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MASTER

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

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JSW

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

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ABSTRACT

A test facility has been designed for investigating many of the impurity control issues associated with fusion reactors. The facility is a steady-state, rf-stabilized mirror with high field and high pumping capability end cells. Analysis indicates that the ICTF should readily produce a plasma with typical parameters of $N_0 = 3 \times 10^{18} \text{ m}^{-3}$, $T_e = 50 \text{ eV}$, 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.

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. 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 exploration.

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 actual tokamak and mirror fusion devices. We believe, however, that many of the issues affecting the edge and scrape-off regions, can be examined with a

much simpler test facility. 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. 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.

A design study of such a facility (ICTF) has been performed at Argonne (ANL) and the University of Wisconsin (UW). The following sections discuss the required conditions for ICTF, the ICTF design, and the experiments that could be performed with such a facility.

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 impurity particle recycling, possibly high hydrogen recycling, near-tangential B-field-to-surface angles, strong B-field, and the presence of a significant 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 limiter and simulated divertor pumping efficiencies. Shallow-angle geometry and characteristic magnetic field strength critically affect the sheath width, electron re-emission, sputtering and reflection coefficients, and charge states of redeposited ions.

TABLE I

Operating Conditions of (1) Ideal Test Facility and (2) ICTF Design

Parameter	Ideal Facility Value	ICTF Value
1. Plasma size, each end (cm diameter)	≥ 20	20
2. Angle between the magnetic field and the test surface (deg)	5-90	5-90
3. Angle at which heat flux = 2 MW/m^2 (deg)	5	15
4. Plasma ion temperature (eV)	10-400	10-200
5. Plasma electron temperature (eV)	10-100	10-50
6. Sheath potential (T_e)	1-3	1-3
7. Plasma density, low-to-moderate recycling (m^{-3})	10^{18} - 10^{19}	10^{18} - 10^{19}
8. Magnetic field (T)	5	4
9. Angle at which hydrogen ion flux = $3 \times 10^{22} \text{ m}^{-2}\text{s}^{-1}$ at 50 eV (deg)	5	15
10. Hydrogen neutral (charge-exchange) flux (% of ion flux)	50	50
11. Helium flux (% of hydrogen flux)	5	5
12. Operation	CW	CW

Table I compares the desired operating conditions of an "Ideal" facility and the parameters for the ICTF design. The ideal parameters are based on the operating conditions expected in future tokamaks and, to some extent, mirror devices. The ICTF values are, in our opinion, a good compromise between the ideal parameters and a cost-effective device which maintains the essential features of the simulation. There are two main areas of compromise. In typical limiter and divertor designs (e.g., INTOR), the (net) magnetic field makes an angle with the surface of ~ 5 deg. This is done to limit the heat flux to a value of $\sim 2 \text{ MW/m}^2$.

Although we can test components at 5 deg in ICTF, the full 2 MW/m^2 heat flux and the corresponding particle flux will be achieved only at 15 deg. This is done to minimize the end cell size and the rf power. The second area of compromise is the magnetic field which is typically $\sim 5 \text{ T}$ in the plasma of typical reactors but will be limited to 4 T in ICTF. The value of 4 T should still be sufficient for realistic testing and, in fact, may correspond to the field in higher beta designs.

ICTF DESIGN

The ICTF design is an rf-stabilized mirror, similar in some respects to the Phaedrus device at UW. The geometry and magnet structure of ICTF are shown schematically in Fig. 1 and some key design parameters are listed in Table II. The operation of ICTF relies on the fact that a stable axisymmetric single-mirror

TABLE II

ICTF Design Parameters

Parameter	Value
Type of device	rf-stabilized mirror
Operating mode	CW
Center cell length	2 m
End cell length (each)	$\sim 1 \text{ m}$
Choke coil field	4.5 T
End cell field	4.0 T
Center cell field	0.4 T
End cell - plasma test region diameter	20 cm
rf power (ICRH)	800 kW

cell plasma can be achieved using only rf heating, as demonstrated by the Phaedrus experiment⁴ and other experiments.⁵⁻⁷ ICTF would 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 pondermotive force stabilization and ion heating. In addition to radial pondermotive force, there may be two other useful contributions to MHD stability. These are partial line tying to the end walls and ECH enhancement of the stability.

Fig. 1. Cross-section schematic of ICTF.

The high field magnets are all superconducting to permit economical steady-state operation. The central cell magnets are room temperature copper coils. The two end cells would be used for the impurity control testing. These are designed to have a fairly uniform, realistic (4-T) field, at least moderate plasma electron temperature, and pumping capability (not shown) sufficient to accommodate the high particle fluxes. The expected plasma parameters in the end cell depend on interactions of recycling particles with particles from the central cell. Plasma parameters listed in Table I were computed with a variety of computer codes, based in part on scaling from the Phaedrus experiment. A higher recycling-lower temperature region could also be employed by controlling the rate of end cell pumping. A variety of diagnostics would be employed in the end cells to characterize the plasma, and experimental results. A key diagnostic, for example, would be laser fluorescence which would permit detailed examination of impurity transport.

Cost estimates have been developed for the ICTF shown and for several variations. The installed device cost is estimated to be between three and four million dollars depending on specific site location, use of existing equipment such as helium liquefiers, and variations in some design parameters such as rf heating power.

EXPERIMENTS

The experiments that could be performed in 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. 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 a 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.

As an example of a pumping experiment, two possible configurations for a limiter geometry are illustrated in Fig. 2. Figure 2-a 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. Figure 2-b 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.

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.

For a typical erosion/redeposition experiment, a flat rectangular plate with dimensions 80×40 cm is oriented at a 15-deg angle to the 4-T magnetic field in one of the end cells of the ICTF as shown in Fig. 3. The plate geometry simulates 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.

Fig. 2. Limiter channel configurations.

Fig. 3. ICTF target plate geometry.

The plate would 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 beryllium, beryllium oxide, carbon, silicon-carbon, tungsten, molybdenum, vanadium, and stainless steel would be tested. The materials will be exposed to different plasma environments ranging from the low edge temperature conditions (≤ 30 eV) to medium edge temperature conditions (≤ 200 eV). The plasma density would also be varied. The tests would be conducted for extended periods (1 h to 1 mo CW) so that sufficient material is eroded for accurate measurements and for comparison with computer codes. Following tests in ICTF, the specimens would be examined in other instruments to determine microstructural changes in the surface.

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 composition, 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.

Erosion/redeposition calculations were performed for ICTF using the REDEP code⁸ and related codes. Figure 4 shows the gross erosion rate of a carbon-coated test plate for the plasma conditions listed in Table I. The rate is shown in cm/yr on the basis of a 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 account. Figure 5 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. 5 thus denotes net erosion with a positive value denoting growth. The results for carbon show a wide range of both gross and net growth rates and should thus provide relevant information for devices such as TFCX and INTOR. The gross rates predicted (Fig. 4) are similar in magnitude to those predicted in the medium-edge temperature regime of INTOR. The net growth rates from -12 to +5 cm/yr would permit experimental analysis of the properties of the redeposited material under a wide range of growth rates.

A variety of computer runs were performed for other materials and a range of edge temperatures and densities. The results for beryl-

Fig. 4. Carbon erosion in ICTF - gross erosion rate.

Fig. 5. Carbon erosion in ICTF - net growth rate.

lium are fairly similar to carbon. Vanadium is similar to carbon at moderate temperatures ($T_e = 50$ eV). For somewhat higher temperatures, vanadium enters an unstable regime where self-sputtering coefficients of the redeposited material approach unity. 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.

CONCLUSIONS

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 an impurity control system design or used for verification and calibration of computer codes. The cost of the facility is estimated to be three to four million dollars.

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