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**AN IMPURITY CONTROL TEST FACILITY (ICTF) FOR THE STUDY OF
FUSION REACTOR PLASMA/EDGE MATERIALS INTERACTIONS**

INTERIM REPORT

by

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May 1984

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ABSTRACT

A test facility for investigating many of the impurity control issues associated with the interactions of materials with the plasma edge is outlined. Analysis indicates that the plasma edge conditions expected in TFCX, INTOR, etc. can be readily produced at the end cells of an rf stabilized mirror, similar in some respects to the Phaedrus device at the University of Wisconsin. A steady-state, Impurity Control Test Facility (ICTF) based on such a mirror device is expected to produce a plasma with typical parameters of $n_e \sim 3 \times 10^{18} \text{ m}^{-3}$, $T_e = 50 \text{ eV}$, and $T_i = 100 \text{ eV}$ at each end cell. A heat load of $\sim 2 \text{ MW/m}^2$ over areas of $\sim 1600 \text{ cm}^2$ could be produced at each end with 800 kW of ICRH power. These conditions would provide a unique capability for examining issues such as erosion/redeposition behavior, properties of redeposited materials, high recycling regimes, plasma edge operating limits for high-Z materials, and particle pumping efficiencies for limiter and divertor designs.

1. INTRODUCTION

1.1 Need for Test Facility

Fusion reactor design studies have identified impurity control as a critical issue for the successful operation of magnetic fusion devices, e.g. Refs. 1-3. In particular, both the lifetime of the impurity control components and the impurity buildup in the plasma will depend on the phenomena that occur at, and near, the materials/plasma edge interface. Unless the proper plasma regimes, and/or materials choices are employed, impurity control may be a very serious limitation to fusion reactor development. For example, for the "medium" edge temperature regime identified for INTOR, the limiter or divertor lifetime would be less than two weeks if acceptable redeposition does not occur. In low temperature regimes the physics of high recycling and impurity transport need thorough explanation. If a divertor is needed in future devices, instead of a pumped limiter, this can make a cost difference of more than \$100 M per device. In a reactor such as TFCX, impurity control presents two crucial issues: (1) achieving adequate helium removal; and (2) having acceptably low plasma contamination from sputtering of the surface.

In general, the impurity control issues can be divided into plasma physics and technology issues affecting (1) the central plasma; and (2) the edge and material boundary (scrape-off) regions. The issues affecting the central plasma and some of the edge region, such as helium containment times, will require the use of tokamak and mirror test devices. Most of the issues affecting the edge and scrape-off regions, however, can be examined with other test facilities. The major issues for the scrape-off region include: (1) sputtering erosion/deposition; (2) materials selection; (3) impurity control component lifetime; (4) vacuum pumping performance; and (5) high plasma recycling regimes.

At present, theoretical models are being constructed and a few tokamak experiments are being conducted to investigate some of these issues. In general, however, the interactions at the plasma/materials interface are poorly understood, and there is a crucial need to establish a technology data base in this area prior to the construction of a long pulse ignition device. Available tokamak and other devices do not exhibit the conditions necessary to adequately study plasma edge/materials interaction issues of interest to the

next generation tokamak and mirror devices. As a result, a separate facility which simulates the expected plasma edge conditions should be very useful and cost effective.

An impurity control test facility would have a number of advantages over large tokamak or mirror systems for examining impurity control issues:

- (1) The facility would be totally dedicated to impurity control issues.
- (2) The facility could be operated cw so that materials-related phenomena (e.g., the buildup of redeposited material) can be readily investigated.
- (3) The test parameters could be varied over a larger range than would be available in any single tokamak system.
- (4) The facility could be operating in a relatively short period of time following approval of the project.
- (5) The cost of such a system is anticipated to be much less than the cost of a tokamak system.
- (6) The facility would provide much of the data that are vital for the designs of future tokamaks and mirror impurity control systems. Components, such as limiters, could be pretested in an ICTF before being employed in an actual reactor.

A conceptual design study of an ICTF is being performed at Argonne National Laboratory. The emphasis to date has been on defining the overall device parameters, and on defining the experimental program. Future work will examine detailed cost estimates of the ICTF, and the size/cost tradeoffs. The initial device design is a mirror about 3 m long having 20 cm diameter end cell plasmas, and a total of about 800 kW of RF power needed to heat the plasma. The following sections will discuss the results of this study. The required conditions for ICTF, the ICTF conceptual design, and the experiments that could be performed with such a facility will be described.

1.2 Required Operating Environment

The test facility should obviously provide plasma conditions as close as possible to those anticipated in tokamak and mirror reactors. Among the key features of tokamak edge conditions are: high heat flux, high hydrogen and impurity particle recycling, near tangential B-field-to-surface angles, high-B

field, and the presence of a sheath potential at the surface. All of these features are important to the erosion and pumping performance of impurity control components. For example, high hydrogen recycling results in a substantial flux of charge exchange neutrals to the surface. In order to obtain a high hydrogen recycling rate, a test surface area large in relation to the mean free path for hydrogen atom ionization and charge-exchange is needed. For similar reasons, a large test area is needed to realistically test pumped limiters and simulated divertor pumping efficiencies. Shallow angle magnetic field geometries critically affect the sheath width, electron re-emission, sputtering and reflection coefficients, and charge states of redeposited ions. Table 1-1 describes the desired operating parameters of an ideal test facility which should be sufficient to simulate tokamak and mirror edge conditions. As will be discussed in the next section, an RF-heated, mirror-based ICTF can provide most, if not all, of these parameters. The proposed mirror device is similar in principle to the central cell of the Phaedrus device at the University of Wisconsin. There are two major differences: (1) the ICTF would be steady state, and (2) the ICTF would be smaller than Phaedrus. However, the operating principles of both devices would be similar. It may be possible to relax some of these parameters and still obtain useful information. For example, a 2 T field could be used in the end cell instead of a 4-5 T field, and a 15 degree angle could be used as the nominal inclination plate angle instead of 5°. We plan to examine the cost and performance trade-offs of an ICTF in the near future.

Table 1-1. Desired Operating Conditions of Ideal Test Facility

Parameter	Value
1. Plasma size (each end)	20-cm diameter
2. Angle between the magnetic field and the test surface	5-90 deg
3. Heat flux (at 5-deg field/surface angle)	2 MW/m ²
4. Plasma ion temperature	50-200 eV
5. Plasma electron temperature	~50 eV
6. Sheath potential	1-3 T _e
7. Plasma density	10 ¹⁸ -10 ¹⁹ m ⁻³
8. Magnetic field	5 T
9. Hydrogen ion flux (at 5-deg field/surface angle)	3 × 10 ²² m ⁻² s ⁻¹ (at 50 eV)
10. Hydrogen neutral (charge-exchange flux)	~1/2 of ion flux
11. Helium flux	~5% of hydrogen flux
12. Operation	CW

2. ICTF CONCEPTUAL DESIGN

As mentioned in the proceeding section, the proposed ICTF will operate in a similar manner to Phaedrus though ICTF will have better plasma performance appropriate to its requirements for impurity control testing. In this section the physics of the Phaedrus mode of mirror operation are reviewed as they apply to the ICTF. In the next section a reference ICTF design is outlined.

2.1 ICTF Operation Based on Phaedrus

The Phaedrus experiment at the University of Wisconsin has demonstrated that a stable axisymmetric single mirror cell plasma can be achieved using only radio frequency heating.⁽⁴⁾ Startup is accomplished using Electron Cyclotron Heating (ECH) followed by Ion Cyclotron Heating (ICRF).⁽⁵⁾ Fueling is achieved by ionizing a gas puff with electrons heated by ICRF, while stability is provided by a radial ponderomotive force by the ICRF.⁽⁴⁾ These techniques have also been utilized at other laboratories. For example, startup using ECH has been achieved on the TMX-U experiment at Livermore,⁽⁶⁾ RF sustained operation was demonstrated on the RFC-XX double cusp + central cell experiment at Nagoya,⁽⁷⁾ and MHD stability by radial ponderomotive force was studied at Kyoto.⁽⁷⁾

Startup densities are limited by the constraint that the cutoff density for fundamental electron cyclotron waves (i.e., where $\omega_{pe} = \omega_{ce}$) is much less than the steady state density. In Phaedrus, startup densities as high as $4 \times 10^{17} \text{ m}^{-3}$ ⁽⁸⁾ or four times the cutoff density are currently achieved. This initial density scales with the square of the magnetic field so initial densities as high as $6 \times 10^{18} \text{ m}^{-3}$ should be achievable in ICTF, which has four times the central cell B field as Phaedrus.

Stability in Phaedrus is achieved by balancing the ponderomotive force associated with the radial gradient of the RF electric field E against the curvature force. Note that the ponderomotive force does not have to balance the plasma pressure. The balance conditions can be written⁽⁴⁾

$$\frac{mv^2}{R_c} = \frac{e^2}{4m_i} v_r \frac{E^2}{\omega^2 - \omega_{ci}^2} \quad (1)$$

where R_c is the radius of curvature of the magnetic field, v is the ion thermal velocity, ω_{ci} is the ion cyclotron frequency, and the gradient is in the radial direction. The presence of the resonance denominator means that the balance depends on how close the RF frequency can be chosen to the ion cyclotron frequency ω_{ci} . There are two possible regimes. In one, the difference is determined by the "adiabaticity condition"

$$\omega - \omega_{ci} > \frac{v}{\ell_z} \quad (2)$$

where ℓ_z is the axial scale length of $\nabla_r E^2$ and v/ℓ_z is the time an ion spends in the region of the radial E^2 gradient. Eq. 2 states that the ion must remain in contact with the electric field gradient for at least several cycles of the RF. It is apparent that a large ponderomotive force can be achieved if there is a long region of uniform axial magnetic field. If the axial field is very uniform, the frequency difference may be limited by Coulomb collisions so that

$$\omega - \omega_{ci} > \nu = A n / T_i^{3/2} \quad (3)$$

where ν is the Coulomb collision frequency.

Solving for the required RF electric field strength gives

$$E^2 > \frac{8\ell_r T_i}{R_c} \frac{B}{ec} (\omega - \omega_{ci}) = \frac{8\ell_r T_i^{3/2}}{R_c} \frac{B}{ec\ell_z \sqrt{m_i}} \quad \text{or} \quad \frac{8\ell_r B A n}{R_c ec T_i^{1/2}} \quad (4)$$

for the two regimes. It is likely that Phaedrus operation lies in between the two limiting values.

Experiments on Phaedrus operating as a tandem mirror (with mirror end cells) have been carried out with central cell ICRF at either $0.8 \omega_{ci}$, or with both frequencies present. When $0.8 \omega_{ci}$ (the "RF sustained" mode) is used, stability is provided by high plasma pressure in quadrupole end cells.^(5,9) When $1.01 \omega_{ci}$ is used (the "SAM" - Stabilized Axisymmetric Mirror mode) stability is provided by central cell radial ponderomotive force which also can anchor axisymmetric end cells.⁽⁴⁾ When both frequencies are used,

stability is enhanced by the radial ponderomotive force while electron heating is provided by the $0.8 \omega_{ci}$ RF. Recent experiments have demonstrated that the two frequency techniques behave as expected and have also demonstrated that stability can be provided by frequencies as high as $1.7 \omega_{ci}$.

For all experiments central cell magnetic fields have been in the neighborhood of 500 G. In steady state, plasma is produced by ionization of hydrogen puffed into the central cell at a gas box which also serves as the limiter for the central cell plasma. In these experiments "steady state" plasma density has only been achieved in the range $1 \times 10^{18} \text{ m}^{-3} < n < 5 \times 10^{18} \text{ m}^{-3}$. RF sustained operation, regardless of the mode that is used, requires that the electron temperature be maintained at a sufficiently high value for ionization of the gas puff, i.e., somewhat greater than 10 eV, and that sufficient power be coupled to the electrons to balance losses (due primarily to dissociation, ionization and excitation of H_2 gas at low plasma densities, and axial losses at high densities). Typical electron temperatures have been in the range of 20 to 40 eV.

Fairly extensive data exist for radial profiles of plasma parameters for the "RF sustained" mode but not for the other modes of operating. Representative results for the radial profiles of density, electron temperature, plasma potential and floating potential are given in Figs. 2-1 to 2-8. A radial density profile that is representative of low power Phaedrus RF sustained data is given in Fig. 2-1. The profile has a general gaussian appearance with an e-folding radius of 12 cm. The corresponding radial electron temperature and plasma potential profiles are given in Figs. 2-2 and 2-3. Note that there is only a slight indication of temperature increase at the edge of the plasma. The plasma potential is seen to be hollow, increasing with radius out to approximately 12 cm. Beyond that radius the potential falls close to zero within approximately 5 cm. The corresponding ballast tank plasma potential and parallel electron temperature profiles are given in Figs. 2-4, 2-5, and 2-6. Note that the horizontal scale in Fig. 2-6 is in terms of "plug" cm which are half as large as central cell cm for the same flux tube. These data show that near the end wall the plasma potential is approximately equal to $1 + 2 T_e/e$ and the floating potential is approximately zero. The increase in the parallel electron temperature at large radii comes from the direct RF electron heating which increases at the plasma edge.

Figure 2-1. Central cell density ($E + 12 \text{ cm}^{-3}$) -vs- R (cm) at 50 milliseconds, shots 31260-31287, swept CS probe gaussian fit with $N_0 = 2.6E + 12 \text{ cm}^{-3}$, $R_0 = 12 \pm 1 \text{ cm}$.

Figure 2-2. Central cell T_e (eV) -vs- R (cm) at 5.0 milliseconds, shots 31260-31287, swept CS probe straight line fit at $T_e = 16.9$ eV.

Figure 2-3. Central cell plasma potential (V) -vs- R (cm) at 5.0 milliseconds, shots 31260-31287, swept CS probe. (A smooth curve is drawn through the data.)

Figure 2-4. Ballast tank potentials (V) -vs- R (plug cm) at 5.5 milliseconds, shots 31301-31641, swept BT probe, O = plasma, x = floating, mapped relative to the end loss axis.

Figure 2-5. Ballast tank electron temperature (eV) -vs- R (plug cm) at 5.5 milliseconds, shots 31301-31641, swept BT probe, mapped relative to the end loss axis; smooth curve drawn.

Figure 2-6. T_e (eV) -vs- plug radius (cm) measured by the end loss analyzers showing evidence for electron pumping.

Figure 2-7. Radial density and pressure profiles. Plasma pressure and density are uniform to $r = 12$ to 15 cm.

Figure 2-8. Central cell operating parameters as a function of RF power.

Radial density and plasma pressure profiles for SAM operation are given in Fig. 2-7. In this mode the plasma density is seen to be approximately flat out to approximately 15 cm, so that it is both flatter and broader than in RF sustained operation. The dependence of electron and ion temperature, line density and beta on coupled RF power is given in Fig. 2-8. Ion temperature is seen to increase faster than linearly while the line density is observed to saturate. In recent experiments ion temperatures as high as 450 eV have been achieved.

It is straightforward to show that the constraint of sufficiently high electron temperature leads to the observed limitations on density. First consider the case of $0.8 \omega_{ci}$ heating. Electron heating comes from parallel electron heating by the z component of the electric field associated with "slow" Alfvén waves. The dispersion relation for electromagnetic waves in the central cell can be approximated by the ($m = 0$) dispersion relation for an infinite medium:⁽¹⁰⁾

$$k_{\perp}^2 c^2 = \frac{\omega^4 \gamma^2 \omega_{ci}^2 - \omega^2 (2\gamma \omega_{ci}^2 k_{\parallel}^2 c^2 + k_{\parallel}^2 c^4) + \omega_{ci}^2 k_{\parallel}^2 c^4}{\omega^2 (\gamma \omega_{ci}^2 + k_{\parallel}^2 c^2) - \omega_{ci}^2 k_{\parallel}^2 c^2} \quad (5)$$

where $\gamma = \frac{\omega_{pi}^2}{\omega_{ci}^2}$.

This equation is graphed in Fig. 2-9 for $\omega/\omega_{ci} = 0.8$ for $n = 4 \times 10^{18} \text{ m}^{-3}$ and $B = 5 \text{ kgauss}$ for a hydrogen plasma. Note that both branches are asymptotic to the curve $k_z^2 = k_{\parallel}^2 = k_A^2/[1 - (\omega/\omega_{ci})^2]$ where $k_A = \omega/v_A$. It is apparent that roots consistent with the plasma column dimensions approximately correspond to $k_{\parallel} \approx k_A$ or to phase velocity $\omega/k_{\parallel}$. Effective heating by Landau damping requires that this phase velocity match the electron thermal velocity, i.e. $\omega/k_{\parallel} = v_e$, so

$$(B^2/4\pi n m_i) \cdot (1 - (\omega/\omega_{ci})^2) \gtrsim T_e/m_e \quad (6)$$

This condition means that the plasma density for fixed T_e is limited to

$$n \lesssim \frac{B^2}{4\pi T_e} \frac{m_e}{m_i} (1 - (\frac{\omega}{\omega_c})^2) \quad (7)$$

Figure 2-9. KX^{**2} versus KZ^{**2} .

and more generally the plasma β is restricted to

$$\beta \lesssim (m_e/m_i) \cdot (1 - (\omega/\omega_{ci})^2) \cdot (T_e + T_i)/T_e \quad (8)$$

These restrictions do not have sharp boundaries. Landau damping can still be effective when the electron velocity does not correspond to the point of greatest slope in the electron distribution function, but then increased power levels are required.

B Field Scaling

The strength of the DC magnetic field enters in several different ways. We must consider the scaling of startup, stability, RF electric fields, and the scaling of the phase velocity of the slow waves and hence electron heating by Landau damping.

Startup

The initial plasma density that can be achieved is determined by the cutoff of the electron cyclotron waves. This scales with B^2 so it is likely that the initial density that can be achieved in ICTF will be as high as $6 \times 10^{18} \text{ m}^{-3}$, and comparable to the steady state operating density. This suggests that fundamental ECH could also play a role in providing steady state electron heating in ICTF. When radial ponderomotive force was present in Phaedrus single central cell experiments startup of a 300 liter plasma was achieved with a minimum of 1.5 kW of coupled input ECH power (and reliably achieved with 5 kW).

Stability

ICTF will be operated with a minimum of two ICRF antennas, one operated at $0.8 \omega_{ci}$ and the other at $1.01 \omega_{ci}$. The former will provide electron heating and ion heating while the latter will provide radial ponderomotive force and ion heating. We can estimate the RF electric field strength required for ponderomotive force stability by rewriting Eq. (4)

$$E_+ \left(\frac{V}{cm} \right) > 0.4 \sqrt{\frac{\ell_r B(\text{gauss})}{R_c \ell_z}} T_i^{3/4} (\text{eV}) \quad (9)$$

with all lengths in cm. For Phaedrus SAM operation this gives $E \gtrsim 2.5$ V/cm while for ICTF this predicts $E > 6$ V/cm. The antenna requirements can be estimated from a code XANTENA⁽¹¹⁾ which calculates the details of the ICRF fields for uniform DC magnetic fields with fairly realistic density profiles. For the ICTF configuration and one set of possible plasma parameters, the results of this code are given in Figs. 2-10 - 2-11. Fig. 2-10a gives results for an assumed radial density profile with a flat $T_e = 35$ eV profile and a flat $T_i = 200$ eV profile and $\omega = \omega_{ci}$. The radial profile for E_z and E_+ (the left circularly polarized field rotating in the ion direction) are shown in Figs. 2-10b and 2-10c. Power absorbed by ions (species 1) and electrons (species 2) are shown in Figs. 2-10d and 2-10e. The B field is taken to be 2 kgauss and a full turn antenna with 1000 A is chosen. Note that the predicted E_+ field is approximately 2.5 V/cm near the edge of the plasma and close to the value predicted to be needed. Since the radius of curvature R_c or the flatness of the B field ℓ_z are not yet determined, this is close enough given the uncertainties in the theory. In any case, higher E_+ can be obtained using more antenna current.

Figure 2-11 gives results of a frequency scan at $z < 15$ cm from the antenna and $r = 30$ cm for the same profile shown in Fig. 2-10. Note that the peak of the power absorbed by the electrons comes at $\omega/\omega_{ci} \approx 0.77$. This sets the frequency for the electron heating antenna to be 2.3 MHz. Note that significant ion heating is predicted when $\omega \gtrsim \omega_{ci}$ (i.e. by the second antenna). Data for Phaedrus parameters are given in Fig. 2-12.

In addition to radial ponderomotive force there may be two other contributors to MHD stability. These are partial line tying to the end walls and ECH enhancement of the stability. Experiments on Phaedrus plasmas⁽¹²⁾ in the presence of stream gun plasma have shown that partial line tying can make a significant contribution to stability. In addition, recent startup experiments have found enhanced MHD stability when the startup ECH was left on during the shot. It was found that the β limit, i.e. the ratio of the central cell to end cell β at the stability boundary, increased from approximately 2 to 40 when the stream guns were on. The key parameter is the plasma density near the end wall, which was comparable or greater than the central cell den-

Figure 2-10. ICTF parameters anticipated in ICTF as a function of radius.

Figure 2-10 (Cont'd). ICTF parameters anticipated in ICTF as a function of radius.

Figure 2-11. ICRF parameters as a function of frequency for $r = 30$ cm and $Z \leq 15$ cm.

Figure 2-12. ICRF fields as a function of radius.

sity with the gun on and approximately one tenth the central cell density with the gun off. Since the end wall plasma density in ICRF will be comparable to the central cell density, this is likely to be an important effect. At the present time the ECH stability enhancement is not understood so it is difficult to scale this effect. However it is likely that both ECH and line tying will enhance the stability and reduce the requirements on the stabilizing RF field.

Scaling of the Ambipolar Potential

The ion confinement time τ_i can be written as the sum of a "Pastukov" (13) time and a "collisional" time (corresponding to a filled loss cone). (14)

$$\tau_i = \tau_{Pas} + \tau_{Col} \quad (10)$$

$$\text{where } \tau_{Pas} = \tau_{ii} \log_{10} R_{eff} = 7 \times 10^{11} \frac{T_i(\text{eV})^{3/2}}{n(m^{-3})} [\log_{10} R_{eff}] \sqrt{\frac{m_i}{m_p}}$$

$$\text{where } R_{eff} = \frac{R_T}{1 + e\phi/E_i}$$

and $\tau_{Col} = 2L_c \left(\frac{m_i}{8T_i} \right)^{1/2} R_T = \frac{1.25 \times 10^{-4}}{\sqrt{T_i(\text{eV})}} L_c R_T \sqrt{\frac{m_i}{m_p}}$. R_T is the mirror ratio (midplane to mirror throat). At low ion temperature and high density the collisional term dominates (e.g. the region of RF sustained operation on Phaedrus). This is also the case for ICTF. Note that while the collisional term scales as $(T_i)^{-1/2}$, the Pastukov term scales like $(T_i)^{3/2}/n$. The electron confinement time can be written:

$$\tau_e = \tau_{ee} \frac{e\phi}{T_e} e^{e\phi/T_e} = 2.1 \times 10^{10} \frac{T_e(\text{eV})^{3/2}}{n(m^{-3})} \frac{e\phi}{T_e} e^{e\phi/T_e} \quad (11)$$

The ambipolar potential ϕ is determined by balancing the electron and ion loss flux out the ends of the device. Since the loss fluxes are given by nV_c/τ , the flux balance is accomplished by equating the electron and ion confinement times. This balance gives:

$$\left\{ \frac{7 \times 10^{11} T_i^{3/2} \log\left(\frac{R_T}{1 + e\phi/E_i}\right)}{n} + \frac{1.25 \times 10^{-4} L_c R_T}{\sqrt{T_i}} \right\} \sqrt{\frac{m_i}{m_p}} \\ = \frac{2.1 \times 10^{10} T_e^{3/2}}{n} \frac{e\phi}{T_e} e^{e\phi/T_e} \quad (12)$$

Typically, values of $e\phi/T_e$ can range from as much as 6 to 3.

We can estimate how much of the potential drop occurs in the end wall region by noting that the plasma must be quasineutral in the uniform B field region near (but still many Debye lengths) from the end wall. Outside of the mirror throat the electron density is given by the Boltzmann relation $n = n_T e^{e(\phi - \phi_T)/T_e}$ where n_T and ϕ_T are the electron density and potential at the mirror throat. The ion density is determined by a combination of flux conservation and decreasing B field to be

$$n_i = \frac{\frac{B}{B_T} \frac{n_0}{2 \tau_i} \frac{V_c}{\pi r_T^2}}{\sqrt{2/m T_0 [R_T - R + e(\phi_T - \phi)/T_0]}} \quad (13)$$

where the subscript T refers to the mirror throat and mirror ratios are with respect to the mirror midplane B field. V_c is the mirror cell volume, and πr_T^2 is the area of the mirror throat. Quasineutrality gives:

$$\frac{B}{B_T} \frac{V_c/(\pi r_T^2)}{2 \tau_i} \frac{n_0}{n_T} \frac{1}{\sqrt{2/m T_0 [R_T - R + e(\phi_T - \phi)/T_0]}} = e^{-(\phi_T - \phi)/T_e} \quad (14)$$

These equations can be solved once the ratio n_0/n_T is given.

For collisional mirror plasmas $n_T = n_0$ but for the general case it is more complicated.

In the Pastukov case it can be shown that

$$v_{T||} \sim \left(\frac{v_{ii}}{\omega_b}\right)^{1/4} v_{0||} \quad (15)$$

where $v_{ii} = \frac{1}{\tau_{ii}}$.

Flux conservation gives

$$\frac{n_o v}{2 \tau_{ii}} = n_T v_{Ti} A_T \quad (16)$$

where V is the central cell volume and A_T is the throat loss area so

$$\frac{n_T}{n_o} \sim \frac{v}{A_T} \frac{v_{ii}}{2 v_{Ti}} \sim \frac{v}{A_T v_{Ti}} \frac{v_{ii}}{2} \left(\frac{\omega_b}{v_{ii}} \right)^{1/4} \sim \left(\frac{v_{ii}}{\omega_b} \right)^{3/4} \quad (17)$$

Outside the throat ion flux conservation gives

$$\frac{n}{n_o} = \frac{\frac{B V_c}{2 B_T \tau_i A_T}}{\sqrt{\frac{2}{n} T_o \left[R_T - R + e \left(\frac{\phi_T - \phi}{T_o} \right) \right]}} \quad (18)$$

Assuming Boltzmann electrons we can equate ion and electron density to give:

$$\frac{.075 \left(\frac{L}{1.20} \right) \frac{n_o}{n_T}}{\sqrt{T_o (eV)} \sqrt{5 + \frac{e \Delta \phi}{T_o}}} = e^{-\frac{e \Delta \phi}{T_o} \frac{T_o}{T_e}} \quad (19)$$

where L is the mirror cell length in m. For $T_o = 200$ eV, $T_e = 30$ eV

$$\frac{5.3 \times 10^{-3} \frac{n_o}{n_T}}{\sqrt{5 + \frac{e \Delta \phi}{T_o}}} = e^{-6.7 \frac{e \Delta \phi}{T_o}} \quad (20)$$

For the collisional case $\frac{n_T}{n_o} \approx 1$ while for the Pastukov case

$$\frac{n_T}{n_o} \sim \left(\frac{v_{ii}}{\omega_b} \right)^{3/4} \sim 1.5 \times 10^{-2} \quad (21)$$

For the collisional case $\frac{n_T}{n_o} = 1$ so $\frac{e \Delta \phi}{T_o} \sim 0.9$ and the ambipolar potential drop outside the mirror throat is $\Delta \phi \sim 180$ V.

For the Pastukov case $\frac{n_T}{n_o} = 1.5 \times 10^{-2}$ we get $\frac{e\Delta\phi}{T_o} \sim 0.28$, $\Delta\phi \sim 56$ V.

We must also consider the potential drop that occurs in the wall sheath. At the wall

$$\frac{n_o}{2\tau_i} \frac{V_c}{B_T A_T} = \frac{n_w}{2} \sqrt{\frac{\pi}{2} \frac{T_e}{m_e}} e^{\frac{-e\phi_w}{T_e}} \quad (22)$$

$$\frac{n_w}{n_T} = e^{\frac{-e(\phi_T - \phi_w)}{T_e}} \quad (23)$$

so

$$\frac{n_o}{2\tau_i} \frac{B}{B_T} \frac{V_c}{A_T} = \frac{n_T e^{\frac{-e(\phi_T - \phi_w)}{T_e}}}{n_o} \frac{1}{2} \sqrt{\frac{\pi}{2} T_e (\text{eV})} 4.2 \times 10^7 e^{\frac{-e\phi_w}{T_e}}. \quad (24)$$

For example, assuming $\tau_i = 7 \times 10^{-4}$ sec gives

$$8.9 \times 10^{-4} = \frac{n_T}{n_o} e^{\frac{-e(\phi_T - \phi_w)}{T_e}} e^{\frac{-e\phi_w}{T_e}} \quad (25)$$

for $\frac{n_T}{n_o} \sim 1$, $\Delta\phi \sim 180$ V so $\frac{\Delta\phi}{T_e} \sim 6$ and $\frac{\phi_w}{T_e} \sim 1.02$ or $\phi_w \sim 31$ V

for $\frac{n_T}{n_o} \sim 1.5 \times 10^{-2}$ $\Delta\phi \sim 56$ V 20 $\frac{\phi_w}{T_e} \sim 0.95$, so $\phi_w \sim 29$ V,

where, for illustration, we considered $T_e = 30$ eV.

End Loss Current

The confinement time also determines the end loss current density which can be written:

$$j_w = \frac{n_o}{2\tau_i} \frac{V_c}{B_T A_T} \frac{B_w}{B_T} \quad (26)$$

Power Balance

The power balance is dominated by RF, electron-ion collisions, and end loss. The electron power balance can be written:

$$\frac{P_e}{V_c} + n v_{ei} \frac{3}{2} (T_i - T_e) = \frac{n^2}{(n \tau_e)} (\phi + T_e) \quad (27)$$

The first term on the left hand side gives the direct power input from the ICRF while the second gives the power input from electron ion collisions. The right hand side gives the end loss power, where ϕ is the total, $\phi_c + \phi_w + \phi_s$, from the central cell, test cell, and sheath. The ion power balance can be written:

$$\frac{P_i}{V_c} = \frac{n^2}{(n \tau_i)} (E_i) + n v_{ei} \frac{3}{2} (T_i - T_e) + n n_o \langle \sigma v \rangle_{cx} \frac{3}{2} T_i \quad (28)$$

The left hand side gives the RF power input. The first term on the right gives the ion end losses, the second gives the losses due to electron-ion collisions, and the last term gives losses from charge exchange with neutral gas. For the ion energy we take

$$E_i = \frac{(2\tau_{Col} + \tau_{Pas})}{(\tau_{Col} + \tau_{Pas})} T_i \quad (29)$$

2.2 ICTF Conceptual Design Parameters

Based on the required plasma characteristics (Table 1-1) and on scaling from the Phaedrus experiment, a conceptual ICTF design has been developed.

Figure 2-13 shows the coil geometry and the resulting magnetic field for this reference conceptual design. Design parameters for this case are shown in Table 2-1.

To estimate the plasma performance in the end cells, we specialize the plasma equations to deuterium ($m_i = 2 m_p$), with $L = 2.0$ m and $R_T = 10.32$; all formulas are MKS and eV:

Table 2-1. ICTF Conceptual Design Parameters

Item	Variable	Value
Center cell volume	V_c	$\approx 0.574 \text{ m}^3$
Center cell length	L_c	2.0 m
Center cell flux radius	r_c	0.302 m
Center cell field	B_c	0.219 T
Center cell area	$A_c = \pi r_c^2$	0.287 m^2
Mirror throat flux radius	r_T	0.0940 m
Mirror throat field	B_T	2.26 T
Mirror ratio in throat	$R_T = B_T/B_0$	10.32
Test region radius	r_w	0.100 m
Test region field	B_w	1.98 T
Test region area	$A_w = \pi r_w^2$	0.0314 m^2
Mirror ratio, test region	$R_w = B_w/B_0$	9.04

Figure 2-13. Flux plot and field strength.

$$\tau_{\text{Col}} = 3.648 \times 10^{-3} T_i^{-1/2}$$

$$\tau_{\text{Pas}} = 9.898 \times 10^{11} \frac{T_i^{3/2}}{n_c} \log \left[\frac{10.32}{1 + \frac{2}{3} \frac{T_e}{T_i} x_c} \right] \quad (30)$$

$$\tau_e = 2.1 \times 10^{10} \frac{T_e^{3/2}}{n_c} x_c e^{x_c}.$$

For illustration, consider central cell values of $T_e = 30$ eV, $T_i = 100$ eV, and $n_c = 8 \times 10^{18} \text{ m}^{-3}$. We find $x_c \equiv e\phi_c/T_e = 5.29$ with an ambipolar (central cell) potential of

$$e\phi_c = 159 \text{ eV} \quad (31)$$

and $\tau = \tau_e = 0.452 \text{ ms}$, $\tau_{\text{Pas}} = 0.087 \text{ ms}$, and $\tau_{\text{Col}} = 0.365 \text{ ms}$. The mirror is collisional.

Assuming a symmetric field geometry, losses out both ends are equal:

$$I_w = \frac{en_o V_c}{2 \tau} = 814 \text{ A} \quad (32)$$

$$j_w = I_w/A_w = \frac{1.618 \times 10^{23}}{\text{m}^2 \text{ s}}.$$

The end cell potential near the test plate, where $B \approx B_w$ is given, for a collisional ($n_T \approx n_o$), by

$$n_o e^{-x_w} = \frac{j_w \sqrt{m_i/m_p} \times 7.22 \times 10^{-5}}{\sqrt{T_i} R_T - R_w + \left(\frac{T_e}{T_i} x_w\right)^{1/2}} \quad (33)$$

where the potential drop, ϕ_w , from the throat to the test region, enters as $x_w \equiv e\phi_w/T_e$. Solving for our illustration we obtain

$$e\phi_w = 57 \text{ eV}. \quad (34)$$

The ion density approaching the test plate is

$$n_w = n_T e^{-x_w} = n_o e^{-x_w}; \quad (35)$$

for our example we have $n_w = 1.22 \times 10^{18} \text{ m}^{-3}$.

The sheath potential, ϕ_s , follows from

$$j_w = 2.63 \times 10^5 n_w e^{-x_s} \sqrt{T_e} \quad (36)$$

where $x_s \equiv e\phi_s/T_e$. In our example, we find $e\phi_s = 72 \text{ eV}$.

The required rf heating power density, p_i , to the central cell ions is given by

$$p_i = n_o [E_i/\tau_i + v_{ei} (T_i - T_e)], \quad (37)$$

with $E_i/\tau_i = (2 \tau_{\text{Col}} + \tau_{\text{Pas}}) \times 1.6 \times 10^{-19} T_i/\tau_i^2$ and $v_{ei} (T_i - T_e) = 1.219 \times 10^{-32} n_o (T_i - T_e)/T_e^{3/2}$. The rf heating power density to the central electrons is $p_e = n_o [(T_e + e \{\phi_c + \phi_w + \phi_s\})/\tau - v_{ei} (T_i - T_e)]$. We find for our example

$$p_i = 845 \text{ kW/m}^3 \text{ and } p_e = 566 \text{ kW/m}^3. \quad (38)$$

So a total rf power $P_{\text{rf}} = 810 \text{ kW}$ is needed, with 485 kW going to deuterons and 325 kW to electrons. The heat flux out each end is $W = P_{\text{rf}}/2A_w$, which is 12.9 MW/m^2 for the case cited. Charge exchange is a negligible power loss from the central cell because of the high plasma density and large size, $r_c \approx 0.3 \text{ m}$. Electron thermal conductivity along the field lines is small provided the electron-electron mean free path for energy exchange, $\lambda_u^{(e)}$, is much larger than the machine dimensions. Assuming a secondary plasma is formed at the test plate with density n_f and temperature $T_f < T_e$, we can compute the mean free path as

$$\lambda_u^{(e)} = \sqrt{2 T_e / m_e} v_u^{(e)} \quad (39)$$

$$= \frac{1.09 \times 10^{16} T_e^2}{n_f [1 - 0.34 (T_f / T_e)]}$$

For $n_f = 3 \times 10^{18} \text{ m}^{-3}$ and $T_f = \frac{1}{2} T_e$ we find for our example $\lambda_u^{(e)} = 3.9 \text{ m}$. Additional operating points are suggested by Table 2-2. We note that there are many independent variables available, which permit a wide range of operation of the plasma source.

Table 2-2. ICTF - Symmetric Operation Deuterium
 $V_c = .574 \text{ m}^3$, $L_c = 2.0 \text{ m}$, $B_c = 0.219 \text{ T}$;
 $A_w = 0.0314 \text{ m}^2$, $r_w = 0.10 \text{ m}$; $n_f = 3 \times 10^{18} \text{ m}^{-3}$, $T_f = \frac{1}{2} T_e$

$B_w(\text{T})$	$B_T(\text{T})$	$n_c(10^{18} \text{ m}^{-3})$	$T_i(\text{eV})$	$T_e(\text{eV})$	$\phi_c(\text{V})$	$\phi_w(\text{V})$	$\phi_s(\text{V})$	$n_w(10^{18} \text{ m}^{-3})$	$\tau(10^{-4} \text{ s})$	$j_w(10^{23} \text{ m}^{-2} \text{ s})$	$I_w(\text{A})$	$P_i(\text{kW})$	$P_e(\text{kW})$	$P_{rf}(\text{kW})$	$W(\text{MW/m}^2)$	$\lambda_u^{(e)}(\text{m})$
1.98	2.26	8	100	50	232	98	127	1.13	4.40	1.66	836	369	784	1150	18.4	10.9
1.98	2.26	8	100	30	159	57	72	1.22	4.52	1.62	814	485	325	810	12.9	3.9
1.98	2.26	7	100	50	227	99	127	0.96	4.52	1.42	712	306	668	974	15.5	10.9
1.98	2.26	6	100	50	222	96	132	0.88	4.67	1.17	591	246	555	801	12.8	10.9
2.00	2.20	8	100	50	231	92	132	1.28	4.29	1.70	857	377	803	1180	18.8	10.0
2.00	2.20	6	100	50	221	95	132	0.91	4.55	1.21	606	252	568	820	13.1	10.9
2.00	2.20	6	100	40	188	74	103	0.94	4.62	1.19	597	271	424	695	11.1	7.0
2.00	2.20	6	100	35	171	64	89	0.96	4.67	1.17	591	287	345	632	10.1	5.4

3. EROSION/REDEPOSITION MODELING

The erosion and redeposition of sputtered material from the surface of a limiter, divertor, or similar device in mirrors (e.g. halo scraper) is one of the most important phenomena expected in fusion devices. Erosion/redeposition would be extensively studied in the ICTF.

As previously discussed, a key goal of the ICTF design is to have as close a simulation as possible of conditions present in the scrapeoff layer of a tokamak or mirror. In particular, a high magnetic field (~ 4 T), and hot electrons ($T_e \sim 50$ eV) are important in the redeposition process. In order to assess the relevance of ICTF for erosion/redeposition purposes, an analysis has been performed of erosion/redeposition on a rectangular plate exposed to the streaming plasma in an end cell of the ICTF. Typical parameters characteristic of the ICTF end cell have been used. The analysis used the Degas Monte Carlo Code to compute the hydrogen charge-exchange (CX) flux incident on the plate, and the resulting CX caused erosion, and the REDEP4 Code to compute the total erosion, impurity transport within the end cell plasma, and the redeposition and net growth rate. Four coating materials were analyzed, beryllium, carbon, vanadium, and tungsten. These represent the range of low-Z, medium-Z, and high-Z materials, most commonly proposed for use as coating materials. The Degas Code is two-dimensional, but was adapted to simulate the three-dimensional geometry of ICTF. The REDEP4 Code is fully three-dimensional.

The experimental geometry is shown in Fig. 3-1. A flat rectangular plate with dimensions 80 x 40 cm is oriented at a 15-degree angle to the magnetic field in one of the end cells of the ICTF. As discussed in Sec. 2, the field in the end cell is designed to be fairly uniform over the test cell region, and has a design value of 4.0 T. The plate geometry simulates rather closely the geometry to be used in future divertor and limiter designs. In these future designs it is necessary to orient the impurity control surface at a shallow angle to the field, in order to keep the heat load to a reasonable value. In a tokamak, the field line structure consists of toroidal and poloidal fields. For the purposes of ICTF, however, it is only necessary to employ a single component magnetic field since this simulates the net field in a tokamak.

Figure 3-1. ICTF target plate geometry.

The plasma parameters used for the erosion/redeposition calculations were derived from the previously described calculations. The main parameters used were: $n_{e_0} = 3 \times 10^{18} \text{ m}^{-3}$, $T_{e_0} = 50 \text{ eV}$, $T_{i_0} = 100 \text{ eV}$, and $\phi = 2 T_e$. Temperature and density were taken as constant within a circular area of radius 20 cm, and decreased linearly outside of this area. In some cases, slightly different parameters were used for the DEGAS Code calculations due to code format. These parameters are typical of those expected to be obtained in ICTF. A key feature of the ICTF end cell is that the 4-T superconducting end cell magnets have a design diameter of 0.5 M and are located outside the vacuum vessel. There is therefore an adequate amount of room and flexibility in the area near the test plate to control the amount of pumping. The plasma parameters used at present would correspond to a case of relatively high vacuum pumping in the region surrounding the test plate, and relatively low-particle recycling. As discussed later, a low-pumping condition would be used to experiment with a high recycling regime.

The exact form of spatial distributions of plasma parameters used for the present calculations are not critical for the erosion/redeposition experiment. An advantage of the steady-state operation of ICTF, besides being necessary to obtain reasonable fluences, is that accurate and detailed diagnostic movements can be obtained of the end cell plasma. These in turn can be employed in the various codes to test their predictive power.

A prediction of the charge-exchange erosion of the plate requires detailed information concerning the neutral distribution function in the vicinity of the plate. For this purpose, use was made of the Monte Carlo neutral transport code DEGAS,⁽¹⁵⁾ which has been applied to a variety of problems involving the edge region of tokamaks.⁽¹⁶⁾ The code incorporates the relevant interaction processes between neutrals and the plasma, such as charge-exchange, ionization, and dissociation. Before striking the plate, incident ions are accelerated through a sheath potential. Reflections from the plate are treated according to the data of Eckstein and Verbeek.⁽¹⁷⁾ Neutrals implanted in the plate are assumed to diffuse to the surface and to desorb as wall-temperature molecules. The sputtering yield from the plate is calculated by the subroutine DISPUT⁽¹⁸⁾ within DEGAS. One of the applications of the ICTF experiment would be to test such wall models.

Since the code is limited to a two-dimensional spatial geometry (but with velocities in three dimensions), the plate was modeled as though it extended indefinitely in the horizontal direction, perpendicular to the incoming plasma. Such models have frequently been employed with the code⁽¹⁹⁾, and the corrections and limitations are reasonably well understood. Reflections from the walls of the surrounding vessel were ignored.

The resulting erosion rates due to neutrals on various plate segments are shown in Fig. 3-2. The solid curve is the DEGAS results as quantized into ten plate sections; the dotted curve is interpolated through the sections. As the figure indicates, the greatest erosion occurs for beryllium, for which the maximum rate is ~ 11 cm/yr. The carbon erosion rate on any particular segment tends to be slightly more than half of the beryllium rate, and the vanadium erosion rate is about one-half that for carbon. The rates for tungsten are two orders of magnitude less than those for vanadium. For each material the erosion rate peaks at about one-third of the distance across the plate, as measured from the leading edge. The maximum rate is about ten times higher than the minimum rate which occurs at the trailing edge.

The erosion rates due to ion impact can be calculated directly from DSPUT, without regard to the neutrals. For carbon, beryllium, and vanadium, these rates are four to five times higher than the above rates, while for tungsten the ratio is nearly eight. If the sputtering yield were independent of the energy and angle of the incident deuteron, the ratio would be expected to be about four, since the neutral flux to the plate turns out to be about 25% of the ion flux.

The results of the charge-exchange calculations, i.e. Fig. 3-2, were incorporated as input into the REDEP4 Code. This code is the latest, three-dimensional version of the REDEP Code⁽²⁰⁾ developed to study erosion/redeposition in tokamak divertors and limiters, and mirror halo scrapers. The REDEP Code contains models of sputtering by hydrogen, helium, and self-sputtering, and computes the transport, ionization, and redeposition of sputtered neutrals in detail.

Figure 3-3 shows the gross erosion rate of a carbon-coated test plate. The rate is shown in cm/yr on the basis of steady-state operation. The gross erosion rate is defined as the erosion rate due to neutral and charged hydrogen sputtering, and self-sputtering, without taking redeposition into

Figure 3-2. Charge exchange erosion of a rectangular plate in ICTF. Square curve is DEGAS code output. Dotted line is smooth curve.

Figure 3-3. Carbon erosion in ICTF.

account. The rate shown includes the effect of the addition of 1% carbon atoms into the deuterium stream in the end cell. This addition of carbon is easily achieved by dropping small pellets into the streaming end cell plasma and represents an important experimental variable. Fig. 3-4 shows the net growth rate for the carbon plate. The net growth rate is the difference between the redeposition rate and the gross erosion rate. A negative value in Fig. 3-4 thus denotes net erosion, with a positive value denoting growth. The results for carbon shows a wide range of both gross and net growth rates and should thus provide very relevant information for devices such as TFCX and INTOR. The gross rates predicted (Fig. 3-4) are similar in magnitude to those predicted in the "medium" edge temperature regime of INTOR. The net growth rates of from -12 to +5 cm/yr would permit experimental analysis of the properties of the redeposited material under a wide range of growth rates. Results for a vanadium-covered surface are shown in Figs. 3-5 and 3-6. For the case shown, vanadium was added to the hydrogen stream. All redeposited material arises initially from the plate.

As with carbon, the gross erosion rate for vanadium is fairly flat over most of the rectangular surface, but falls off near the plate boundaries where the deuterium flux falls off. The net growth rate for this case varies from slightly positive to ~ 8 cm/yr. The rate can be controlled by adding material to the hydrogen stream.

A variety of computer runs were performed for other materials and a range of edge temperatures and densities. The results for beryllium are fairly similar to carbon and vanadium. For somewhat higher temperatures, vanadium enters an unstable regime where self-sputtering coefficients of the redeposited material approach unit. The resulting sputtering rates increase drastically. The same result was found for tungsten. Thus, it appears that the predicted plasma temperature limitations of non-low-Z materials could be thoroughly tested in ICTF.

Figure 3-4. Carbon erosion in ICF.

Figure 3-5. Vanadium erosion in ICTF.

Figure 3-6. Vanadium erosion in ICTF.

4. EXPERIMENTAL PROGRAM

4.1 Introduction

The need for an ICTF is related to its ability to aid in the study of critical issues identified for impurity control. Before describing the experiments that could be performed in the ICTF, the critical issues will be briefly reviewed.

For near term devices, like TFCX, the most important issues are whether the impurity control system will adequately remove impurities and helium from the plasma and whether the system itself will be a significant source of impurities. The first issue involves the pumping capabilities of the system, and the latter issue involves the erosion and redeposition behavior of the plasma side material. At present, a pumped limiter is being considered for impurity control but recent calculations indicate that it may be difficult to get adequate pumping with the limiter.⁽²¹⁾ Therefore, a poloidal divertor may also be considered for impurity control. The pumping capabilities of limiters and divertors have not been tested under prototypical operating conditions. The concentration of impurities in the plasma will depend to a great extent on the recycling behavior of sputtered particles within the plasma edge and the degree to which these properties redeposit upon plasma side surfaces. Calculations indicate that recycling and redeposition processes will significantly reduce the level of impurities within the plasma,⁽²²⁾ but there are no experimental data available to provide verification of the models.

For long term devices, like DEMO or a commercial reactor, the major issues are related to the lifetime and reliability of the impurity control system. Not only must the system be capable of controlling impurities, but it must also have a lifetime that makes it economically attractive in high availability devices. The specific lifetime issues are the erosion rates of plasma side materials and the impact of radiation damage on the material properties. The ICTF would be utilized to examine the lifetime erosion issues. The erosion/redeposition behavior is the key area that affects the net erosion rate. In addition, the properties of redeposited material, which could build up to significant thicknesses, are of particular interest for determining the lifetime of plasma side components.

The experiments to be performed in the ICTF can be divided into two classes: pumping related experiments and materials related experiments. The pumping experiments would address many of the near term issues, while the materials experiments would address both the near term and long term issues. The experiments would provide information that could either be used directly to qualify specific impurity control concepts or which could be used to verify computer codes that would in turn be used to design and analyze future impurity control systems. The ICTF must provide an environment which simulates the plasma edge conditions of future devices and also has diagnostic capabilities that provide proper characterization of the experimental conditions. The remainder of this section will describe specific experiments and diagnostic needs for the ICTF. Some specific tests anticipated for the facility are:

- (1) Testing of erosion/redeposition properties of candidate surface materials under high fluence conditions with plasma conditions approximating those of a real fusion reactor.
- (2) Testing of the physics of high recycling -- low temperature/high density regimes which are being predicted to be theoretically obtainable for divertor channels.
- (3) Testing of the particle pumping efficiencies of limiter and divertor designs.
- (4) Determination of the operating limits for high-Z materials. Present models indicate that high-Z materials (e.g., tungsten, tantalum) are limited in use to plasma edge temperatures ≤ 50 eV because of self-sputtering considerations. The point at which self-sputtering exceeds unity under reactor conditions is important in determining the upper energy limit for high-Z materials.
- (5) Evaluation of the sputtering and redeposition behavior of alloys and compounds. Multi-component systems will behave differently from single elements when exposed to the plasma edge. The modeling of these systems is complex, and therefore experimental information is needed. New concepts such as developing self-sustaining, erosion-resistant low-Z surfaces will be tested.

- (6) Testing of self-pumped helium removal concept (using vanadium, molybdenum, niobium, or other metals to trap helium in a continuously deposited surface layer).
- (7) Hydrogen (tritium) retention and permeation.

4.2 Pumping and Recycling

Experiment: Limiter/Divertor Channel Behavior

Objectives: Determine pumping efficiency of pumped limiter channels and divertors.

Study recycling behavior of neutralized particles and possible decrease in plasma temperatures and increase in plasma densities at the neutralizer plates.

Calibrate computer codes for plasma behavior in limiter and divertor channels.

Test Environment:

Limiters

In general, the environment in ICTF is expected to cover the projected conditions of the limiter channels in TFCX and INTOR, as shown in Table 4-1. The particle flux is potentially very high in ICTF, which will be useful in examining the possibility of "choking" in the channel at high gas pressures. Particle energies and channel geometry are expected to be similar to those in TFCX or INTOR. However, the channel width is limited by the dimensions of the end cell magnets (~ 0.5 m), in contrast with the continuous opening of a toroidal belt limiter. Also, the end cell magnetic field may be somewhat reduced from the values expected in TFCX and INTOR.

Important parameters for the limiter pumping are the incoming particle fluxes and energies, the limiter geometry, duct conductance and pumping speed, and magnetic field strength and orientation. It is anticipated that ICTF will have the capability to vary all of these parameters. Two possible configura-

Table 4-1. Operating Environment for the Limiter Channel Experiment

Parameter	Value		
	ICTF	TFCX	INTOR
Plasma Flux ($s^{-1} m^{-2}$)	$\lesssim 1.3 \times 10^{23}$	9×10^{20}	6×10^{20}
Ion Energy (eV)	< 150 eV	< 50 eV	< 50 eV
Electron Energy (eV)	~ 50 eV	< 50 eV	< 50 eV
Channel Height (cm)	5 - 15	~ 10	10
Channel Width (cm)	~ 40	∞	∞
Channel Length (cm)	40 - 80	~ 100	75
Magnetic Field (T)	4	~ 4.5	5.5

tions for the limiter geometry are illustrated in Fig. 4-1. Fig. 4-1a is a configuration similar to that proposed for TFCX where plasma particles are neutralized on the back side of the limiter blade at a considerable distance from the duct. Fig. 4-1b shows another geometry where the particles are neutralized at a plate located adjacent to the pumping duct. The neutralizer plate is placed to maximize the ballistic scattering of neutrals into the duct. Other important geometric parameters are the channel height to length ratio and the channel width. The channel width should be greater than the mean free path for the particles so that interactions with the channel side walls do not become the dominant source for neutral particles. The particle fluxes and energies will have a significant effect on the pumping behavior, and it is desirable to be able to examine the pumping over a range of fluxes and energies. It appears that by adjusting the RF power input and particle density in the central cell, these parameters can be adjusted in ICTF. The duct conductance and pumping speed can be adjusted by placing valves within the pumping duct.

Divertors

Test environment considerations for the divertor channel are similar to those for the limiter. Table 4-2 shows that the environment in ICTF is expected to cover the projected conditions in the INTOR divertor. Channel width is again limited by the dimensions of the magnet in the end cell. A possible configuration for the divertor channel is illustrated in Fig. 4-2. Particles enter the channel and strike the collector plate where they are neutralized and recycled.

Information to be Obtained:

Data that would be obtained from this experiment are neutral density as a function of position in the channel, the percent of incoming particles that are pumped into the vacuum system, the backflow of particles out of the channel, the pressure within the vacuum pump duct, particle energies as a function of position in the channel and duct, and impurity levels within the channel and duct. This data would provide information concerning the recycling behavior in the channel and the reflection coefficients for low energy particles.

Figure 4-1. Limiter channel configurations.

Figure 4-2. Divertor channel geometry.

Table 4-2. Divertor Channel Parameters

Parameter	INTOR	ICTF
Scrape-off Temperature (eV)	120	50 (electrons) 150 (ions)
Scrape-off Plasma Density (m^{-3})	5×10^{19}	3×10^{19}
Ion Flux Coating Divertor ($\text{s}^{-1} \text{m}^{-2}$)	1.5×10^{22}	3×10^{22}
Channel Length	60 cm	50 cm
Channel Width	> 1 m	30 - 40 cm
Angle of Collector Plates with Field Lines	15 - 30°	15 - 90°

4.3 Erosion/Redeposition

Experiment: Erosion/Redeposition

Objectives: Measure net erosion rates on exposed surfaces.

Determine impurity behavior in plasma region above specimen surface.

Calibrate computer codes.

Test Environment:

A flat plate specimen is placed at an angle $\lesssim 15^\circ$ with respect to the magnetic field lines, as shown in Fig. 4-3. The plasma conditions are close to those anticipated in the scrape-off region of future devices like TFCX and INTOR, as shown in Table 4-3, and as described above. The plate is to be actively cooled to maintain the exposed surfaces at reasonable temperatures. The surface temperatures will be adjusted so that the potential impact of chemical sputtering can be studied. A variety of materials, including Be, BeO, C, SiC, W, Mo, V, and stainless steel will be tested. The materials will be exposed to different plasma environments ranging from the low edge temperature conditions ($\lesssim 30$ eV) to medium edge temperature conditions ($\lesssim 200$ eV). The plasma density will also be varied. The tests will be conducted for extended periods (> 1 h CW) so that sufficient material is eroded for accurate measurements and for comparison with computer codes. Following tests in ICTF, the specimens will be examined in other instruments, to determine microstructural changes in the surface.

Information to be Obtained:

Data that would be obtained from this experiment are the neutral density as a function of position above the specimen plate, particle energies as a function of position, impurity concentration in the plasma above the plate, and net erosion rates. Post test examinations would provide information about the surface morphology of eroded and redeposited surfaces, surface composi-

Figure 4-3. ICF target plate.

Table 4-3. Plasma Conditions for Erosion/Redeposition Testing

Parameter	TFCK	INTOR	ICTF
Edge Temperature (eV)	150 (est)	20 - 600	10 - 200
Edge Density (m^{-3})	-	$2.5 \sim 5 \times 10^{19}$	$\sim 3 \times 10^{19}$
Surface Heat Flux (MW/m^2)	3.5	1 - 4	~ 2
Ion Flux to Surface ($m^{-2} s^{-1}$)	-	$\sim 2.5 \times 10^{22}$	$\sim 3 \times 10^{22}$
Neutral Flux ($m^2 s^{-1}$)	-	$\sim 1 \times 10^{22}$	-
Magnetic Field	~ 4.5	5.5	4

tion, and the microstructure of redeposited material. Comparison with computer codes would provide indirect information concerning mean free paths for sputtered particles, ionization cross sections, and interactions of charge exchange neutrals with the plasma.

4.4 Redeposited Materials

Experiment: Properties of Redeposited Materials

Objective:

Measure surface and bulk properties of redeposited materials under conditions where a large flux of gas particles is co-deposited.

Test Environment:

Test conditions would be similar to those described for the erosion/redeposition experiments. The ICTF would be utilized to build up thick layers ($\sim 1 \mu\text{m} - 1 \text{mm}$) of redeposited material which would then be examined and tested. Long term continuous operation is required to achieve reasonable thicknesses. The plasma conditions and substrate temperatures would be varied to produce different microstructures in the redeposited materials. The materials of interest are Be, BeO, C, SiC, V-15Cr-5Ti, austenitic stainless steel, W, Ta, and Mo. They would be examined in the as-deposited condition, or there would be further specimen preparation for the bulk property testing.

Information to be Obtained:

Surface

Examination in beam devices (e.g., SEM, AUGER, electron or ion microprobes) would provide information on the surface morphology and chemical composition.

Bulk Microstructure

Optical metallography, SEM or AUGER analyses would provide information on grain size, pore (bubble) distributions, and compositional variations from the

top surface. The structure at the interface between the substrate and redeposited materials would also be examined.

Bulk Properties

Thick layers of redeposited materials would be prepared to determine their physical properties (e.g. thermal conductivity, thermal expansion, specific heat) and mechanical properties (e.g. ductility, tensile strength). In addition, the adhesion to the substrate would also be tested.

This information is vital to determine if redeposition can be counted upon for extended lifetimes of impurity control components. If the properties of redeposited materials are poor and the bond is weak then the redeposited material is likely to flake off from the surface.

4.5 Limits for High-Z Materials

Experiment: Plasma Temperature Limits for High-Z Materials

Objective:

A sputtering cascade is predicted when the self-sputtering coefficient of the plasma side material exceeds unity. This experiment would determine the plasma conditions where high-Z materials cannot be utilized in fusion devices.

Test Conditions:

The test conditions would be similar to those described for erosion/redeposition experiments. A test plate with a high-Z material on the surface would be exposed to the plasma. The plasma temperature would be increased to the point where sputtered high-Z particles which are recycled to the surface produce sputtering coefficients greater than one. This condition is predicted to occur at a plasma-edge temperature of ~ 50 eV. Materials to be tested are V-15Cr-5Ti, Mo, and W.

Information to be Obtained:

The plasma temperature where a sputtering cascade occurs would be obtained. This information is important in determining if high-Z materials are

viable for the impurity control system. The plasma behavior above the test plate would also be examined. Plasma temperature and density would be measured. The impurity control in the plasma would also be monitored. The amount of power radiated from the plasma would be determined as the plasma temperature is increased.

4.6 Self-Pumping Systems

Experiment: Self-Pumping Behavior of Co-Deposited Materials

Objective:

Several materials are known to preferentially trap helium over hydrogen.⁽²³⁾ Continuous deposition of such a material on the plasma side surface could provide an alternative method for helium removal. The experiment would test the feasibility of this concept.

Test Conditions:

The test conditions would be similar to the conditions for erosion/redeposition experiments. In addition, provision would be made to inject particles of specific medium to high-Z materials into the plasma stream. The particles would be deposited on the test plate where they would provide trapping sites for the helium in the plasma. The materials of interest are V, Ta, Mo, and Ni. The plasma would be limited to temperatures of ~ 50 eV to prevent the self sputtering coefficient from exceeding unity.

Information to be Obtained:

The helium pumping of the co-deposited material would be measured as a function of plasma temperature and density along with the rate of material injection into the plasma. The adhesion and structural integrity of the co-deposited material would be determined following the test.

4.7 Tritium Permeation and Retention

Experiment: Permeation and Retention of Hydrogen through Duplex Materials

Objective:

Tritium permeation and retention in plasma side structures is recognized as a generic problem for DT devices. Measurements in prototypical test environments have not been made. This experiment would measure permeation and retention rates and compare the results with code calculations.

Test Conditions:

The plasma environment would be similar to the environment for the erosion/redeposition tests. The test plate would be constructed to allow in-situ measurements of permeation during transient and steady state conditions. The plate temperatures and the heat and particle flux would be varied in the experiments. These tests would incorporate the synergistic effects of surface implantation, thermal gradient in the structure, the duplex structure (plasma side material bonded to a heat sink), and the surface recombination kinetics at the plasma side and coolant side surfaces. Deuterium would be used as the permeating species.

Information to be Obtained:

Permeation rates into the coolant would be monitored during the tests. The retained hydrogen in the test plate would be measured following the test.

4.8 Diagnostics

The ICTF would incorporate an array of diagnostics to obtain the necessary experimental data. The types of diagnostics would include those that are presently utilized on tokamak and mirror devices plus others that would be aimed specifically at erosion/redeposition issues, pumping issues, and recycling issues. Post test examinations would make use of other diagnostic equipment. A list of possible diagnostic techniques and equipment is given in Table 4-4.

Table 4-4. Possible Diagnostics for ICTF

Category	Purpose	Examples
Plasma	Define plasma conditions in end cell	Langmuire probes Calorimeter probes Carbon probes Laser fluorescence
Thermo-mechanical	Determine specimen response to plasma	Infrared thermography Coolant calorimetry Thermocouples Strain gauges Halography
Surface	Establish erosion/redeposition and recycling behavior	Laser fluorescence Implanted tracer markers Implanted erosion markers Optical reflectometer
Vacuum	Determine pumping capability of impurity control concepts	Vacuum gauges Residual gas analyzers Tracer gases
Post Test Examination	Determine properties of surface and redeposited material	Scanning electron microscope Energy dispersive x-ray analysis Optical metallography Scanning AUGER microprobe Mechanical testing equipment

5. USE OF EXISTING DEVICES

There are several other facilities which could be considered for performing experiments for impurity control. Of the existing facilities, the ones that have the greatest potential are PISCES (UCLA), PHAEDRUS (UW), and RF test facilities such as those at PPPL and ORNL (scheduled). The plasma in PISCES is produced by a reflex arc discharge from a large area lanthanum molybdenum disc cathode to a cooled cylindrical anode,⁽²⁴⁾ whereas the other facilities are based upon mirror operation. At the present time, none of these facilities meet all the requirements of an ICTF. Except for PISCES, the main purpose of these facilities is something other than impurity control, and therefore it is questionable whether they could be considered for the experiments described in Section 4. The operating parameters of existing facilities also appear to be inadequate in certain areas, such as low electron temperature, low plasma density, small specimen size, or low duty factor. Upgrades to meet the operating needs of ICTF could be considered for existing devices. Obviously the costs and benefits of such upgrades would need to be weighed against the costs and benefits of a totally dedicated device. Additional effort will be put into this area in the near future.

6. CONCLUSIONS AND FUTURE EFFORT

The work to date indicates that a facility based upon an RF stabilized mirror will produce a plasma which closely simulates the scrape-off region of a tokamak. In addition, CW operation should be possible with such a device. The mirror facility could be used to examine many important impurity control issues including impurity removal efficiency, recycling behavior, pumping, and erosion/redeposition. The plasma environment and the specimen responses would be closely monitored with an array of diagnostic equipment, and the results could then be used directly for impurity control system design or used for verification and calibration of computer codes. The cost of the facility is estimated to be \$2-5 M.

The work for the remainder of FY 1984 will emphasize design and cost tradeoffs. Specifically, three options will be explored in greater detail.

- 1) A facility with a 20 cm diameter end cell plasma, similar to the facility described in this report.
- 2) A facility with only a 10 cm diameter end cell plasma, which is considered to be a minimum cost device that could still meet most of the objectives.
- 3) A device which would utilize the FELIX magnets and power supplies.

Conceptual designs and cost estimates will be made for the different options. The possibility of constructing a minimum cost device with the potential for future upgrades will also be considered. In addition, the possibility of making use of other facilities outside of ANL will be explored in greater detail.

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