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Proposal for a Major Modification to the ISX Tokamak to Study Equilibrium and Stability of Externally Heated High β Plasmas in Noncircular Geometry

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FUSION ENERGY DIVISION

PROPOSAL FOR A MAJOR MODIFICATION TO THE ISX TOKAMAK
TO STUDY EQUILIBRIUM AND STABILITY OF EXTERNALLY
HEATED HIGH β PLASMAS IN NONCIRCULAR GEOMETRY

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ABSTRACT

The eventual use of tokamaks as economical fusion power generators will require the confinement of plasmas with values of $\beta \geq 5\%$. The ISX tokamak at Oak Ridge National Laboratory can be modified to study the equilibrium, stability, and transport of a plasma with β in the 3-5% range. This can be achieved by adding up to 3 MW of neutral beam injection power on the machine and by replacing the vacuum chamber and poloidal field coils. The design, cost, and schedule for the modification are covered in this proposal, in addition to the reasons for performing these experiments.

1. INTRODUCTION

Recent theoretical analyses of high β equilibrium configurations in tokamaks, including the flux conserving tokamak (FCT) equilibria, have indicated that plasma states exist with values of β up to 20%.¹ However, theoretical estimates of ideal magnetohydrodynamic (MHD) stability of these equilibria yield maximum β limits of 1.5-12%, depending on the plasma shape.² Because the economic validity of a tokamak power reactor depends strongly on attaining a β of 5-10%, an experimental study of β limitations in a "hot" tokamak (i.e., in the collisionless regime, $v^* < 1$) with variable plasma shape is of utmost importance. In addition, the scaling up of neutral beam injection experiments is also necessary, because the technique of neutral injection appears to be the most promising one for auxiliary heating of tokamaks.

The machine we propose for testing of β limitations and neutral beam heating is a modification of the present Impurity Study Experiment (ISX) machine which is designated as ISX-B. The funding requested in this proposal covers the design, fabrication, and installation of a new vacuum vessel, a new poloidal field (PF) coil system, and a new torque restraint system for the toroidal field (TF) coils. It also covers the installation of two neutral beam injectors. A comparison of the present ISX-A parameters with the proposed ISX-B parameters is shown in Table 1.

The torque restraint system consists of two fiberglass cylinders that serve a dual function of withstanding the overturning moment from the TF coils and supporting the PF coils. A higher pressure plasma requires a larger vertical field, B_v , to contain it radially. The torque on the TF coils is proportional to $B_v B_o$, where B_o is the toroidal field. Mechanical limitations of the fiberglass cylinders limit the permissible value of $B_v B_o$ and hence, for a fixed B_o , the plasma pressure. For $B_o = 1.2$ T, the maximum plasma pressure that can be contained would yield an average β of 5%. Note that the maximum β due to mechanical limitations scales approximately as B_o^{-3} (for constant plasma current), so for $B_o = 1.0$ T, $\beta = 9\%$.

Table 1. Comparison of ISX-A and ISX-B

	ISX-A	ISX-B
Size		
R_o (major plasma radius)	92 cm	92 cm
a (horizontal semi-minor axis)	26 cm	26 cm
b (vertical semi-major axis)	26 cm	49.5 cm
Plasma volume (circular/elliptic cross section)	$1.2 \times 10^6 \text{ cm}^3$	$2.3 \times 10^6 \text{ cm}^3$
Design features		
Maximum bake-out temperature	400°C	200°C
Wall material	304L stainless steel	304L stainless steel
Bundle divertor	No	Yes ^a
Number of toroidal field coils	18	18
Design limit of vertical field $\times$ toroidal field, $B_v B_o$		1.2 T^2 ^b
Performance		
Maximum magnetic field at centerline (B_o)	1.8 T	1.8 T
Maximum volt-seconds	0.9	0.9
Maximum plasma current, I_{max}	200 kA	200 kA
Injectors		
number	0	2
energy		40 keV
neutral beam power into the plasma		1.8 MW ^c
pulse duration		>200 msec
Current pulse duration	>200 msec	>200 msec

^aA bundle divertor is being designed for ISX-B and ports are being provided on the vacuum vessel for it. Divertor costs and schedules are not included in this proposal.

^bDesign limit due to stresses in toroidal coil support system.

^cUsing 40-kV, 60-A source of the type now used for PLT beam lines. Power supply capability exists to upgrade these to 40 kV and 100 A, if source development permits at a later date. This would provide 3 MW to the plasma.

The vacuum vessel is being designed to accept two neutral beam injectors of the Princeton Large Torus (PLT) type, with minor modifications. The injectors are being constructed for ORMAK Upgrade [an improved version of the Oak Ridge Tokamak (ORMAK)] but can be made available initially for the proposed experiments in ISX-B. These injectors should deliver 1.8 MW to the plasma. The ion sources can probably be upgraded in the future to provide 1.5 MW each, or a total of 3 MW into the plasma. In addition, provision is being made for a bundle divertor to be placed on the machine.

In an earlier design for ISX-B, we considered putting ports on the vacuum vessel to accept four injectors. We have allowed provision for two in this design, for the following reasons. First, the installation of four beam lines plus a bundle divertor and necessary pumping ports severely limited the amount of room available for diagnostic access. Second, the original injector design was capable of providing 0.9 MW/injector, for a total power input to the plasma (with four injectors) of 3.6 MW. As mentioned above, the two ion sources can probably be upgraded to provide 1.5 MW/injector, for a total of 3 MW into the plasma, at a fraction of the cost of adding two more beam lines. The 17% loss of power seemed to be more than offset by the advantages of better access and reduced cost.

Preliminary design work on the divertor is presently being carried out to determine the ports necessary on the vacuum vessel for attachment of the divertor. Neither the beam line nor the divertor construction is covered in the funds requested in this proposal.

The experiments that we propose to perform in ISX-B are described briefly in Sect. 2. Because much of the theoretical analysis of high β equilibrium which was presented in the ORMAK Upgrade proposal³ is also applicable here, it is not repeated in this proposal. However, a brief discussion concerning the potential of ISX-B to create FCT equilibria is presented in Appendix A. A more detailed design description of the machine is given in Appendix B. A schedule and a cost estimate are presented in Appendixes C and D. Manpower requirements are discussed in Appendix E.

2. PROPOSED EXPERIMENTS

The ISX-B vacuum vessel is designed with a simple rectangular cross section, which allows for considerable flexibility in the size and shape of the plasma. The excellent experimental access available in this design will be essential in measuring the properties of the high β , noncircular plasma we propose to produce. Table 2 gives a list of some of the diagnostic apparatus. Most of this apparatus is now on ISX-A, and one of the criteria of the new design is that a large fraction of these diagnostics could be put on ISX-B with little modification.

Table 2. Major diagnostics for ISX-B

Diagnostic	Parameters measured
Multichannel Thomson scattering (ruby, 90°)	T_e, n_e
Charge-exchange parallel and perpendicular analyzers with a diagnostic neutral beam	T_i, n_i , ion momentum
Multichannel microwave interferometer	$\int n(x,y) dy$
Magnetic flux pickup loops (including diamagnetic loops)	Plasma position, shape, β
Optical and VUV spectroscopy, spatially resolved	Impurity species and transport
Surface physics analyzers	Sputtering rates, material on walls

The proposed experiments can be divided into four categories:

- 1) noncircular equilibrium and transport studies (no injection),
- 2) neutral beam injection studies,
- 3) impurity and surface physics studies, and
- 4) electron cyclotron heating (ECH) studies.

These are discussed briefly in the remainder of this section.

2.1 NONCIRCULAR EQUILIBRIUM AND TRANSPORT STUDIES

The poloidal field design that is currently planned will allow the creation of plasmas with elongation $\sigma = b/a$ of up to $\sigma \approx 1.9$. In addition, the plasma can be changed from D-shaped to approximately elliptical. The first experiments that will be carried out on ISX-B will study the equilibrium, stability, and transport of these low β plasmas with no beam injection. These experiments will be a logical extension of the work carried out on ISX-A (which has a circular plasma) to plasmas with noncircular cross section.

As in ISX-A, the vacuum vessel, vacuum pumps, and diagnostics in ISX-B will be of essentially all-metal construction, with a minimal number of O-ring seals. Thus, we shall be able to carry out an extension of the impurity studies of ISX-A to plasmas with elongated cross section.

2.2 NEUTRAL BEAM INJECTION STUDIES

The main purpose of the proposed modification is to carry out neutral beam injection experiments. A review of the theory of neutral beam heating and a comparison of the theory with experiments on ORMAK and other machines have been covered in detail in the ORMAK Upgrade proposal.³ We outline the proposed program here and refer the reader to the ORMAK Upgrade proposal³ for a comprehensive review.

The major difference between the ORMAK Upgrade machine of Ref. 3 and ISX-B is that in ISX-B we will have the ability to change the plasma shape from circular to elliptical and D-shaped. This provides a new degree of flexibility in the experiments and one that appears particularly significant in light of the recent theoretical results on stability of high β plasmas.

The two ORMAK Upgrade neutral beam injectors, of the same internal designs as the PLT injectors, will be able to provide a total of 1.8 MW of neutral beam power into the ISX-B plasma. Upgrading of the ion sources from the present 40 kV, 60 A to 40 kV, 100 A can raise the input power to 3 MW. Work toward this objective is presently being carried out by the Plasma Technology Section in the Fusion Energy Division at Oak Ridge National Laboratory (ORNL). Plasma parameters expected for these amounts of injection power are shown in Table 3.

Table 3. Plasma parameters expected for ISX-B for three values of neutral beam injection power

	$P_{inj} = 0$	$P_{inj} = 1.8 \text{ MW}^a$	$P_{inj} = 3 \text{ MW}^a$
B_t (T)	1.2	1.2	1.2
$\sigma = b/a$	1.0	1.5	1.5
I (kA)	130	200	200
$q_{limiter}$	3.5	3.5	3.5
$\bar{n}_e$ (cm ⁻³)	2×10^{13}	5×10^{13}	10^{14}
$\bar{T}_e$ (eV)	400	900	1000
$\bar{T}_i$ (eV)	200	1000	1000
Z_{eff}	≤ 2	≤ 2	≤ 2
τ_E (msec)	8	20	25
$\bar{\beta}$ (%)	1.3	2.5	5

^aThe numbers for the $P_{inj} = 0$ case fall within ranges of parameters (n_e , B_t , a , etc.) that have been measured, so estimates of those numbers are probably fairly accurate. Scaling laws for a noncircular plasma with $P_{inj} \gg P_{OH}$ have not been experimentally investigated. Consequently, the numbers for $P_{inj} = 1.8$ and 3 MW are estimates based on extrapolations of present knowledge.

With 1.8 MW of power available, we should be able to carry out experiments extending the average β of the plasma by a factor of 3, from the present values of 1% to over 3%. One significant problem in calculating the results of these injection experiments is the uncertainty of the applicable scaling laws for maximum density, energy confinement time, etc., in this entirely new experimental regime of $P_{inj}/P_{OH} \gg 1$. Extrapolations of the present experimental results indicate that the

plasma β could reach at least 2.5% of a magnetic field of 1.2 T, provided the resulting equilibrium is stable. With 3 MW of injection power, an average β of 5% should be attainable. (For details of scaling calculations for plasmas with a circular cross section, the reader may refer to Appendix A of the ORMAK Upgrade proposal.³)

The most crucial question that the proposed experiments will attempt to answer will be the MHD stability of plasmas as β is increased. Recent theoretical work has indicated that plasmas with circular cross sections may be unstable for β greater than about 1.5%, while plasmas with elongated and/or D-shaped cross sections may be stable to MHD perturbations for β over 5%. If tokamaks are to be viable power producers, then the achievement of β in the 5-10% range is necessary. We propose to test experimentally the stability limitations and to try to maximize the β of the plasma by altering the elongation, the shape, and (to a lesser degree) the pressure profile of the plasma as the injection power is increased. The plasma expected in ISX-B should be in the collisionless regime, so that results obtained from these experiments would have direct relevance to future reactor designs.

These neutral beam injection experiments will heat the plasma (and consequently raise β) on a time scale comparable to τ_s , the slowing down time of a fast beam particle (≈ 5 msec, depending on the density). For the experimental conditions we expect to obtain on ISX-B (see Table 3), the plasma may behave like a flux conserving tokamak. The FCT concept, proposed by John Clarke at ORNL, has generated considerable interest as a possible way to attain high β equilibria. It is difficult to extrapolate the physics of present-day experiments to the plasma conditions expected with high injection power in ISX-B. However, the possibility for FCT behavior appears promising, particularly in the hot central region of the plasma. We will search for the existence of this behavior in the course of our high β injection experiments. A further exposition of FCT behavior on ISX-B is contained in Appendix A.

In addition to and in direct conjunction with the high β experiments, there will be a study of confinement and transport of plasma particles and energy in this machine. These experiments will extend the plasma

scaling results from ORMAK and other machines into a regime associated with much higher injection power and will also allow the study of the effects of elongation and shape on the transport.

2.3 IMPURITY AND SURFACE PHYSICS STUDIES

The initial purpose of the ISX experiment was to study the source and transport of impurities in tokamaks and to attempt to minimize their detrimental effect on the plasma. By the time ISX-B is ready for operation, ISX-A should have been in operation long enough to yield valuable information on impurities.

We propose to continue and extend these studies on ISX-B, using the same equipment and personnel that were used on ISX-A. This will give us the opportunity to observe and study impurity production and transport in a "clean" (i.e., almost all-metal construction with oilfree cryogenic and turbomolecular pumps) machine that is subject to increased sputtering and desorption of neutral hydrogen and impurities from the walls due to the much larger power flux to the walls. [For 3 MW of injection, the average wall loading of the vacuum chamber will be 0.15 MW/m^2 , which is a substantial fraction (~20-50%) of the wall loading expected in a fusion reactor.] The surface physics studies, which are an integral part of the impurity production research, will also be continued.

The injection of the large amounts of neutral beam power that we are proposing may cause a large influx of impurities from the walls. To allow for this possibility, the machine is being designed with ports that will allow the use of a bundle divertor. This divertor is being designed using the same criteria that were used to design the Divertor and Injection Tokamak Experiment (DITE) divertor and in close collaboration with Culham Laboratory. The design of this divertor is just starting but should be sufficiently far along to define the port requirements on the vacuum vessel by the time the information is needed. The divertor will not be installed when ISX-B first begins operation but will be added later.

2.4 ELECTRON CYCLOTRON HEATING (ECH) STUDIES

There are six ports on ISX-B designed for the injection of microwave radiation for ECH experiments. The delivery of one 200-kW source at 28 GHz is expected by October 1978; successful tests of ECH with this single tube can lead to the addition of up to ten tubes of similar design, providing up to 2 MW of power. This experimental program is covered in detail in a proposal for this work, which was submitted in June 1977.

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APPENDIX A. ISX-B AND FLUX CONSERVATION

A1. TIME SCALES FOR FLUX CONSERVATION

The primary objective of ISX-B is to test the effects of strong beam heating ($P_b/P_{OH} \lesssim 10$) in an elongated tokamak plasma. Theoretical estimates of MHD instability limits and of the transport properties of a low field, moderate size plasma such as that in ISX-B arrive at β of a few percent. With an aspect ratio $A = 3.5$ and expecting $q^2 \sim 0(10)$ one obtains from the naive formula

$$\begin{aligned}\beta_p &= \beta q^2 A^2 \\ &\approx \beta \times 10 \times 12.5 \gtrsim A,\end{aligned}$$

commonly called the "high β regime." Numerical calculations yield similar results for β_p .

With $P_b \leq 3$ MW, the heating time $\tau_h = p/(P_{OH} + P_b)$ is expected to fall below the poloidal skin time $\tau_s = 4\pi a^2/\eta_{\parallel} c^2$, so that from

$$\frac{\tau_h}{\tau_s} \approx \frac{\beta_p}{8} \frac{1}{1 + P_b/P_{OH}}$$

it appears that the plasma is taken from its initial ohmic heating equilibrium with $\beta_p \leq 1$ to $\beta_p \leq A$ in a time less than the skin time. Thus one concludes from a very simple theoretical viewpoint that in ISX-B the stage is set for a flux conserving transition to high β tokamak equilibria.

It is worthwhile to investigate this time ordering in more detail, including the effect of the finite energy confinement time τ_E and the delay in heating due to the finite slowing down time τ_f . The simplest model for the pressure rise is

$$\frac{\partial p}{\partial t} = -\frac{p}{\tau_E} + P_b(1 - e^{-t/\tau_f}) + P_{OH}, \quad (1)$$

where τ_f is the slowing down time of the fast injected particles, P_b and P_{OH} are the power/volume into the plasma due to beam injection and ohmic heating, respectively, and we define $\tau_E^{-1} = \kappa/4a^2n$, κ being the effective perpendicular heat conductivity. [We note from Eq. (1) that only in a purely ohmic steady-state discharge does τ_E equal the usual definition $\tau_E = p/P_{OH}$.]

From the solution of Eq. (1),

$$\frac{p(t)}{p(0)} = e^{-t/\tau_E} + \frac{\tau_E P_{OH}}{p(0)} \left(1 - e^{-t/\tau_E}\right) + \frac{\tau_E P_b}{p(0)} \left[\left(1 - e^{-t/\tau_E}\right) + \frac{e^{-t/\tau_E} - e^{-t/\tau_f}}{1 - \tau_E/\tau_f} \right], \quad (2)$$

one obtains for the fractional pressure rise at $t = \tau_E$

$$\frac{\Delta p}{p} = \frac{p(\tau_E) - p(0)}{p(0)} = 0.632 \left[\frac{\tau_E (P_{OH} + P_b)}{p(0)} - 1 \right] + \frac{\tau_E P_b}{p(0)} \left(\frac{e^{-1} - e^{-\tau_E/\tau_f}}{1 - \tau_E/\tau_f} \right), \quad (3)$$

and we recall that for $P_b = 0$, $\frac{\tau_E P_{OH}}{p(0)} - 1 = 0$. In the desirable limit $\tau_f/\tau_E \ll 1$, Eq. (3) becomes

$$\frac{\Delta p}{p} \approx 0.632 \left[\frac{\tau_E (P_{OH} + P_b)}{p(0)} - 1 \right], \quad (4)$$

an expected result, whereas in the opposite limit of poor confinement $\tau_f/\tau_E \gg 1$, there results

$$\frac{\Delta p}{p} \approx 0.632 \frac{\tau_E P_{OH}}{p(0)} \left(1 + 0.582 \frac{\tau_E P_b}{\tau_f P_{OH}} \right) - 0.632, \quad (5)$$

indicating that beam heating will not occur unless $P_b/P_{OH} > (\tau_f/\tau_E) \times 2.72$.

Specifically, for the ISX-B parameters volume (Vol.) = $1.23 \times 10^6 \text{ cm}^3$ (for circular cross section), $n = 2 \times 10^{13} \text{ cm}^{-3}$, $T_e + T_i = 0.6 \text{ keV}$, and $\tau = 8 \text{ msec}$ (from Table 3 in the text), one has $\frac{p(0)}{\tau_E} \times \text{Vol.} = 300 \text{ kW}$.

It then follows from Eq. (4) that to produce a $\Delta p/p = 2$ in one energy containment time takes 1.26 MW, quite within the range of available beam

power. This confirms the statement made previously that ISX-B may have the injection power to heat in a flux conserving manner, provided the energy confinement time τ_E does not strongly deteriorate with high power injection, which ultimately only the experiment can show. (Model-dependent transport simulations, including trapped ion mode anomalous diffusion and beam driven impurity effects, lead to less optimistic predictions at 3 MW of beam power.)

A2. THEORETICAL EXPECTATIONS OF FLUX CONSERVING BEHAVIOR

A2.1 Equilibrium

Briefly, one expects a pressure-driven plasma current rise $\propto \bar{p}^{1/3}$, conservation of the $q(\psi)$ profile, and absence of the separatrix equilibrium limit as $\beta_p \rightarrow R/a$. The current rise may be less than calculated because of plasma expansion and scrape-off by the limiter but should show up in the magnitude of the poloidal field system currents needed to provide equilibrium. Concerning $q(\psi)$, $q(0) = 1$ is expected to be maintained via sawtooth oscillations at any rate. In a non-flux-conserving tokamak, however, $q(a)$ would rise sharply as the temperature profile peaks in the center, as shown by Fig. A1 (Ref. 1). In contrast, a flux conserving tokamak $q(a)$ will remain at its initial low β value. (The minor radius a stands for the largest plasma radius still obeying flux conservation.) Concerning β_p , we recall that under flux conserving conditions the naive proportionality formula $\beta_p = \beta A^2 q^2$ is replaced by a weaker function $\beta_p = \epsilon^{-2/3} \beta_{p,in}^{1/3}$ (where $\beta_{p,in}$ is the average pressure normalized by the square of the initial plasma current). The onset of a separatrix equilibrium limit is predicted at about the same pressure that drives the ballooning mode unstable (see Sect. A2.2), so that it might be difficult to tell these effects apart.

A2.2 Stability

With the gross $q(0) < 1$ instabilities eliminated by the sawtooth oscillations, independent of flux conservation, there remain the pressure-driven ballooning modes for

Fig. A1. Comparisons of FCT ($\bar{\beta} = 4.6\%$, $\beta_{\max} = 17.3\%$, $\beta_p = 3.03$, $p_0 = 162.1$) and non-FCT ($\bar{\beta} = 3.3\%$, $\beta_{\max} = 16.9\%$, $\beta_p = 5.21$, $p_0 = 211.3$) profiles: (a) FF' and (b) q . Small changes in FF' result in significant differences in q .

$$\beta > \beta_{\text{marg}} \approx \epsilon/q^2. \quad (6)$$

The tendency of high β , FCT equilibria to establish an absolute minimum B configuration may stabilize modes susceptible to the sign of $\nabla_{\perp}|B|$ but cannot influence the ballooning modes driven by the bad curvature $(\underline{b} \cdot \nabla)\underline{b}$ on the outside of any tokamak configuration. Note, however, from Eq. (6) and the remarks on $q(\psi)$ in Sect. A2.1 that if flux [and therefore $q(a)$] is conserved at its modest low β value, substantially larger marginal β 's should result in flux conservation. Finally, there might be a destabilizing effect due to the additional bootstrap current which may drive the total plasma current over the Kruskal-Shafranov limit. (Unfortunately, the theory of the bootstrap current in a high β , impure plasma has not been worked out so far.)

A2.3 Transport

The theoretically predicted pressure-driven current increase in an FCT should improve confinement. As an absolute minimum B configuration is established, the number of trapped particles decreases [by at most $\leq 30\%$ (see Ref. 2)], thus reducing neoclassical and trapped particle mode transport. In the same vein, the minimum B equilibrium changes the drift resonance as a "maximum J" ($= \oint d\ell v_{\parallel}$) geometry is established. These improvements of confinement in a high β flux conserving tokamak may be balanced by the enhanced internal MHD activity (incipient ballooning modes, β -driven "flutter" transport).

The central question in the overall context of this note, however, is the question of possible anomalous flux diffusion. Consider the flux diffusion equation

$$\frac{\partial \psi}{\partial \tau} = \eta_{\parallel} R j_{\phi} - \underline{v} \cdot \nabla \psi, \quad (7)$$

where the boundary conditions for Eq. (7) may be nonideal (because the metallic limiter will not only remove particles and energy but also flux on the time scale of interest). Transport theory links j_{ϕ} and the diffusion flux $\underline{v} \cdot \nabla \psi$ to the driving "forces" ∇n and ∇T of all species

and the electric field $\underline{E}$, thus revealing possible effects of impurities on $\partial\psi/\partial t$, beyond Z_{eff} contained in $\eta_{\parallel}$ (see Ref. 3). This points to a natural confluence of the ISX-A and ISX-B missions, in that auxiliary heating and impurity studies are inseparable in the achievement of high β .

A3. EXPERIMENTAL POSSIBILITY OF MEASURING FLUX CONSERVATION

Until one can obtain good profiles for the poloidal magnetic field in any tokamak, it is not possible to resolve the FCT-related local properties of $j_{\parallel}$, $j_{\perp}$, and $q(\psi)$. (Because flux diffusion in a flux conserving tokamak has not reached its steady state, the measured temperature profile will not determine the $j_{\parallel}$ profile!) Global properties such as current, β_p , plasma shift, and vertical field index will carry a limited amount of FCT information. Combined with measurements of temperature, density, and poloidal field (from pickup loops outside the elongated plasma), one hopes to obtain an indication of possible flux conserving behavior from a combination of measurement and numerical equilibrium calculations.

The ultimate direct measurement is that of the global lifetime of the high β configuration, supported by gas puffing, beam heating, and poloidal field control. Because the classical skin time of the hot plasma may reach 300 msec, this theoretical lifetime would exceed the pulse time of the machine.

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1. Y-K. M. Peng (ORNL), private communication, 1976.
2. Y-K. M. Peng and J. A. Rome (ORNL), private communication, 1977.
3. J. T. Hogan, "Injection, Compression, and Transport Studies," paper presented at the Annual Meeting on Theoretical Aspects of Controlled Thermonuclear Research (Project Sherwood), San Diego, California, May 4-6, 1977.

APPENDIX B. DESCRIPTION OF THE ISX-B DESIGN

A reference design has been established for ISX-B which involves a major modification to the existing ISX-A device. These modifications will allow ISX-B to have a larger, elongated plasma; a higher vertical field; more diagnostic access; and two neutral beam injectors. In an effort to ensure low costs, short schedules, and maximum reliability, as many components as possible will be reused from ISX-A and the design of new components will utilize only existing technology. The principal parameters and dimensions for ISX-B are shown in Table B1.

The physics goals for ISX-B require replacement of the ISX-A vacuum vessel, the poloidal coil system, and the torque restraint system which resists the overturning moment of the toroidal field coils and serves as a support for the poloidal coils. Existing ISX-A components which will be reused are the iron transformer core, toroidal field coils, bucking ring (for toroidal field coil centering forces), torus vacuum system, main support base, and toroidal field power supply. In addition, most of the instrumentation, data acquisition, and data coordination equipment will be retained.

A new austenitic stainless steel vacuum vessel will be fabricated to confine the plasma in ISX-B (see Figs. B1 and B2). The vessel will be of all-welded construction with a 67.9- by 111.7-cm cross section and will provide room for a plasma with a 26-cm radius and an elongation, σ , of approximately 1.9/1. The vessel will have a maximum wall thickness of 1.9 cm and will not require reinforcing structure. Two 27.9-cm-long rectangular bellows will be provided to give the vessel a resistance of 3.7 m Ω . The bellows will be fabricated from 0.025-cm-thick Inconel and will be differentially pumped for structural reliability.

The ISX-B vacuum vessel will have approximately 172 diagnostic ports to provide top, bottom, and radial access. Port sizes will range from 5 to 30 cm in diameter with nine special 5- by 48-cm rectangular ports. Neutral beam injection will be through two 28-cm-diam injector ports located 60° apart in Sectors 1 and 16 (see Fig. B1). The entire vacuum vessel will be bakeable to 200°C; however, present plans call for discharge

Table B1. ISX-B baseline design parameters

Plasma

Cross section	Circular, elliptical, or D-shaped
Major radius (nominal)	92 cm
Plasma size (nominal)	26×49.5 cm
Centerline field	<1.8 T
Plasma current (maximum)	200 kA
Safety factor at limiter	≥ 3

Toroidal field system^a

Magnetic field at plasma axis	1.8 T
Repetition rate	1/min
Flat top length	250 msec
Current (maximum)	120 kA
Number of turns	72
Number of turns per coil	4
Ampere turns	8.6×10^6
Maximum driving voltage	1000 V
Inductance	2.2 mH
Resistance	2.6 m Ω
Flat top power	30 MW
Energy per pulse	24 MJ
Coil shape	Rectangular
Conductor material	Copper
Maximum temperature rise per pulse	12°C
Coolant	Water
Vertical aperture	132 cm
Radial aperture	105 cm
Conductor cross section	2.5×12.7 cm
Current density	4000 A/cm ²
Conductor weight	12,000 kg
Centering force (per coil)	10,700 N
Overturning moment (per coil)	74,700 N-m

Table B1 (continued)

Transformer core and yoke ^a	
Material	M-6 grain oriented silicon steel
Lamination thickness	0.356 mm
Core radius	29 cm
Flux change	0.9 V-sec
Weight of core and yoke	26,000 kg
Ohmic heating system ^b	
Coil material	Copper
Current	25 kA
Turns ratio	8:1
Driving voltage	400 V initial; 50 V flat top
Conductor area	10 cm ²
Temperature rise per pulse	0.5°C
Coolant	Water
Vertical field system ^b	
Coil material	Copper
Maximum current	5 kA
Total turns	36
Driving voltage	1000 V
Coolant	Water
Shaping field system ^b	
Coil material	Copper
Maximum current	25 kA
Total turns	6
Driving voltage	36 V
Coolant	Water
Vacuum pumping	
Pumps	Two 450-l/sec turbomolecular pumps; two 2000-l/sec cryosorption pumps
Design base pressure	6×10^{-9} torr

Table B1 (continued)

 Vacuum vessel

Type	All welded
Material	304L stainless steel
Wall thickness	
maximum	1.9 cm
minimum	1.27 cm
Bake-out temperature	200°C

Limiter

Inside, outside, top, and bottom	Fixed rail
Material	Variable ^c

^aSame as used for ISX-A.

^b"Standard" design; may be changed to alternate concept utilizing same power supplies.

^cLimiter design will allow testing of different materials; first limiter probably molybdenum.

Fig. B1. Top view of ISX-B.

Fig. B2. Cross section of ISX-B.

cleaning only. The vessel weight will be supported in 18 locations, but no radial restraint of the vessel is necessary due to its monolithic, thick-wall design.

The entire ISX-A ohmic heating and vertical field coil system will be replaced. Coils will be fabricated from 1.65- by 6.2-cm hollow copper conductor on hand at ORNL. The 36-turn vertical field system will provide a maximum vertical field of 0.1 T. The vertical field system will be powered by a 1000-V, 5000-A solid state power supply which is being purchased for ORMAK Upgrade. The 8-turn ohmic heating system will be driven at 72 V, 25 kA by the ohmic heating battery bank supply used for ORMAK and ISX-A. The system will utilize the existing ISX-A transformer core and will provide a plasma current up to 200 kA. Ohmic heating and vertical field coils will be mounted on new fiberglass/stainless steel cylinders which provide resistance to toroidal coil overturning moments.

In addition to the vertical field and ohmic heating windings, a set of four coils mounted on the fiberglass cylinders and two coils mounted on the vacuum vessel will provide a shaping field. These shaping coils will vary the index of the vertical field to allow a change in the plasma elongation from $\sigma = 1.0$ to $\sigma = 1.9$ and will be driven by a 36-V, 25-kA supply (the ISX-A vertical field supply). Present plans call for the shaping system to have a baseline design capable of creating a D-shaped plasma with a triangularity of approximately 0.5. The device will be designed so that the plasma triangularity may be changed by changing the interconnection of the coils generating the shaping field.

Detailed calculations are presently under way to optimize the poloidal coil locations for our proposed experiments. In addition, Y-K. M. Peng of ORNL Fusion Energy Division has proposed a different coil system which provides the same fields but has the coils connected to the power supplies differently. In this concept, the poloidal coils on the inner fiberglass cylinder are connected to one supply, those on the outer cylinder to a second supply, and the coils at the top and bottom of the vacuum vessel to a third supply. Preliminary calculations indicate that this system may have some significant advantages for containing and shaping a high β plasma. Both concepts will be given further study with a decision on the final poloidal coil design to be made by July 31, 1977.

Because the toroidal coil overturning moments of 25,000 N-m per coil in ISX-A will increase to 74,000 N-m in ISX-B, and radial access to the vacuum vessel will be different, new torsion cylinders must be fabricated. The cylinders will be fabricated from fiberglass with stainless steel in the load transfer area to ensure maximum dependability. The 2.5-cm-thick fiberglass cylinders will be fabricated using a woven roving glass fabric with a room temperature curing epoxy. Holes will be cut in the outer cylinder for diagnostic access. The stainless steel sections of the cylinders will be provided with dielectric breaks. Poloidal field coils will be bolted directly to the cylinders prior to installation on the device.

Two neutral beam injectors being fabricated for ORMAK Upgrade will be installed initially on ISX-B. These injectors represent a slight modification to the PLT injector design and will use 40-kV, 60-A sources to impart approximately 0.9 MW each to the plasma. The ORMAK Upgrade project will provide the beam lines, sources, and all other electrical gear and power supplies with the exception of the accel power supply, which will be time-shared with the existing 150-kV test stand. Injector costs incurred by the ISX-B project will be for injector installation, modifications for time-sharing the 150-kV accel power supply, and utility services to the injectors.

The design, fabrication, and installation of diagnostics are not included in this proposal. Many of the ISX-A diagnostics will be modified for use on ISX-B. Funding requirements for diagnostics will be covered separately.

Design efforts are being made to utilize existing ISX lifting fixtures and installation equipment for ISX-B. Assembly procedures are being formulated to minimize the disassembly of reusable components. Existing services and utilities will be used when possible.

APPENDIX C. PROJECT MILESTONES AND SCHEDULES

Figure C1 is a schedule of the design, procurement, fabrication, and assembly of ISX-B. Assuming Energy Research and Development Administration (ERDA) approval of the directive request by August 1, 1977, and availability of funds as outlined in Appendix D, we estimate that ISX-B and its injectors can begin operation by June 1, 1978. This estimate is based on ISX-A being made available for modification on February 1, 1978. A list of project milestones and a schedule for their completion is given in Table C1. These dates are achievable by a disciplined priority effort, but there is very little allowance for contingencies. We should emphasize that the goal is to be prepared for ISX-A shutdown on February 1, 1978. If the ISX-A shutdown date is delayed, then all dates subsequent to that will be delayed by an equal amount.

Fig. C1. Design, procurement fabrication, and assembly of ISX-B.

Table C1. Project milestones

Milestone	Schedule
Vacuum vessel	
All drawings complete	August 1977
Vessel completed and leak checked	March 1978
Torque cylinders	
Complete design	August 1977
Cylinders delivered	November 1977
Poloidal coils	
Design and location specified	July 1977
Complete fabrication	November 1977
Coils installed on cylinders	March 1978
Neutral beam systems	
Beam line power supplies (except accel supply) delivered by ORMAK Upgrade program	January 1978
Modification of 150-kV test stand supplies for use as beam line accel supplies complete	January 1978
Beam lines 1 and 2 (without cryopumps) delivered by ORMAK Upgrade program	March 1978
Cryopump for beam line 1 installed	April 1, 1978
Cryopump for beam line 2 installed	June 1, 1978
Machine installation	
All design for electrical components and utilities complete	October 1977
ISX-A shutdown	February 1, 1978
Tokamak installation complete; begin operation	June 1, 1978
Beam system 1 installation complete; begin test operation	June 1, 1978
Beam system 2 installation complete; begin test operation	June 1, 1978

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APPENDIX D. COST ESTIMATES

Cost estimates are based on analysis of the conceptual design and the proposed construction schedule; the estimates include the cost of detailed design, procurement, fabrication, and assembly. The estimates were made by the Union Carbide Corporation Nuclear Division (UCC-ND). Engineering estimating organization based on drawings and bills of material produced by UCC-ND engineers and specialists and using standard estimating procedures. Because this project is a modification of ISX-A, much of the equipment exists and is operational. The neutral beam injector systems, poloidal coil copper, and poloidal field power supply are available from the ORMAK Upgrade project. A contingency of 30% is included to allow for uncertainties in the design and cost estimation.

Table D1 gives a breakdown of the total estimated cost of the ISX-B device. There are no plans to procure additional capital equipment. Table D2 indicates the operating expenses related to ISX-B. These costs are not covered by the funds requested in this Major Device Fabrication (MDF) proposal, but will be obtained from ISX Operations and High Beta Experiment Operations budgets.

Table D1. Total estimated cost by subsystem and fiscal year

Description	Estimated cost (in thousands of dollars)		
	1977	1978	Total
Tokamak			
Vacuum vessel	125	175	300
Torque structure	70	8	78
Poloidal coil system	9	30	39
Installation	0	75	75
Engineering	<u>271</u>	<u>110</u>	<u>381</u>
Subtotal	475	398	873
Neutral beam system installation			
Electrical and instrumentation	110	241	351
Piping (water and cryogenics)	95	219	314
Platforms and enclosures	20	45	65
Installation	0	40	40
Engineering	<u>65</u>	<u>40</u>	<u>105</u>
Subtotal	290	585	875
TOTAL	765	983	1748
Contingency (30%)			524
Escalation (9% on FY 1978)			<u>88</u>
GRAND TOTAL			2360

Table D2. Related operating costs other than device fabrication
(in thousands of dollars)

	FY 1977	FY 1978	FY 1979
Management and technical support	120	240	240
Engineering services	70	100	60
Diagnostics			
Design and construction	660	710	200
Capital equipment	200	350	250
Bundle divertor	50	250	100
Operating expenses	<u>0</u>	<u>2000</u>	<u>4000</u>
TOTAL (excluding capital)	900	3300	4600

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APPENDIX E. PERMANENT MANPOWER AND MANAGEMENT PLAN

The ISX-B modification will be designed by the UCC-ND Engineering Organization. Components will be procured from industry or fabricated in ERDA-ORO facilities; assembly will be performed by UCC-ND or a cost-plus-fixed-fee contractor; and overall project management and specialized technical input will be provided by ORNL Fusion Energy Division permanent staff. No increase in permanent staff is required for these activities.

ISX-B will be operated by the Tokamak Experimental Section of the ORNL Fusion Energy Division using personnel who presently operate ISX-A. It will not be necessary to recruit additional operating staff, and training requirements will be minimal. Following the shutdown of the ISX-A (tentatively scheduled for February 1978), approximately half of the staff will be engaged in development and preparation of diagnostics for use with ISX-B. The full operating staff will be needed by July 1978. No incremental permanent manpower is required.

The Tokamak Experimental Section will be responsible for the construction and operation of the ISX-B device. A program manager representing the Tokamak Experimental Section will manage design and construction of the device in accordance with Division of Magnetic Fusion Energy MDF procedures. The development and execution of the experimental program will be guided by an experimental planning committee composed of senior physicists from the Tokamak Experimental Section.

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