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A REVIEW OF THE ISX-B EXPERIMENTAL PROGRAM*

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ABSTRACT

The ISX-B tokamak is a modestly sized device that is designed to explore questions closely related to the design of future tokamak devices, such as the Engineering Test Facility (ETF). The major program emphasis is on high-β plasma operation using neutral beams to heat plasmas with noncircular cross sections. In addition, substantial efforts are under way or planned in the areas of plasma-wall interactions, coated limiter studies, electron-cyclotron heating, ripple injection, impurity flow reversal, pellet injection, and the application of a bundle divertor. The current status and future plans for these programs are reviewed.

I. INTRODUCTION

The past decade or more has seen considerable advances in the study of magnetically confined fusion experiments, particularly with tokamak devices. The effort in the U.S. is now beginning to focus on the design of an Engineering Test Facility (ETF), which would be expected to operate with a burning D-T plasma for tens of seconds. Before such a design can be completed, however, a number of crucial issues must be resolved. The ISX-B tokamak[1] at Oak Ridge National Laboratory (ORNL) has been built specifically to investigate some of the more important of those issues. In this paper we give a brief description of the ISX-B facility itself and outline some of the more important elements of the current experimental program. These program elements include studies related to high-β plasma operation, plasma impurities and their control, the application of rf heating to a plasma, refueling of a plasma using pellets of frozen hydrogen, the effect of toroidal field ripple on the plasma properties, and the application of a bundle divertor for impurity control.

II. THE ISX-B FACILITY

The ISX-B (Impurity Studies Experiment), which became fully operational in the fall of 1978, is a modestly sized, iron-core tokamak with the major parameters given in Table I. The machine is an upgraded version of the successful ISX-A tokamak[2] modified primarily for the investigation of high-β plasma operation. The range of values of some of the major plasma parameters that have been obtained to date is given in Table II.

The major changes involved in the upgrade were the addition of two ORNL neutral beam injectors[3] similar to those produced for the Princeton Large Torus (PLT), a new poloidal coil system to permit control over the plasma cross section, and a larger rectangular cross-section vacuum vessel to accommodate elongated plasmas. The design improved upon the excellent diagnostic access that was a feature of ISX-A and has some 164 access ports built into the vacuum vessel, including a large flange for the future addition of a bundle divertor. Figure 1 shows a top view of the vacuum vessel, which is made of 304L stainless steel. The vessel was fabricated in two halves that were joined by two insulated breaks with a fluorocarbon O-ring vacuum seal in each. The design of the breaks is shown in Fig. 2, and follows closely the system used in the MACROTOR device.[4] These insulating breaks are required to stop large toroidal currents from flowing in the vacuum vessel. The decision was taken to adopt this system, rather than high resistance bellows sections similar to those used on ISX-A, because of fabrication difficulties with rectangular cross-section bellows. Initial fears that the O-ring system might result in significantly higher low-Z impurity levels in ISX-B plasmas have proved to be unfounded. Most other vacuum flanges on the machine are all metal with copper gaskets, but some of the larger rectangular flanges and those designed for large windows also use fluorocarbon O-rings. Following the ISX-A pattern, no heavy metals are permitted inside the vacuum system, the limiters being made from type 304L stainless steel. As a result, the plasmas do not suffer large radiation losses from heavy metal ions.

The pumping is identical to that used on ISX-A. An aspirator and cryosorption pumps are used for roughing the system and two 450-liter/sec turbopumps provide the normal pumping. Base pressure is typically $\approx 1 \times 10^{-7}$ Torr.

A large variety of diagnostics and other ancillary equipment is now attached to the tokamak, as shown in Fig. 3, a photograph taken in July 1979. Some of the major items indicated are one of the neutral beam injectors, the Thomson scattering laser system for measuring electron temperature, the surface analysis station, the pellet injector, visible and vacuum ultraviolet (VUV) spectrometers, the fast x-ray analyzer, the 2-mm microwave interferometer system, and the waveguide for injection of 35-GHz rf power. Within the next few months a number of additions will be made, including higher power beam lines to increase the injection power to 3 MW, a multipoint Thomson scattering system, a diagnostic neutral beam, more charge exchange analyzers, and a host of smaller diagnostics. This multitude of equipment poses a major problem in preserving the vacuum integrity of the machine and has required the use of a complex vacuum interlock system to lessen the risk of contamination of the vacuum vessel.

The poloidal coil system is shown in Fig. 4, with some representative plasma shapes which can be obtained according to theoretical modeling computations.[5] The coil system can be configured in a number of ways into four separate systems, each with its own power supply. The combination of the "inner," "outer," and "shaping" coils generates the plasma ohmic heating (OH) current, produces the vertical field that controls the plasma position, and controls the shape of the plasma cross section. The inner and outer power supplies are controlled by a feedback system[6] that holds the plasma current and position constant, while the current in the shaping coils is preprogrammed to provide the required plasma shape. A second feedback system controls the current in the radial field coils to keep the vertical plasma position constant.

III. ISX-B EXPERIMENTAL PROGRAM

A. High-β plasma operation

As mentioned in the introduction, the experimental program on ISX-B is directed toward some of the crucial questions that must be resolved before a successful ETF design can be completed. Foremost amongst these is the question of high-β plasma operation. Beta is defined as the ratio of plasma pressure to the pressure exerted by the confining magnetic field and may therefore be regarded as a measure of the efficiency with which this field is utilized. A major fraction of the capital cost of a tokamak-based reactor will be associated with the production of the toroidal confining field,[7] so operation at high values of β ($\gtrsim 5\%$) is recognized as a prerequisite for an economically viable tokamak reactor. The significance of β is also illustrated by the expression[8] for the fusion power density,

$$P_f = \text{const} \times \frac{\beta^2 B^4 \langle\sigma v\rangle E_f}{(T_i + T_e)^2} ,$$

where B is the confining field strength, $\langle\sigma v\rangle$ is the averaged product of fusion reaction cross section (σ) and particle velocity (v), E_f is the energy release per fusion event, T_i is the ion temperature, and T_e is the electron temperature.

Thus, at a given operating temperature and magnetic field strength the power density scales as β^2. However, magnetohydrodynamic (MHD) stability calculations indicate that a circular plasma will be unstable above a value of $\beta^* = \sqrt{\langle\beta^2\rangle} \approx 2\%$ due to the so-called ballooning instabilities.[9] The limiting value of β^* can be increased by tailoring the plasma cross section and profile. For shapes attainable in ISX-B the limit is predicted to be $\approx 6\%$, while for other configurations it may be as high as 12%. The experiments on ISX-B therefore have the following objectives.

(1) To use the massive neutral beam heating power ($P_b > P_{OH}$) to attain the high temperatures and densities needed to reach β^* values of $\gtrsim 2\%$ in circular cross-section plasmas.

(2) To identify any β^* limits on plasma stability.

(3) To explore the dependence of the β^* limit on the plasma cross section.

During the past year stable, approximately circular plasmas have been studied with injected powers up to 1 MW. The value of β^* has so far shown a steady, roughly linear increase with beam power up to a level of ≈3%, exceeding the theoretical limit of β stability as shown in Fig. 5. No evidence of any gross instability has been seen, a very encouraging result. However, collimated x-ray and poloidal loop diagnostics have indicated the presence of plasma oscillations as the level of beam heating, and thus β^*, has increased. Some of the diagnostic systems needed for making definitive mode assignments are not yet operational, so we are unable to say whether the observed modes are evidence of the ballooning instabilities predicted by the theory. This program will continue this year with higher power injection from the existing beam lines. The beam lines will be upgraded to give a 3-MW capability at the beginning of 1980, permitting the exploration of even higher values of β^*.

B. Ripple studies

The effects of ripple in the toroidal magnetic field are being explored in collaboration with Princeton Plasma Physics Laboratory (PPPL). Special coils, installed on the machine during its construction, are capable of inducing up to 1.5% ripple on the axis. The program has two major objectives: first, to understand the effects of ripple on a hot plasma so that the tolerable level of ripple can be assessed, and second, if the tolerable level is high enough, to utilize a ripple field as a means of lowering the injection energy required for neutral beam heating of a plasma. The allowable level of ripple has important consequences in the design of tokamak reactors. If it is large, the number of toroidal field (TF) coils can be reduced, thus reducing the cost and improving access for maintenance or auxiliary equipment. Furthermore, it is now generally accepted that a bundle divertor offers great promise for control of plasma impurities and helium ash buildup in a long pulse tokamak. As such a device will necessarily impose some toroidal ripple, it is vital to know what level can be tolerated. Experimental measurements are required in order to model with confidence the effects of the divertor field on gross plasma properties. To date, the results of the ISX-B program have indicated no catastrophic effects of the maximum achievable ripple (1.5%) on the background plasma characteristics; e.g., a reduction of ≈10% in ion or electron temperature has been seen. However, a large loss of scattered fast ions from the injected beams has been observed. The effective loss of injected beam power is ≈10-20%, comparable with other loss mechanisms such as charge exchange. Much more detailed work is needed to understand these effects and to extrapolate them to ETF conditions.

The second part of the ripple program, the study of ripple injection[10] as a technique for using lower injection energies, will commence toward the end of 1979, when a vertically injected neutral beam system (10-40 keV) will be added to ISX-B. Reactor-size tokamaks require an injection energy of ≈200 keV to allow the neutral beams to penetrate and heat the plasma center. Existing neutral beam systems rely on the production of positive ion beams that are neutralized in a gas cell, and the efficiency of this process drops drastically above 100 keV.[11] Thus, fusion reactors must either be heated with very poor efficiency, or some other technology, such as negative ion beams or direct recovery systems, must be developed. If, however, the plasma can tolerate a reasonable level of toroidal ripple, then the ripple injection technique may allow the use of much lower injection energies, $\lesssim$100 keV. The idea is illustrated in Fig. 6. A ripple field with significant top-bottom asymmetry is induced in the plasma volume. A vertically injected neutral beam is directed from the side of stronger ripple, and fast ions produced by the beam are trapped in the resultant magnetic well. They drift upwards until they enter a region where the ripple field is small enough that they can escape from the well and enter normal fast ion trajectories nearer the plasma center. In this manner the injected power is deposited nearer the plasma core than would be the case for injection without ripple (also shown in Fig. 6).

C. Pellet fueling studies

The normal mode of refueling in existing tokamaks is gas puffing, that is, the injection of cold gas into the torus. However, this method is unlikely to be successful in reactor-scale devices that employ a divertor, because the gas will be ionized in the divertor scrape-off region and pumped by the divertor itself rather than by the plasma. The most promising method of refueling the plasma core appears to be the high speed injection of frozen D-T pellets, which can traverse the confining magnetic field and deposit the fresh fuel near the plasma center. Experimental work which began on ISX-A[12] is continuing on ISX-B with the following objectives:

(1) to investigate pellet ablation physics in ohmically and neutral beam heated plasmas to derive pellet ablation scaling laws,

(2) to investigate transient transport and stability properties of plasmas subjected to massive localized density and temperature perturbations, and

(3) to establish the pellet size, velocity, and frequency requirements for refueling present and future tokamak devices.

The pellet injector presently in use on ISX-B is an improved version of the pneumatic device[13] first used on ISX-A. Pellets ≈1 mm in diameter have been injected at velocities up to 1.1 km/sec. The studies in ohmic discharges have shown good agreement with the neutral gas shielding model[14] for pellet ablation, although modifications to the model have been required to explain the enhanced ablation at the plasma edge due to the fast ion heat flux from neutral beam injection. Figure 7 shows the results of two pellet shots into ohmically heated discharges. In the first case, the pellet is stopped near the magnetic axis, resulting in a 500% increase in average density and a decrease in MHD amplitude. In the second case, the pellet is stopped before reaching the center, resulting in large amplitude MHD oscillations but no disruptions. Transport calculations are able to model the evolution of the central electron temperature following pellet injection. In low density discharges, the efficiency of neutral beam heating has been shown to improve after pellet injection, as expected from the observed density increase. Measurements of energy loss from the pellet-plasma interaction zone have shown losses of less than 4 eV from the plasma per injected particle (other than ionization energy), and half of this is accounted for by Lyman-α line radiation. The program involves a collaborative effort with the Massachusetts Institute of Technology (MIT) in the application of ruby laser holographic interferometry, pulsed laser shadowgraphy, and high speed photography to study the pellet ablation. Figure 8 is a photograph of a pellet penetrating a plasma and striking the inside wall of the torus.

Future work in this program will involve the application of a new mechanical injection apparatus capable of repetitive pellet injection into the ISX plasma. Pellets 0.8 mm in diameter will be injected into the plasma at a velocity of ≈300 m/sec and at rates up to 150 pellets/sec in a study of continuous plasma refueling.

D. Impurity studies

Impurities enter tokamak discharges via plasma-wall and plasma-limiter interactions. The enhanced radiative losses due to the presence of these impurities can dominate the power balance of the discharge, with high-Z impurities being particularly troublesome even at very low concentrations ($\lesssim 10^{-5}$). A comparison of spectroscopic measurements of the radiated power from the Oak Ridge Tokamak (ORMAK), which employed tungsten limiters, and ISX-B is given in Table III. The beneficial effect of excluding high-Z materials from the ISX-B torus is evident. The major impurities in ISX-B discharges are the low-Z elements (C, O, N), which are present as adsorbed species on the surfaces of the vacuum vessel, and the moderate-Z elements (Fe, Ni, Cr), which form the bulk material of the walls and limiters. While for existing tokamaks the low-Z species make up the dominant impurities, the moderate-Z impurities are potentially more damaging for reactor-scale devices with their high plasma temperature and wall power loadings. The ISX-B impurity program includes investigations into the release of impurities from the walls and their transport to the plasma, as well as methods for controlling impurity release and transport. In addition, the recycling of hydrogen (or deuterium) between the plasma and the wall is being studied.

One possible source of the release of impurity atoms from the wall and limiter surfaces is the phenomenon of unipolar arcing. Experiments performed on ISX-A have shown that arcs occur most frequently during plasma initiation, quenching, and disruptions.[15] Optical and scanning electron microscope (SEM) analysis of the arc tracks indicates that 10^{16}-10^{17} atoms may be ejected for each arc. Although this is easily sufficient material to explain the observed levels of metal impurities in the plasma, little is known as to the proportion of ejected particles that is transported into the plasma. Associated laboratory studies have shown that surface conditioning occurs, leading to a reduction in arcing frequency with time, and that surface conditions such as the presence of oil contamination can greatly increase the probability of arcing.[16] *In situ* measurements of total surface erosion, due to arcing or other mechanisms such as sputtering or vaporization, are planned. A sample composed of a thin coating of stainless steel on a silicon substrate will be exposed to the plasma, and ion backscattering techniques will be used to measure the film thickness before and after exposure.

The transport of particles in the plasma-wall interaction region is being investigated with a variety of techniques to measure particle mass, charge, flux, and energy. Silicon surfaces exposed in the plasma-wall interaction region have been employed as collectors, using ion backscattering, nuclear reaction analysis, and radiation damage analysis to measure the depth profiles of implanted hydrogen and deuterium atoms and thereby to derive fluxes and energy estimates.[17] Similar techniques are being used to measure the fluxes of impurity ions and neutrals in the interaction region. A charge state analyzer has been installed in the shadow of the limiter to provide *in situ*, time-resolved measurements of impurity fluxes, and dye laser fluorescence and dye laser Doppler shift measurements are planned to give particle density and velocity measurements.

The control of impurity levels in the ISX-B plasma during normal operation is achieved by regular discharge cleaning of the vacuum system to remove adsorbed species from the chamber walls. The cleaning procedure currently employed is a combination of low power discharge cleaning in deuterium (the normal working gas), usually done overnight, together with an ≈30-min glow discharge in helium before tokamak operation. The first part of the procedure was used successfully on ISX-A, while the second was adopted after encouraging results had been reported for this technique by Brevnov *et al.*[18] Table IV shows the typical status of the vacuum wall surface before a day's operation as measured by Auger analysis of a stainless steel sample exposed to the plasma at the wall position, together with a residual gas analyzer (RGA) scan of the gas in the torus taken at a similar time. From these results and the spectroscopy data in Table III, it can be seen that the surface contamination and the background of gaseous impurities are kept sufficiently low by the discharge cleaning procedure currently employed. However, it is unlikely that this procedure is optimized for cleanup after a vacuum opening. Discharge cleaning studies on the tokamak itself are very desirable in this regard but are severely constrained by the need to keep the machine operating in a clean condition for the other programs.

Three major experiments to control impurities are planned for ISX-B: impurity flow reversal studies, coated limiter studies, and a bundle divertor program. The impurity flow reversal studies started on ISX-A[19] are being continued in collaboration with a group from General Atomic Company (GA). This technique uses a poloidally asymmetric source of hydrogen gas injection to reverse the normal inward flow of impurity ions in the plasma.[20,21] The ISX-A results have already shown qualitative agreement with the theory. A more quantitative study of the effect is under way on ISX-B, using the improved array of edge diagnostics discussed above to provide better information about the outer plasma regions (where the effect occurs). These studies will be followed by a test of the theoretically predicted impurity flow reversal induced by momentum from neutral beam injection,[22] also in collaboration with GA.

The second impurity control technique to be tested is a study of coated limiter materials undertaken in collaboration with Sandia Laboratories. It is hoped that a suitable low-Z coating/substrate combination can be found that will withstand the thermal shock during a plasma shot, have good erosion characteristics, and reduce the influx of moderate-Z impurities into the plasma. The first phase of this experiment will be the exposure to the plasma of small samples of graphite with coatings of TiB_2 and B_4C to ensure that no catastrophic effects in the plasma will occur. In the fall of 1979, a vacuum interlock system capable of inserting cleaned, instrumented, water-cooled limiters will be installed on the machine, and the first tests of coated limiters will be made in ISX-B discharges. If successful coating/substrate combinations can be developed, they will have immediate applications as limiters in existing tokamaks and potential applications for reactor first walls.

Finally, the third and most ambitious attempt at impurity control is the installation of a bundle divertor on ISX-B, currently scheduled for the end of 1980. The bundle divertor concept was first applied to a tokamak on the Divertor and Injection Tokamak Experiment (DITE)[23] with encouraging results. The ISX program will be the first test of such a device within the U.S. fusion program and will complement the poloidal divertor program planned on the Poloidal Divertor Experiment (PDX) at PPPL.

The principle of the bundle divertor is illustrated in Fig. 9, which shows how the divertor coils act to detach a flux bundle from the outer regions of plasma and bring it outside the TF coils into a separate chamber, where the plasma ions are neutralized and then pumped away. The divertor acts to reduce plasma impurities in three ways: first, by removal of the need for a material limiter within the torus to constrain the size of the plasma; second, by providing an outer layer of relatively cold plasma that can ionize incoming neutral impurities and sweep them out of the discharge region; and third, by transporting ions escaping from the plasma out to the divertor chamber before they can sputter material from the first wall. In a reactor application, the bundle divertor would also act to exhaust the helium produced by the fusion burn. If the bundle divertor concept should prove to be workable, it would have a number of advantages over a poloidal divertor system. In particular, the bundle divertor components are external to the TF coil system and are therefore accessible for maintenance and repair, whereas the poloidal divertor components are inside the main torus and represent a major problem for remote maintenance and handling. Furthermore, the possibility of using more than one bundle divertor around a tokamak reactor would permit repairs or regeneration without shutting down the whole facility.

The ISX-B divertor program will offer the opportunity to investigate the effect of divertor operation on the impurity level in the plasma, the effectiveness of the scrape-off layer in shielding the plasma core from incoming neutrals and in transporting the impurities out of the plasma core, the particle confinement times in the absence of large recycling of hydrogen from the chamber walls, the effects of ripple on plasma operation and neutral beam injection, and questions related to the technology of bundle divertors themselves, such as the development of suitable target materials for intercepting the diverted plasma.

E. Electron-cyclotron heating

The use of rf power to provide electron cyclotron heating (ECH) of a plasma is of interest for three reasons. First, rf power provides an alternative to neutral beam injection for auxiliary heating of plasmas. Second, because the region of power deposition for a given frequency occurs in a region of constant magnetic field strength, the power deposition profile can be tailored to provide some control over the electron temperature profile and thus optimize the plasma equilibrium and stability. Third, ECH power could assist in plasma startup by preionizing the plasma and thus lowering the volt-seconds that must be expended in establishing the plasma current. This would mean less iron required in the transformer core of a tokamak for a given burn time, saving both money and the valuable space on the inside of the torus.

Two parallel efforts are being mounted on ISX-B. A 35-GHz gyrotron capable of 100 kW for 10 msec developed at the Naval Research Laboratory has been installed on the machine and is now being tested with plasmas. In addition, an ORNL group has almost completed an installation using a 28-GHz, 200-kW gyrotron that will be capable of pulse lengths up to 100 msec.

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REFERENCES

[1]D. W. Swain, T. E. Smith, D. J. Sigmar, and M. Murakami, ORNL/TM-5985, Oak Ridge National Laboratory, Oak Ridge, Tennessee (1977).

[2]R. J. Colchin and T. C. Jernigan, J. Nucl. Mater. 63, 83 (1976).

[3]J. Kim *et al.*, ORNL/TM-6896, Oak Ridge National Laboratory, Oak Ridge, Tennessee (1979).

[4]R. Taylor, private communication to D. W. Swain (Oak Ridge National Laboratory), 1977.

[5]R. A. Dory *et al.*, *Plasma Physics and Controlled Nuclear Fusion Research 1978* (International Atomic Energy Agency, Vienna, Austria, 1979), Vol. 1, p. 579.

[6]L. A. Charlton, D. W. Swain, and G. H. Neilson, to be published in IEEE Trans. Plasma Sci.

[7]D. Steiner *et al.*, ORNL/TM-6720, Oak Ridge National Laboratory, Oak Ridge, Tennessee (1979).

[8]D. Steiner, Nucl. Sci. Eng. 58, 107 (1975).

[9]A. M. M. Todd *et al.*, Nucl. Fusion 19, 743 (1979).

[10]R. J. Goldston *et al.*, PPPL-1398, Princeton Plasma Physics Laboratory, Princeton, New Jersey (1978).

[11]K. H. Berkner, R. V. Pyle, and J. W. Stearns, Nucl. Fusion 15, 249 (1975).

[12]S. L. Milora, C. A. Foster, P. H. Edmonds, and G. L. Schmidt, Phys. Rev. Lett. 42, 97 (1979).

[13]S. L. Milora and C. A. Foster, Rev. Sci. Instrum. 50, 482 (1979).

[14]S. L. Milora and C. A. Foster, IEEE Trans. Plasma Sci. PS-6, 578 (1978).

[15]P. Mioduszeski, R. E. Clausing, and L. Heatherly, to be published in J. Nucl. Mater.

[16]R. E. Clausing (Oak Ridge National Laboratory), private communication, June 1979.

[17]S. P. Withrow, R. A. Zuhr, J. B. Roberto, H. Vernickel, and B. R. Appleton, to be submitted to Nucl. Fusion.

[18]N. N. Brevnov, R. E. Clausing, A. A. Kazachov, and N. V. Kuznetsov, IAE Report No. 3100, I. V. Kurchatov Institute of Atomic Energy, Moscow, U.S.S.R. (1979).

[19]K. H. Burrell *et al.*, Phys. Rev. Lett. 41, 1382; M. Murakami *et al.*, *Plasma Physics and Controlled Nuclear Fusion Research 1978* (International Atomic Energy Agency, Vienna, Austria, 1979), Vol. 1, p. 269.

[20]T. Ohkawa, Kaku Yugo Kenkyu 32, 67 (1974).

[21]K. H. Burrell, Phys. Fluids 19, 401 (1976); Phys. Fluids 20, 342 (1977).

[22]W. M. Stacey and D. J. Sigmar, ORNL/TM-6575, Oak Ridge National Laboratory, Oak Ridge, Tennessee (1979); P. B. Parks, K. H. Burrell, and S. K. Wong, GA-A15247, General Atomic Company, San Diego, California (1978).

[23]P. E. Stott, C. M. Wilson, and A. Gibson, Nucl. Fusion 17, 481 (1977); Nucl. Fusion 18, 475 (1978).

FIGURE CAPTIONS

Fig. 1. Top view of the ISX-B vacuum vessel.

Fig. 2. Construction of the insulating breaks on the ISX-B vacuum vessel.

Fig. 3. The ISX-B tokamak and some of its associated diagnostic equipment (as of July 1979). The following items of auxiliary equipment are indicated: (1) one neutral beam injector, (2) rf waveguide and launching system for the 35-GHz gyrotron, (3) Thomson scattering system, (4) surface physics station, (5) 2-mm microwave interferometer, (6) fast x-ray analyzer, (7) visible and VUV spectrometers, and (8) pellet injector.

Fig. 4. Various possible configurations of the poloidal coil system on ISX-B and representative plasma shapes that can be obtained, according to theoretical modeling computations.[5] The centerline of the tokamak is to the left. The dotted lines indicate the coils that make up the inner, outer, and shaping coil sets for the four configurations.

Fig. 5. Values of β^* as a function of the injected power and plasma density attained in ISX-B for circular plasma cross sections. The β stability limit for circular plasmas is taken from Ref. 9.

Fig. 6. Flux contours in ISX-B with examples of fast ion orbits corresponding to a vertically injected 20-keV neutral beam with (solid line) and without (dotted line) ripple.

Fig. 7. Representative diagnostic signals from two pellet injection shots for which the density increased by a factor of ≈5. At left is a shot for which the pellet reached the plasma center and the MHD level ($\dot{B}_0$) decreased. At right is a shot for which the pellet stopped before reaching the center, resulting in a large increase in MHD amplitude.

Fig. 8. A pellet penetrating the plasma and striking the inner wall.

Fig. 9. The bundle divertor concept. The divertor coils act to detach a flux bundle from the outer plasma region and bring it outside the torus, where the plasma can be neutralized and pumped away.

Table I. Major parameters of ISX-B

Major radius, R	0.93 m
Minor radius, a	0.27 m
Plasma elongation, b/a	≤ 1.8
Toroidal field, B_T	≤ 1.8 T
Plasma current, I_p	≤ 200 kA
Neutral beam power, P_b	≤ 1.8 MW

Table II. Range of plasma characteristics in ISX-B

Elongation, b/a	1-1.4
Toroidal field, B_T	0.9-1.5 T
Plasma current, I_p	90-185 kA
Central electron temperature, $T_e(0)$	0.5-2.2 keV
Central ion temperature, $T_i(0)$	0.3-1.5 keV
Line average electron density, $\bar{n}_e$	$0.5-9 \times 10^{13}$ cm^{-3}
Injected power, P_b	0-1 MW
Ohmic heating power, P_{OH}	50-250 kW
Effective charge, Z_{eff}	1-3
Energy confinement time, τ_E	6-25 msec
Shot length, τ	150-250 msec

Table III. Fraction of input power radiated from various impurities in ORMAK and ISX-B

Impurity species	Radiated Power	
	ORMAK	ISX-B
O	24%	16%
C + N	0.4%	4%
Fe, Ni, Cr	0.3%	5%
W	60%	—
Total	85%	25%

Table IV. Wall surface auger analysis and residual gas analysis (RGA) for ISX-B

Auger Analysis		RGA Scans		
Element	Surface concentration (%)	Mass number	Assumed source	Partial pressure (10^{-9} Torr)
Fe	55	1-6	H, D	910
Cr	15	14	N, CH_4	2
Ni	12	16	H_2O, CH_4	1.1
N	2	17	H_2O, CH_4	2.5
C	1	18	H_2O, CH_4	8
C (mostly carbide)	15	19	H_2O, CH_4	5.5
		20	H_2O, CH_4	8
		28	N, CO	40
		44	CO_2	1.3