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PLASMA HEATING R & D ASSESSMENT

PLASMA PHYSICS LABORATORY

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PLASMA HEATING R & D ASSESSMENT

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DISCLAIMER

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1. INTRODUCTION

1.1 OBJECTIVES

The purpose of this report is to compare the heating requirements of INTOR with the present state-of-the-art of tokamak plasma heating technology and demonstrated heating performance, and also with the technology expected by 1983-84 according to development and testing programs in place. This comparison results in a set of recommendations for a heating technology development program for the 1980s.

The range of possible heating scenarios is covered in the companion report by the Energy and Particle Confinement Team. Heating requirements have been established by that team and by the INTOR Concept Advisory Group.

1.2 POSSIBLE HEATING METHODS FOR A TOKAMAK REACTOR PLASMA

Table 1-1 lists the preferred methods for heating tokamak plasmas. To date, the most widely successful method has been energetic neutral beam injection. In addition, ion cyclotron wave heating has generated promising results in large tokamaks, wave heating in the lower-hybrid frequency range has experienced mixed results in a large number of experiments in smaller tokamaks, and electron cyclotron heating has proved effective in several small tokamaks. Adiabatic compression heating has also been successful in a number of relatively small tokamaks. Other proposed methods, such as relativistic electron beam injection, have barely been attempted so far.

TABLE 1-1
PREFERRED PLASMA HEATING METHODS

METHOD	ADVANTAGES	DISADVANTAGES
Neutral-Beam Injection	Demonstrated effectiveness	Significant development needed. Neutron streaming problems. High cost.
Radio-Frequency Waves	Less development needed. Reduced neutron problems. May be cheaper.	Heating effectiveness uncertain. Launching structures near plasma (in some cases).
Major-Radius Compression*	Demonstrated effectiveness	Oversized vacuum vessel. Very large EF power.

*Used to multiply effectiveness of NBI or RF heating.

The application of neutral-beam injection heating to reactor plasmas seems feasible from the plasma physics point of view, but the method faces a host of technological problems which are summarized in this report. It is these problems, and the high capital cost of beam injectors, that provide the continuing incentive to investigate alternative heating techniques. But these alternatives, too, have problems. Adiabatic compression requires oversized vacuum chambers and rapid changes in large magnetic energies; ion cyclotron heating faces difficulties in developing wave launching structures for the reactor environment; the physics of lower-hybrid wave heating in tokamak plasmas is still not understood and its effectiveness uncertain; the power tubes for electron cyclotron heating face substantial development problems, and this heating method has undergone only limited testing.

The effectiveness of high-power neutral-beam injection heating has now been demonstrated in several tokamaks. Despite apparent disadvantages, this method offers at the present time the best assurance of providing adequate heating for the INTOR plasma. Consequently, neutral-beam heating has been adopted as the reference heating method. Whether beam heating will be displaced eventually by another technique as the reference method depends to a great extent on the results of heating experiments during the next five years. In the present concept, radio-frequency (rf) heating methods are retained as important backup options.

2. HEATING REQUIREMENTS FOR INTOR

2.1 POWER LEVELS

The power required to heat the INTOR plasma to ignition is not known definitively, because of the continued uncertainty in high-temperature transport coefficients (although upper limits can be given), and because a variety of schemes for reaching ignition has been devised. These schemes are discussed by the Start-Up Team and result in power requirements in the range 40 - 75 MW. It seems advisable to specify a heating capacity at the upper end of this range, as any margin in this specification will be useful when one or more heating stations is forced out of action for maintenance or by breakdown.

2.2 PULSE LENGTH

2.2.1 Ignition Scenario

Depending on transport scaling and on the heating scenario adopted, a heating pulse length of 3 to 8 s is required for INTOR. However, provision for utilizing a heating system during the burn is required if an ignited plasma becomes unignited — for example, because of impurity (or alpha) buildup, or minor plasma disruptions that reduce plasma temperature or density — and then must be reignited or driven at high Q to complete the 100-s burn pulse. The required heating capacity for this purpose could come from other than the primary heating system, but it is likely to be at least 10 MW.

2.2.2 Driver Systems

In the unlikely event that ignition cannot be attained at all, then the heating systems must be capable of long-pulse operation in order to provide the neutron fluence requirement of INTOR in high-Q operation ($Q \geq 5$). Maintenance of the reacting plasma just below ignition conditions may actually turn out to be desirable for thermal stability control.

Thus for any operating mode, the capability of long-pulse operation of at least a significant fraction of the INTOR heating system capacity is an important basic requirement.

2.3 SUMMARY OF NEUTRAL-BEAM HEATING STATUS RELATED TO INTOR

Heating of the INTOR plasma to ignition conditions seems feasible using neutral-beam energies achievable with positive-ion-based technology (120-200 keV per deuteron), but with significant advancements beyond TFTR beam technology required to obtain long pulses at high duty factor, as well as high currents at the upper end of this energy range. Other development requirements are an increase in overall efficiency to be obtained primarily by direct energy recovery, and increased reliability and component lifetime.

Potentially serious disadvantages of the beam heating method are the exposure of the injector internals to the fusion neutron flux, and the expected high capital cost of beam systems. The difficulties incurred by the direct exposure of injector components to neutrons can probably be overcome, especially if the beam injectors can be closed off after start-up.

2.4 SUMMARY OF RADIO-FREQUENCY HEATING STATUS RELATED TO INTOR

The continued attraction of pursuing the many different rf heating schemes derives in part from the perceived disadvantages of neutral-beam injection mentioned above. To date, only ion cyclotron wave heating (ICRF) has provided a convincing demonstration of substantial plasma heating in tokamaks of significant size. RF heating in the lower-hybrid frequency range still has uncertain effectiveness. The power sources for electron cyclotron heating face formidable problems of development and cost. [Low-frequency ($\omega \ll \omega_{ci}$) methods are not addressed in this report.]

The greater success of ICRF heating, the reasonably straightforward extrapolation of the ICRF technique to INTOR, and the relative inexpensiveness of ICRF systems, make it the favored rf technique. Still, before any rf method can displace neutral-beam injection as the preferred heating technique for INTOR, that method must demonstrate the multi-keV temperature increases already obtained with neutral-beam injection, and convincing designs for reasonably efficient and neutron-resistant wave launching structures must be devised and tested.

1. PHYSICS OF NEUTRAL BEAM INJECTION

1.1 BEAM PENETRATION

The penetration of a neutral beam into a tokamak plasma is determined by impact-ionization collisions with the plasma electrons and ions, and by charge-exchange collisions. As far as the effect of impurity ions is concerned, the most recent measurements and theoretical analyses indicate that charge-exchange trapping on impurity ions dominates over impact ionization up to beam energies of at least 100 keV (10^8), and that the charge-exchange cross-section scales nearly as $Z^{1.2}$ [1,2]. Thus for the values of Z_{eff} which are permitted by the need to obtain ignition in INTOR (i.e., $Z_{eff} \leq 2$), impurity-ion trapping is expected to be relatively unimportant (see Fig. 3-1). Most experiments to date are consistent with "classical" beam penetration, but there is some indication that in high-density PLT plasmas the penetration of neutral beams is somewhat poorer than expected.

Various start-up scenarios for INTOR have been devised with near-perpendicular injection in the horizontal plane, and indicate that penetration of neutral beams with energy E_b in the range 120 to 200 keV per deuteron is satisfactory [3]. Until further experimental data is available on penetration, however, one must be cautious about specifying an E_b in the lower end of this range.

A potential method of providing enhanced penetration of large reactor plasmas is the ripple-injection method [4] which makes use of a vertically asymmetric ripple configuration. When neutral beams are injected vertically from the side of stronger ripple, energetic ions formed from the beam are trapped in the ripple magnetic well and drift upward to the central plasma region where the ripple is small and the fast ions are untrapped. This method has many potential problems, such as a possible enhanced loss of the tail of the bulk-ion distribution in the ripple field, and the need to provide space in the blanket for special ripple coils. A test of ripple injection will take place in the ISX-B tokamak in 1979-80.

1.2 FAST-ION LOSS CHANNELS

The efficiency of neutral-beam heating may be reduced by several loss channels as for the fast ions:

1.2.1 Charge-Exchange Loss

For $E_b = 120$ to 200 keV, the concentration of neutrals introduced into the plasma by charge-exchange trapping of the beams is relatively small, and is in fact negligible for $E_b > 150$ keV. However, the fast ions can charge exchange with the ambient neutral population and also with other injected beam neutrals; some of the resulting fast neutrals may escape the plasma. In present experiments this loss can be important at lower plasma densities, and it might be significant if tangential injection at lower E_b were used in the INTOR plasma.

Fig. 3-1 Effective trapping cross section versus beam energy, defined as

$$\sigma_{\text{eff}} = \frac{Z-Z_{\text{eff}}}{Z-1} (\sigma_{\text{cx}}^{\text{H}} + \sigma_{\text{p-ion}}^{\text{H}}) + \frac{\langle \sigma V \rangle_{\text{e-ion}}}{V_{\text{beam}}} + \frac{Z_{\text{eff}}-1}{Z(Z-1)} (\sigma_{\text{cx}}^{\text{imp}} + \sigma_{\text{ion}}^{\text{imp}}),$$

for a deuterium neutral beam traversing a deuterium plasma with an impurity of charge Z [1, 2]. (792225).

3.2.2 Orbit Loss

When counter injection or perpendicular injection is used, a fraction of the beam ions will be trapped banana orbits that intercept the limiter or divertor scrape-off layer. Initially untrapped ions could also be lost by pitch-angle scattering into an escape orbit while slowing down, but this effect is expected to be unimportant at the high-plasma currents needed in INTOR.

3.2.3 Ripple Loss

The discrete toroidal-field coils set up a periodic toroidal field ripple which may result in local magnetic mirrors between TF coils. Fast ions injected or scattered into these mirrors experience a vertical drift which may carry them out of the plasma. This loss process was especially severe on the FFR device with near-perpendicular injection [5], and must be avoided on INTOR by careful choice of the injection angle.

3.3 MICROINSTABILITY EFFECTS

Microinstabilities that arise during neutral-beam injection can in principle cause an anomalously rapid thermalization of the fast ions or an enhanced radial diffusion rate. The first effect is unimportant for heating a thermonuclear plasma such as INTOR, and could even be desirable in the face of competing fast-ion loss processes, but the second effect must be avoided. Theoretical analysis predicts that ion cyclotron waves may arise during intense beam injection and in general the amplitude of drift-wave turbulence may be enhanced, but no adverse effect on plasma heating or on fast-ion confinement is expected [6].

In experiments to date strong density fluctuations have often been observed by microwave scattering measurements, and were especially pronounced during high-power co- and counter-injection at low densities in the PLT device [7]. However, the characteristic temperature scaling $\Delta T_i \propto P_{\text{beam}}$ observed at low- and moderate-beam powers continued throughout the higher-power regime where the intense fluctuations became evident. Thus these instabilities, still unidentified, had only a minor effect on the bulk-ion power loss. Examination of the fusion-neutron production rate showed also that the fast-ion slowing-down process continued to be classical in the presence of this instability.

Microinstabilities are also expected to arise when the beam velocity exceeds the Alfvén speed [6], but this condition will not occur in the range of INTOR injection energies.

3.4 PREFERRED INJECTION MODE FOR INTOR

As indicated above, ignition conditions in INTOR can apparently be attained with $E_b = 120$ to 200 keV, using near-perpendicular injection in the co-direction ($\phi_b \approx I_p$) in the horizontal midplane. This injection mode results in greater beam penetration for a given E_b , and the co-injected trapped ions are carried further inward on their banana orbits. Near-perpendicular injection is also

more favorable for physical implementation, as less blanket need be penetrated by the beam duct, and the shielding around the beamline is more easily installed.

The direction of injection must be at least 10° off-normal to minimize losses due to ripple trapping and to limiter-intercepting orbits. The possibility of beam-induced instabilities arising due to intense perpendicular injection must still be investigated experimentally. Before the injection direction can be fixed, an analysis of the effect of the INTOR ripple is required, as well as stability and heating results from the perpendicular injection experiments on the PDX tokamak in 1980-81.

3.5 EXPERIMENTAL STATUS

The slowing-down process of injected fast ions has been investigated on at least 7 tokamaks so far, using charge-exchange and neutron diagnostics, and found to be in accordance with classical theory [8]. Heating has been more effective with the bulk ions than with electrons, as expected with the relatively low E_b (≤ 40 keV) used to date, where most of the beam energy is given up to the bulk ions. In smaller tokamaks, counter-injection has often resulted in an influx of impurities which clamped the electron temperature and thereby limited the ion temperature. In higher-power experiments with co-injection alone, and in PLT with co- and counter-injection together, it has been possible to control impurity influx, and there has been apparently little if any deterioration of electron or ion energy confinement time.

Table 3-1 summarizes the heating results obtained with neutral-beam injection during the last several years. The most spectacular results have so far been obtained in PLT, where at low density the central T_i was raised from 1.0 to 6.5 keV and the central T_e from 2.6 to 4.0 keV [9]. Smaller temperature increases were measured at higher density, with $\Delta T_i \propto P_{\text{beam}}^{1/3}$. Significant electron heating could be obtained only with water-cooled graphite limiters, and with titanium gettering of the vacuum vessel wall. On ISX-B, large electron heating has also been observed with stainless-steel limiters and without any gettering [10].

TABLE 3-1
RECENT EXPERIMENTAL RESULTS WITH NEUTRAL-BEAM INJECTION

Device	Year	Beam Power (MW)	Beam Energy (keV)	Pulse Length (ms)	Average Density (10^{13}cm^{-3})	Max ΔT_i at $r = 0$ (keV)	Max ΔT_e at $r = 0$ (keV)	Average Toroidal Beta
ORMAK	1976	0.36	30	50	2.8	1.0	0.4	0.01
TFR	1976	0.45	34	50	5.0	1.0	0.25	0.01
T-11	1978	0.5	35	30	3	0.8	0.2	0.014
PLT	1978	2.5	39	150	1.5	5.5	1.5	0.01
PLT	1978	2.0	39	150	4.0	1.0	1.0	0.
ISX-B	1979	1.0	40	100	5.0	0.9	0.9	0.

In these experiments the relative fast-ion pressure has been at least as high as that expected in INTOR, with apparently no adverse affects. Average toroidal betas of 1.8% and peak betas of 9% have now been reached in beam-heated circular plasmas in ISX-B [10].

Table 3-2 summarizes the beam-heating experiments that are planned for the next four years in U.S. tokamaks. The results of these experiments should provide the more detailed data base necessary for the confident application of neutral-beam injection at longer pulse lengths to the heating of INTOR plasmas.

TABLE 3-2
PLANNED NEUTRAL-BEAM INJECTION EXPERIMENTS ON U. S. TOKAMAKS

Device	Year	Beam Power (MW)	Beam Energy (keV)	Pulse Length (msec)
PLT	1979-80	3.2	40	200
ISX-B	1979	1.8	40	100
	1980	3.0	40	100
PDX	1980-81	6.0	50	500
Doublet III	1980-81	7.0	80	500
	1982-84	20	30	1000
TFTR	1982	27	120	500
	1983-84	35	120	1500

4. NEUTRAL-BEAM TECHNOLOGY

4.1 STATES AND EFFICIENCY OF NEUTRAL-BEAM SYSTEMS

Table 4-1 gives the state-of-the-art parameters of neutral-beam injectors. On operating tokamaks, the PLT/ISX neutral-beam systems [11] have achieved the following parameters: $E_b = 45$ keV, $t_b \leq 0.4$ s, $P_b = 600$ kW injected H^0 per beamline with overall power efficiency $\eta_I = 0.25$ to 0.35, and $P_b = 850$ kW injected D^0 per beamline with $\eta_I = 0.45$. The effective transmission efficiency $\eta_t = 0.55$ ($\eta_t = \eta_I/\text{neutralization efficiency}$). The six-source 2X11B injectors [12] have achieved $E_b = 18$ keV, $t_b = 10$ ms, $P_b = 3.5$ MW injected D^0 per beamline with $\eta_I = 0.42$ and $\eta_t = 0.50$.

TABLE 4-1

NEUTRAL-BEAM INJECTOR TECHNOLOGY BASED ON POSITIVE IONS

	Present Status		Expected Status in 1983 on Fusion Machines	INTOR Requirement
	on Fusion Machines	on Test Stands		
<u>Ion Source</u>				
Current Density (A/cm ²)	0.4 at 40 keV	0.4 at 40 keV 0.3 at 120 keV	0.3 at 120 keV	*
Extracted Current (A)	65 at 40 keV	100 at 50 keV 65 at 120 keV	80 at 80 keV 65 at 120 keV	*
Monatomic Fraction	0.85	0.85	0.90	≥ 0.90
Gas Efficiency	0.50	0.50	0.50	0.50
Maximum Pulse Length (s)	0.4 at 30 keV	1.5 at 120 kV 10 at 25 kV	1.5	5-100
Lifetime-Source (s) -Accelerator (s)	10 ⁴ - 10 ⁵ 10 ³	10 ⁴ - 10 ⁵ >10 ⁵	10 ⁵ >10 ⁵	3 × 10 ⁵ 3 × 10 ⁵ (In neutron field)
<u>Beamline</u>				
Beam Energy (keV)	45	120	120	150-175
Neutral Beam Current** (Amp-equivalent D ⁺)	30 at 40 keV 250 at 20 keV	40 at 50 keV	90 at 120 keV 190 at 80 keV	120 at 150 keV†
Pumping Speed (1/s)	4 × 10 ⁵	3 × 10 ⁶	3 × 10 ⁶	1 × 10 ⁷ †
Transmission Efficiency	0.55	0.75	0.75	≈ 0.8
Overall Power Efficiency **	0.45 at 45 keV		0.35 at 120 keV	0.35 at 150 keV
Direct Recovery Efficiency	0	See Table 4-2		≥ 0.6
Power Density at Torus Port (kW/cm ²)	1.5	2.0	4.0	*

*To be determined.

**Including full and fractional-energy beam neutrals.

†Assuming each beamline supplies 15 MW of neutral beam power.

In the present beam development program, LBL and ORNL have operated ion sources at 110-120 kV and 15-20 A for $t_b \geq 0.25$ s [13,14]. The TFTR development source [15] has operated at 120 kV and 65 A up to the power supply limit of 20 ms. The TFTR prototype neutral beam system [16], which is just beginning operation at LBL, will have one of three sources installed and powered by a 120 kV, 65 A, 0.5 s power supply. The projected η_I is 0.24 for 120 keV D^0 and $\eta_I = 0.37$ including fractional-energy components. By the end of 1982, three of these beamlines should be operating on TFTR, each delivering 6.5 MW of 120 keV D^0 and 8.4 MW including all D^0 components.

4.2 SPECIES MIX

Figure 4-1 shows the fraction of neutral-beam power at full energy for the INTOR probable energy range of $E_b = 120$ to 200 keV. The monoatomic-ion fraction, f_{D^+} , of the extracted ion beam is as large as 0.80 to 0.85 in operating systems [17] and test stands [18]. A design goal for systems operating in 1983 should be $f_{D^+} = 0.90$. This value or higher is desirable to insure adequate plasma penetration by most of the neutral beam and to maximize the usefulness of direct energy recovery systems.

4.3 DIRECT ENERGY RECOVERY SYSTEMS

Figure 4-2 shows projections of η_I versus E_b for INTOR-like-beam systems with various efficiencies of direct recovery [19,20,21]. This figure is the result of an analysis which accounts for losses due to beam optics, fractional energy components, reionization of beam neutrals, less-than-equilibrium density in the neutralization cell, and charge exchange, ionization, and secondary-electron production in the grid gaps. Operating and planned beamlines on tokamaks do not yet include direct recovery systems. The results that have been obtained on test stands [22] are shown in Table 4-2. There has so far been no demonstration of direct recovery with the combination of beam energies, beam power densities, and pulse lengths characteristic of existing or future injector systems.

TABLE 4-2
DIRECT ENERGY RECOVERY EXPERIMENTS

Laboratory	Beam Energy (keV)	Pulse Length (s)	Total Current* (A)	Total Current Density* (mA/cm ²)	Ion Current (A)	Ion Current Density (mA/cm ²)	Recovery Efficiency
Livermore	100	0.5	2	7	1.5	5	0.6
Oak Ridge	37	0.1	20	200	7	70	0.6 - 0.2
Fontenay	70	1.0	20	250	13	150	0.45
Needed for INTOR	150 to 175	5-100	>>20	>250	>>13	>150	>0.6

*Ions + Neutrals

Fig. 4-1 Fraction of neutral-beam power in the full-energy component for various species mix in the ion beam. (793571)

Fig. 4-2 Power efficiency, defined as neutral beam power delivered into the torus divided by total electrical power into the injector. h^+ fraction = 90%. (796064)

Direct recovery is desirable for reducing power supply capacity (and cost), as well as for reducing power handling requirements in the beamline. The higher efficiency obtained with direct recovery (see Fig. 4-2) is also a valuable step toward commercial reactor requirements.

4.4 NEGATIVE-ION BEAMS

Consideration of negative-ion-based neutral beams is motivated by potential advantages of power efficiency at $E_b \geq 160$ keV and by the potential option of efficient beam production at $E_b > 200$ keV (see Fig. 4-3). This advantage would be useful if higher energies are eventually thought to be desirable for penetrating to the center of denser INTOR plasmas, or if the required number of beamlines is to be reduced by going to higher energies at the same beam current. However, the development of D^- systems [23] is expected to be insufficiently advanced by 1983 to allow a commitment to a D^- -based neutral-beam system for INTOR. Nevertheless, there remains the possibility of replacing one or more D^+ -based systems with a D^- system at some later date during INTOR operation.

4.5 LONG-PULSE BEAM DEVELOPMENT

While the reference pulse length for heating the INTOR plasma to ignition is 4 to 8 seconds, it is highly desirable to have a 100-s pulse length capability (the duration of the burn). This requirement implies essentially steady-state operation for the INTOR duty factor of 0.8, but in practice only a fraction of the heating capacity is likely to be needed for an extended period throughout the burn. On present-day test stands, ORNL has operated a PDX ion source [24] at 40 kV and 140 A for 0.5 s, and also a relatively small ion source [25] at 25 kV and 8 A for 10 s. The Lawrence Berkeley Laboratory has operated a fractional area TFTR ion source [14] at 120 kV and 8 A for 1.5 s. Neutral-beam systems operating by 1983 on PDX, Doublet III, TFTR, and MFTF are expected to have 1 to 2 s pulse capability. No insuperable problems are foreseen in extending the pulse length as required, but a major development program is needed.

While the INTOR reference beam energy has been chosen as 150 keV, the ability to operate at somewhat higher energy (> 175 keV) is desirable to obtain moderate improvements in penetration and in beam-duct transmission efficiency. Consequently, the beam development program should concentrate on the range 140 to 175 keV.

4.6 DELIVERABLE POWER DENSITY

The PLF neutral-beam system [11] delivers 1.5 kW per cm^2 of torus port area. By 1983 the TFTR beam system [16] is expected to be delivering 3 kW/cm^2 and the Doublet III system [26] about 4 kW/cm^2 at the torus.

For INTOR, the power per unit area should be maximized in order to minimize the size of the injection ports, and thereby minimize the neutron streaming problem. A high power density is more difficult to achieve with the long duct necessary to penetrate the INTOR blanket and shield [27].

Fig. 4-3 Comparison of overall neutral-beam power efficiencies for various production methods of negative-ion-based beams and a positive-ion-based beam with 60% efficiency of direct recovery. Full-energy beam component only. (793570)

Fig. 4-4 Neutron flux including backscattered neutrons along the wall of a duct perpendicular to the vacuum vessel, for a fusion neutron wall loading of 1 MW/m^2 . (793573)

Because of the ambient magnetic field, beam neutrals that are ionized by collisions with gas molecules in the beam duct are deflected to the duct walls. Bombardment of the wall may liberate more gas, leading to an enhanced rate of beam ionization and partial blocking of the beam. The duct pressure build-up observed in the PLT experiments [28] must be further understood because this phenomenon may impose design limits on power density and duct length.

The potential for significant duct reionization losses can be reduced markedly by designing high conductance ducts, insofar as such design is compatible with the need to minimize neutron streaming problems, and also by insuring a low gas evolution rate from the duct wall through the proper choice of wall materials and coatings. It is important that neutral beam test facilities have the capability to simulate tokamak magnetic fields in the duct region, so that the extent of reionization losses and the effectiveness of countermeasures can be determined.

Specific values of power densities and the number of ports on INTOR will require specific neutral-beam designs and total layouts, and will require more extensive prototype testing than in current development practice to establish relationships between power density and reliability/availability.

4.7 EFFECTS OF NEUTRON ENVIRONMENT

There is no experimental data and little analysis on problems resulting from neutron streaming into neutral-beam systems. There will not be any on-line damage data by 1983. It has been speculated that there may be degradation in performance of filaments, grids, and insulators. Figure 4-4 shows the fall-off in neutron flux, including backscattered neutrons, with distance up to the beamline [29]. While a definitive analysis needs to be made, a comparison of the streaming neutron flux with the damage thresholds for various materials indicates that damage effects may be tolerable [30] if the ion source and accelerator structure can be replaced semi-annually, for example. Heating of cryopanels may be so significant [29, 30, 31] that the use of cryosorption pumps or getter pumping should be examined carefully as alternatives.

If radiation damage of the ion source and/or accelerator structure is a problem, a potential solution is a beam bending magnet to allow the source to be out of direct line-of-sight from the plasma. Other possible advantages of a bending magnet include the reduction of streaming gas and of lower energy beam components going to the plasma. However, a magnet introduces problems with beam optics and control which can more than offset any advantages, so that it should be avoided if possible.

A more likely solution to the problem of damage caused by neutron streaming is provision of a 30-cm thick plug valve which is closed after ignition is attained. The plug valve would also result in a large reduction in the heat load on the cryopumping system. This bulky valve would have to be opened and closed reliably once every 120 secs, probably by rotation. However, the valve could not be used with any injectors that may have to operate for a substantial fraction of the burn pulse (see Sec. 2.2).

In any event it appears that the lifetime of ion source components will be limited by the present-day problems of erosion, thermal cycling, and so forth, rather than by neutron bombardment.

4.1 BEAMLINE PUMPING SYSTEM

Figure 4-5 shows cutaways of the PDX beamline, revealing the important components. A PLT beamline cryopump [32] has a speed of 4×10^5 liters/second (l/s) and requires 2.5 hours for regeneration. The TFTR cryopump [16] has a design speed of 3×10^6 l/s and a regeneration time of eight hours. While the TFTR pumps could operate for 10,000 one-second shots from the viewpoint of hydrogen-explosion safety, the need to limit tritium accumulation results in more frequent regeneration.

For the INTOR beamlines, cryopumping could provide sufficient pumping speed and acceptable regeneration intervals, but it has two potential disadvantages. Build-up of tritium inventory may require unacceptably frequent regeneration, and cryopanel heating by neutrons may be significant. Zirconium-aluminum getter pumping [33] should be investigated as an alternative, as it would not release the tritium in the event of a vacuum leak and is not sensitive to small temperature rises.

If significant amounts of helium must be pumped in the beamlines, special cryosorption [34] or cryotrapping [35] pumps may have to be developed.

4.2 IMPURITY INFLUX FROM INJECTORS

Species analysis of the ion beams extracted from an ORMAK source [36] and a fractional area TFTR source [37] have indicated the presence of up to 2% oxygen. There is also evidence that neutral-beam injectors have introduced oxygen into 2XIIB [38] and copper into PLT [17], but the injected impurity levels were too small to degrade plasma performance. Before a specific development program in this area can be justified, more information on beam impurity levels is needed from test stands and tokamak heating experiments.

4.10 MAGNETIC SHIELDING

The stray magnetic field at the ion source must be restricted to about 1 G. As the field outside the TF coils may be hundreds of gauss, magnetic shielding must be provided for the ion source, the beam acceleration region, and the beam extraction region [16, 32]. The neutralizer region must be shielded to reduce the stray field here to about 10 G. The Doublet III and TFTR neutral-beam system designs include additional finger shielding [39] between the neutralizer and the bending magnet to reduce the stray field in this region while allowing high pumping speeds. The INTOR beamlines may also require magnetic shielding of the direct recovery subsystem.

Fig. 4-5 Elevation view of
the PDX neutral-beam injector
firing into the PDX plasma.
(791032)

4.11 INNOVATIVE POWER SUPPLY REQUIREMENTS

By 1983 several additional high-voltage power systems should be operational: 120 kV, 65 A for TFTR and 80 kV, 80 A for Doublet III and MFTF. Because of long lead times, these systems were specified at a time when little or no operational experience was available with very high voltage ion sources. Thus, in many respects, these systems are similar to those used on test stands, with good regulation and wide ranging voltage selection, and capability for frequent interrupts on all supplies. For the reactorlike conditions of INTOR, the sources will have to have evolved to a higher level of reliability, and certain simplifications which may be possible in the power system [40] should be investigated: elimination of high-voltage regulation by feedback regulation of the arc power supply to maintain optimally focused beam at various accelerator voltages; no fast high-voltage switching, relying instead on star-point regulators and contactors in the primary lines; elimination of fast-restarting-after-a-fault requirement; elimination of wide-selection-range for accelerator voltage; etc.

None of these options has yet been tried. They will require interactive development of sources and power systems.

4.12 AVAILABILITY, MAINTENANCE, AND RELIABILITY

4.12.1 Availability

The requirement that INTOR have a high availability (up to 50% is presently the goal) is a stringent requirement on systems such as the neutral beams which require routine maintenance. With a start-up time of 5 to 10 seconds and as many as one pulse every 120 s, each injector will be required to operate for up to 10^3 s per month. (Those injectors with longer on-times during each pulse would have a correspondingly greater requirement.)

4.12.2 Maintenance

Simple hot filaments used as cathodes generally have a lifetime in the range 10^4 to 10^5 seconds, and it may well be necessary to replace even more advanced cathodes and other components subject to erosion, such as the grids, on a monthly basis. Neutron irradiation experiments are required to determine the effect (if any) of neutron bombardment on normal replacement intervals.

4.12.3 Reliability

Each component must have very high reliability in producing the rated power during the rated lifetime (e.g., if there were 20 ion sources each with a 99.5% probability of successful operation on each pulse, then on average the system would provide only 90% of the maximum rated capability).

4.12.4 Redundancy

It is important that components requiring regular maintenance (e.g., the ion source and recovery system) can be isolated, so that the rest of the system

can continue to operate. If the required power of 60 MW is produced by four beamlines each with three ion sources all working perfectly, it would be advisable to have an extra beamline since beamline failure would lead to an extended downtime. In addition, to permit regular maintenance during operation, it would be desirable to have an extra ion source per beamline. Thus we arrive at five beamlines, each with four injectors, rated for 100 MW total. To give 60 MW, there must be four beamlines always operating (rated for 80 MW maximum), and requiring 98% reliability of each of the 12 ion sources.

4.13 FEASIBILITY OF TRITIUM INJECTION

Tritium beams would be needed if penetration of tritons to the plasma center could not be obtained by pellet injection or gas puffing, or if certain beam-driven operating modes were to be adopted [8]. However, none of these cases is appropriate to the reference INTOR operating scenario.

To give the same penetration as 150 keV D⁰ beams, the T⁰ beam energy must be 215 keV, but the power efficiencies of the T⁰ and D⁰ beams would be about the same. The principle problem encountered in the tritium injector is excessive build-up of tritium inventory in the ion source and elsewhere, which would hamper maintenance. Hydrogen could possibly be used as the neutralizer gas.

In view of the questionable need for T⁰ beams in INTOR, there is no strong case at this time for a special development program.

4.14 COST OF NEUTRAL-BEAM SYSTEMS

The reference INTOR neutral-beam system has been costed as installed hardware dollars per delivered watt of full energy neutral-beam power [21]. This cost is estimated from the projected costs for individual TFTR beam system components by scaling in power, power loading, current, voltage, or gas loading, as appropriate. The cost of a prototype including system engineering is scaled from the projected \$12 M for the prototype TFTR 6.5 MW, 120 keV D⁰ system.

Assuming five 15 MW beam systems (possibly one redundant) at E_b = 150 keV for INTOR, with 60% efficient direct recovery, the total system cost including prototyping and engineering is estimated as \$90 M. It should be emphasized that this estimate is based on projected costs for the TFTR beams. Estimates based on the actual costs of PLT and ISX beams are significantly higher.

Development funds will be required for at least one beam test stand. Consideration should be given to upgrading the Livermore 200 keV test stand, or the Berkeley NSTF, if one of these facilities can be made available in time for developing the INTOR beams.

5. PHYSICS OF RADIO-FREQUENCY HEATING

5.1 PHYSICS AND EXPERIMENTAL STATUS OF HEATING IN THE ION-CYCLOTRON RANGE OF FREQUENCIES (ICRF)

Since ICRF heating was first introduced into tokamaks in 1973, experiments at power levels up to about 400 kW have been conducted on the ST, ATC, PLT, TFR,

T-4, and T-4-1 Vch tokamaks [41-46]. The present experimental status is summarized in Table 5-1. There are current plans to carry out megawatt-level experiments on PLT, Alcator, TFR, and T-10 tokamaks. The main purpose of the high power experiments is to determine the feasibility of employing ICRF heating in hot reactor-sized plasmas. Consequently, the emphasis of these studies is being placed on evaluating the physics of the wave-plasma interaction to permit extrapolation to reactor conditions.

Once the physics base is established, the effort will shift to the development of wave launching structures suitable for the reactor environment (Sec. 6.1).

5.1.1 Fast Wave Heating

In principle, a variety of heating schemes can be utilized in the ion-cyclotron frequency range [47-55]. Primary heating regimes include (a) the two-ion regime for which minority fundamental cyclotron damping, majority second-harmonic cyclotron damping, and electron Landau damping contribute to the absorption of rf energy according to the dictates of the ion-ion hybrid resonance effect on the dispersion relation; (b) the pure second harmonic resonance regime; (c) possibly the pure fundamental resonance regime; and (d) under certain conditions the regime for which mode conversion into ion-Bernstein waves allows electron Landau and transit-time absorption mechanisms to become substantial.

In early heating experiments, heating efficiencies of about 20-40% were achieved but were hampered by substantial plasma density rise, indicating surface heating and loss of energetic ions. As the machine size and Ohmic heating current have been increased, a systematic improvement in energetic ion confinement has resulted, as expected, and the mode spectrum has improved so as to increase the heating efficiency and considerably reduce the influence on recycling characteristics.

The best data presently available is for regime (a), the two-ion regime. Recent experiments have been performed for damping dominated by direct minority fundamental cyclotron-resonance damping in PLT [43] and for damping dominated by two-ion hybrid mode conversion in TFR [44]. In the case of PLT, the minority distribution is made very energetic and serves as a conduit of the rf energy to the majority ion distribution. This regime is quite similar to ECRH and scales very favorably to reactor conditions. In the case of TFR, sufficient minority concentration is maintained to permit mode conversion so that a large fraction of the power heats the electrons directly. In both cases the heating efficiencies appear to be $\geq 70\%$ and minimal perturbation of the density is found.

5.1.2 Power Deposition Profile

For optimization of ICRF heating in large tokamak systems, control of rf power deposition profile is essential. Adequate control will assure that the rf power is delivered to the core plasma and that surface heating resulting in enhanced impurity influx is negligible. Since the strength of damping processes is approximately proportional to the beta of the species involved [51,52],

TABLE 5-1

EXPERIMENTAL STATUS OF ICRF

Device	f (MHz)	P_{rf} (kW)	Launching Structure	Coupling Efficiency % (Z)	Heating Efficiency % (Z)	ΔT (eV)	Pulse Length (ms)	Heating Mechanism
ST	25	<450	Two 1/2-turn coils	90	20	200 D^+	1	Fundamental two-ion hybrid
ATC	25	<200	1/3-turn cop- per strip	50	<40	200 D^+	10	Two-ion hybrid
T-4	23	50	3/4-turn coil			200 D^+	1	Direct minority heating
TFR	50-90	80	Tandem 1/2- turn coils	80	70	150 D^+	30	Two-ion hybrid
PLT	25	350	1/2-turn coils	70	270	500 (up to 60 keV)	100	Two-ion hybrid, D^+ (direct minority heating, H^+)
TM-1 Veh	7-35	150	Two asymmetric coils	80-90	30-40	100	0.8	Two-ion hybrid
DIVA	50	<100	1/2-turn coil	65	20	100 H^+	1	Second harmonic
Macrotron	1-20	5200	Rectangular loop			200	2-4	Off resonant
TO-1	48	<5	Coil			50	20	

*These efficiencies are defined differently from those used in Section 5.

considerable central peaking of the power deposition profile should ensue regardless of the wave k spectrum excited. However, selection of the k spectrum, as well as to avoid waves which have large fields and field derivatives in the plasma surface, should serve to virtually eliminate surface heating [49]. Some k selection is inherent in the dispersion relation for the fast wave, since waves relatively near onset couple more strongly to the antenna. Additional k selection is easily incorporated into the antenna design. Finally, under conditions for which distinct toroidal eigenmodes can exist, mode tracking can be used to obtain a very favorable power deposition profile.

Much of the experimental effort over the next few years will be directed toward determining the power deposition profiles and the control which can be exercised over them for the several ICRF regimes. In all experiments to date, these profiles have been centrally peaked even under unfavorable plasma conditions, so that adequate control is expected to be achieved for reactor conditions.

5.1.3 Future Plans

The goals of the larger planned experiments are given in Table 5-2. These plans are aimed at producing significant heating in the hottest plasmas available to assess the physics of the wave damping, the damping profile and means to control it, and the potential of the several heating schemes for extrapolation to reactor conditions. As indicated by the initial results on PLT and TFR, the physics of ICRF heating and its relevance for reactor conditions should become clear during the next few years from the experiments performed on the larger, higher-current tokamaks. With the recommended development program, these results will be available in time to impact the choice of primary heating technique for INTOR.

TABLE 5-2

PLANNED LARGE TOKAMAK EXPERIMENTS

Device	Expected rf Power	Launching Structure	f (MHz)	Pulse Length	Heating Mechanism
Macrotron	1 MW	Rectangular Loops	1-20	10	Stochastic Heating
Alcator A	200 kW	Tandem Coils	150-200	10	Second Harmonic
Alcator C	>4 MW	Tandem Coils	150-200	10	Second Harmonic
PLT	5 MW	Multiple 1/2-Turn Coils	25	200	Two-Ion Regime
			55	200	Second Harmonic
TFR	1 MW	Tandem Coils	60	10	Two-Ion Regime
T-10	1 MW		60	100	Two-Ion Regime

5.2 PHYSICS OF LOWER HYBRID HEATING

5.2.1 General Features

Lower hybrid heating (LHH) offers one of the most attractive engineering outlooks of any of the present options for auxiliary heating of INTOR, although the viability of this choice is dependent upon a definitive experimental demonstration of its effectiveness. The anticipated flexibility of lower hybrid heating is derived from the ability to heat electrons or ions, and to control the spatial deposition profile of the energy.

5.2.2 Coupling

The lower-hybrid wave utilized for both ion and electron heating is an electrostatic wave which requires a slow wave structure for launching. Since the launched wave is absorbed in a single pass across the plasma, eigenmode structure is not involved. However, the choice of the parallel wavelength is restricted by accessibility and penetration requirements. That is, the imposed parallel refractive index ($n_{||} = k_{||}c/\omega$) is bounded from below, in order to couple to a lower-hybrid mode which has access to the plasma center. An upper limit on $n_{||}$ arises from the need to control linear mode conversion to thermal modes and/or electron Landau damping to achieve energy deposition near the plasma center. Successful launching of the lower-hybrid waves has been demonstrated by a variety of techniques: simple loops [56], slow wave structures [57], and phased waveguide arrays [58-62]. The phased waveguide arrays are the most desirable for reactor applications, and offer the potential for tracking the parallel wavelength if required. These waveguide array couplers are often described by a theory developed by Brambilla [63]; quantitative agreement with the basic features of this theory was first demonstrated on the linear device H-1 at Princeton [64]. Some tokamak experiments have also verified this theory, but the heating effects on Alcator A were independent of phase [65], a result that disagrees with Brambilla theory. Concern remains about high power nonlinear effects on coupling [66].

5.2.3 Wave Propagation

Several tokamak experiments have provided indirect evidence of wave propagation to the center through the observation of spatially localized power absorption. Probe measurements up to 4 cm inside the limiter in Doublet IIA confirmed the existence of resonance cone propagation (i.e., a LH energy propagation channel) and the waves were observed to have the proper phase velocity as determined by the slow wave launch structure [67]. Recent CO₂ laser scattering experiments in Alcator A with a two waveguide array launcher have detected the lower-hybrid waves inside the plasma, but with slower phase velocity than expected [68].

5.2.4 Ion Heating

One physical process for transferring the lower-hybrid wave energy to the plasma ions is ion Landau damping of either the incident slow LH wave or the warm plasma waves produced by linear mode conversion. This heating mechanism

was most clearly demonstrated by ion heating in the Wega [56] and JFT-2 [61] experiments. (However, it is not clear whether the bulk ions were heated directly or whether an ion tail was produced with subsequent thermalization.) A competing heating mechanism is the damping of short-wavelength products of parametric decays. Evidence of this heating was observed in NTC [69] and JFT-2 [61], although the JFT-2 experiments indicate a saturation of the heating due to the decay waves. Ion heating mechanisms still need extensive documentation.

5.2.5 Electron Heating

By choosing the frequency of the lower-hybrid wave large enough to avoid linear mode conversion inside the plasma, the energy can be transferred preferentially to the electrons by electron Landau damping. Strong quasi-linear Landau absorption of the wave energy by electrons also requires the parallel wavelength be chosen to satisfy $\omega/k_{\parallel} v_e \sim 3$. A test of this scheme on toroidal devices is presently in progress on both the Doublet IIA [70] and Octupole [71] devices at General Atomic Company.

5.2.6 Applications for INTOR

5.2.6.1 Bulk Heating

Lower-hybrid waves can be utilized to provide bulk heating directly to the ions or through the electron channel since electron-ion collisional coupling is sufficiently strong. Energy deposition at the center is difficult with reactor plasma parameters due to convergence of accessibility and penetration limits on n_H , but ignition scenarios have been devised. An ignition scenario was illustrated by Harvey and Rawls [72] for a Doublet TNS (The Next Step) device utilizing a quasi-linear electron Landau damping model of the wave absorption coupled with a one-dimensional transport code.

5.2.6.2 Current Profile Control

The potential to spatially control the energy deposition with electron Landau damping provides a method of modifying the current profile during plasma build-up and to some degree during ignition. However, considerable heating power may be required, which would adversely affect the reactor power balance.

5.2.6.3 Other Applications

It has been proposed [73] that unidirectional lower-hybrid waves can be used to drive the plasma current, thus allowing steady-state tokamak operation. By proper control of the wave spectrum, the auxiliary heating and plasma current can be provided from the same source.

Another interesting application of lower hybrid heating is the possibility of enhancing fusion reaction rates by ion tail heating near the plasma center. Enhanced neutron production has been observed in the Alcator A experiments. A similar possibility exists by coupling lower hybrid heating with neutral beam injection. The ATC lower-hybrid experiments observed a strong interaction with the neutral beam which resulted in increased perpendicular energy of the injected beam [74].

5.2.7 Future Plans

The present U. S. program plan includes two major lower hybrid heating experiments, on PLT and Alcator C, which will hopefully provide the required demonstration of significant heating along with a confirmation of the physics basis which are necessary before lower hybrid heating can be a viable choice for INTOR. A test of lower-hybrid current drive is also planned on PLT and Versator II. At present no approved plans exist for a follow-on to these experiments.

5.3 PHYSICS OF ELECTRON CYCLOTRON HEATING (ECH)

The physical model of high frequency heating is that of electron-cyclotron resonance. The absorption of the radiation should be dominated by classical processes. Bulk heating of $kT_e \sim 100$ -200 eV was obtained in the TM-3 tokamak at $\omega/\omega_c = 1$ and also $\omega/\omega_c = 2$, with 60 kW of 60 GHz radiation [75].

The limits on plasma parameters for ECH are determined by wave penetration requirements. Extraordinary wave heating at $\omega = \omega_c$ gives the lowest frequency requirement for a given plasma density. However, in this case the radiation must penetrate from the inboard side of the plasma, where access may be very difficult.

5.3.1 Application to INTOR

Ordinary wave heating at $\omega = \omega_c$ can be obtained with heating from the outboard side of the plasma. The penetration requirement for ordinary heating at $\omega = \omega_c$ is that $\omega_p^2/\omega_c^2 < 1$, which sets a limit on $\langle \delta_e \rangle$. At the reference design toroidal field of 5.2 T at the center, heating at $\omega = \omega_{ce}$ requires a frequency of about 145 GHz. The cutoff for ordinary wave heating of $\omega_p < \omega_c$ limits the density to $2.6 \times 10^{20} \text{ m}^{-3}$ at the center or $1.3 \times 10^{20} \text{ m}^{-3}$ line average, or slightly less than the reference design of $1.4 \times 10^{20} \text{ m}^{-3}$. To avoid refraction, a somewhat lower density is required — less than about $2.3 \times 10^{20} \text{ m}^{-3}$ at the center. For this density, the volume average of δ_T for a central temperature of 20 keV is limited to about 5%.

Gyrotrons could also play an important role in the startup of the INTOR plasma, by preionization and initial heating (to about 10 eV) with consequent reduction in the required loop voltage.

A possible difficulty with gyrotron heating is that most of the power goes to the electrons, and thus is especially susceptible to loss by impurity or cyclotron radiation. On the other hand, the frequency dependence of gyrotron heating provides a possible means of electron temperature profile control. For this purpose some control of gyrotron frequency is desirable.

If INTOR operates at a significantly lower magnetic field value, such as 4.0 T, the gyrotrons must have a two-frequency capability in any event, or two sets of tubes must be supplied.

5.3.2 Near-Term Experiments

An experimental ECH program on ISX-B will start in the summer of 1979, using 200 kW, 100 msec pulses at 28 GHz, and 100 kW, 10 msec pulses at 35 GHz. Proposals have been made for tests on ISX-B at the 1 MW level in 1981. Other experiments may possibly be carried out during 1981-83 on PDX, Doublet III, and Macrotron.

Table 5-3 compares some of the parameters of the forthcoming ECH experiments on T-10 and ISX-B.

TABLE 5-3

T-10 VERSUS ISX-B ECH EXPERIMENTS

T-10	ISX-B
$B_0 = 3 \text{ T}, \nu = 86 \text{ GHz}$	$B_0 = 1 \text{ T}, \nu = 28 \text{ GHz}$
Ordinary Wave	Extraordinary Wave
Launch at Large R	Launch at Small R
Polarize at Gyrotron	Polarize at Tokamak
$P_{rf} < P_{oh}$	$P_{rf} \sim P_{oh}$
$\tau_{rf} \sim \tau_e$	$\tau_{rf} \gg \tau_e$
$n_e \leq 8 \times 10^{13} \text{ cm}^{-3}$	$n_e \leq 10^{13} \text{ cm}^{-3}$

6. TECHNOLOGY OF RADIO-FREQUENCY HEATING

6.1 ICRF TECHNOLOGY SCALING TO INTOR

An ICRF heating system for startup and ignition of a device such as INTOR consists of a radio-frequency source, power transmission and matching system, and the wave launching structure or antenna. Some relatively minor extensions of the present technology of the power source and transmission system will be required for high-power long-pulse (or CW) operation. However, a concerted R and D effort will be required to evaluate the possible launching structures

suitable for reactor operation under conditions which will permit optimization of the antenna design. A startup modeling and reactor design study has been conducted for ICRF for the tokamak conceptual tokamak reactor [76] and a conceptual ICRF system and antenna designs have been considered for the EPR [77]. Many of the problems which must be addressed for reactor operations, such as neutron and plasma heat loading of the launching structures and the design and placement of external rf components, are discussed in more detail in Refs. [76] and [77].

6.1.1 External Component Technology

External component technology, which embraces all components outside the neutron shield (i.e., excluding the wave launching structure and its power feed), is well in hand but will benefit from some modest improvements. Megawatt tetrodes and coaxitrons suitable for ICRF heating sources currently exist with efficiencies of ~ 90%, but are now rated for pulse lengths of only 0.1 - 0.35 s at peak power. Although CW operation is now possible at the MW level for a single tube, development may be desirable to obtain the long pulse 5 - 100 s (essentially CW) operation envisioned for INTOR, at the highest possible power per tube. Power transmission can be accomplished easily with 10 - 20 MW capacity, 23 cm diameter lines which are commercially available. Furthermore, in the ICRF frequency range transmission over hundreds of feet incurs negligible loss of power, permitting the source to be located external to the reactor containment building, removing the need for remote handling maintenance of the source. Although radiative cooling of the line may suffice, liquid cooling with, for example, borated water should be investigated. Finally, double stub tuners are presently used to match the antenna feed impedance to the source and are likely to work well with minimal modification for the INTOR device. However, an optimal scheme should be developed during the next decade.

6.1.2 Launching Structures and Feeds

The one area for which the technology is only crudely developed is that of launching structures and feeds suitable for reactor operation. To date, work has concentrated on establishing the physics base with structures and feeds already proven reliable for current devices. In all previous ICRF experiments and those planned for the near future, coupling loops have been and will be employed to launch the waves.

Efficient wave coupling > 90% is easily achieved with loop antenna systems located in the shadow of the plasma limiter, when such coupling systems are suitable. This includes all devices for which the heat load does not seriously increase the resistance of the loop elements. It is possible that loop systems can be developed for INTOR with adequate cooling of the system elements. The all-metal antennae recently employed on TFR [78] and T-4 [79] suggest that such cooling may be feasible. (All elements, including shields, are of sufficient size to permit liquid cooling.) Furthermore, insulators within the radiation shield may be eliminated, thereby avoiding unnecessary neutron damage of the loop system.

In addition to loop coupling, other wave launching structures need to be seriously explored for the reactor regime. Flush-mounted cavity apertures, dielectric filled waveguide, ridged waveguide, and strip line couplers are possible contenders. As the machine size is scaled up, these structures become more attractive, but remain virtually unexplored experimentally.

The feeds to the launching structure in INTOR can also be similar to the coaxial transmission lines now in use. These can be routed around magnetic field coils and at angles through the radiation shield to minimize neutron flux escape. Placement of the vacuum break will be crucial to avoid excessive neutron damage.

Power levels through ports on ICRF experiments have ranged from 1 - 10 kW/cm² without breakdown, and peak electric field intensities of 10 kV/cm have been obtained in the loop region. It is expected that these levels will remain at comparable values as scaling to INTOR is achieved, depending on the success of the R & D effort envisioned.

6.1.3 Launching Structure Research and Development

Detailed design and testing of launching structures on large-size high-power-density tokamaks over the next decade will be required to advance the technology sufficiently to assure a suitable design for INTOR. Such an rf testing facility (RFTF) could be operated under long-pulse operation to provide large wall heat loading to simulate such conditions in INTOR and serve for the development of rf launching structures generally. In particular, PLT could be upgraded to give several keV plasmas for 1 - 3 s to serve as a RFTF. In addition to studies on such a facility and materials testing, the effect of INTOR neutron flux levels on the launching structures should be investigated during the same time frame.

6.1.4 Reliability

Reliability of the external components is already very high, since many of these components have been in service for many years. Also, rapid maintenance is insured by the remote placement of these components. The main concern will be the launching structure reliability. A demonstration of both suitability and reliability must be the goal of the RFTF effort.

6.1.5 Cost Factors

Large ICRF power systems including all of the elements discussed above (once developed) are expected to cost significantly less per delivered watt than neutral-beam injectors. (One notes that the cost of the PLT ICRF system was \$0.75 per watt.) The proposed RF Testing Facility to develop suitable launching structures would cost a few million dollars per year and should be used for several years.

6.2 TECHNOLOGY OF LOWER HYBRID HEATING

6.2.1 General Features

In the area of technology, lower hybrid heating appears to be the most attractive of all present heating options. The existing technology is capable of providing the required power in the appropriate frequency range with high conversion efficiency. The ability to remotely site the power supplies, transport power large distances with waveguides through convenient paths with bends, and the simple, potentially radiation resistant, coupling structure provided by waveguide arrays considerably simplify the engineering requirements on the auxiliary heating system. The engineering design of a complete lower-hybrid heating system is discussed in Refs. [80] and [81]. An estimate of the overall dc to rf efficiency of the system is about 60% without thermal recovery.

6.2.2 Source Technology

At present 500 kW klystron tubes at 2.4 and 2.1 GHz with conversion efficiency of 53% and 55 dB gain can be obtained from Varian Associates. A modest development program can provide a 1 MW tube around 2.0 GHz with conversion efficiency of 80% and 55 dB gain. It is estimated that five 1 MW tubes will be driven from a single high voltage power supply of conventional design rated at 75 kV, 83.3 A, and 6.25 MW [80]. Ion heating for INTOR at 5 T would require a frequency of about 2.5 GHz while electron heating would require about 5.0 GHz.

6.2.3 Power Transmission

The major part of the waveguide run from the klystron to the launcher will be pressurized with SF_6 to avoid breakdown, and the windows separating the pressurized and evacuated regions should be removed from the direct neutron flux. A dc break is required in the free waveguide to prevent possible ground-loop interference and for personnel safety. It may be necessary to have a pressurized waveguide section through the point of electron-cyclotron resonance (near the TF coil). Waveguide constructed of beryllium-coated copper appears to be the desired choice of waveguide material with direct exposure to the plasma. Window material will probably be BeO or Al_2O_3 unless radiation resistance is the most important consideration, for which Y_2O_3 would be preferred [80]. The waveguides and grills should operate at low temperatures to avoid radiation-induced swelling; hence, areas exposed to considerable neutron irradiation will require cooling.

6.2.4 Launching Structures

The waveguide array grills have been tested with power densities up to 5 kW/cm² without breakdown, and the planned Alcator C experiment is designed for 10 kW/cm². Possible nonlinear effects, including parametric decays and ponderomotive effects, which can modify the wave coupling and propagation may restrict power densities to values lower than the technological limits. Coupling efficiencies of greater than 90% have been achieved with a four waveguide array [82].

6.2.5 Reliability and Cost

The reliability of klystrons is known to be quite good, and availability of the lower-hybrid systems on Doublet IIA up to the 0.5 MW level has been excellent. The required heating power for INTOR can be obtained from a lower-hybrid heating system at an estimated cost per delivered watt comparable with that for neutral-beam injectors.

6.3 TECHNOLOGY OF ELECTRON CYCLOTRON HEATING

Assuming 80% efficiency of transmission from the tubes to the plasma, 75 MW of gyrotron output power will be required for delivery of 60 MW into the INTOR plasma. This could be done with 500 gyrotrons of 150 kW each. Assuming that 4 kW/cm^2 can be transmitted through the port windows, a port area of 1.88 m^2 is required. Port area should be doubled to about 3.75 m^2 to accommodate supports for window pieces. The gyrotron radiation should reflect from mirrors that would allow the angle of launch to vary.

It may be possible to use optical waveguides to conduct the gyrotron radiation to the reactor environment. The use of mirrors will facilitate access and prevent neutron streaming into the gyrotrons. The positioning of a polarizer which transforms circularly polarized gyrotron radiation to linearly polarized radiation could be a major problem. The proposed efficiency of gyrotron heating in INTOR should be around 80 - 90% (efficiency of absorption of the high-frequency radiation).

The main obstacle to the application of electron cyclotron heating is the present primitive level of tube technology at $f \geq 35 \text{ GHz}$. High-frequency ($\geq 150 \text{ GHz}$) gyrotron tubes with power levels on the order of 100 kW and efficiencies $\geq 20\%$ must be developed by the mid to late 1980s. This requirement should be compared with the state-of-the-art, namely, 20% efficiency for a long-pulse tube delivering 200 kW at 28 GHz. The expected large cost of the tube together with the cost of the power supplies and the transmission system result in a cost per delivered watt that is considerably larger than that for neutral-beam injectors.

Contracts have been let by ORNL for the development of a 200 kW, 110 GHz CW tube for delivery in 1982-83.

7. OTHER HEATING METHODS

In addition to neutral-beam injection and the various rf techniques discussed in the preceding sections, a number of other methods for heating tokamak plasmas have been proposed, and are listed in Table 7-1.

7.1 ADIABATIC COMPRESSION

A tokamak can be compressed and heated either by increasing B_z in time with the plasma at fixed R (minor-radius compression), or by increasing the vertical field at fixed $B_z(r)$, thus displacing the plasma inward in R to a

TABLE 7-1

OTHER METHODS OF PLASMA HEATING

	Tokamak Experiments Performed	Major Results	Physics Outlook	Technology Prospects (Reactors)
1. Major-Radius Compression	ATC	$T_i \propto C^{4/3}$, $T_e \propto C$	Excellent	Very Good
2. Minor-Radius Compression	TUMAM TOSCA	$T \propto C^{4/3}$, Enhanced τ_E (Both machines)	Very Good	Poor
3. Turbulent and Shock	TTT Various small tokamaks	$\Delta T_e \sim 0$, Energetic ion tails $\Delta T_e \sim 0.5$ keV Energetic ion tails τ_E maintained	Poor	?
4. Relativistic Electron Beam	SPAC V	Injection and trapping $\Delta I_p \sim 30$ kA $T_i \sim 300$ eV $q \ll 1$	?	Good
5. Cluster Injection	None		?	Fair
6. Plasma-Gun Injection	None		?	Good
7. Laser-Pellet Hot Plasma Formation	None		?	Fair

new equilibrium at higher B_t (major-radius compression) [83,84]. Minor-radius compression has the serious disadvantage that a large fraction of the toroidal field energy must be changed in a time smaller than τ_E , which is impractical in the largest present-day normal-coil tokamaks, and certainly out of the question for reactors with superconducting coils.

Major-radius compression was proven to be effective in the ATC device [85]. However, a significant technological requirement for a reactor is the large pulsed power (~ 500 MW) that must be provided for the EF system. Nevertheless, because R-compression is a proven and well-understood heating method that can multiply the energy input of neutral beams or rf, and because of the prospect of using it for thermal stability control, this technique continues to play a potential role in tokamak reactor design [86]. Further experimental investigation of this method is planned for TFTR and for JET.

7.2 TURBULENT HEATING

Two important questions for the application of turbulent heating to reactor-sized plasma are whether the rapidly rising electric field can penetrate the plasma, particularly as T_e increases from the Ohmic-heating value, and whether the normally good plasma energy confinement will be degraded by turbulence. In a large plasma, the bulk of the heating seems likely to occur at the edge [87] with subsequent energy loss to the wall and inefficient heating of the central plasma region.

7.3 RELATIVISTIC-ELECTRON-BEAM INJECTION

The most impressive REB experiment to date in a toroidal device is the injection of 26 kJ of a 0.6 MeV REB into the SPAC V device at Nagoya ($a_0 \sim 10$ cm, $B_t \sim 5$ kG) [88]. Penetration to the center was achieved, and a T_e of 300 eV at n_e up to 10^{-4} cm $^{-3}$ was obtained with plasma stability apparently retained for the experimental time of 1 ms. Critical issues that must now be addressed experimentally are whether the REB can penetrate into the center of a tokamak plasma of interesting size, whether the fast electrons give up all their energy before they escape the plasma, and whether instabilities produced by REB injection will degrade the energy confinement time.

The technology for REB generation and transmission is well advanced. At least 2 MJ per pulse have been produced at very high efficiency (more than 65%) [89]. But a practical method of coupling the REB to a reactor plasma is still speculative.

7.4 CLUSTER INJECTION

Cluster injection has the advantages over neutral-beam injection that at high cluster energies no neutralization should be required, so that production efficiency can be very high, and steady-state operation is readily achieved [90]. But the apparatus is extraordinarily complicated, and it is difficult to see how the formidable energy requirement for heating a reactor-sized plasma can be satisfied. No experiments have yet been performed. It is likely that this method would find use only for fueling.

7.5 SUMMARY

Of all the "other heating methods" that have been proposed, only one — adiabatic compression — has been demonstrated to work at all in a tokamak of significant size with significant energy confinement times. Compression appears to have practical benefit only when employed in conjunction with some other auxiliary heating method.

At this time it seems likely that REB injection will be tested in one or more small tokamaks, but mainly to investigate its potential for current driving. Turbulent heating experiments will proceed in small toruses outside the U. S. But major-radius compression is the only "other heating method" that is planned for use in TFTR, JET or JT-60, or which could be confidently scaled to INTOR.

8. SUMMARY OF ADDITIONAL RESEARCH AND DEVELOPMENT NEEDS FOR NEUTRAL BEAM HEATING AND TECHNOLOGY

8.1 NEEDS PRIOR TO INTOR AUTHORIZATION — MAJOR DEVELOPMENT REQUIREMENTS

The following items are listed in approximate order of urgency:

- Ion Beam. Development of high-current D^+ beams at 150 to 175 keV is required.
- Pulse Length. Development of at least 10-second pulse length capability is required (ion sources, beam dumps, and power supplies).
- Direct Recovery. Development of direct recovery of unneutralized beam energy with at least 50% efficiency is required.
- Monatomic Fraction. At least 90% D^+ fraction in the ion beam should be attained reliably.
- Radiation Hardening. Adequate lifetime of all injector components against the expected neutron bombardment must be insured. Neutron irradiation programs for sensitive materials such as seals and accelerator grids should be initiated. Beamline pumping systems (cryopanel or getter pumps) must be able to operate in the anticipated neutron flux. If it is determined that the ion source must remain out of the line-of-sight of fusion neutrons, then a development program resulting in the demonstration of magnetic deflection of high-intensity ion beams is required.
- Reliability. Continued improvement in the reliability of critical injector components is needed.
- Negative Ion Beams. If a certain amount of beam power at $E_b > 200$ keV is eventually found to be required to heat the center of denser plasmas, then high-current negative-ion beams must be developed. At present no clear need is foreseen for this capability.

- Beam Test Stands. The most important task during the next few years is the further development of beam injector parameters. An investigation should be initiated immediately to determine whether one of the Beam Test Stands at Berkeley, Oak Ridge, or Livermore can be upgraded in time to pursue most of the above development items.

8.2 NEEDS PRIOR TO INTOR OPERATION

By 1983-84, the problems of achieving adequate reliability and ease of remote maintenance will become the most vital issues.

- Maintenance. Intensified work on neutral-beam reliability, and design modifications to facilitate remote maintenance will be required. A detailed assessment is needed to determine the reliability and maintenance requirements.
- Conditioning. To meet the reliability requirement, considerable effort will be required to determine the best materials and assembly techniques to minimize the need for time-consuming conditioning of the ion source.
- Power Supplies. Significant impact might be made on injector costs and operational simplicity by investigating proposed schemes for elimination of high-voltage regulation, high-voltage switching, and fast-restarting after faults (see Sect. 4.11).

9. SUMMARY OF ADDITIONAL RESEARCH AND DEVELOPMENT NEEDS FOR RADIO-FREQUENCY HEATING AND TECHNOLOGY

9.1 ICRF DEVELOPMENT NEEDS PRIOR TO INTOR AUTHORIZATION

The heating test program on the current generation of tokamaks should continue until the prospects for ICRF heating are fully established. Subsequent heating experiments for the attractive heating modes should then be performed on the next generation of tokamaks (e.g., TFTR and JET), where the problems of excessive heat loads and neutron deterioration of performance will have to be faced. The next step would be the application of ICRF heating to an ignition device such as INTOR.

In order to assure the timely development of the components needed both for TFTR and INTOR, a Radio-Frequency Test Facility (RFTF) should be established as soon as the physics base warrants, probably within two years' time.

The optimization of the loop coupling structures on PLT, TFR, etc., will serve to delineate the degree of wave spectrum selectivity required. This information can then guide the design of new types of coupling structures for testing on suitable devices, preferably on an RFTF. In this manner, all of the development required to optimize operation on TFTR, and subsequently on INTOR, can be done off-site and can lead to highly reliable operation on INTOR.

9.2 ICRF DEVELOPMENT NEEDS PRIOR TO INTOR OPERATION

Prior to INTOR operation, the following development needs are required. In addition to continued testing of structures in the plasma environment, materials testing relevant to ICRF heating is needed. Radio-frequency conductivity and the effects of surface blistering should be measured for candidate conductors (copper, vanadium, molybdenum) at neutron and plasma flux levels comparable to those expected in INTOR. Methods must be found so that all insulators (sintered beryllia, alumina, yttria) can be located out of the direct view of the neutron flux, and their compatibility with conductors, thermal stress, and neutron bombardment should be studied.

Finally, optimization of the external components should be pursued. In particular, 1 - 5 MW sources in the frequency range 25 - 75 MHz which are capable of long-pulse operation (practically CW) should be developed.

9.3 LOWER-HYBRID HEATING DEVELOPMENT NEEDS PRIOR TO INTOR AUTHORIZATION

Clearly, the pressing need for the LHH method is a definitive demonstration of significant heating in a tokamak and the corollary confidence in the physics understanding. The planned experiments for PLT and Alcator C are aimed at this purpose. It is also essential to determine in these experiments any nonlinear limitations on the coupling and heating to ensure optimum choice of the required frequency, parallel wavelength, and port area for INTOR.

The primary additional need for lower-hybrid heating before INTOR authorization is the demonstration of long-pulse high-power operation, perhaps on TFTR or Doublet III, to determine if impurity influx is significant, and to provide a test of the engineering reliability of the complete system. An additional need exists for engineering optimization of coupling structures, tests of phase tracking, and studies of nonlinear effects; these tests can be done more efficiently and economically on a smaller dedicated rf facility.

Test of the other potential applications of lower-hybrid waves, such as current drive and current profile control, would be desirable before authorization for INTOR.

9.4 LOWER-HYBRID DEVELOPMENT NEEDS PRIOR TO INTOR OPERATION

The selection of lower hybrid heating as the primary auxiliary heating system for INTOR would make modest development work on the klystron sources beneficial. The development of long-pulse klystrons of 1 MW or larger and improved efficiency are the key objectives. A continued need for system optimization, reliability studies, and material selection will exist during this period.

9.5 ELECTRON-CYCLOTRON DEVELOPMENT NEEDS PRIOR TO INTOR AUTHORIZATION

Experiments to optimize ECH for plasma breakdown and ionization, and tests of bulk electron heating, including profile control, should be carried out on tokamaks of significant size, such as ISX-B and PDX.

Research is required into achieving high-efficiency ($> 20\%$) high-power gyrotron operation at frequencies in the 120 - 200 GHz range. Such research will require new techniques for optimizing gyrotron resonator design and on techniques for eliminating mode competition in high mode resonators.

The requirement of a 10^6 pulse lifetime in 5 s pulse operation has not been tested even in low frequency gyrotrons. Techniques for resonator and collector cooling and for gun operation under long-pulse operation must be investigated.

Improved techniques for efficiently transforming the working mode of the gyrotron into a linearly polarized mode for plasma heating should be developed. Systems, including quasi-optical systems, for efficiently transmitting gyrotron radiation from the tube to the reactor should be studied. Window materials for the gyrotron and ECH port must be developed.

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REFERENCES

- [1] ALLEN, R. E., et al., Phys. Rev. Lett. 41 (1978) 163.
- [2] GRANDALL, D. H., in Use of Small Accelerators, (Proc. 4th Conf. Denton, TX, 1976).
- [3] INTOR Energy and Particle Confinement Team Report (May 1979).
- [4] JASSBY, D. L., GOLDSION, R. J., Nucl. Fusion 6 (1976) 613.
- [5] TFR Group, in Plasma Physics and Controlled Nucl. Fusion Research (Proc. 6th Int. Conf. Berchtesgaden, 1976) Vol. 1, IAEA, Vienna (1977) 85.
- [6] BERK, H. L., HORTON, W., JR., ROSENBLUTH, M. N., RUTHERFORD, P. H., Nucl. Fusion 15 (1975) 919.
- [7] SUBANK, H. P., et al., in Plasma Physics and Controlled Nucl. Fusion Research (Proc. 7th Int. Conf. Innsbruck, 1978) I (IAEA, Vienna, 1979) 167.
- [8] JASSBY, D. L., Nucl. Fusion 17 (1977) 309.
- [9] SUBANK, H. P., et al., Phys. Rev. Lett. 43 (1979) 270.
- [10] SWAIN, D. W., et al., in Controlled Fusion and Plasma Physics (Proc. 9th Europ. Conf., Oxford, 1979) to be published.
- [11] MENON, M. M., et al., Oak Ridge National Lab. Report ORNL/TM-6658 (1979).
- [12] HIBBS, S. M., in Proc. 6th Symp. on Eng. Prob. of Fusion Res. (IEEE, New York 1976) 868.
- [13] MENON, M. M., et al., Bull Am. Phys. Soc. 23 (1978) 746.
- [14] BERKNER, K. H., et al., in Proc. 7th Symp. on Eng. Prob. of Fusion Res. (IEEE, New York, 1977) 1405.
- [15] BERKNER, K. H., et al., Bull Am. Phys. Soc. 23 (1978) 845.
- [16] PITTINGER, L. C., et al., in Proc. 7th Symp. on Eng. Prob. of Fusion Res. (IEEE, New York, 1977) 535.
- [17] GRISHAM, L. R., et al., Princeton Plasma Phys. Lab. Report PPPL-1484 (1978).
- [18] TSAI, C. C., et al., Oak Ridge National Lab. Report ORNL/TM-6360 (1978); BROWN, I. G., et al., Bull Am. Phys. Soc. II, 21 (1976) 1136.
- [19] STEARNS, J. W., et al., Lawrence Berkeley Lab. Report LBL-4492 (1976).
- [20] FINK, J. H., et al., Lawrence Livermore Lab. Report UCRL-52173 (1976).
- [21] STEWART, L. D., Conference Record - 1979 IEEE Int. Conf. on Plasma Science (IEEE, New York), paper 1F3, p. 19.

- [22] KIM, J., et al., Conference Record - 1979 IEEE Int. Conf. on Plasma Science (IEEE, New York), paper 5C9, p. 136; BARR, W. L., in Lawrence Livermore Lab. Report UCRL-50051-79-1 (1979); FUMELLI, M., et al., in Proc. Workshop on Neutral Injection, Culham, 1978, to be published.
- [23] Proc. of Symp. on the Production and Neutralization of Negative Hydrogen Ions and Beams, (Brookhaven, 1977), Brookhaven National Lab. Report BNL 50727 (1978).
- [24] MENON, M. P., Conference Record - 1979 IEEE Int. Conf. on Plasma Science (IEEE, New York) paper 5C7, p. 135.
- [25] HASELTON, H. H., et al., Oak Ridge National Lab Report ORNL/TM-5405 (1977) 130.
- [26] COLLERAINE, A. P., et al., in Proc. 7th Symp. on Eng. Prob. of Fusion Res. (IEEE, New York, 1977) 549.
- [27] HASELTON, H. H., in Proc. of the Plasma Heating Requirements Workshop, CONF-771241 (1977) 163.
- [28] STEWART, L. D., Princeton Plasma Phys. Lab. Report PPPL-1469 (1978).
- [29] JUNG, J., and ABDOU, M. A., Nucl. Technol. 41 (1978) 71.
- [30] FAULKNER, J. E., et al., Westinghouse Electric Corp. Report WFPS-TME-022 (1975); LEE, J. D., et al., Lawrence Livermore Lab. Report UCID-18017 (1975).
- [31] SANTORO, R. T., et al., Nuc. Sci and Eng. 70 (1979) 223.
- [32] JOHNSON, R. L., et al., in Proc. 7th Symp. on Eng. Prob. of Fusion Res. (IEEE, New York, 1977) 539.
- [33] BROMBERG, L., and JASSBY, D. L., in Proc. 7th Symp. on Eng. Prob. of Fusion Res. (IEEE, New York, 1977) 971.
- [34] SCHWENKLY, S. W., in Proc. 7th Symp. on Eng. Prob. of Fusion Res. (IEEE, New York, 1977) 1793.
- [35] BATZER, I. H., Lawrence Livermore Lab. Report UCRL-50051-77-4 (1978) 30.
- [36] STEWART, L. D., et al., in Proc. of 3rd Int. Symp. on Toroidal Plasma Confinement, Garching (1972) E12.
- [37] BERKNER, K. H., et al., Bull. Am. Phys. Soc. 23 (1978) 913.
- [38] DRAKE, R. P. and MOOS, H. W., Nuc. Fus. 19 (1979) 407; DRAKE, R. P., et al., Lawrence Livermore Lab. Report UCID-17864 (1978).
- [39] COLLERAINE, A. P., et al., in Proc. 10th Symp. on Fusion Tech. (Padua, Italy, 1978).

- [40] HOPKINS, D. B. and BAKER, W. R., in Proc. of the Plasma Heating Requirements Workshop, CONF-771241 (1977) 377, also Lawrence Livermore Lab. Report UCID-3989 (1977).
- [41] ADAM, J., et al., Proc. 5th Int. Conf. on Plasma Physics and Controlled Nucl. Fusion Research (IAEA Vienna, 1975) 66; HOSEA, J., 2nd Symp. on Plasma Heating in Toroidal Devices (Varenna, 1974) 61; Princeton Plasma Phys. Lab. Rept. MATT-1129 (1975).
- [42] TAKAHASHI, H., 3rd Int. Congress on Waves and Instabilities in Plasma (Ecole Polytechniques, Palaiseau, June 1977); Princeton Plasma Phys. Lab. Report PPPL-1374 (1977); TAKAHASHI, H., et al., Phys. Rev. Lett. 39 (1977) 31.
- [43] PLT Group, Bull. Am. Phys. Soc. 23 (1978) 7.
- [44] TFR Group, Assoc. EURATOM-C.E.A. Report EUR-CEA-FC-988 (Fontenay-aux-Roses, Jan. 1979).
- [45] BUZANKIN, V. V., et al., 6th Int. Conf. on Plasma Physics and Controlled Nucl. Fusion Research (IAEA Vienna, 1977) 61; IVANOV, N. V., Kovan, I. A., and SOKOLOV, Yu. A., JETP Lett. 24 (1976) 374.
- [46] VDOVIN, V. L., Shapotkovskii, N. V., and RUSANOV, V. D., Proc. 3rd Int. Meeting on Theoretical and Experimental Aspects of Toroidal Plasmas (C.E.A. Grenoble, 1976) Vol. 2 p. 349; VDOVIN, V. L., RUSANOV, V. D., and SHAPOTKOVSKII, N. V., JETP Lett. 24 (1976) 374.
- [47] ADAM, J. and SMAIN, A., Assoc. EURATOM-C.E.A. Report EURA-CEA-FC-579 (Fontenay-aux-Roses, 1971); ADAM, J., Report EUR-CEA-FC-711 (Fontenay-aux-Roses, 1973).
- [48] STIX, T. H., Nucl. Fusion 15 (1975) 737.
- [49] HOSEA, J., Proc. 3rd Symp. and Workshop on Plasma Heating in Toroidal Devices (Varenna, 1976); Princeton Plasma Phys. Lab. Rept. PPPL-1309 (1976).
- [50] ADAM, J. and JACQUINOT, J., Assoc. Euratom-C.E.A. Rept. EUR-CEA-FC-886 (1977).
- [51] PERKINS, F. W., Nucl. Fusion 17 (1977) 1197.
- [52] JACQUINOT, J. MCVEY, B. D., and SCHARER, J. E., Phys. Rev. Lett. 39 (1977) 88; JACQUINOT, J., Proc. Joint Varenna-Grenoble Int. Symp. on Heating in Toroidal Plasmas, Vol. 1 (CEA Grenoble, 1978) 127.
- [53] WEYNANTS, R. R., Phys. Rev. Lett. 33 (1974) 78.
- [54] SWANSON, D. G. and NGAN, Y. C., Phys. Rev. Lett. 35 (1975) 517; NGAN, Y. C. and SWANSON, D. G., Phys. Fluids 20 (1977) 1920; SWANSON, D. G., Phys. Rev. Lett. 36 (1976) 316.
- [55] SCHARER, J. E., MCVEY, B. D., and MAU, T. K., Nucl. Fusion 17 (1977) 297.

- [56] BLANC, P., et al., 7th Int. Conf. on Plasma Physics and Controlled Nuclear Fusion Research (Innsbruck, 1975) IAEA CN37/A-4-3.
- [57] MOELLER, C., et al., Proc. 3rd Topical Conf. on Radio-Frequency Plasma Heating (Pasadena, 1978) A4 (also GA-A14610).
- [58] BERNABEI, S., et al., Proc. Symp. on Plasma Heating in Toroidal Devices (Varenn, 1976).
- [59] PORKOLAB, M., et al., Proc. 3rd Topical Conf. on RF Plasma Heating (Pasadena, 1978) E10.
- [60] FAIRFAX, S. A., et al., Bull. Am. Phys. Soc. 23 (1978) 764.
- [61] KONOSHIMA, S., et al., 7th Int. Conf. on Plasma Physics and Controlled Fusion Research (Innsbruck, 1975) IAEA CN37/A-4-2.
- [62] BRIAND, P., et al., 7th Int. Conf. on Plasma Physics and Controlled Nucl. Fusion Research (Innsbruck, 1975) IAEA CN37/A-4-1.
- [63] BRAMBILLA, M., Nucl. Fusion 16 (1976) 47.
- [64] BERNABEI, S., et al., Phys. Rev. Lett. 34 (1975) 866.
- [65] SCHUSS, J. J., et al., Bull. Am. Phys. Soc. 23 (1978) 765.
- [66] BERS, A., Bull. Am. Phys. Soc. 23 (1978) 765.
- [67] MOELLER, C. and CHAN, V. S., Bull. Am. Phys. Soc. 23 (1978) 821.
- [68] SLUSHER, R. E., (private communication).
- [69] PORKOLAB, M., et al., Phys. Rev. Lett. 38 (1977) 230.
- [70] STAMBAUGH, R. D., et al., Bull. Am. Phys. Soc. 23 (1978) 788.
- [71] CHAN, V. S., CHIU, S. C., and OHKAWA, T., General Atomic Co. Rept. GA-A15264 (1979).
- [72] HARVEY, R. W. and RAWLS, J. M., General Atomic Co. Rept. GA-A15234 (1979).
- [73] FISCH, N. J., Phys. Rev. Lett. 41 (1978) 873.
- [74] HOOKE, W. M., Proc. 3rd Topical Conf. on Radio-Frequency Plasma Heating (Pasadena, 1978) A2.
- [75] ALIKAEV, V. V., et al., in Plasma Physics and Controlled Nucl. Fusion Research (Proc. 5th Int. Conf., Tokyo, 1974) I (IAEA Vienna, 1975) 241.
- [76] SCHARER, J. E., CONN, R. W., and BLACKFIELD, D. T., EPRI Rept. ER-268, Proj. 237-3 Topical Rept., (Sept. 1976); SCHARER, J. E., BLACKFIELD, D. T., BEYER, J. B., and MAU, T. K., UWFD-227 (Univ. of Wisconsin, June, 1978) to be published.

- [77] REED, B. W., et al., Princeton Plasma Phys. Lab. Rept. PPPL-1410 (1977).
- [78] TFR Group, EUR-CEA-FC-988 (Jan. 1979).
- [79] BUZANKIN, V. V., et al., Proc. 5th Int. Conf. on Plasma Physics and Controlled Nucl. Fusion Research (IAEA Vienna, 1977).
- [80] BROOKS, N., et al., Proc. 3rd Topical Conf. on Radio-Frequency Plasma Heating (Pasadena, 1978) E4.
- [81] REED, B. W., et al., Princeton Plasma Phys. Lab. Rept. PPPL-1410 (1977).
- [82] BERNABEI, S., et al., Proc. Symp. on Plasma Heating in Toroidal Devices (Varenna, 1976).
- [83] ARTSISOVICH, L. A., Nucl. Fusion 12 (1972) 215.
- [84] FURTH, H. P., YOSHIKAWA, S., Phys. Fluids 13 (1970) 2593.
- [85] BCL, K., et al., Phys. Rev. Lett. 32 (1974) 661.
- [86] COHN, D. R., JASSEY, D. L., KREISCHER, K., Nucl. Fusion 18 (1978) 1255.
- [87] Plasma Physics and Controlled Nucl. Fusion Research (Proc. 5th Int. Conf., Tokyo, 1974) III (IAEA Vienna, 1975) 271-307.
- [88] MOHTI, A., et al., in Plasma Physics and Controlled Nucl. Fusion Research (Proc. 7th Int. Conf., Innsbruck, 1978) paper CN-37-X-4-2 (to be published).
- [89] Physics Internat. Co. (private communication, 1978).