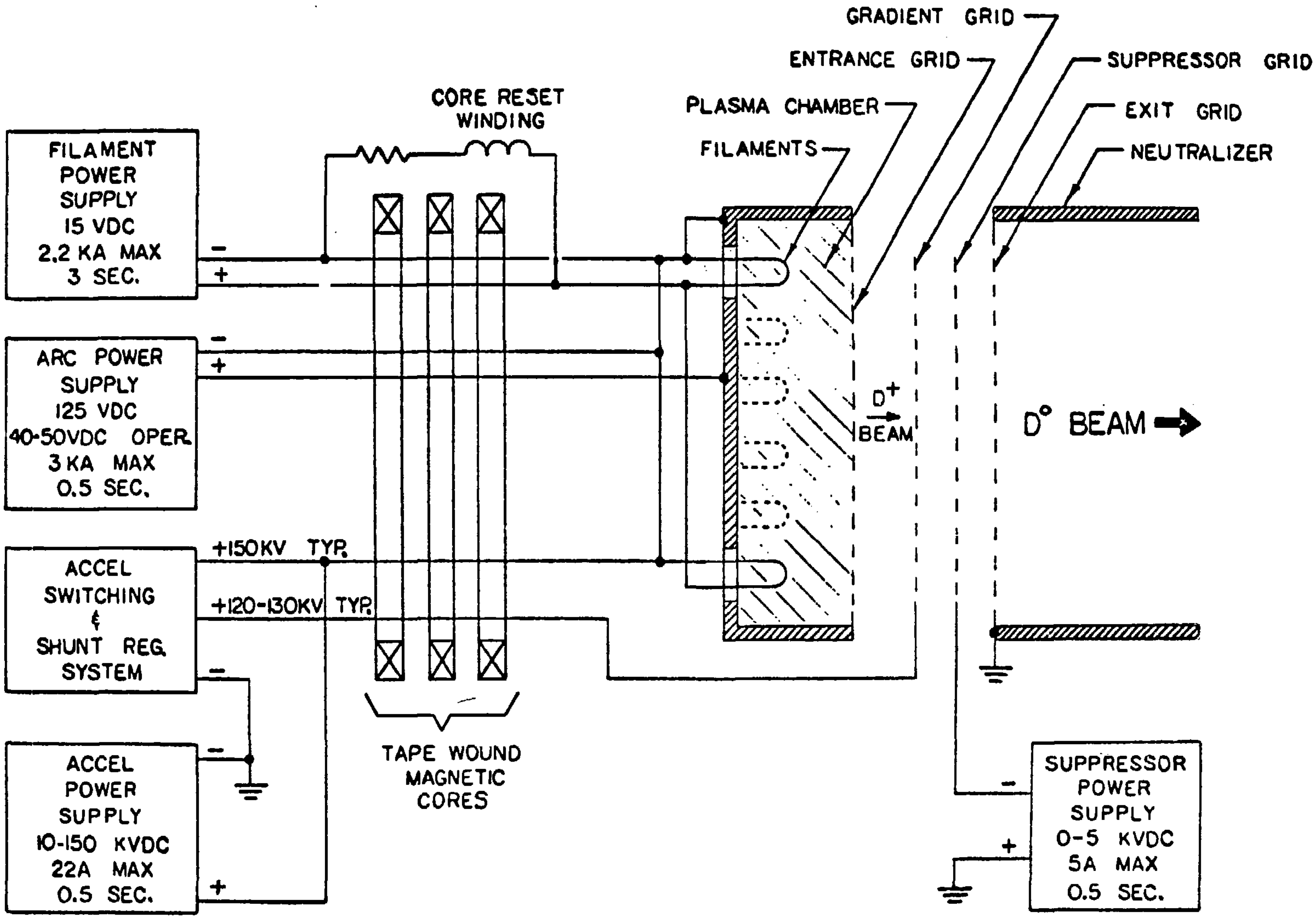
Figure 1 shows a simplified diagram of a NB source and the power supply system it requires. It happens to pertain specifically to the LBL 10 x 10 cm (extraction aperture) NB source and the 120-kV, 20-A, 0.5-sec Test Stand III A facility.[1,2] Our intent here is not necessarily to emphasize or advocate any NB source and power system approach over another, but rather to discuss general principles.

The arrangement shown in Fig. 1 is typical for many positive ion extraction systems presently in use. As can be seen, the plasma chamber is operated at high positive accel potential and requires arc and filament power supplies insulated for the full accel voltage. Note also that the suppressor power supply (for reflecting back-streaming electrons) is ground-referenced and that the neutralizer is grounded.

Various approaches being pursued at several locations differ in NB source design, in the magnitude of filament and arc currents required, and in the details of switching and regulating the accel power supply. (The latter is discussed more fully in a later paragraph.) For LBL/LLL-type NB sources, the tape wound magnetic cores indicated in Fig. 1 have been found to be decidedly beneficial to successful operation for accel voltages as low as 20 kV and absolutely essential at voltages above 40 kV. Since this issue is increasingly important as accel voltage is raised, it is discussed in some detail in the next paragraph.

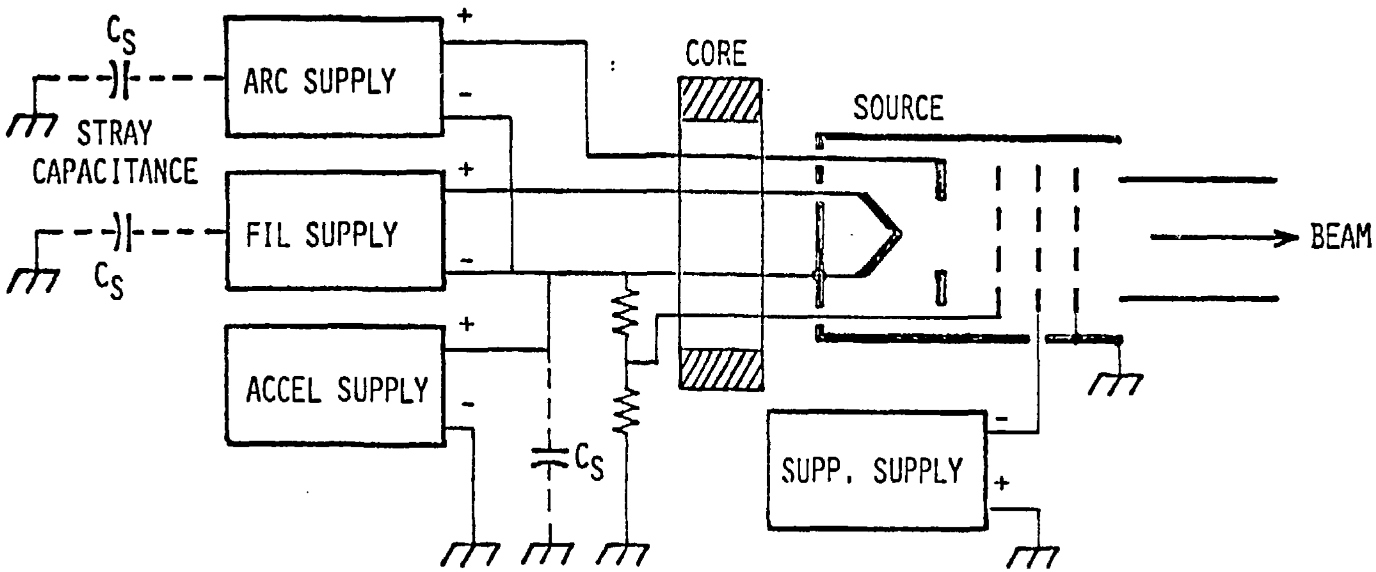
The Stored Energy Problem

Figure 2 shows a simplified diagram of a system indicating the stray capacitance to ground from points at full accel potential.

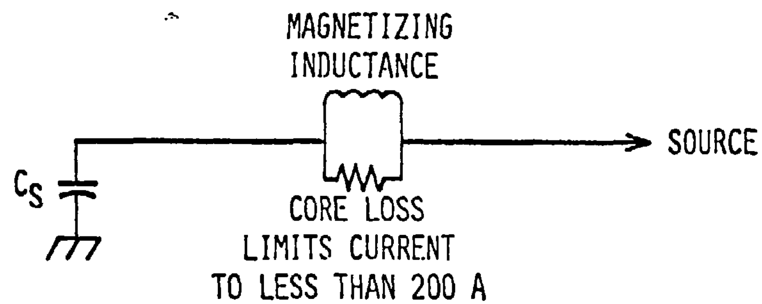


XBL 766-1996

FIGURE 1



EQUIVALENT CIRCUIT



CORE TECHNIQUE TO ABSORB STRAY-FIELD ENERGY

XBL 7710-10076

FIGURE 2

Without the core snubber, a spark in the NB source would rapidly ($\ll 0.1\,\mu$sec) discharge these capacitances. A high peak current would flow (> 1 kA) and most of the energy stored in the stray capacitance would be dissipated in the NB source spark and at electrode surfaces, causing pitting. Structures operating in high vacuum have been found capable of tolerating sparking where the stored energy dissipated is of the order of 50 Joules. The initial discharge in a vacuum device such as a tube is space-charge limited for the first 1 to 2 μsec.[3] The energy deposited on the anode during this phase is less concentrated so that electrode damage is very little. On the other hand, when sparking occurs in structures operating in a low-pressure gas environment, e.g., NB sources, a low impedance arc can be established in a very short time, perhaps < 50 nsec, resulting in equally rapid voltage collapse and high peak currents. The process is totally different from that of a vacuum spark and has been observed to occur in two different modes in LBL-type NB sources. The first is a "soft" high-impedance spark which may exhibit a voltage drop of > 1 kV and take $\gg 1\,\mu$sec to develop. The second is the "hard" spark mentioned above which can develop in < 50 nsec when current is allowed to exceed ~ 200 to 300 A. The "hard" sparks have been found to cause rapid deconditioning of LBL sources, apparantly by excessively pitting electrode surfaces. A preliminary study by one of us (W.R.B.) indicates that for currents $\gtrsim 200$ A, conditions might be approximately correct for establishing a high current-density "pinch"-type discharge which could cause such pitting.

LBL tests performed at 20, 40, and 120 kV indicate that the maximum tolerable energy stored in stray capacitance decreases with voltage and is only 3 to 5 J at 120 kV. At this voltage, the stored energy of the LBL source itself is approximately 1 J. Since capacitive energy increases as the square of the voltage, it is evident that new techniques will be required for successfully operating these large sources at voltages much in excess of $\sim$150 kV. Because of the energy stored in the source itself, it is obviously difficult to scale the results from one type of source to another. A source geometry tested successfully in a fractional-size source may not perform identically or successfully in a full-size version.

Referring to Fig. 2 again, the function of the magnetic core snubber is to limit the peak current during a NB source spark to $\lesssim$200 A, limit the energy delivered to an NB source spark to a few Joules, and to directly absorb the remainder of the stray capacitance energy in core losses governed by time-dependent eddy-current and hysteresis processes.[4] There are still unknowns associated with stored energy. For example, a recent test at LBL showed that at 120 kV, a capacitively stored energy (on the power supply side of the core snubber) of $\sim$25 J was barely tolerable while $\sim$40 J was intolerable; i.e., it caused rapid source deconditioning so that it no longer would hold voltage. In principal, it would seem possible to design a core snubber to "handle" <u>any</u> value of "upstream" capacitance. In practice, the above test indicates there is a limit, for unknown reasons, even when simulating a core snubber properly designed for the increased capacitance.*

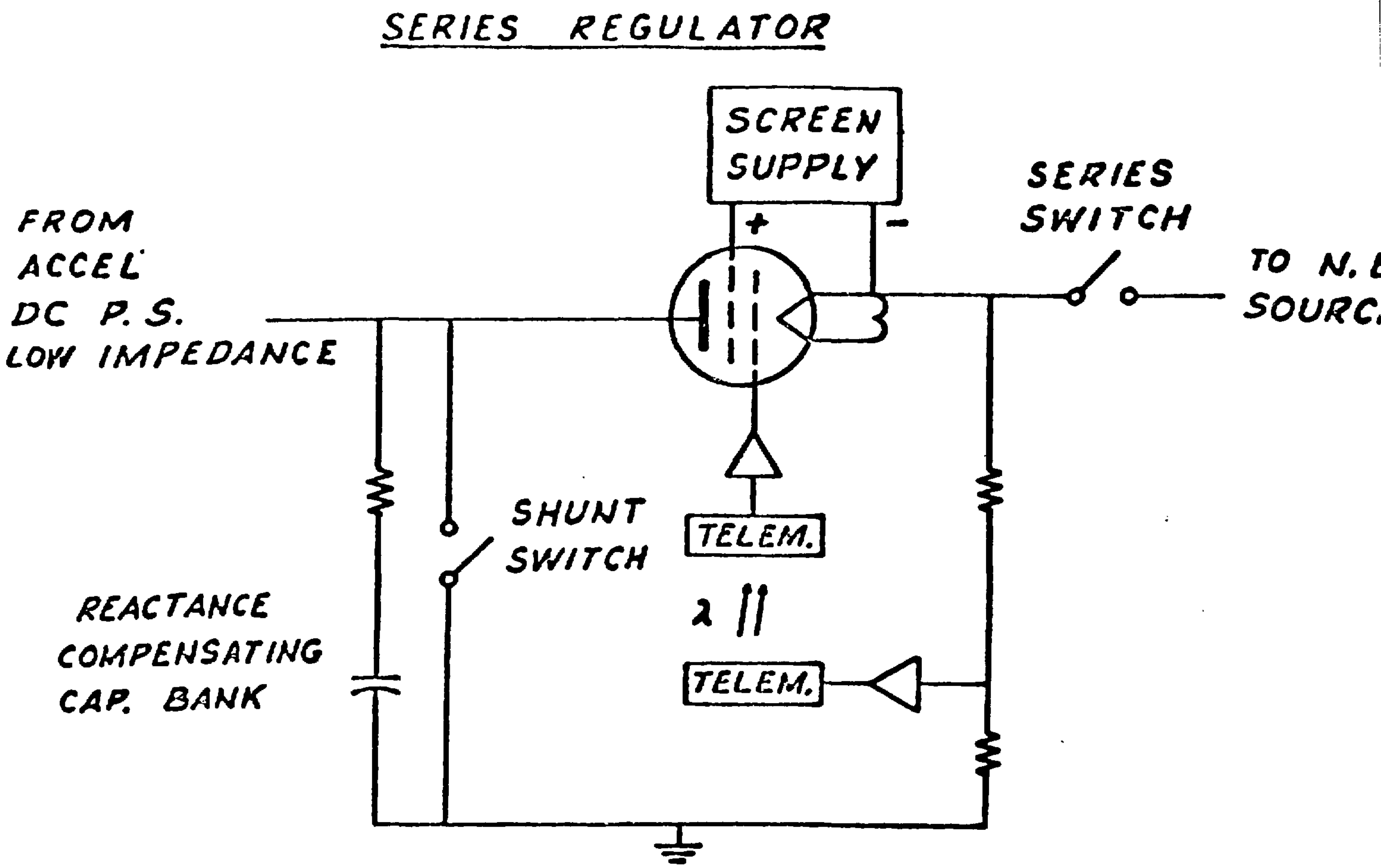
*It is likely that stored energy >40-50 J could be tolerated if, <u>in addition</u> to the core snubber, a fast-acting ($\sim$2 to 3 μsec) crowbar is provided and triggers on a source spark.

Certainly one limit is the fact that the snubber itself has stray capacitance to ground which increases with its size, obviously setting a practical limit on its size. These limits should be studied and defined, and thorough, systematic measurements of the effects of stored energy on NB sources should be made so that intelligent design of future NB systems can proceed.

Accel Power Supply Switching and Regulating

At the multimegawatt level of present NB systems, the extracted ion beam must be properly focussed so that only a negligibly small percent of the beam is permitted to intercept other NB source electrodes; otherwise, extreme heating, sputtering, sparking, and possibly damage to the source occurs. Until recently, the risetime of the application of accel voltage to LBL NB sources has been made short ($\lesssim 20\,\mu$sec). This minimizes the time period during which electrode voltages are improper and incorrectly focussed extracted beam is "spraying" electrodes. Also, to date, it has been necessary to have the accel voltage well regulated (to $\sim 1\,\%$, typically) to minimize neutral beam energy variations which would result in a variable beam divergence and possible beam interception.

Figure 3 shows simplified diagrams of the two most-used types of accel regulating systems now in use at various locations. The series regulator approach is in use at LBL for systems operating up to 40 kV, 80 A, 10 msec,[5] and being designed for operation up to 120 kV, 70 A for the Princeton TFTR systems,[6] for 120 to 150-kV operation at ORNL,[7,8] and for 80 to 200-kV operation in the LLL High Voltage Test Stand.[9]


SERIES REGULATOR
SCREEN SUPPLY
FROM ACCEL DC P.S. LOW IMPEDANCE
SERIES SWITCH
TO N.E SOURC
SHUNT SWITCH
TELEM.
TELEM.
REACTANCE COMPENSATING CAP. BANK

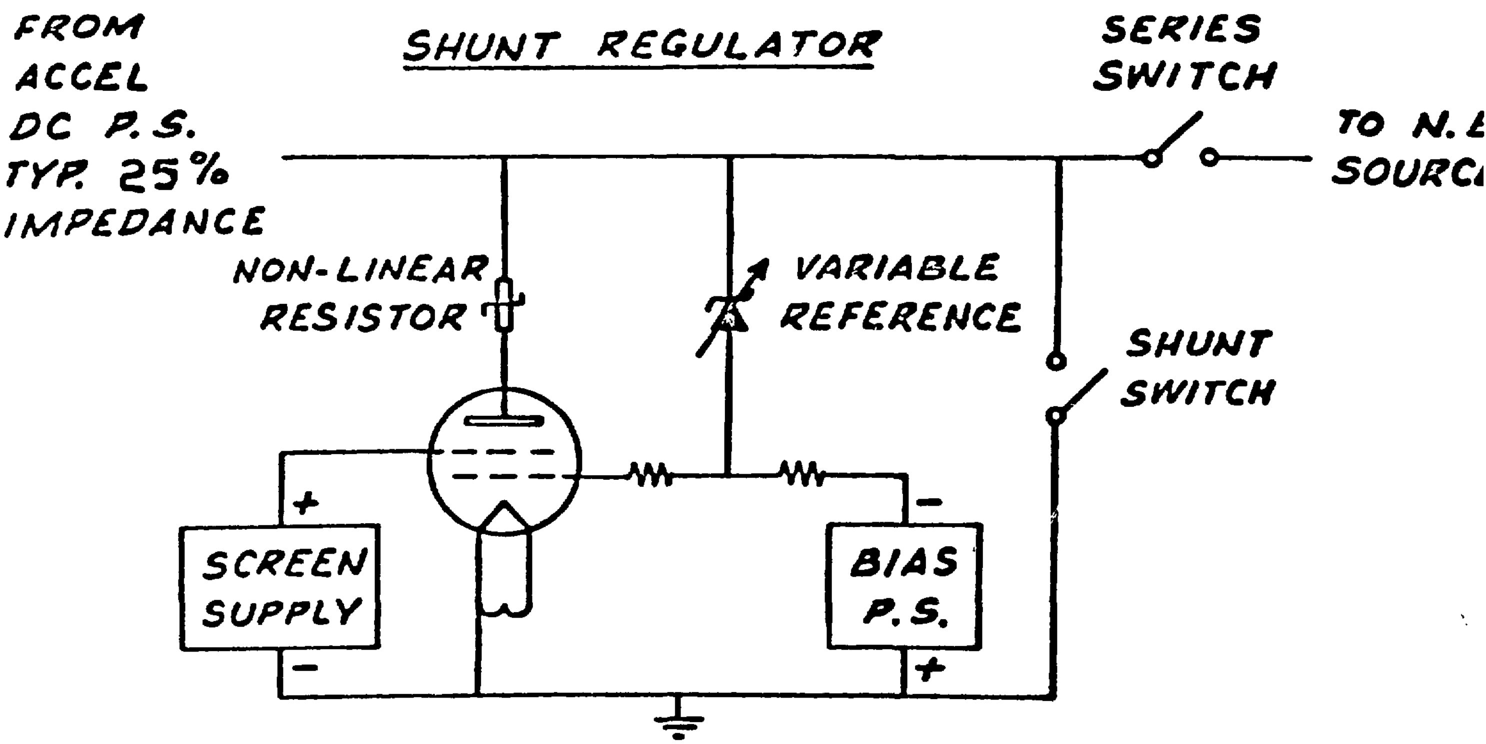
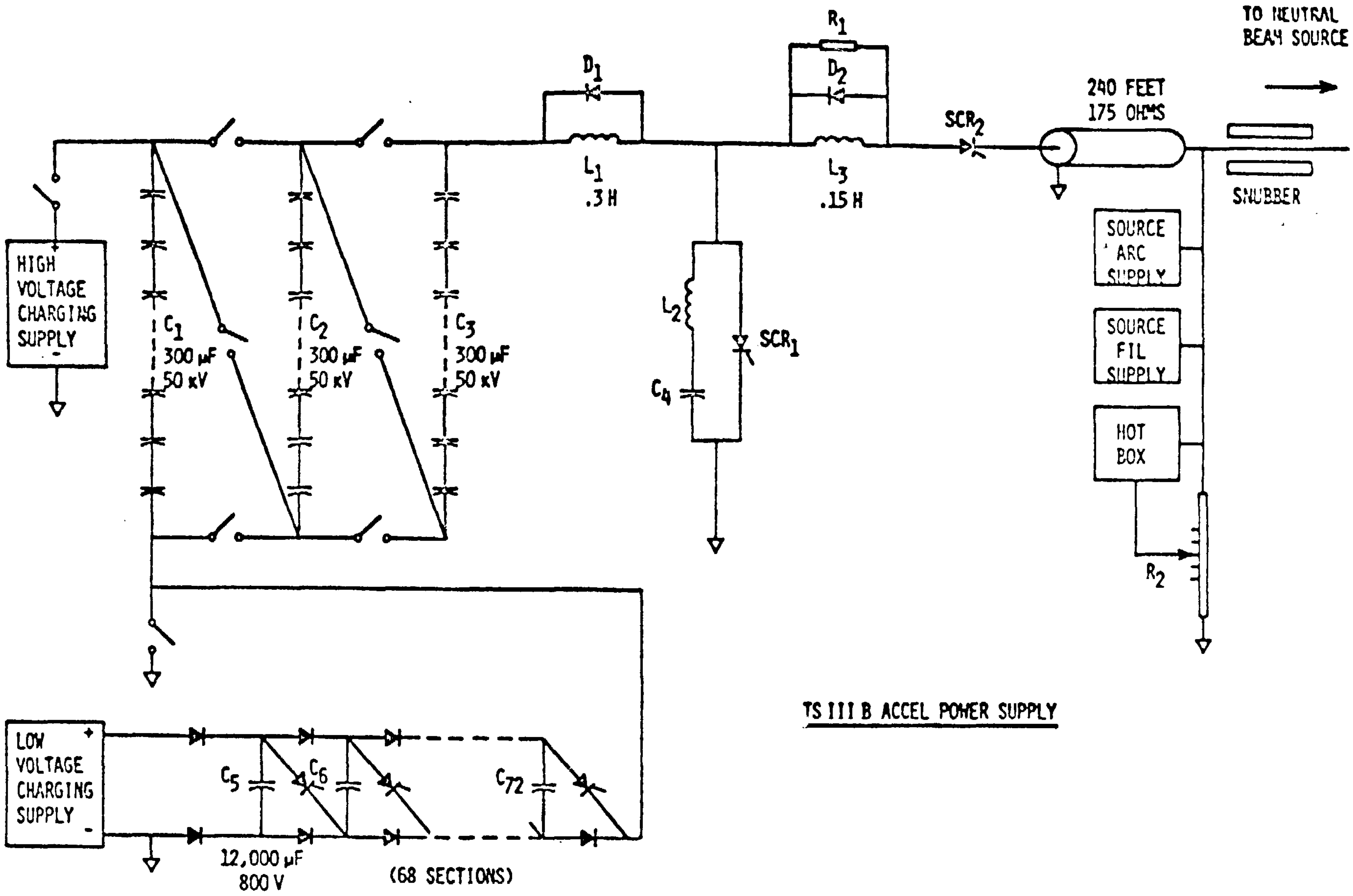

SHUNT REGULATOR
FROM ACCEL DC P.S. TYP. 25% IMPEDANCE
SERIES SWITCH
TO N.E SOURC
NON-LINEAR RESISTOR
VARIABLE REFERENCE
SHUNT SWITCH
SCREEN SUPPLY
BIAS P.S.

FIGURE 3

The shunt regulator approach is fully operational in the LBL 120-kV, 20-A, 0.5-sec Test Stand III A facility,[2] and will be used in the LBL 120-kV, 65-A, 0.5-sec NBSTF facility for TFTR NB source development and testing.[10] The 200-kV high voltage switch tube, under development by both RCA and Eimac, is intended to serve as a combined series switch and regulator in the Princeton TFTR and LLL High Voltage Test Stand power supply systems. In the LBL 120-kV test facilities, shunt regulation is provided by parallelling existing Machlett DP-15 triodes with nonlinear plate resistors so that the tube anode voltage is always $< \frac{1}{2} V_{accel}$. This system requires a separate series switch so that the full current and voltage can be first established in the shunt regulator before switching to the NB source with a fast risetime. Series assemblies of silicon controlled rectifiers (SCR's) have been developed[11] for this purpose and are in routine use in LBL test facilities.

Two other configurations should also be mentioned. The first is an ORNL series switching and regulating system, nearing completion, which has three 60 kV floating decks in series to provide $\sim$150-kV, 50-A pulses.[8] It is our understanding that this system is now undergoing initial tests in a switching mode, only. The second configuration of note is that developed by H.M. Owren at LBL for $\sim$50-msec pulse testing of the TFTR NB sources at a 120-kV, 65-A level. Figure 4 shows a simplified diagram of this 1.2-MJ electrolytic capacitor bank stored energy system. It has three main capacitance modules, shown in Fig. 5, which are charged in parallel to 40 to 50 kV, then mechanically switched in series. A series SCR assembly provides fast-



TO NEUTRAL BEAM SOURCE
240 FEET
175 OHMS
SNUBBER
R_1
D_1
D_2
L_1
.3 H
L_3
.15 H
SCR_2
SCR_1
L_2
C_4
HIGH VOLTAGE CHARGING SUPPLY
C_1
300 µF
50 kV
C_2
300 µF
50 kV
C_3
300 µF
50 kV
SOURCE ARC SUPPLY
SOURCE FIL SUPPLY
HOT BOX
R_2
LOW VOLTAGE CHARGING SUPPLY
C_5
C_6
C_{72}
12,000 µF
800 V
(68 SECTIONS)
TS III B ACCEL POWER SUPPLY

FIGURE 4

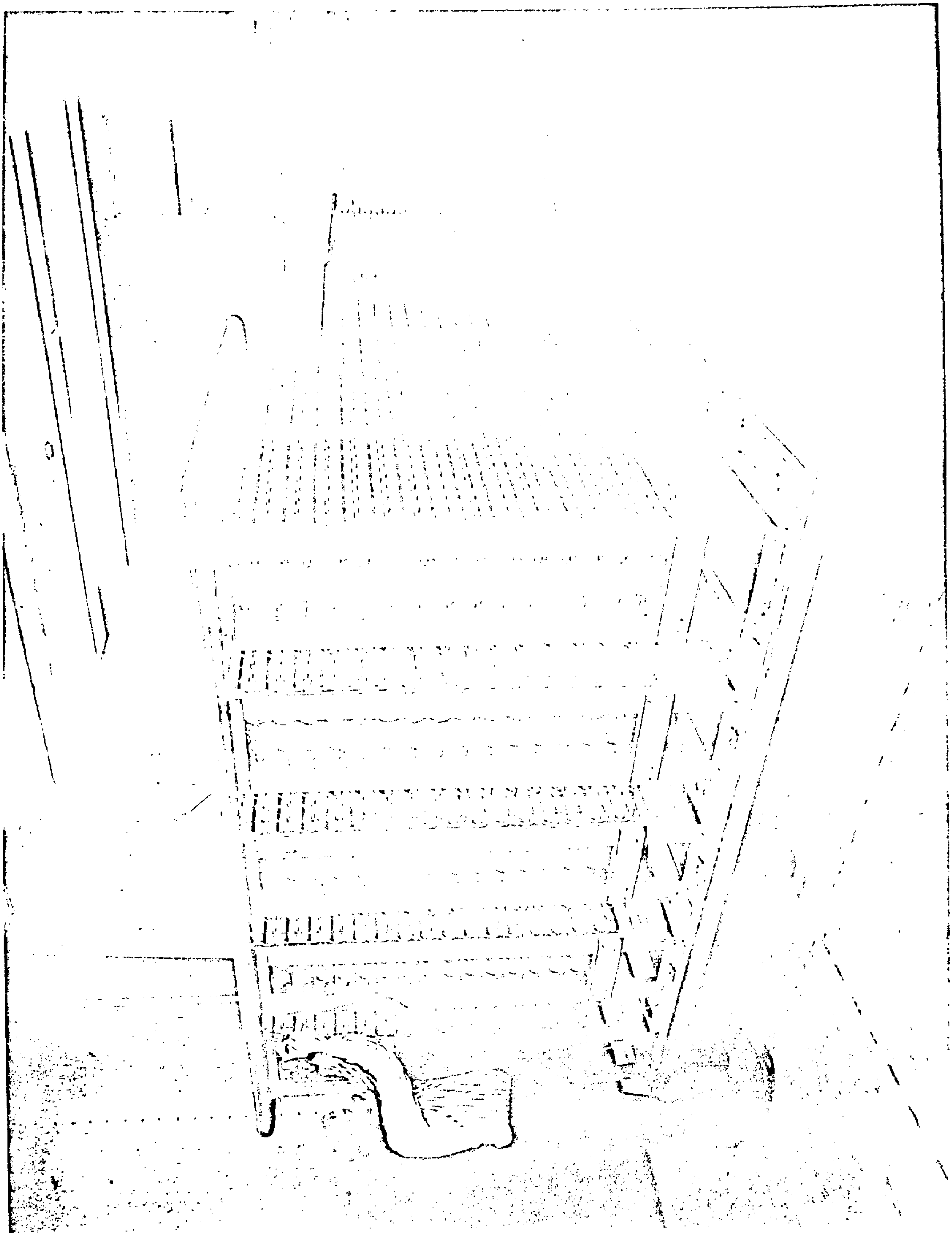


FIGURE 5

risetime switching to the NB source. In order for this to operate, a one-Henry inductor and a free-wheeling diode in shunt with it provides essentially constant-current from the bank during the SCR commutation. A shunt SCR assembly and LC commutating network permit interrupting the accel current during source sparkdown. The SCR and LC assemblies are shown in Fig. 6. A unique feature of this system is its ability to regulate (i.e., flat-top) the output voltage, while the main capacitor bank voltage is sagging, by sequentially switching in series up to 68 low voltage charged capacitor increments in the "low" lead of the main bank. This results in efficient usage of the main bank energy; as much as 50% is delivered to the NB source during a pulse.

USE OF EXISTING POWER SUPPLIES FOR FUTURE REQUIREMENTS

E.B. Hooper of LLL has spoken of the desirability of using existing equipment as much as possible in future projects in the interest of minimizing costs and construction time. Time and space do not permit a detailed study or survey of the possibilities for extended usage of existing power supply systems, but the following brief comments may be of benefit. We assume the trend will be toward higher voltages, possibly negative output, possibly bipolar output, longer pulse widths, and higher duty factor.

Larger Pulse Widths; Duty Factors

From 0.5 sec on to pulse widths of $\sim$ 30 sec, nearly all transformers should be limited only by thermal considerations. Therefore, such longer pulses are likely to be possible. Because of the long thermal time constants of large transformers, however, the duty factor

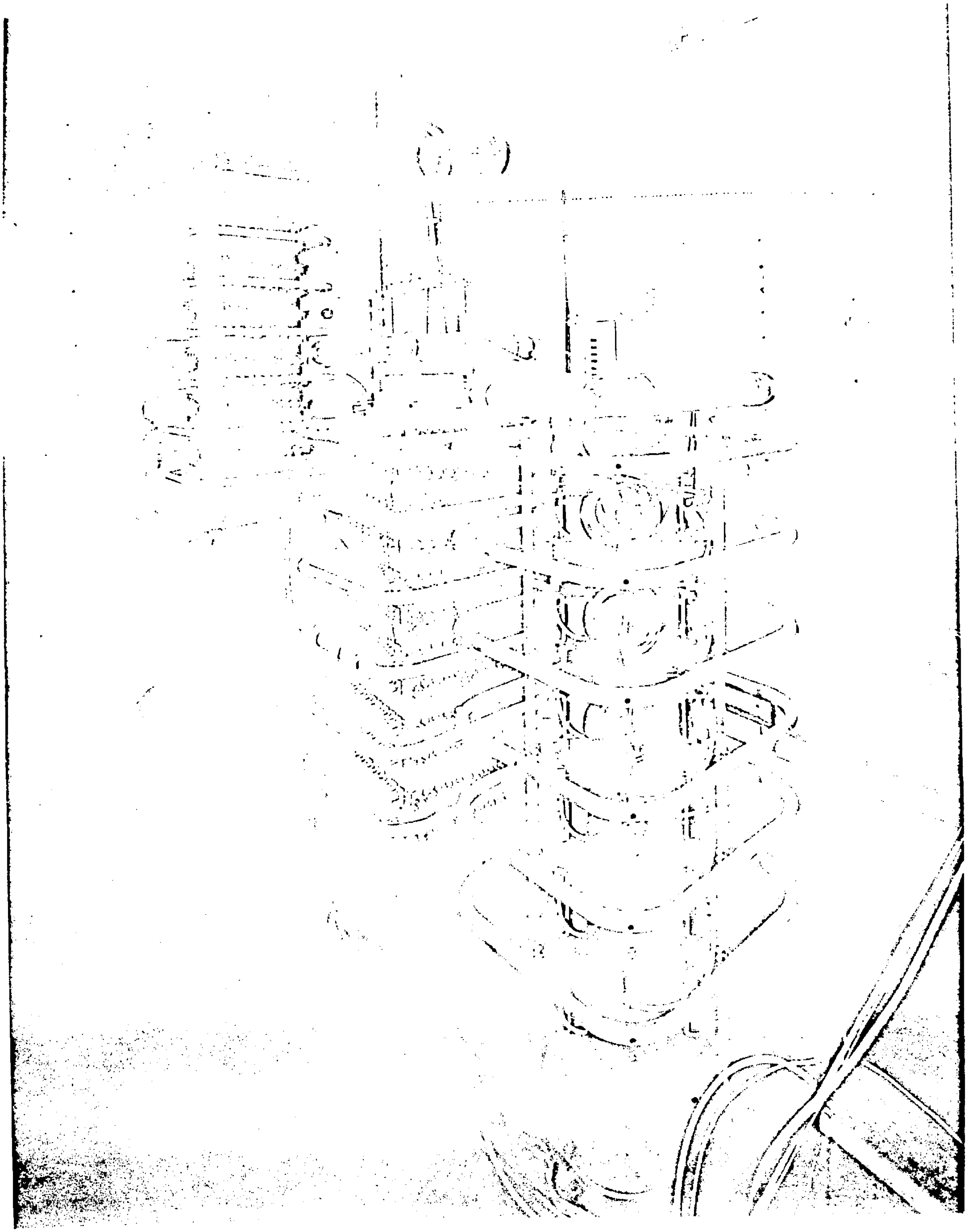


FIGURE 6

may need to be reduced. Alternatively, a short series of pulses may be permitted if this is followed by a relatively long cooling period. The addition of cooling coils, heat exchangers, oil agitators, and/or forced flow systems to existing tanks could alleviate this problem if it becomes a limiting factor. In any case, monitoring the temperature of such components is in order.

Solid state rectifiers and SCR's designed for 0.5-sec operation are operating at nearly their dc ratings. Except for possible required cooling changes, no problems at longer pulse widths should exist. Tubes, and particularly SCR;s appear to be a better choice for series switches used in long pulse operation, e.g., 30 sec, than ignitrons or thyratrons. Ignitrons have limited coulomb ratings, typically $<$ 300 A-sec/pulse. Thyratrons for such service are beyond the present state of the art. Additionally, they would have cathode life limitations.

Some auxiliary power supplies and components such as power resistor dividers for gradient grid supplies may require modification and additional cooling.

Finally, systems having tubes and other dissipative components such as linear and nonlinear resistors, water resistors, etc., which have been designed for a shorter pulse will usually require either a full rebuild with increased cooling, additional parallel elements, or a drastic change in the system configuration or operating procedure (see section below on Simplifying Future Systems).

Higher Current

Higher voltage systems are expected to require accel currents no larger than present rated currents. Required accel current may even be less than present rated currents where a limit on available power exists.

Negative Output

Usually it is relatively easy to reverse connections to rectifiers in order to obtain a negative power supply output for negative ion extraction. It also appears to be relatively easy to reconnect either a series or shunt regulator and associated switching and crowbar systems for negative output. Minor changes in some areas may be required; e.g., inverting voltage dividers, relocating current monitors, modifying ingitron trigger circuitry, and insulating the shunt regulator tube filament transformer for high voltage. Some of the tube control electronics may also require modifications.

Bipolar Output

Some conceptual designs for future NB systems based on negative ion sources require bipolar power supplies. The negative-output supply described above can be combined with a standard positive-output supply to produce such a system. A preliminary study of this configuration reveals no major problems.

If the NB sources have a well-defined ground point, say a grounded intermediate grid as described by E.B. Hooper, it should be possible to prevent long-duration coupling of the two power supplies during a source spark which could otherwise develop troublesome twice-voltage levels. Short-duration overvoltage transients may be produced but these are easily clamped and damped by surge suppressors. LBL has successfully and routinely used many series-assemblies of the relatively new metal-oxide varistors (nonlinear resistors) for this purpose. Larger units with energy ratings in the kJ range are just now appearing in the commercial market for use in power distribution surge arrestors.[12]

TABLE I

SOME LBL/LLL P.S. CAPABILITIES FOR FUTURE USE

LBL	TEST STAND IIIA	+ 170-kV, 20-A, 0.5-SEC ± 80-kV, 20-A, 0.5-SEC
LBL	N.B.S.T.F.	± 170-kV, 16-A, DC ± 150-kV, 32-A, 30-SEC
LBL	1.3-MJ CAP. BANK (UPGRADED)	(+ OR -) 1-MV, 20-A, 30-mSEC (+ OR -) 500-kV, 40-A, 60-mSEC
LLL	H.V.T.S. + 80-kV P.S.	± 80-kV, 80-A, 30-SEC -200-kV, + 80-kV, 20-A, 30-SEC

Coordinated control of the two power supplies is obviously necessary. The regulating, switching, and crowbarring functions all appear to be relatively easily implemented by control through the reliable optical coupling links now available. These can be operated in such a manner as to permit the checking of their proper operation before a pulse, and inhibiting firing if any are malfunctioning. If regulators are used, an additional feedback link will probably have to be added if close tracking of the outputs is required.

With bipolar output power supplies, overall voltages up to 340 kV are obtainable from supplies now built or being constructed. The LBL NBSTF (Neutral Beam System Test Facility) transformer-rectifiers would be capable of supplying $\pm$ 170 kV, 16 A dc, or $\sim\pm$ 150 kV, 32 A for 30-sec pulses. E.B. Hooper has stated that, at LLL, the High Voltage Test Stand power supply could be combined with an 80-kV, 80-A, 30-sec power supply now under construction to provide a range of bipolar outputs from $\pm$ 80 kV, 80 A, 30 sec to -200 kV, +80 kV, 20 A, 30 sec. These capabilities are summarized in Table I. The capacitor bank is discussed in a later section.

COSTLY, COMPLEX FEATURES OF PRESENT POWER SUPPLIES

Table II lists the features of existing power supplies which contribute significantly to high cost and complexity (over and above the required transformer-rectifier systems). Several of these features are probably with us for some time; e.g., magnetic core arc snubbers, computers, and the need for repetitive interrupts and low available stored energy in the accel system stray capacitance. All of the others, however, appear to show promise for eventually reducing, relaxing, or eliminating altogether. The potential for greater simplicity and

TABLE II

FEATURES CONTRIBUTING TO HIGH COST AND COMPLEXITY

Hardware		Prime Candidate for Simplification
regulators		*
tubes		
power varistor assemblies		
fast high voltage switches		*
tubes		
SCR assemblies		
crossed-field interrupters		
magnetic core arc snubbers		
computers for control and diagnostics		
Operational Requirements		
fast-rise switching	(typ. < 25 μsec)	*
tight regulation	(typ. < 1 %)	*
wide ranging	(typ. 5 to 100%)	*
low available stored energy	(typ. < 5 J)	
repetitive interrupts		
large number of short circuits		?
small voltage-selection resolution		*
"load" rather than "no load" tap-changing of transformers		*

reliability is high, while the possible cost savings in the many systems being planned for construction in the next several years is enormous.

Although it would be informative to have specific cost impact figures for the above features, it has been our experience that it is nearly impossible to obtain and present these in any kind of self-consistent manner which is not fraught with possibilities for misinterpretation. Accounting procedures vary from location to location. Private industry costs differ greatly from those in laboratories. Cost division between "Operating" and "Capital" accounts vary from project to project. Overhead, escalation, assignment of contingency, and the effects of various managerial decisions further confound the issue. It would certainly be beneficial if there were standard procedures, used by fusion research laboratories at least, for cost estimating and accounting.

The present dual, and different, funding procedures for "Capital" and "Operating" accounts leads to arbitrarily different divisions of funding at different laboratories, and also between various projects within a given laboratory. Although "Capital" funds committed in one fiscal year can be spent in the following year(s), the expenditure of "Operating" funds must cease on the last day of the fiscal year. This has always promoted inefficiency and, when projects have underspent, wasteful near-panic spending at the end of the fiscal year. On the nationwide fusion research spending scale, this effect could easily account for tens of millions of dollars' worth of ill-conceived spending. Since it is possible to carry Capital expenditures over into the next fiscal year, it should be possible to conceive a system which has only one category of funding and the ability to carry expenditures over into the next fiscal year for some reasonable period.

SIMPLIFYING FUTURE SYSTEMS FOR OPERATION TO $\sim$300 kV

In future systems, a close look should be taken at the need for small voltage selection resolution and wide range control of accel systems. Instead of each system being fully variable, it may be acceptable to have one or more fully variable "conditioning" power supplies for initial testing and conditioning of NB sources. If the remaining on-line "production" systems only require a small operating voltage range, or a few selected fixed operating points, a very large cost savings in the power conditioning transformers would result. Moreover, since most of the repetitive short circuits would occur while conditioning, the rated number of short circuits specified for the "production" system transformers could probably be in the industry-standard range, rather than being abnormally high and non-standard as is the case with some systems under construction. This would also result in large cost savings.

Simply settling for changing variable voltage transformer taps between pulses, under "no-load", rather than specifying "load" tap changing can result in large savings for these costly components. Settling for coarser voltage selection resolution, say 7 to 10% steps rather than continuous variation, can also result in large savings.

We now come to a discussion of the area having the greatest potential for simplifying future systems and saving a great deal of money, perhaps tens of millions of dollars over the next 5 to 10 years. This subject centers on the need for accel voltage regulation and fast-rise switching.

If the NB source arc current were properly controlled so as to always maintain beam divergence at a minimum (e.g., by a feedback signal derived from the accel beam current), the accel voltage regulation requirement could be relaxed to the point where the neutral beam energy variations become unacceptable. Preliminary inquiries indicate that variations of at least $\geq 10\%$ are tolerable from the standpoint of fusion device physics. This implies that accel power supplies could be operated unregulated on nearly every conceivable ac power source as long as an appropriate arc modulator system is provided. Furthermore, since such a system can be made to maintain proper beam divergence over the full range of accel voltage, it should become possible to operate with reasonably long risetimes for the accel voltage, perhaps in the millisecond range. This, then, may make it possible to turn the accel power supply on and off with conventional contactors in the primary ac lines. These can also momentarily interrupt the ac primary when a NB source spark and subsequent crowbar occur, then re-energize the system to continue the pulse.

To date, some physicists have stated that only a few-msec period of "wrong-energy" injected beam is tolerable, e.g., that injected during a long accel voltage risetime. If this remains as a fixed limitation, it probably will not be possible to eliminate the need for fast high-voltage accel switching; i.e., tubes or SCR's would still be required. If it were found permissible to have a 10 to

$\sim$ 40-msec risetime, the series switches could likely be eliminated, saving a great deal of money. (An installed developmental High Voltage Switch Tube with its associated pure water system probably costs $150 to 200 k. An SCR system would probably cost about half of this when manufactured by private industry.) The resulting accel power supply would be extremely simple, involving as major components only primary contactors, voltage-variable transformer, transformer-rectifier, and dc crowbar. Preliminary studies indicate such a configuration is relatively easily adapted to pairing for bipolar outputs and/or running several smaller NB sources from one large power supply. This statement is also true if fast high voltage switching is still required.

DC arc modulators are already required in existing systems for depressing the arc current prior to switching on accel voltage. This causes beam focussing to be approximately correct during the $\sim$25-μsec risetime of accel voltage. Therefore, it would seem that by increasing the control and modulating capabilities of the arc power supply system (involving only 150 to 250 kW of power) the following significant advantages could be obtained for multimegawatt systems:

1. Accel voltage regulators can be eliminated.
2. Fast accel switching tubes or SCR;s can be eliminated.
3. Standard ac primary switchgear can be used for controlling the accel supply.
4. Fast-transient problems associated with fast switching can be eliminated.
5. Neutral beam divergence can automatically be maintained at a minimum over a wide range of conditions.

6. Cost and complexity can be greatly reduced.

7. Reliability can be greatly improved.

Table III summarizes the desirable features of simplified power supply systems.

An attractive technique for reliable and flexible control of the arc power supply is the use of a "star-point controller" at the neutral point of the wye-connected ac primary. One version of this circuit is the Kerns Actuator, shown in Fig. 7a, connected in the primary side of the rectifier transformer. This circuit makes it possible to satisfy an ac control requirement by a dc control means. In this case, the tube can provide easily variable full-range ac primary excitation as well as a fast turn-off function (typically << 1-msec). If the arc supply transformer has a 12 or 13.8-kV rated primary winding, a standard 10 or 250-kW tube could be used and would be conservatively operating within its ratings. Such variable control permits the correct matching of ion density for achieving minimum beam divergence during the accel voltage risetime as well as during the pulse.

In principle, a Kerns Actuator could be used to control the accel power supply; however, this may be impractical because of the large magnitude of power involved. A different arrangement of the circuit, shown in Fig. 7b, appears to have many advantages and intriguing possibilities for overall power supply simplification.[13] In this version, an inductor replaces the tube and SCR's (or ignitrons) replace the diodes. A varistor limits possible inductively generated overvoltage transients to safe levels. It achieves several simultaneous functions:

TABLE III

SIMPLIFIED FUTURE PS SYSTEMS

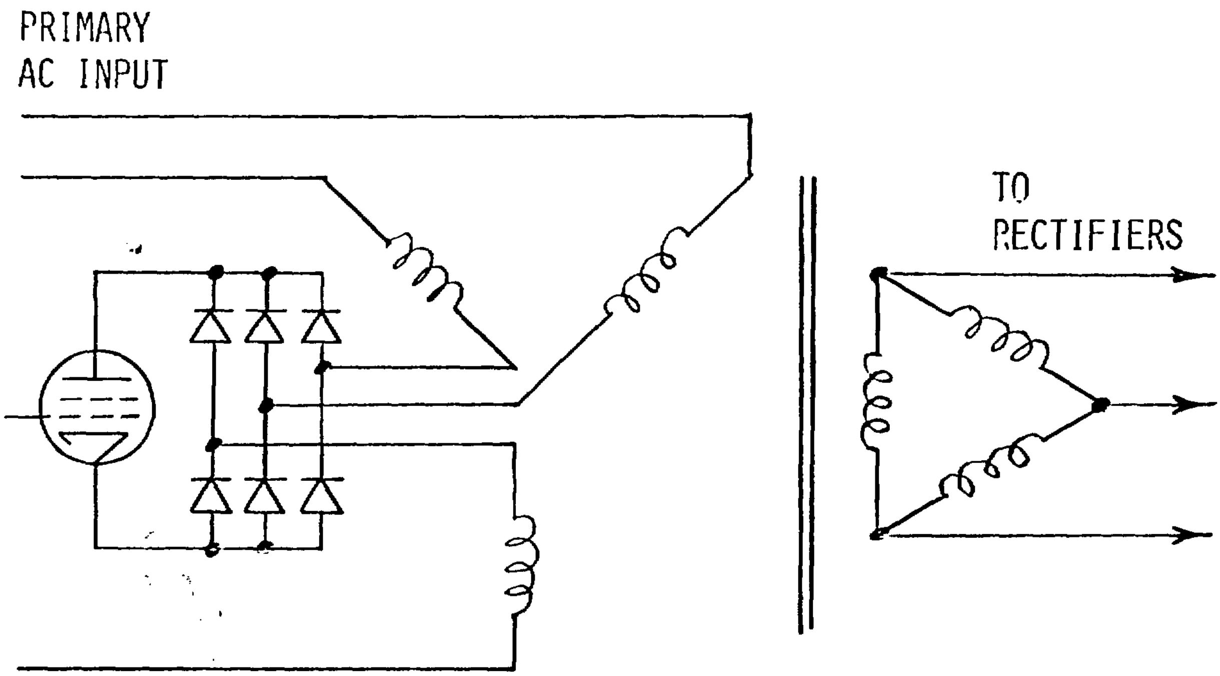
1. ACCEL RISETIME > 8-mSEC.
2. FULLY CONTROLLED ARC PS TO MATCH REQ'D ION DENSITY.
3. CONTINUALLY OPTIMIZED BEAM DIVERGENCE.
4. NO ACCEL REGULATOR.
5. NO FAST HV DC SWITCHING
6. OPERATES OFF STANDARD POWER LINES.
7. CONTROLLED BY STANDARD AC PRIMARY SWITCHGEAR.
8. ONLY MAJOR COMPONENTS:

   AC PRIMARY CONTACTORS

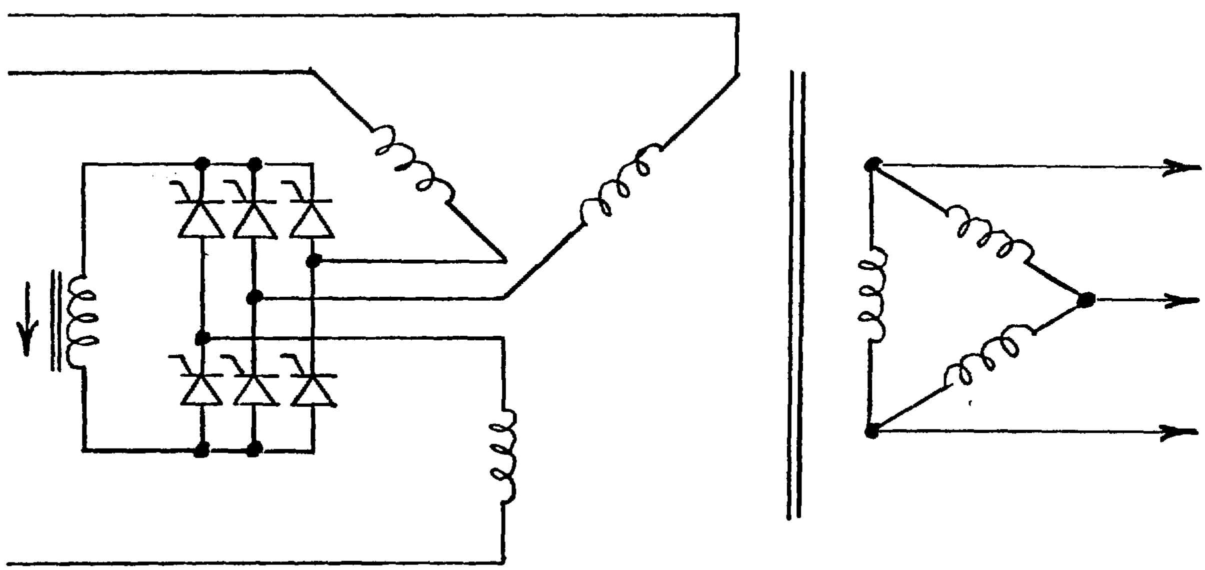
   VOLTAGE - VARIABLE TRANSFORMER

   TRANSFORMER - RECTIFIER

   DC CROWBAR



A. Kerns Actuator



B. Mondino's Circuit

Figure 7. Star-Point Controllers

1. Provides output dc ripple filtering equivalent to that provided by HV dc output choke, but at somewhat lower cost because of the reduced insulation requirement;
2. In the process of providing energy for ripple filtering, some of the inductor's energy is returned to the power mains reducing line current harmonics and required telephone interference filtering;
3. The SCR's can provide the pulse-to-pulse ON-OFF switching and probably a small amount of voltage control, with the primary contactor serving as a backup interruptor or disconnect switch;
4. Eliminates the need for a "step-start" by limiting inrush currents, even in the presence of "asymmetrical offset" effects, to acceptable levels (perhaps < 4 times normal);
5. During crowbars, the transient current rate-of-rise is limited to reasonable levels while the primary power is being interrupted;
6. While it has a high transient impedance, the system can have a relatively low steady-state impedance, permitting essentially full dc output; this is in contrast to the somewhat lower output available from systems where impedance is increased by ac line reactors or higher transformer reactance.

For the LBL 160-kV, 75-A NBSTF Accel power supply, it appears that an inductor storing only 20 to 30 kJ should be adequate. These circuits have only been studied in a preliminary way, so far, but we believe they can play an important role in simplifying future NB power systems.

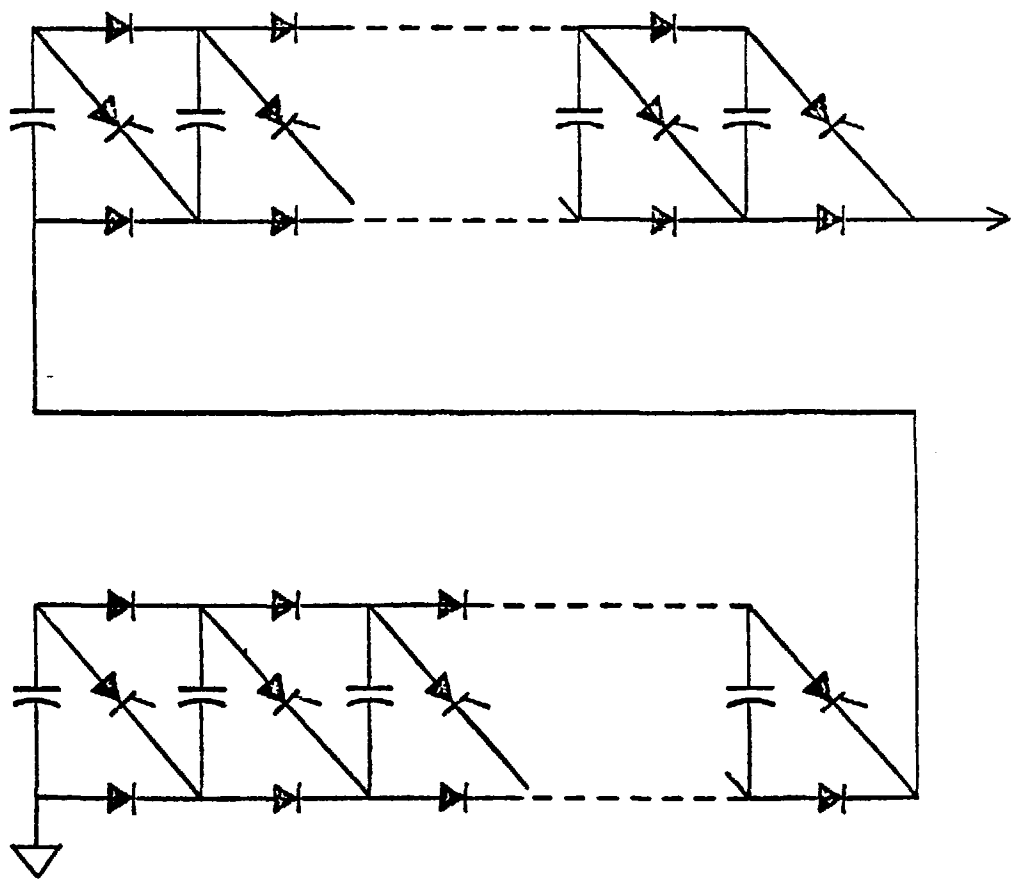
OPERATION TO $\sim$ 1-MV

To our knowledge, J. H. Fink of LLL has devoted the most time to conceptual thinking and design related to possible future 1-MV neutral beam source and power supply systems.[14]

A primary concern is the self-capacitance of NB sources and its stored energy which increases as the square of the voltage. Extending the use of present NB source designs to the $\sim$ 1-MV range appears to be impractical, not only from the electrical standpoint but because of a beam neutralization efficiency which is rapidly decreasing at these levels. Negative ion sources are being proposed for use at these higher voltage levels since they enjoy a much larger neutralization efficiency. These <u>may</u> be constructed in such a manner as to effectively separate the plasma source from the accelerator grid system permitting high vacuum pumping in between. This may mean that any high voltage sparking present could be of the vacuum-spark type (rather than the low pressure gas type) which is much more tolerant of the magnitude of stored energy delivered to the spark.

It may be possible to construct NB source and accelerator systems with magnetic core stacks or other types of arc snubbers built into the structure, inside the vacuum chamber. J. H. Fink has mentioned that it may be possible to design an accelerator structure which could suffer a local sparkdown in a single section, and recover from it, without initiating an overall breakdown. The potential impact of this on improving the reliability and "ON-time" of fusion power reactors is obvious. To date, the effort spent on conceptual designs aimed at solving these problems appears to be very small.

It may not be difficult or too costly to conduct initial experiments in the 300 to 1000-kV range. When the LBL 120-kV, 65-A, 0.5-sec power supply becomes operational (mid to late 1978), the 1.3-MJ capacitive energy-storage system shown in Figure 5 should become available. It can be relatively easily reconfigured in a number of different ways for various voltage levels. Figure 8 shows an arrangement capable of supplying a pulse of approximately 20-A, 30-msec at 1-MV, voltage-regulated (flat-topped) to 1 %. If good regulation is not a prime requirement, higher current and/or longer pulses could be supplied. The metal clad room now housing the capacitor bank is large enough to accommodate such equipment operating at voltage levels in excess of 500-kV, possibly up to 1-MV. The shunt SCR and LC commutating system now used for repetitively interrupting the current during source sparking could be expanded to provide the same function at higher voltage levels. Alternatively, a



1 MV CAPACITOR BANK

e.g., 1-MV, 20-A, 30-msec, ∿ 1% REG.

FIGURE 8

relatively inexpensive crowbar could be provided for terminating the pulse if sparking occurs.

SUGGESTED TASK AREAS FOR EARLY R & D EFFORT

We believe the following tasks should receive early R and D effort in order to help guide the design and minimize the costs of not only the next generation but the long-term power system needs of neutral beam injectors. Without this, there will very likely be a tendency to proliferate the construction of the probably unnecessarily complex and expensive earliest designs of these systems.

1. Feasibility testing of > 1-msec accel voltage risetime, appropriately controlling the arc current, aimed at eliminating the need for accel regulators and fast switches.
2. Tests of star-point controllers in ac primary systems.
3. Low-level hardware circuit modeling and testing of power supply configurations.
4. Long-pulse (e.g., 30-sec) testing of NB sources.
5. Development of a dummy load for accel power supplies which simulates the < 50-nsec sparking of NB sources. (It has been demonstrated that resistive dummy load testing leads to a false sense of security; it is no guarantee of proper power supply operation with NB source loads.)
6. Cost studies of various possible power system configurations.
7. Determination of the acceptability of 10- to 40-msec periods of "wrong-energy" beam injection during accel voltage risetime.
8. A thorough study and optimization of construction and assembly

procedures aimed at reducing costs of SCR assemblies (as was done earlier for electrolytic capacitor bank modules.)

9. Thorough tests to determine stored-energy limits of NB sources.
10. A literature search for voltage-holding and sparking data and experience for vacuum high voltage systems operating up to 1-MV.
11. Establishing a test facility for basic measurements on vacuum voltage holding and sparking up to 1-MV.
12. Conceptual design studies of higher voltage NB sources and power system geometries, possibly incorporating internal arc snubbers.

CONCLUSIONS

For the near-term, we see no great difficulty in adapting existing power supply systems to the 150- to 300-kV level.Fears of power supply system unreliability due to mysterious transient voltage problems have been calmed by the learning experience of the past few years.

Above all, we hope we have successfully argued that NB source and power supply systems are as yet in an early stage of development. It should be clear that there is much room for simplification and improvement. The consequences of implementing these changes may well be cost savings measured in tens of millions of dollars over the next several years, as well as significantly improved reliability.

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Ion Cyclotron Heating Technology

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A brief survey of ion cyclotron heating technology is presented. Emphasis is placed on the minor developments needed on the short term and possible major improvements which could be beneficial on the long term. Much of the existing technology and possible extrapolations required for large tokamak systems has been reviewed in considerably more detail in "Preliminary Report on the Development of RF Auxilliary Heating Systems for TEPR-1" by B. W. Reed et al. (PPPL-1410, 1977) to which the reader is referred for a general classification of the important technological considerations.

I. INTRODUCTION

The major components of ion cyclotron heating technology include:

1) Sources

2) Transmission systems - low loss lines, DC breaks, vacuum breaks, etc.

3) Matching elements

4) Coupling elements

5) Mode locking circuits

A diagram illustrating these elements for the PLT system is given in Fig. 1. For the near term, all of the elements exist although minor development may be required to refine certain elements, especially the matching and mode locking circuits.

However, for the long term (reactor) application, optimization of sources, transmission systems, matching elements, and couplers is desirable to produce the maximum power input at the minimum cost and to adapt the system to the mode physics and the reactor environment.

II. NEAR TERM

For the near term, multimegawatt sources are available from existing technology. However, these must be adapted through transmission, matching, mode locking, and coupling systems to deliver rf power to the plasma waves over a reasonably broad band of frequencies (~10%) to assure adequate control over the power deposition through mode selection. In addition, the entire system must be adapted to the environment peculiar to a tokamak. Critical areas which are apparent at this time include:

1) Coupling structures must sustain the plasma bombardment. Loops are to be used in the near term to firmly establish the mode propagation and damping (heating) properties to permit reliable extrapolations to reactor conditions. These are to be surrounded initially with Faraday shields composed of the vessel material which then will offer the same recycling properties as the vessel. Alternative schemes - bare loops, ceramic clad loops, possibly waveguide structures, etc. - will be employed subsequently to determine the feasibility of their use in larger tokamaks.

2) Matching and mode-locking systems depend critically on the mode loading of the antenna. This in turn depends on the antenna geometry and the plasma regime (density, temperature, ion

species present, etc.). Choices of these systems for the near term do not necessarily correspond to the systems which will be suitable for a reactor. However, the optimal systems for the near term will undoubtedly be more complex since mode-locking is thought to be important only in the smaller devices and their development should result in establishing the development needs for a reactor. In principal, matching circuits could be eliminated with proper design of the coupling array and near term development will be directed toward achieving this once the mode properties are adequately known.

3) Conventional transmission systems are adequate for handling the power requirements but wide-band DC breaks must be developed to isolate the rf system from the tokamak vessel. Conventional breaks should prove adequate but care must be exercised to minimize the rf power radiation outside of the tokamak-rf system to prevent interference with tokamak diagnostic systems.

III. LONG TERM (REACTOR)

The most critical area requiring development for a reactor is the coupling antenna system. Although it is possible that loops appropriately constructed and possibly recessed in vessel cavities will suffice, optimal couplers based on mode physics and coupler integrity in the reactor environment must be developed. Such couplers could be coils, waveguides, strip lines, or some hybrid mix of these. As the machine size increases toward that of a reactor, wave guide coupling is facilitated. An illustration of such a system which could be used for TEPR-1 is given in Fig. 2.

As indicated earlier, it is suspected that matching devices can be simplified by introducing an antenna array, each element of which is excited separately, so that the loading resistance can be made approximately equal to that of the line. Also, mode tracking should not be necessary since the mode density will be large.

A second area for which development is needed is large power sources. Although individual 2.5 MW sources are now available, even larger sources could be beneficial to reactor applications for which (a) wideband circuitry, (b) efficient anode cooling, and (c) long pulse - high duty factor operation will be required. The tube which meets these requirements is of the coaxitron type. This tube has an integral cavity output which assures wideband capability and ease of rf power transfer to the coupling antenna system. RCA has developed a coaxitron amplifier operating at ~400 MHZ and delivering ~5MW of power for ~26μsec (F. S. Keith et al., American Institute of Electrical Engineers, General Meeting, New York City, 1962). Therefore, the development of a long pulse tube of this type would facilitate ICRF heating over a wide range of frequencies and hence plasma conditions. Note that wideband operation in a reactor could be used to select the power deposition profile in much the same manner as one would apply ECRH.

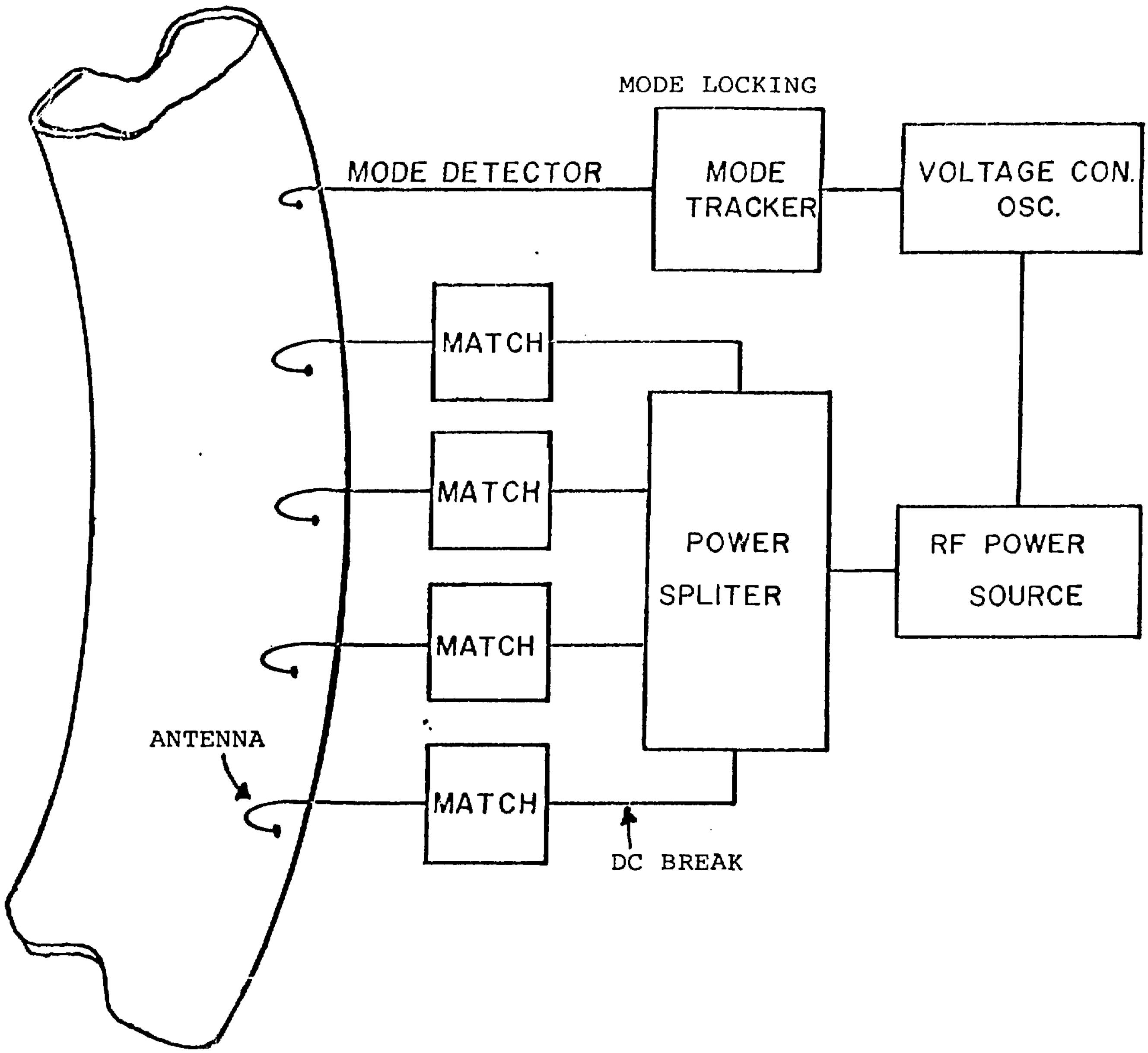
SUMMARY

Sufficient technology now exists to determine the feasibility of employing ion cyclotron heating to push the plasma of a tokamak into the reactor regime. However, technological development should be undertaken to assure that

a) optimal couplers, and

b) large power sources

will be on hand for large machine and then reactor use. First priority development at this time should be allocated to the large power, long-pulse source. Optimal coupler development can then follow after the mode properties have been more adequately determined for the hot, dense plasmas being studied in the ongoing ion cyclotron heating programs.



COUPLING SYSTEM

FIG. 1

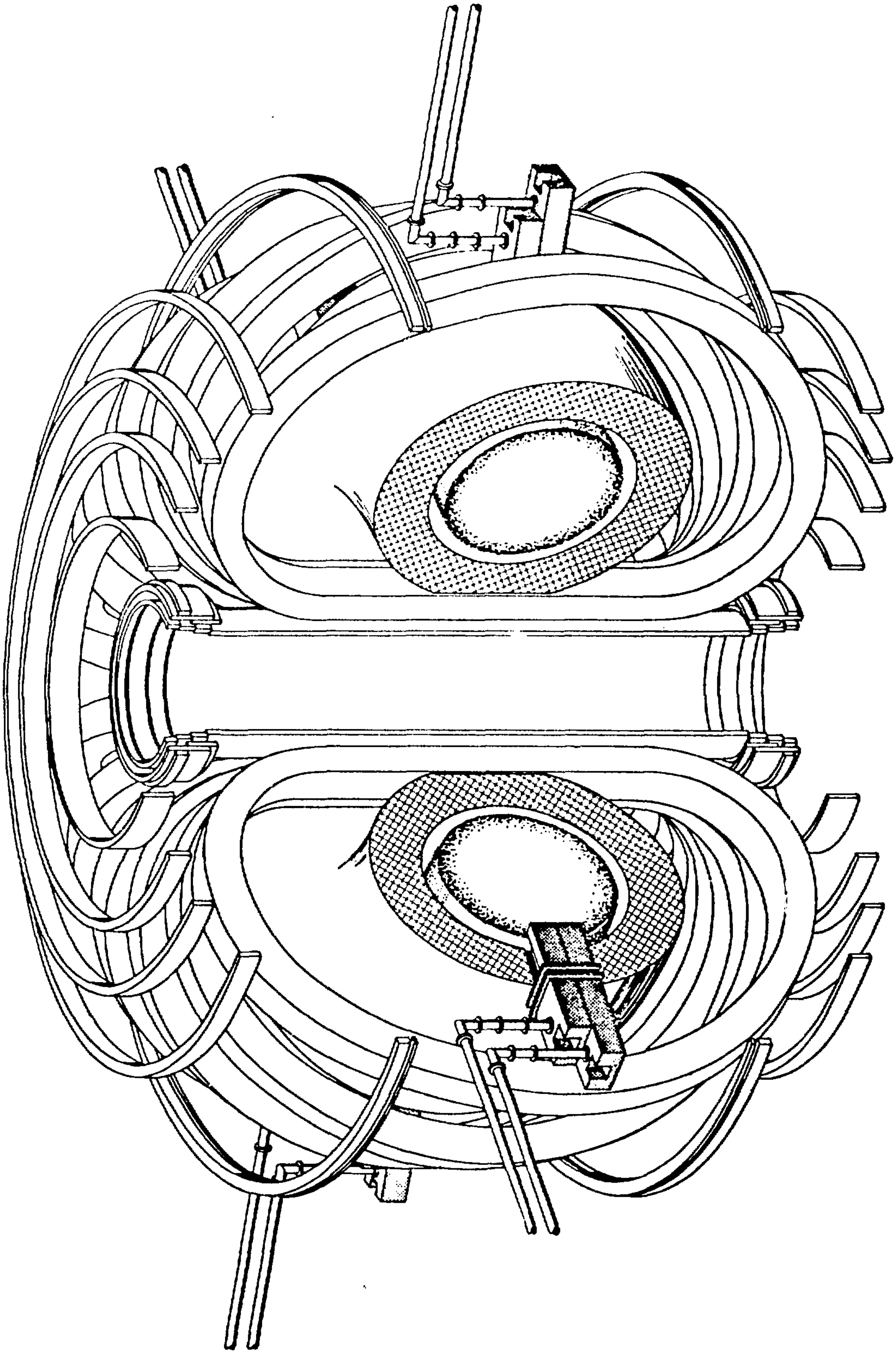


FIG. 2

ORNL MICROWAVE TECHNOLOGY PROGRAM

PRESENTATION TO DOE-DMFE WORKSHOP ON
PLASMA HEATING DEVELOPMENT REQUIREMENTS

DECEMBER 5-7, 1977

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*RESEARCH SPONSORED BY THE DEPARTMENT OF ENERGY UNDER CONTRACT WITH UNION CARBIDE CORPORATION.

ORNL MICROWAVE TECHNOLOGY PROGRAM

December 1977

H. O. Eason

General Discussion

Electron cyclotron heating (ECH) technology has developed to its present state largely through the efforts of R. A. Dandl. The present ORNL Microwave Technology Program is an outgrowth of a supporting effort for the ELMO Bumpy Torus (EBT) and the succession of open-ended confinement experiments which preceded it. The method used for ECH in EBT, and which is basically applicable to almost any magnetic confinement system, involves feeding CW microwave power into a multimode cavity in which the dc magnetic field is adjusted for electron cyclotron resonance. Microwave power is efficiently absorbed in the cyclotron-resonant magnetic field regions of the device, and, since the magnetic field is nonuniform, plasma heating can occur simultaneously in different regions of the device through application of power at more than one frequency. In normal operation, $\omega_{pe} < \omega_{ce}$ so that propagation readily occurs through the plasma. Plasma density is limited by the propagation cutoff which occurs when $\omega_{pe} \rightarrow \omega_{ce}$. ECH application at high plasma density thus implies the use of strong magnetic fields and ECH power at a correspondingly high microwave frequency, e.g., at $B_o = 43$ kG, and $f_o = 120$ GHz ($\lambda_o = 2.5$ mm), $n_e(\max) \approx 10^{14}$. These conditions, which are parameters of the EBT-II design, and which also represent interesting regimes in other confinement systems, approach limitations imposed by superconducting magnet technology in which B_{max} at the conductor is limited to $\approx$80 kG.

Up to the present time, ECH experiments have largely been conducted with the use of microwave power output tubes which were developed in response to other needs and which are commercially available. The current limitation upon extension of the method, to which the ORNL Microwave Technology Program is primarily addressed, lies in the lack of millimeter wave power devices (tubes) suitable for multi-megawatt sources at the required frequencies. Applicable device types for ECH are generally limited to those with basic suitability for cw operation, since power output durations of milliseconds or greater are required, and since such pulse durations exceed practical values of critical thermal time constants for millimeter wave devices. The principal limitation to extension of power output capability of most microwave devices at higher frequencies is that of power density in the rf interaction structure, since the dimensions of such structures are proportional to wavelength. This limitation is considerably relaxed in the Cyclotron Resonance Maser (CRM) device class, to which the gyrotron oscillator and the gyroklystron amplifier belong, since the rf interaction structures in these devices may be made quite large. CRM devices have been studied extensively for a number of years in the Soviet Union and are now the object of intensive development efforts both in this country and in other countries abroad. Operation of these devices involves cyclotron resonance interaction of a hollow electron beam having large transverse energy with TE_{0n1} cylindrical resonators.

Amplifier and oscillator devices are both readily applicable to ECH systems, although amplifier devices are generally preferred due to the relative ease of simultaneous control of the power output of many tubes in a large system by variation of the rf drive power. Oscillator tubes must be controlled by variation of supply voltages. Power output of amplifiers is also generally less subject to variation due to rf load mismatch. No requirement exists for spectral purity, mode purity, or coherence of the output of ECH sources, and size, weight, and auxiliary equipment requirements are of secondary importance. High efficiency of conversion of dc power to microwave power is of great importance, not only because of operating costs but also because of the large capital costs of the associated power supplies. Gyrotrons have demonstrated efficiencies above 30% in fundamental mode operation. Maximum unit power output is desirable for minimizing capital costs and greatest reliability in large multiple-tube systems.

Present Program

The present ORNL Microwave Technology Program is divided into two distinct parts: 1) Microwave tube development, constituting the bulk of the effort, is conducted by private industry under DMFE sponsored subcontract. This portion of the effort presently involves expenditures of ≈\$900K/yr., and 2) Development of compatible microwave transmission and plasma coupling systems and of associated power supply and control systems for ECH is conducted at ORNL. This portion of the effort involves expenditure of ≈\$430K/yr. The near-term objectives of this program are the development of a 200-kW, cw, 28-GHz gyrotron and to demonstrate its application in an ECH system. The system will be utilized as the primary ECH system for EBT-Scale.

The 200-kW, cw, 28-GHz gyrotron is currently being developed by Varian Associates, Palo Alto Microwave Tube Division, Palo Alto, CA, U.S.A., under ORNL subcontract. Both oscillators and amplifiers are under investigation. This program is intended to serve as the first step toward the development of devices capable of operation at similar power levels at a frequency of 120 GHz. Feasibility of the 120-GHz device is required to be demonstrated wherever possible in the design of the 28-GHz device.

The 28-GHz gyrotron is designed to operate from a negative-polarity 80-kV beam supply at a [illegible] cu[illegible]t of 8 amperes. It employs a crossed-field electron gun whic[illegible] ad[illegible]on requires an adjustable well-regulated low-current 40-kV positiv[illegible]y referenced to the cathode. The 11-kG water-cooled copper magnet required for beam focussing and for the cyclotron resonance interaction requires a well-regulated low voltage, high current supply of ≈70 kW capacity. Small auxiliary coils located in the region of the electron gun provide field shaping and control over the transverse energy of the hollow electron beam. The output waveguide has a 2.5-in.-diameter circular cross section, and the output power progagates primarily in low-loss circular electric modes. A 200-kW, cw, rf dummy load has been developed for testing of the gyrotron. Demountable waveguide vacuum windows required for vacuum interface of the waveguide to the plasma device have been adapted from the design used on the gyrotron.

A total of three (3) pulsed amplifiers employing different interaction structures and one (1) pulsed oscillator have been built and tested by Varian. Pulse testing has been used through the early stages of development to simplify construction and to permit separate consideration of the thermal design. The gain of the first two amplifiers (and hence the power output) was limited by interfering oscillations. The third amplifier, although exhibiting adequate stable gain ($\approx$40 dB) had a maximum peak power output of $\approx$75 kW corresponding to an efficiency of 9%. It is believed that the reasons for this relatively low efficiency are now understood, and a fourth modified pulsed amplifier is now under construction.

Best results have been obtained from the single cavity gyrotron oscillator which demonstrated peak 28-GHz power output up to 248 kW at an efficiency of 35%, average power output of 10.7 kW at >200-kW peak with 31% efficiency in high duty cycle operation, and long pulse operation up to 1-msec pulse duration. In addition, it was demonstrated that power output could be controlled over a range of 18 dB by adjustment of a single parameter, viz., the current in one of the auxiliary coils surrounding the electron gun which primarily determines the transverse energy of the electron beam. The oscillator demonstrated moderate sensitivity to rf load impedance variation up to VSWR = 2:1.

As a consequence of the results described above, it was decided that the cw tubes would be constructed as oscillators. The first cw gyrotron oscillator has now been completed and evacuated. The device will undergo pulse testing at increasing duty cycle and average power output through mid-December 1977, with full cw tests and delivery to follow immediately thereafter.

The portion of the ECH development effort being conducted at ORNL is currently addressing the problems of operation of the 28-GHz gyrotron in an ECH system. Specifications for the power supply system were developed for procurement from an industrial source. Final testing of this supply at the manufacturer's plant is scheduled for mid-December. Since power supply system procurement has had to proceed concurrently with the gyrotron development, the system could not be optimized for the gyrotron requirements and the emphasis was instead upon versatility. As a consequence, the supply has somewhat greater capacity (100 kV at 10 amps) than will actually be required for gyrotron operation.

Additional ORNL effort has involved the development of a quasi-optical oversized waveguide transmission system and plasma coupling interface for connection of the gyrotron output to EBT-S. The 2.5-in.-diameter circular multimode waveguide output of the gyrotron is necessary because of power-handling and efficiency considerations. Since a waveguide of this diameter can simultaneously support ≈100 different propagating modes at 28 GHz, extreme care is required to provide good impedance match for all modes everywhere in the system and especially to avoid trapped-mode resonances which could absorb large amounts of power. Maximizing transmission efficiency is, of course, a principal design consideration in ECH systems. Since multimode operation is involved, many additional constraints are introduced as compared to more conventional single-mode operation of oversized waveguide systems. Minimum component requirements, in addition to the waveguide vacuum windows and rf dummy loads mentioned above, include means of negotiating changes in direction (bends), and means of monitoring rf power flow during operation (directional couplers). Components required for the EBT-S system have been developed and tested at low power. Quasi-optical methods have also been employed in the design of the 28-GHz ECH power distribution system for EBT-S, which utilizes a redesigned vacuum manifold for coupling to all of the 24 regions of the torus.

Future Program

The subcontracted gyrotron development effort will be continued following the completion of the 200-kW, cw, 28-GHz phase, with the objective of developing amplifier devices capable of producing 200-kW, cw, at 120 GHz. The initial device development is expected to commence in the spring of 1978 and will require approximately 2-1/2 years at an expenditure of ≈$1M/yr.

The subcontracted device effort is expected to be continued beyond this initial development phase for improvement of efficiency and attainment of the necessary reliability for operation of large numbers of tubes in multi-megawatt ECH sources. Initial application of the 120-GHZ devices will be as elements of a 2-MW, cw, 120-GHz ECH source for EBT-II requiring ten (10) tubes. An additional objective during this period will be the development of gyrotrons at intermediate frequencies between 28 GHz and 120 GHz suitable for EBT-II profile heating and other applications. Scaling of microwave power tubes to lower frequencies is relatively straightforward and can be accomplished with a high level of confidence since dimensions of the rf interaction structure become larger and thermal considerations are thereby relaxed.

The ORNL in-house activity for development of ECH transmission systems, plasma coupling systems, power supply systems, and control and monitoring systems will be continued as a complement to the subcontracted 120-GHz device effort. The primary objective during this period will be the development of components, systems, and techniques required for realization of the 2-MW, cw, 120-GHz ECH source for EBT-II. Modest expansion of the in-house effort to a level of ≈$650K/yr. is required due to the complex control aspects of operating large numbers of tubes in a multi-megawatt ECH system. It is expected that microprocessors will be employed to facilitate system start-up, control, protection, monitoring, and shut-down.

Granting successful development of the 120-GHz gyrotron, for which a high level of confidence exists, and its successful and timely employment in the 2-MW, cw, EBT-II system, one can reasonably extrapolate that ECH systems up to 20 MW, cw, and employing up to 100 such tubes can be realized by the mid-1980's. Due to considerations of capital cost, complexity, and reliability, it is evident that significantly larger systems beyond 20 MW will require yet further advance in the state-of-the-art for microwave tubes. The necessary advance will involve increase in unit power output in order to attain required source output with a manageable number of devices.

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Promises and Problems of Lower Hybrid Heating

R. W. Motley

Like the other speakers in this marathon, I will look into my cloudy crystal ball and enumerate some of the promises and pinpoint the most troublesome problems for lower hybrid heating of reactor plasmas. To begin, let us discuss the choice of RF frequency. Since the lower hybrid resonance depends on the plasma density, the magnetic field, and the temperature, which are still vaguely defined, one can specify only a range of frequencies extending from 1 to 4 GHz. The lower part of this range would be appropriate for low density tokamaks, such as TFTR, while the higher frequencies would be appropriate for high field, dense Alcator-type reactors. The choice of frequency is also influenced by the absorption process within the plasma, whether ion heating at the LH resonance or electron Landau damping is desired.

This frequency interval falls right into the radar and high power cooking range. <u>Powerful CW klystron sources up to 1 MW exist or can easily be designed by upgrading existing tubes</u>. No billion dollar development program is necessary. For example, the Varian X-3070 klystron delivers 1/2 MW at 2.4 GHz and costs approximately $100K per (bare) tube. If we were to decide on a frequency where no tube exists, then a low cost development program of less than $1 M would produce the tube. Varian has indicated to our engineering group, for example, that they could develop a 1.5 MW, 1.5 GHz klystron without any extrapolation of existing technology.

It is not clear at this time what the optimum power per tube will be. A design study for TEPR-1 by J. Lawson's group at Princeton assumes the need for 50 MW of RF power at 1.2 GHz, which could be supplied by 50 1 MW klystrons, one tube for each waveguide. At higher frequencies, e.g., 3.6 GHz, each individual waveguide could not accept 1 MW, so a larger number of lower power tubes may be more cost effective. Recently, Eastland has suggested that very cheap power could be delivered by a large battery of efficient low power magnetrons, provided one developed a simple method of phase locking all of the oscillators. We believe that this is an appropriate time to fund cost studies, the object of which would be to minimize the cost of high power rf power sources.

Now let us pass from the RF power sources to the transmission system that transfers power from the klystrons to the reactor plasma. The first stage of this system will be a strictly conventional 3" or 6" copper coaxial transmission line, perhaps filled with high pressure gas (SF_6) to minimize breakdown problems and cost. At some point within the strong field of the reactor we will convert the coaxial line to waveguide. This point is an excellent place to locate the vacuum seal, both to avoid cyclotron resonance breakdown and to utilize the reactor blanket to minimize neutron bombardment of the seal. Here we have a second great advantage of lower hybrid heating, one that it shares with other RF systems. Namely, <u>the seals and power sources can be located around a bend and need not look into the awful heart of the reactor</u>. In this way one can avoid the cumbersome moveable shutters that will probably be required to shield nuetral beam sources during the main burn cycle. Since the seal must accept some neutron radiation, however, we would welcome studies of beryllium oxide and other RF insulators to determine how their physical properties degrade when subject to intense radiation.

Next, we pass on to the final stage of the transmission line, which will be a multielement phased waveguide, perhaps one like that designed for 1.2 GHz by the Lawson group (Fig. 1). Detailed engineering design studies for this structure are certainly needed. The waveguide near the first wall must accept fast neutral, neutron and x-ray bombardment while maintaining high conductivity. Stainless steel will probably be satisfactory at the lower frequencies (1 GHz) provided it is well cooled. If high frequencies (4 GHz) are required, then pure stainless may be unsatisfactory (~0.3 db loss/10 ft. at 200°C), especially if radiation enhances the resistivity. Perhaps a layered structure, consisting of stainless steel coated to a few skin depths with a high conductivity metal coated with a few monolayers of titanium to avoid secondary electron emission, will be necessary.

Let me at this point bring up a problem shared by all RF systems. RF power must be delivered to the plasma through the blanket via waveguides or coax lines. Inside these apertures will be a background gas whose density will be determined by refueling and recycling of the working gases. These gases will be subject to intense bombardment by radiation, especially within about 1/2 m of the first wall, and low density plasma will build up in these areas. If the plasma frequency approaches the wave frequency, either complete or partial reflection of the RF power will result. At the lower hybrid frequencies the critical density will be in the 10^{10}- $10^{11}cm^{-3}$ range, but at lower frequencies, e.g., 50 MHz, the critical density will be only $2 \times 10^{7}cm^{-3}$ with a 1 meter skin depth. There will definitely be a certain frequency below which it is not possible to propagate RF energy. I believe that it is a challenge to the reactor designers to calculate this residual plasma that must exist in the hot region near the first wall so that this critical frequency may be evaluated.

The total number of elements in the launching array will depend more on the ability of the surface plasma to accept the power than on engineering constraints. At a certain power level, depending on the surface plasma density and temperature, a high powered wave will parametrically decay into daughter waves which will heat the surface plasma. In present day tokamaks we believe that the limiting power density is of order 5 kW/cm^2, but this number depends on the details of the surface plasma which are not well defined in a reactor. However it is not at all certain that decay instabilities will be a serious limitation to power flow. In a sense the problem is self-correcting because the decay threshold rises rapidly with electron temperature ($\sim T^{3/2}$).

To answer this important question, we need further experiments at moderate power levels (~ 1 MW). At this moment only two such experiments are being planned or executed in the USA.

(1) A 600 kW experiment on Doublet 2A is in progress, but it will be difficult to analyze for parametric blockage, since the power density radiated along special antenna structure is not well defined.

(2) In addition to this experiment, however, a 1 MW, 4 GHz experiment on Alcator C has been recently proposed.

Other types of nonlinear wave blockage at the plasma surface, such as cavitons, are unlikely to be of importance, since the power densities involved should be well below threshold ($E^2/8\pi nkT \ll 1$).

At last we come to the plasma interior. What can lower hybrid heating accomplish here? There are at least four possible roles for LHRH. Obviously, <u>direct main body heating of the plasma would</u>

be the most ambitious program. What are the results to date in this area? Resonance heating of ions near the lower hybrid has been observed at the 150 kW level and less in ATC and has been verified conclusively by a long, detailed series of measurements on the Vega Tokamak at Grenoble. Electron heating by Landau damping of slow waves has been proposed and attempted, but as yet has not been verified in a toroidal plasma. The credibility of LHRH as a prime heater relies primarily on a firm scientific identification of the relevant resonant absorption mechanisms and their appropriate scaling to a reactor environment.

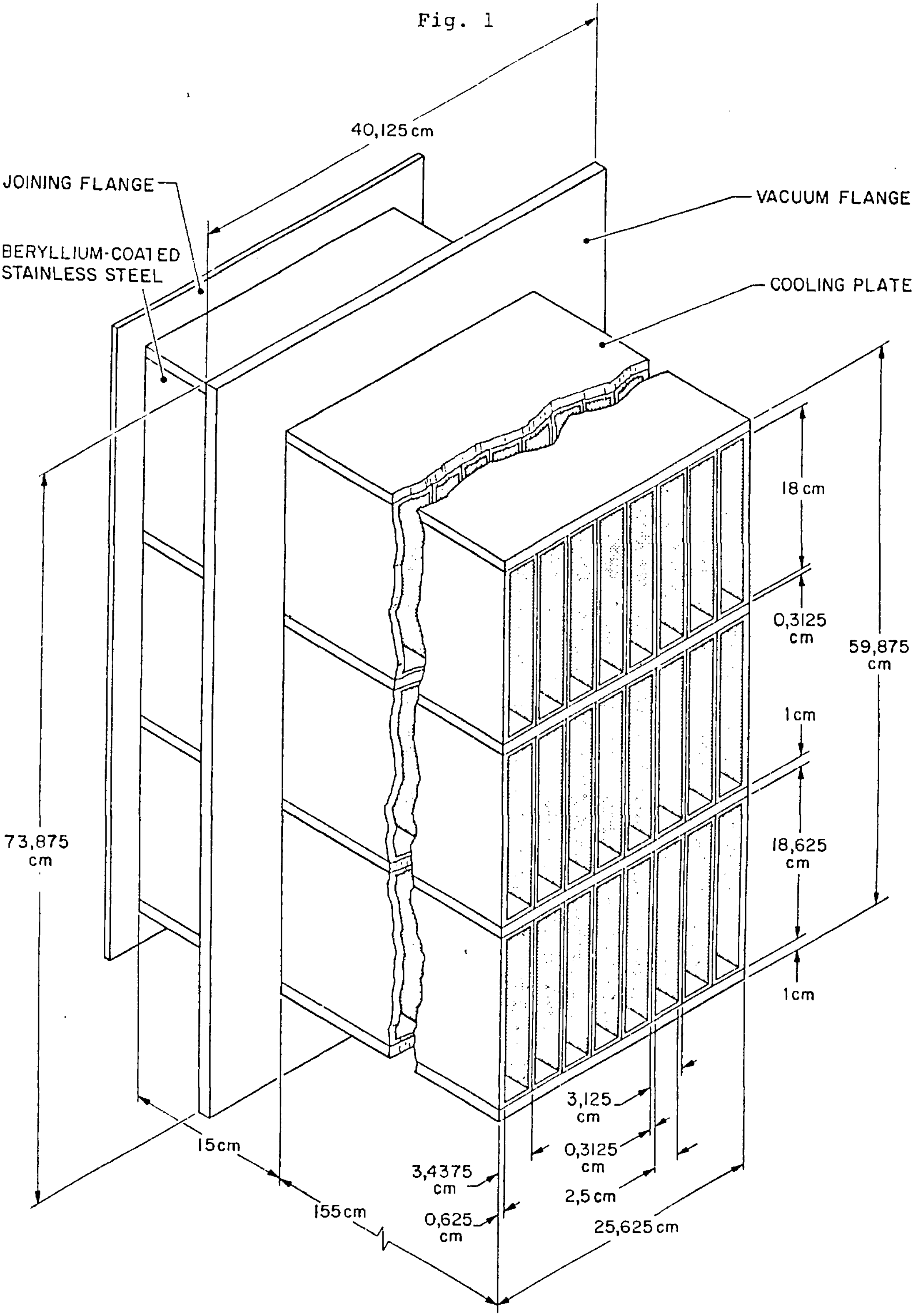
A second, and subsidiary, role for LHH is as an energy clamp for the trapped ion beam resulting from neutral injection. This is at this moment a much more credible role for LHH. Since the orbits of the injected ions are large, their (harmonic) cyclotron interaction with the lower hybrid waves is strong (if $k_\perp r_L \sim n$). One can therefore use the RF beam to pump the trapped ion beam, thereby maintaing the beam energy and neutralizing electron drag. This process has been observed on ATC and hopefully will be studied further at Princeton.

Profile modification of the reactor plasma may also be a reasonable task for LHH. In a tokamak with a normal temperature profile, one should be able to control the Landau heating zone (and thereby the energy deposition profile) by appropriate phasing of the waveguide elements. This type of control is now being attempted on Doublet 2A at G. A.

One further possibility has recently been emphasized by Bers, i.e., the maintenance of a DC tokamak discharge by the interaction of directed LH waves with the tail of the electron distribution. Needless to say, this idea at this time is far out and needs much further study.

In conclusion, we believe that the problems facing lower hybrid heating are just the reverse of those facing neutral beam heating. The engineering problems are relatively straightforward. We need some engineering studies of conductors and insulators under intense neutron and x-ray bombardment, and we need studies to improve the cost effectiveness of high power RF sources. We also require the funding of additional moderate power experiments to test surface heating limitations and to identify conclusively the mechanisms of LH wave absorption in hot plasmas. We believe that funds spent now to establish the scientific credibility of RF heating, either in a primary or secondary role, will be good disaster insurance should neutral beam heating falter.

Fig. 1



LHRF COUPLER DETAIL

SESSION - IMPLICATIONS/EFFECTS OF REACTOR-LIKE ENVIRONMENT
[Direction (Where to from Here) Discussions]

Moderator: John Sheffield, ORNL

Opening remarks reflecting some of the previous day's discussions:

1. Viability of neutral beam heating and RF?

- The simple answer is that they are viable.
- The question should rather be
  Which energy(ies) for NBH?
  Which frequency(ies) for RF?

2. Reactor requirements/limitations

- Conventional wisdom suggests that we require a power $P \lesssim 100$ MW.

  RF and NBH power density limits one in the range $5\ \mathrm{kW/cm^2} \begin{cases} \times 2 \\ \div 2 \end{cases}$

  Allowing for redundancy, this suggest a port area $\sim 4 \times 10^4\ \mathrm{cm}^2$.

  e.g., 10 ports each 65 cm × 65 cm — <1% of wall area
  Toroidal wall area $40\ a \times R$ — if $a \gtrsim 65$ cm

  Therefore, it should be possible to have adequate blanket penetration.

- NBH limits set by beam heating $\alpha \dfrac{L}{D^3}$ of port.

  RF limits set by electric-field for breakdown.

- Direct radiation effects

  NBH $\begin{cases} \text{Heating of cryopanels} \\ \text{Irradiation of high voltage electrodes, insulators} \end{cases}$

  RF Irradiation of launching structure

- Remote handling - easy maintenance, two main categories

  Components requiring routine maintenance in radiation area
  NBH sources, dumps . . .
  RF launchers

  Components not requiring routine maintenance in radiation area but which might fail
  Cryopumps
  RF matching equipment

- Simulaneous requirements $\begin{cases} \text{Reliability} \\ \left.\begin{array}{l}\text{Low cost/watt} \\ \text{Low cost/particle}\end{array}\right\} \text{efficiency} \end{cases}$

Comments: It takes a long time to establish reliable components. Power supplies, power feeds, remote components are critical cost items, the least stringent requirements should be determined for them.

Session 1 was concerned mainly with systems. In Session 2, techniques were discussed which might reduce heating requirements; Session 1 was, therefore, devoted primarily to looking at a range of heating systems and identifying their status in respect of the following factors:

Efficiency

What were the expected direct radiation effects: What were the routine maintenance requirements in the radiation area?

What was, or is, expected to be the first date of operation?

Which types of reactor was the system considered for?

General comments on the system.

The point of identifying the first date of operation was to indicate when there could be information on reliability and ease of handling.

In reference to the question of which type of reactor they were considered for applied to tokamaks and mirrors; it was pointed out that most of the methods had been considered for one or other of the alternative concepts (as discussed on Monday, December 5).

The systems considered were:

Neutral beam heating $\left\{ \begin{array}{l} \text{Low energy } \lesssim 50 \text{ keV} \\ \text{Medium energy } \lesssim 80\, A_b \text{ keV} \\ \text{High energy } \gtrsim 80\, A_b \text{ keV} \end{array} \right.$

A_b = atomic mass number

RF heating $\left\{ \begin{array}{l} \text{ICRH} \\ \text{LHRH} \\ \text{ECRH} \end{array} \right.$

A. Low Energy NBH $\lesssim$ 50 keV

The presently applied <u>high power</u> systems fall into this range. The systems involve positive ion acceleration and neutralization. The best system efficiencies are ~50%. Lower achievements are usually a result of poor systems integration, owing to application to a device not designed for injection.

- The dates of first operation were ~1960's for mirrors 1973 for tokamaks
- It was considered that problems of direct radiation would be soluble with careful design, but that more study was needed. It was clear that routine maintenance and remote handling were required. A problem was to establish the typical operational time before maintenance would be required.

Comments: It takes a long time to establish reliable components. Power supplies, power feeds, remote components are critical cost items, the least stringent requirements should be determined for them.

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The systems considered were:

Neutral beam heating
- Low energy $\lesssim 50$ keV
- Medium energy $\lesssim 80\ A_b$ keV
- High energy $\gtrsim 80\ A_b$ keV

A_b = atomic mass number

RF heating
- ICRH
- LHRH
- ECRH

A. Low Energy NBH $\lesssim$ 50 keV

The presently applied high power systems fall into this range. The systems involve positive ion acceleration and neutralization. The best system efficiencies are ~30%. Lower achievements are usually a result of poor systems integration, owing to application to a device not designed for injection.

- The dates of first operation were ~1960's for mirrors 1973 for tokamaks

- It was considered that problems of direct radiation would be soluble with careful design, but that more study was needed. It was clear that routine maintenance and remote handling were required. A problem was to establish the typical operational time before maintenance would be required.

These energy beams have been proposed for both mirror and tokamak startup, and it would be desirable to isolate them from radiation during the burn phase. In the case of large tokamaks, because of their poor penetration capability, it would be necessary to employ either

Ripple injection or
Layer-by-layer plasma buildup.

B. Medium Energy Beams $\lesssim 80\ A_b$ keV ($\gtrsim 50$ keV)

- To obtain efficiencies $\sim 50\%$, it would in general be necessary to employ either positive ions + direct recovery or negative ions.

- The questions of radiation effects and maintenance requirements are similar to those for low energy beams.

- Beams using positive ions in this range are under development and are proposed for application to tokamaks ~ 1979, to mirrors ~ 1981-1982.

- They are proposed for application in mirror and tokamak reactors and in hybrid reactors.

C. High Energy Beam $>80\ A_b$ keV

- It is necessary to use negative ion systems to achieve efficiencies $\sim 50\%$.

- The radiation effects and maintenance requirements are similar to those above.

- The date of first application to both mirrors and tokamaks is sometime from ~ 1982 onwards.

- They are proposed for heating in both tokamak and mirror reactors.

Summary of Comments on (A), (B), and (C)

(1) Useful application on present experiments does not necessarily mean that a system may be used on a reactor.

(2) We should establish reliability requirements.

(3) The radiation damage to the reactor wall etc will be worse than to the injector.

(4) There is no evidence that neutral beams are not viable for use on reactors.

D. Ion Cyclotron Resonance Heating

The power source is very efficient, and in principle overall system efficiencies of 50-60% are possible; however, more detailed experimental evidence is needed on the coupling efficiency of power to the plasma.

The problems of radiation effects and routine maintenance are confined primarily to the coupling structure. Further studies of coupling structures and radiation effects on conductors and insulators are needed to establish the optimum designs for routine maintenance and remote handling. In principle, the insulators can be hidden.

The first operation on a tokamak was 1973.

ICRH is proposed for heating tokamak reactors (some possibilities exist for mirrors).

E. Lower Hybrid Resonance Heating

Power source efficiencies $\sim$60% are possible. The overall system efficiency depends on the coupling to the plasma which has not yet been fully studied; overall $\sim$50% is in principle possible.

A coupling structure composed of a simple waveguide array (grill) is being tested, and if it is proven to be effective would suffer only limited radiation damage problems. Though the effects of radiation on waveguide wall conductivity should be established, the coupler should require only limited routine maintenance.

The first application to a tokamak was made in 1975.

LHRH is proposed for heating tokamak reactors.

F. Electron Cyclotron Resonance Heating

The present best power source efficiencies are $\sim$30%. Theoretical calculations suggest that higher levels may be possible. The efficiency of coupling to the plasma requires further investigation.

The waveguide launching structures used with this system should have only limited radiation damage and maintenance problems.

The technique was first applied to a tokamak in $\sim$1975, but because of the developmental status of the power supply has not yet received much experimental attention.

The system is proposed for EBT, tokamaks, and mirrors. It has also been proposed for use on minimizing startup problems and in profile shaping in a tokamak.

Summary of Comments on (D), (E), and (F)

1. Useful application on present experiments does not necessarily mean that a system may be used on a reactor.

2. We should establish reliability requirements.

3. The main uncertainties relate to coupling of power to the plasma, and require application of each technique.

4. Studies are required of reactor relevant launching structures.

5. Potentially these systems have a low cost/watt, if the coupling is good.

6. There is no evidence that RF systems are not viable for use on reactors.

Review of Discussions by John Sheffield

<u>Heating and Fueling Systems for Reactors</u>

A few remarks were made on fueling, these are summarized first:

<u>Fueling</u>

Mirrors use neutral beams

Tokamaks may use:
- neutral beams (low-medium energy)
- (clusters)
- pellets
- gas puffing

- There is limited experience in fueling. Future tokamaks will be larger and may need new fueling techniques.

Pellets
Clusters
Low-medium energy beams with:
- Ripple injection
- Plasma buildup

<u>Heating</u>

- Both neutral injection and RF appear feasible.

- The main questions are:

Which neutral beam energy(ies)?
Which RF frequencies(ies)?

Radiation effects on power launchers:
- Damage?
- Lifetime?
- Maintenance and Remote Handling?

- There is only limited experience at high power with a few systems - low-medium energy beams, some RF.

- Most systems will not be adequately tested until the early 1980's.

  Therefore, it will not be easy to select "optimum" systems until after that time.

- Low-medium energy beams exist and have operated for sometime, but there is not enough effort on improving

  Reliability
  Ease of handling
  Cost/watt { Recovery systems, Negative ions }

  given the total proposed near-term application.

- High energy beams at high power are being developed. They are needed in the mirror program and would be used for heating to the plasma center in the tokamak program. They require the use of negative ions.

{ The problem is to achieve a sensible balance between consolidating the present position with respect to positive ion systems while continuing an active negative ion development program. }

- RF systems have great potential, power sources are relatively cheap (ECRH sources require further development). The coupling of power to the plasma needs further study.

{ The problem is to have a balanced program of application of the various systems so that in the not too distant future there will be a good basis for choosing between one system and another. }

- We should be careful not to restrict the options too early in case we omit the one system which in the end might turn out to be the most relevant one.

- Studies of reactor heating systems should continue in order to establish the constraints due to the radiation environments.

- Studies should continue which will help to minimize the power supply costs.

- There should be a careful balance between meeting the needs of the near-term program while preparing for long-term requirements.

SUMMARY OF DISCUSSION SESSION

on

POTENTIAL METHODS FOR REDUCING PLASMA HEATING REQUIREMENTS

by

D. L. JASSBY
Princeton Plasma Physics Laboratory
Princeton, NJ 08540

DOE/DMFE Workshop on Plasma Heating Development Requirements,
Gaithersburg, MD, 5-7 December 1977.

SUMMARY OF DISCUSSION SESSION

on

POTENTIAL METHODS FOR REDUCING PLASMA HEATING REQUIREMENTS

by

D. L. JASSBY
Princeton Plasma Physics Laboratory
Princeton, NJ 08540

In this session, the following questions were addressed:

- How can heating power be reduced?
- How can heating energy be reduced?
- If neutral beams are used, how can the beam voltage requirement be reduced?

Most of the discussion applied specifically to tokamaks, and was heavily weighted toward the issues of neutral-beam injection. The basic concern was how to facilitate the heating of large toroidal plasmas to ignition temperature. Some consideration was given also to mirror machines. Nevertheless, the questions considered are expected to be rather similar to those arising for other magnetic confinement schemes with plasma densities in the range 10^{13} to 10^{15} cm^{-3}, such as stellarators, bumpy tori, reversed-field pinches, and multipoles. It should be kept in mind that low-Q fusion-neutron sources may have somewhat different heating requirements than those of ignited plasmas — for example, neutral-beam injectors that operate at relatively low voltage, but with steady-state cooling and pumping systems, and with special neutron-resistant construction.

2. METHODS FOR REDUCING INJECTED HEATING POWER

The power required to heat a toroidal plasma of minor radius a_p and major radius R_o to a temperature T at density n is

$$P_{heat} = \frac{3nT\times(\text{Vol.})}{\tau_E} \propto \frac{(na_p)^2 TR_o}{n\tau_E} \qquad (1)$$

where τ_E is the energy confinement time. For a specific device performance (e.g., ignition), $n\tau_E$ and T can be given definite values. Certain tokamak experiments show that $n\tau_E \propto (na_p)^2$; for this dependence one would expect $P_{heat} \propto TR_0$. Then the most straightforward way of reducing P_{heat} is to strive for plasmas of the smallest possible R, while keeping na_p large enough to achieve the required plasma performance.

One can write $P_{heat} = P_{e\ell} + P_{inj} + P_{\alpha}$, where $P_{e\ell}$ includes ohmic and compressional heating, P_{α} is fusion-alpha heating, and P_{inj} is the injected power, such as by neutral beams or electromagnetic waves. Table 1 lists several potential methods for reducing P_{inj}, which includes only the power injected into the vacuum vessel. Note that as far as the power supply requirements are concerned, an increase in the efficiency of production of a given P_{inj} is as valuable as a reduction in P_{inj} itself.

For devices that are to obtain ignition conditions, it seems clear that maximization of P_{alpha} (by using various start-up scenarios) is the generally most useful method for reducing P_{inj}. Alpha heating is also increased by the use of reacting ion beams, that is, deuteron or triton beams that produce significant energy by beam-target or beam-beam reactions, while thermalizing with the bulk plasma. The production of alpha particles by reacting beams is enhanced if an energy clamping technique can be employed, such as a slow major-radius compression, a strong toroidal electric field, or RF heating at a harmonic of the ion cyclotron frequency.

3. METHODS FOR REDUCING INJECTED HEATING ENERGY

Reduction in the total energy required for heating the plasma to a given temperature is important if this energy, E_{inj}, is to come from an energy storage system. (This consideration is, of course, not applicable to steady-state plasmas.) Table 2 suggest several methods for minimizing the total energy requirement. Again, the maximization of alpha-particle heating is the generally most useful technique. If very high power can be made available for short periods $t_{heat} \ll \tau_E$, then this approach is advantageous, because the only energy to be supplied corresponds to the specific heat of the plasma — that is, no heat is lost in radiation or conduction, losses that are incurred during more leisurely heating approaches. In tokamak plasmas, E_{inj} thus adds linearly to the temperature level obtained by Ohmic heating; during slow

Table 1. Methods for Reducing Injected Heating Power.

METHOD	FEASIBILITY TEST	DISADVANTAGES
Maximize Fusion-Alpha Heating	TFTR (1983) JET (1986) ITR (1986)	Ignition or near-ignition plasmas only.
Reacting Ion Beams (D^+, T^+)	PLT (1977) (classical neutron production)	
Compression	ATC (1972) TFTR (1982)	(i) Ultra-large EF power. (ii) Oversized coils and vacuum chamber.
Enhanced Ohmic Heating	Alcator C (1979) FT (1979)	(i) Need favorable τ_E scaling law. (ii) Uncertain extrapolation to large temp.

Table 2. Methods for Reducing Injected Heating Energy.

METHOD	FEASIBILITY TEST	DISADVANTAGES
Maximize Fusion-Alpha Heating*	TFTR (1983) JET (1986) ITR (1986)	
Ultra-High Power	2XIIB (1976) TFTR (1982)	(i) High capacity power supplies. (ii) More numerous and larger injection ports.
Peaked Profiles	Characteristic of most tokamaks	May be MHD unstable.

*For example, by overheating a small plasma, which then reaches ignition size by expanding into a dense "blanket".

heating, on the other hand, the energy input from ohmic heating is lost. In toroidal devices with superconducting coil sets, rapid heating may be impractical, however, because of the associated large rates of charge of poloidal fields.

Plasmas with peaked profiles have much higher fusion reactivities than flat-profile plasmas with the same energy content, so that in the former case, ignition or a given Q-value can be obtained with smaller total heating energy. Penetration of the injected energy to the plasma center is usually necessary to obtain a peaked temperature profile.

4. METHODS OF REDUCING OHMIC-HEATING REQUIREMENTS

There is considerable incentive for reducing the role of the conventional ohmic-heating system in large toroidal devices, inasmuch as the primary coils and space for the transformer flux greatly increase the device size, and must be powered by energy storage systems with power capabilities of the order of 1 GW or more (for an air-core transformer). While some toroidal schemes such as Z-pinches, and tokamaks with ultra-high fields and current densities, can in principle reach large temperatures with ohmic heating only, it would seem reasonable to take greater advantage of nonohmic heating methods in those fusion devices where such heating methods must be applied in any event. The following are some examples:

- Electron cyclotron resonant heating (ECRH) could be used for breakdown of the filling gas and initial heating of a small volume of plasma to $T_e \sim 5$ to 250 eV, depending on the power available and the degree of optimism of one's heating model. Then the peak OH voltage required for plasma start-up could be reduced by a large factor. Such ECRH experiments are planned for ISX-B. The principal development problem for reactor-sized tokamaks is to obtain high power at high efficiency at frequencies $\gtrsim$ 100 GHz, (e.g., 1-sec pulses of 200 kW per tube).

- The "resistive volt-seconds" required from the transformer can be reduced markedly if neutral beams or electromagnetic waves are injected as early as possible during start-up, to raise T_e to the keV range as rapidly as possible. The OH system would have to induce a plasma current of I_p = 200-300 kA before neutral beams could be injected.

• The "inductive volt-seconds" for current start-up could be reduced significantly if a large beam-induced current were to be set up very early in the discharge. At this early stage, the plasma opacity is small, so that tangentially injected 80-keV D° beams would penetrate adequately.

• Start-up scenarios should attempt to maximize the flux swing obtainable from the equilibrium-field coils. For example, a large current increase can be obtained from the EF system by utilizing major-radius compression, as demonstrated in the ATC device, but an oversized vacuum vessel is required.

5. METHODS FOR REDUCING THE BEAM VOLTAGE REQUIREMENT

The chief drawback in the use of neutral beams for heating large plasmas is the difficulty of producing efficient high-power, long-pulse beams at high voltage. Consequently, a number of different injection methods and operating scenarios have been proposed, principally for tokamaks, which would minimize the voltage requirement. These methods are summarized in Table 3. (Note that beam "energy" is used interchangeably with beam "voltage" in this context.)

A controversial issue at present is whether direct heating of the central plasma region is required, or whether it is sufficient to heat the outer plasma region and depend on inward heat diffusion to heat the center. In this discussion, the term "good penetration" signifies a neutral-beam trapping profile, H(r), that is constant or monotonically decreasing with plasma radius, r. Even with H(r) = constant, 75% of the beam energy is deposited between $r = a_p/2$ and $r = a_p$. The following arguments indicate that it is preferable to deposit heat directly into the center region, if in fact it can be done:

(1) With good penetration, the efficiency of plasma heating is increased, since the diffusion length from the source of heat to the cold plasma boundary is maximized.

(2) Fast ions deposited by neutral-beam trapping or RF acceleration in the outer region of the plasma are more susceptible to loss by charge exchange and large-orbit excursions.

(3) Enhanced sputtering of the vacuum wall or liner by fast neutrals from an overheated plasma edge may lead to an intolerable influx of impurities into the plasma.

Table 3. Methods of Reducing the Beam Voltage Requirement in Toroidal Devices.

METHOD	BEAM ENERGY FOR IGNITION	FEASIBILITY TEST	DISADVANTAGES
Perpendicular injection	$\gtrsim$ 250 keV	TFR (1976)	Possibly enhanced orbit and ripple-induced losses.
Injection + major-radius compression	150 keV ($C \gtrsim 1.5$)	ATC (1975) TFTR (1982)	Oversized TF coils and vacuum vessel
Ripple-assisted injection	80-120 keV	ISX-B (1979) TFTR(?)(1982)	Extra space required for ripple coils and vertical injection.
Edge heating (and fueling) of expanding plasma	80-175 keV (depending on scenario)	TFTR (1982) JET (1983)	(i) Less efficient. (ii) Enhanced charge-exchange loss of fast ions. (iii) Possibly enhanced wall sputtering.

(4) A peaked temperature profile results in the highest fusion power production for a given plasma pressure. With peaking, alpha production is maximized, and the $n\tau_E$ required either for ignition or a given Q-value is minimized. It is difficult for a peaked T(r) to be established unless external heating of a large central region is maintained until the power density of fusion-alpha production and thermalization in this region exceeds 1 watt/cm^3.

The following is a brief discussion of the methods listed in Table 3.

(1) Perpendicular Injection. Perpendicular injection allows a factor of 1.5 reduction in E_b, compared with tangential injection. Presently used "empirical scaling laws" indicate that $\bar{n}\tau_E \propto (\bar{n}a_p)^p$, where $p = 3/2 - 2$. For a slightly peaked beam trapping profile in a plasma with a parabolic density gradient, and with an opacity $\bar{n}a_p$ sufficiently large to satisfy the $\bar{n}\tau_E$ required for ignition at $T(0) \sim 15$ keV, the required beam energy is $E_b \sim 250$ keV. Larger beam energies will be required if the present "empirical scaling" turns out to be optimistic for higher-temperature plasmas, or if $Z_{eff} > 1$. (The effect of impurities on beam penetration has been determined recently to be rather weak.)

(2) Injection Plus Compression. The opacity $\bar{n}a_p$ of a tokamak plasma increases by a factor $C^{3/2}$ upon major-radius compression, where $C = R_{initial}/R_{final}$. An ignited plasma can be formed by compression of a beam-heated plasma whose $\bar{n}a_p$ is determined by neutral-beam penetration into the center. As shown in Fig. 1, a compression C = 1.6 is adequate to bring E_b into the range of TFTR-type injectors. "Compression boosting" is attractive only if $n\tau_E$ increases strongly with $\bar{n}a_p$ (as it apparently does). When compression was performed on the ATC device, the increase in plasma temperature was sometimes less than expected, but the expected change in plasma opacity always occurred. Compression boosting is at this time the only proven method for reducing E_b, but the penalties to be paid are oversized coils and vacuum vessel, an increase in stored magnetic energy, and a large pulsed power requirement (~ 500 MW) for the equilibrium-field system.

(3) Ripple-Assisted Injection. In this method, a ripple with significant top-bottom asymmetry is created in the toroidal magnetic field. When neutral beams are injected vertically from the side of stronger ripple, energetic ions formed from the beams are trapped in the ripple magnetic well, and drift upward to the central plasma region, where the ripple becomes small and the fast ions are detrapped. Deuteron beams with energy in the range 80 to 125 keV can thus

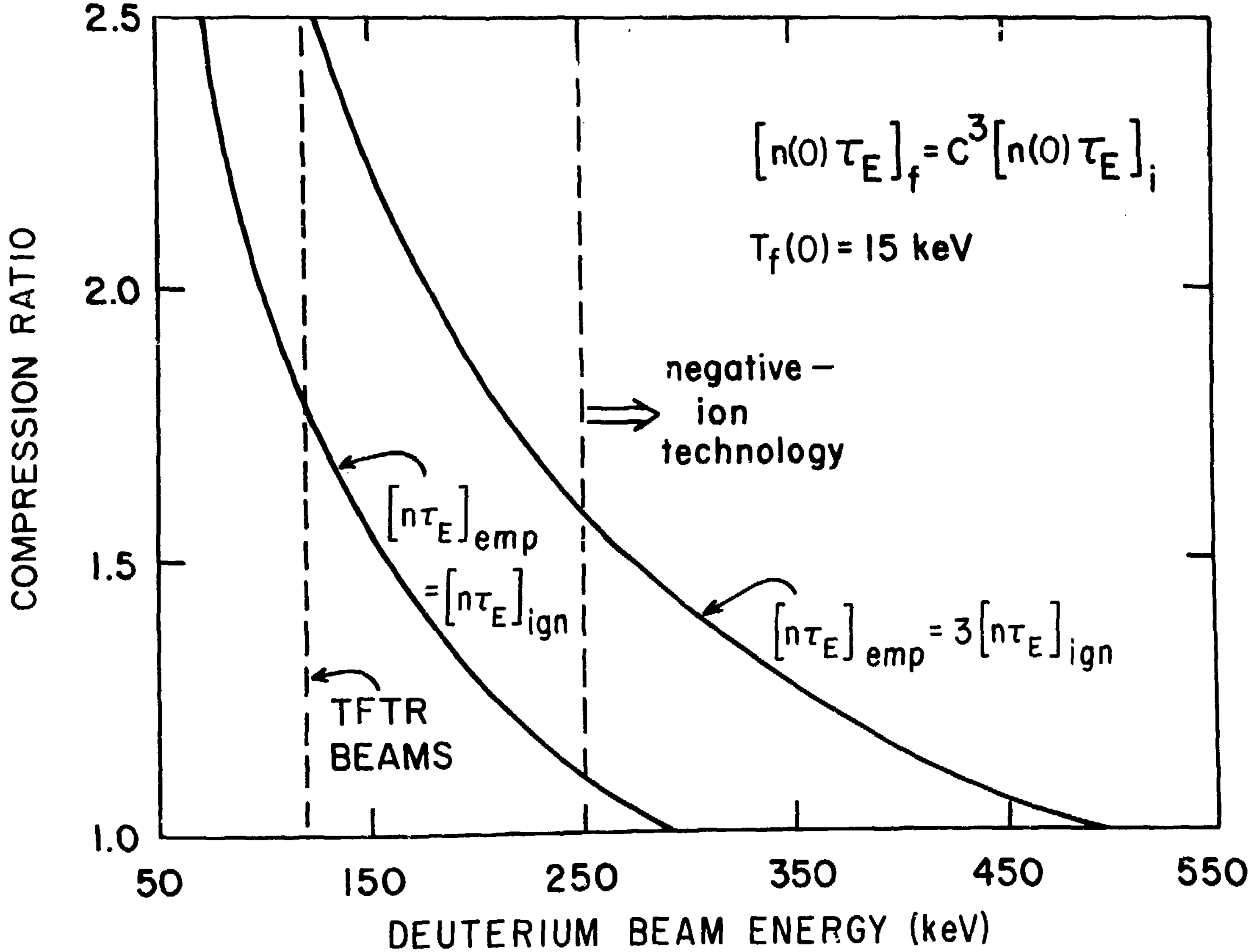

Figure 1. Dependence on beam energy of compression ratio required to reach the ignition $n\tau_E$, following beam heating. It is assumed that $\tau_E \propto na_p^2$.

penetrate into the center of toroidal plasmas of the highest opacities likely to be encountered.

The first test of ripple injection will take place on the ISX-B device at ORNL in late 1978 or 1979. However, the injection current will be rather small ($\lesssim$ 5 A), so that potentially deleterious effects such as electric field build-up and velocity-space instabilities may not arise until higher current experiments are performed.

Practical difficulties in applying ripple injection include access for near-vertical injection, space for "ripple coils" for inducing the vertically asymmetric ripple, and perhaps increased structural support for the toroidal-field coils in the vicinity of the ripple coils.

(4) Edge Heating with Layer-by-Layer Build-up. If limitations on beam technology make it difficult to heat the plasma center -- or if the reservations on edge heating listed at the beginning of this section turn out to be unfounded — then various edge heating scenarios may be attractive. In one set of techniques, the plasma density is built up, layer by layer, by deceleration of injected energetic ions with energy in the range of 20 to 100 keV. (Present and planned mirror-machine plasmas are started-up in an analogous fashion.) This scheme can thus solve simultaneously the problems of initial fueling and heating an ignition-sized plasma. In practice beam energies closer to 100 keV would be utilized, because of the problems of charge-exchange loss and need for large access areas at low E_b.

(5) Outer Plasma Heating. Another scenario envisages heating the center of a relatively tenuous ignition-sized plasma to the 5 to 10-keV range. The density is then rapidly increased by gas puffing or pellet injection. Alpha particle heating maintains the central temperature, while increasing plasma density causes the beam heating to become restricted to the outer plasma region. Numerical calculations have demonstrated the viability of heating to ignition with 150 to 200-keV beams — in the absence of impurity influx.

An urgent task for transport-code evaluations is to examine the effectiveness of the edge heating schemes when impurity influx from wall sputtering is included. Present tokamak experiments indicate that it is important to keep the edge of the plasma cold, to minimize impurity influx. The effectiveness of outer plasma heating in maintaining the center hot can be tested in TFTR and JET, although alpha-particle heating of the centers of these plasmas is likely to be only weak.

Several other relevant points were raised in the discussions:

(1) Elongated plasmas can have smaller half-widths for a given $n\tau_E$, in principle, so that beam penetration to the center is easier.

(2) Helium neutral beams can be produced with much higher efficiencies than positive-ion-based D° beams in the energy range above 150 keV (although D° penetration is somewhat better). Pumping of He gas by liquid-helium-cooled surfaces covered by frozen argon seems feasible. A serious drawback of helium injection is dilution of the plasma by thermalized helium ions, a problem that aggravates the ash build-up resulting from D-T burning.

6. METHODS FOR FACILITATING RF HEATING

• On the basis of the TM-3 experiments, ECRH seems attractive for electron heating in toroidal plasmas, but present tubes are incapable of delivering high power at high efficiency for long pulses in the frequency range of interest ($f \gtrsim 100$ GHz). Again, major-radius compression could be used to advantage, as the cyclotron frequency in the precompression plasma is reduced by a factor C from that in the compressed plasma.

• In experiments with RF injection in the lower hybrid and ion cyclotron frequency ranges, much of the RF energy has sometimes gone into producing energetic-ion populations near the plasma edge. If the RF energy were injected vertically into a region of significant ripple, fast ions would have a time-averaged vertical drift. To exploit this phenomenon, the same vertically asymmetric ripple configuration would be used as for ripple injection with monoenergetic neutral beams, but in the RF case, only ions that have been accelerated above a certain energy would reach the central region (i.e., $\Delta z \propto E^{5/2}$).

• Ion cyclotron heating occurs in a frequency range which is generally too small to permit the use of waveguide couplers, especially in fusion devices such as tokamaks where there has been a strong trend toward more compact design.

To enable the use of waveguides, there is incentive to inject at higher harmonics of the ion cyclotron frequency. Alternatively, the wavelength can be reduced by dielectric loading of the waveguides, but the tolerance of the dielectric to the fusion-neutron flux is questionable.

7. SUMMARY OF DEVELOPMENT REQUIREMENTS

7.1 Electromagnetic Waves

Adequate power sources for RF injection are available now at frequencies $\lesssim$ 10 GHz, but the following two requirements are outstanding:

- The development of efficient, high-power, long-pulse, millimeter-wave sources in the frequency range $\gtrsim$ 75 GHz should be expedited.

- Neutron-resistant coupling structures should be developed for lower-frequency RF injection schemes for which waveguides are impractical.

7.2 Neutral Beams

By using special start-up scenarios, it appears possible to heat ignition-sized toroidal plasmas to ignition temperature with 120-keV TFTR-type beams, but certain penalties must be paid:

- The TFTR injector efficiency is only 20 to 25%, so that for injection powers of 50 MW or more, the power supply costs may be prohibitive.

- The larger power at the fractional beam energies (~ 35% of the total) can result in overheating of the plasma edge and enhanced impurity influx by wall sputtering.

- 120-keV beams require more access at the torus than needed for higher energy beams of the same power.

- If the start-up scenario involves major-radius compression, or ripple injection, significant modifications to the 'standard' tokamak device configuration are required.

• If the start-up scenario calls for preferential heating of the outer region of an expanding plasma, then even more stringent impurity control measures may be called for.

The following are the most urgent improvements required for neutral-beam injectors, listed in order of priority.

(1) Increase of the D^+ fraction of the extracted ion beam to at least 90%. This improvement is essential for achieving (i) the highest production efficiency of full-energy neutral beams, (ii) the minimum wall sputtering by hot neutrals originating from a plasma edge heated by fractional-energy ions, (iii) the minimum density increase in the outer plasma region, by deposition of fractional-energy ions, and (iv) maximum possible direct recovery efficiency (if employed).

(2) Development of efficient energy recovery systems for beamlines. As shown in Fig. 2, with a 90%-D^+ beam and 80% recovery efficiency of the energy of unneutralized deuterons, ideal injector efficiency would be 60% at E_b = 150 keV (an appropriate energy for use with major-radius compression) and 30% at E_b = 250 keV (an appropriate energy for heating the center of an ignition-sized plasma). These efficiencies do not take into account transmission loss in the beam duct or use of a neutralizer of less than ideal length, effects which reduce the TFTR injector efficiency, for example, from 29% to 22%. The practical upper energy limit with D^+ beams and direct energy recovery may be about 200 keV, however, because the required area of the ion source becomes impractically large when the neutralization efficiency drops below about 20%.

(3) Long-pulse ($\gtrsim$ 5 sec) operation. Actively cooled, highly transparent electrodes and high-throughput beamline pumping systems must be developed for multi-second operation.

(4) Development of high-current ($\gtrsim$ 20 A) negative-ion beams. Heating of the center of ignition-sized plasmas would then be possible with injector efficiencies of at least 50% at $E_b \gtrsim$ 250 keV, even using a gas (rather than plasma-cell) neutralizer. However, the main justification for the development

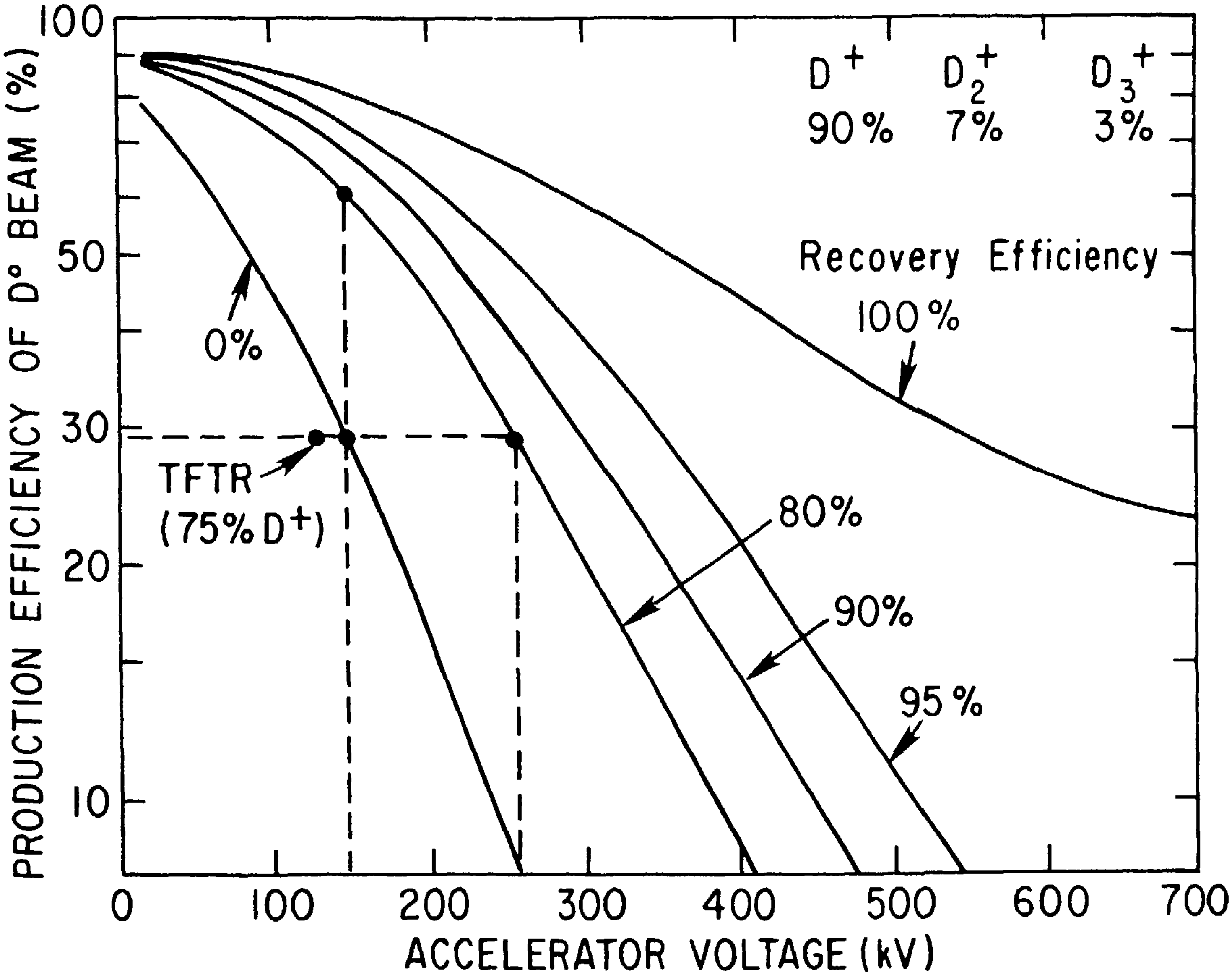

Figure 2. Efficiency of production η of D° beams, where η = (power in full-energy neutral beam)/(total power in ion beam, less recovered power). Transmission loss in beam duct not included. [From J. W. Stearns, K. H. Berkner, and R. V. Pyle, in The Technology of Controlled Nuclear Fusion (Proc. 2nd Top. Mtg., Richland, WA, 1976) IV, 1221.]

of negative-ion beams at this time is for their potential use in certain types of magnetic mirror machines. Although injection energies up to 400 keV (D°) could eventually find use in toroidal devices, the various operating scenarios that have been discussed in this summary would allow satisfactory performance with beams in the energy range 80 to 200 keV (D°).

Conclusion. The most important near-term task of injector development for tokamak applications is to perfect long-pulse, reliable, efficient, and (hopefully) inexpensive D^+-based beams in the energy range of 80 to 200 keV. These considerations hold for ignition devices as well as for low-Q fusion neutron sources. For the latter application, and also for mirror machines beginning with the MFTF, steady-state injector operation must be achieved.

Summary of L. Stewart Presentation and of the Discussion Following the L. Stewart, D. Jassby, and G. Sheffield Presentations

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The efficiencies which can be achieved with neutral beam systems based on positive ions, negative ions and direct recovery are shown in Fig. 1. These can be matched to the necessary efficiencies as determined by reactor design studies to give the following required neutral beam technology:

(1) For standard mirror reactors and most driven tokamak reactors, $\gtrsim$ 200 keV and $\gtrsim$ 70% efficiency are needed, and these requirements demand negative ions with direct recovery.

(2) For hybrid mirror reactors and CIT-type driven tokamak reactors, ~ 120 keV and $\gtrsim$ 50% efficiency are indicated. Negative ions with or without direct recovery or positive ions with direct recovery would be required.

(3) For ignited tokamak reactors, $\gtrsim$ 150 keV and 50% efficiency may be sufficient, although below ~ 250 keV there will be straightforward penetration to the center, so untested techniques such as low-density startup or edge heating must be used. The ~ 50% efficiency requirement is determined by the allowed capital investment in the neutral beam system rather than by power balance, and it is not well known at this time. Negative ions with or without direct recovery would be sufficient. Positive ions with direct recovery would be sufficient for energies below ~ 200 keV, so techniques which do not

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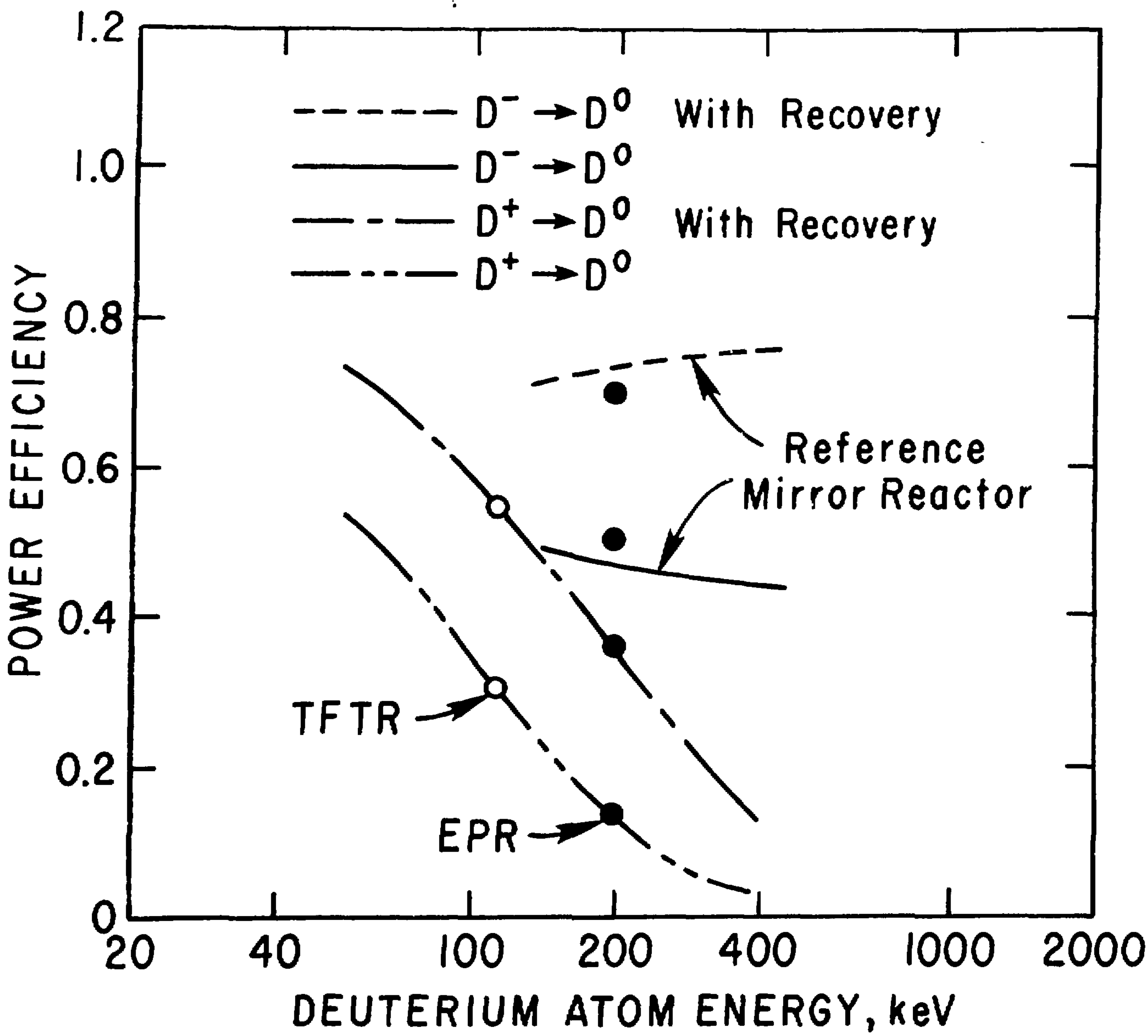
require full penetration must work for this case. Positive ions by themselves would require too much capital investment in the neutral beam system.

(4) For ripple-injected, ignited tokamak reactors, 80-120 keV and ~ 50% may be sufficient. Ripple-injection is again an untested concept, and again the efficiency requirement is set by capital investment. Ripple-injected, ignited tokamaks are in a class by themselves because at 80 keV and 50% efficiency, existing positive ion technology is viable. For energies higher than ~ 80 keV, negative ions with or without direct recovery or positive ions with direct recovery would be sufficient, and existing positive ion technology would be sufficient if the efficiency requirement is relaxed.

The required neutral beam technologies as indicated by reactor design studies are summarized in Table 1. The injection concepts which require little or no extension of present technology need to be tested. At the same time, in case these concepts do not work, negative ions and direct recovery (and RF heating) must continue to be developed.

The lower-energy injection concepts should be tested as soon as possible. Ripple-injection tests on ISX-B have been partially funded through TFTR and are planned for FY79. Edge-heating, low-density startup, and injection-compression scenarios require large-bore machines for testing. The first definitive tests for these concepts may have to be on TFTR in the early 80's, although some tests of edge-heating may be possible with PLT at high densities and using D° injection.

Funding of development and technology is given in Table 2, and is determined by considering near-term commitments vs. the long-range requirements discussed above. Neutral beams from positive ions receives the bulk of available funding because of its proven effectiveness, because it is required for almost all of the upcoming machines, because it is the basis for future negative ion systems, and because it may be used in the future by itself or with direct recovery. Relatively more funding will go to technologies other than positive ions if more funding becomes available.



1.2
1.0
0.8
0.6
0.4
0.2
0
POWER EFFICIENCY
$D^- \rightarrow D^0$ With Recovery
$D^- \rightarrow D^0$
$D^+ \rightarrow D^0$ With Recovery
$D^+ \rightarrow D^0$
Reference
Mirror Reactor
TFTR
EPR
20
40
100
200
400
1000
2000
DEUTERIUM ATOM ENERGY, keV

TABLE 1

Required Neutral Beam Technologies
As Indicated by
Reactor Design Studies

	D^- & Direct Recovery	D^-	D^+ & Direct Recovery	D^+
Mirrors & Most Driven Tokamaks	Maybe	No	No!	No!!
Hybrid Mirrors & CIT-Type Driven Tokamaks	Yes	Yes	Yes	No
Ignited Tokamaks	Yes!	Yes	Maybe	No (High Cost)
Ripple Injected Ignited Tokamaks	Yes	Yes	Yes	Maybe (High Cost)

TABLE 2

D&T FUNDING OF SELECTED PROGRAMS IN FY1978.

Postiive Ions	8 M$
Negative Ions	2 M$
Positive Ion Direct Recovery	0.45 M$
ECRH Tube Development	0.9 M$
Fueling (Not Neutral Beams)	0.7 M$