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TRANSPORT OF NEUTRAL ATOMS AND MOLECULES IN TFCX

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ABSTRACT

The distribution of neutrals in the proposed reactor TFCX has been modeled by the 3-D Monte Carlo neutral transport code DEGAS, which has been run in conjunction with the 1-1/2-D time-dependent plasma transport code WHIST. The former code contains the best available treatment of neutral-particle physics, including a selection of wall reflection models. The latter code has a comprehensive set of plasma transport coefficients, an MHD equilibrium package, and provision for source terms such as those involving neutrals. It has a simple scrape-off model involving sound-speed flow to the neutralizer plates. The codes are run in iteration, so that the flux-surface averaged particle and energy sources due to interactions with neutrals are consistent with the plasma profiles. The design considered here has a bottom limiter with a pumping plenum. Results are given for the power balance, the mutually consistent plasma and neutral distributions set up in the edge region, the neutral density in the plenum, and the charge-exchange power deposition and erosion along the limiter.

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I. INTRODUCTION AND MODEL

The Tokamak Fusion Core Experiment (TFCX) is a next-generation tokamak reactor currently under design at Princeton and other laboratories [1]. The particular design considered here has the parameters $R = 325$ cm, $a = 131$ cm, $\kappa = 1.6$, $I_p = 10$ MA, $B_t(\text{axis}) = 4$ T, $\beta_t = 0.06$, and an alpha power of about 50 MW. Heat removal and particle control are effected by a bottom pump limiter, illustrated schematically in Fig. 1.

Three important issues connected with the bottom pump limiter design are (1) the mutually consistent plasma profiles and neutral distributions set up in the edge region, (2) the neutral pressure established at the entrance to the pump duct, and (3) the charge-exchange flux, and concomitant erosion, along the limiter and wall. These issues have been investigated with the aid of the plasma transport code WHIST [2,3] and the 3-D Monte Carlo neutral transport code DEGAS [4]. WHIST advances the plasma profiles in time, in the presence of prescribed neutral sources (as well as other sources such as fusion heating, which are calculated internally), while DEGAS calculates the instantaneous neutral sources in the presence of prescribed plasma profiles. Since the issues of concern here pertain to the steady-state portion of the burn, an ignited plasma with full current was assumed to be present initially. The profiles were then allowed to evolve according to WHIST, with the neutral sources updated by DEGAS as often as necessary. In the presence of appropriate mechanisms for burn control and refueling (cf. below), this eventually led to a consistent steady-state configuration, the properties of which are discussed in this paper.

DEGAS has, by now, been employed in a number of calculations [5,6]. It incorporates the full set of relevant plasma/neutral processes such as charge exchange, ionization, dissociation, and hydrogen radiation. In addition, it

provides a number of wall models. In the calculations reported here, reflections from the graphite neutralizer plate and stainless steel walls were handled via the model of Tabata *et al.* [7]. Neutrals implanted in the wall were assumed to diffuse to the surface and to desorb as molecules at wall temperature. The entrance to a pump duct was treated as a surface with an appropriate sticking probability. In the TFCX configuration modeled here, the array of ducts and pumps connected to the plenum gave rise to an effective pumping speed of about 8×10^4 l/s, which translated into an atomic and molecular sticking probability of $\sigma = 0.05$ for the outer 24 cm of the plenum floor.

The plasma transport code WHIST is one and one-half dimensional, in that the flux surfaces are calculated from a full 2-D MHD equilibrium [8] and 1-D transport takes place normal to these surfaces. The transport laws employed here included empirical transport, diagonal neoclassical transport (multiplied by a factor of three), the Ware pinch (multiplied by a factor of two), the ripple trapping and ripple plateau contributions to the ion thermal conductivity, and sawtooth oscillations. In the scrape-off region, the transport model included neoclassical effects with an additional diagonal constant of 10^4 cm²/s added to the diffusion matrix to simulate Bohm diffusion. These transport assumptions are quite standard in the description of tokamak reactors. Burn control was maintained by line radiation from a diffusing iodine profile, fed from the center of the discharge to simulate impurity pellet injection. This method of burn control was selected here because there has been experience in using it with WHIST [9]. The final TFCX design, however, is more likely to employ another mechanism, such as a beta limitation in the transport. The pumped or burnt fuel was replenished by pellet injection, with the pellet ablation calculated from a standard model [10]. The pellet radius was 0.11 cm, and the injection speed was 2×10^5

cm/s.

II. RESULTS

It is sufficient to consider neutral transport only in the bottom half of the tokamak, since virtually no neutrals find their way into the top half. The flux surfaces within the plasma proper, as calculated by WHIST, are shown in Fig. 1. The flux surfaces within the scrape-off region correspond to a particular configuration of external coils [11]. The horizontal position of the limiter was adjusted so that about 85% of the plasma particle flux would be received on the top face; this required trial and error, since the profiles themselves were influenced by the position of the limiter. The depth of the plenum floor was chosen so as to minimize the fraction of plasma hitting the floor without at the same time allowing a significant escape probability from the plenum. The depth finally selected was 21 cm.

The steady-state plasma parameters and power balance are listed in Table 1. About half of the alpha power is removed by radiation, and about 70% of the radiative power is line radiation from the iodine impurity. A different burn control mechanism might lead to a greater heat load on the limiter. The present power split seems more or less realistic, however, in that current large machines tend to radiate about half the heating power. Ionization and dissociation account for about 4% of the power balance, and charge-exchange losses are negligible.

The plasma profiles near the edge are shown in Fig. 2. In particular, note that the electron and ion temperatures in the scrape-off region fall off very rapidly from values of about 100 eV to values less than 10 eV. The density profiles, on the other hand, remain high and quite flat. In the first zone within the scrapeoff, the total fuel density is almost $2 \times 10^{13} \text{ cm}^{-3}$, and

at the edge the density is still $7 \times 10^{12} \text{ cm}^{-3}$. The combined effect of the density and temperature profiles is to deposit virtually all the transport power onto the top face and leading edge of the limiter. The heat load on the leading edge turns out to be approximately 90 W/cm^2 , and the maximum on the top face is about 250 W/cm^2 .

Figures 3 and 4 show some sample contours of the neutral D and DD densities, which, of course, were confined almost entirely to the vicinity of the limiter. The T and TT densities were similar to these, and the DT density was close to the sum of the DD and TT densities. In the scrape-off region along the left and right walls, the neutrals penetrated vertically only about 100 cm above the limiter. In this region, the typical D density was in the range of $5 \times 10^9 \text{ cm}^{-3}$, while the DD density was two or three times higher. In the plasma below the limiter, the average atomic density increased to about $5 \times 10^{10} \text{ cm}^{-3}$. Above the pumping duct in the plenum, the molecular densities exceeded the atomic densities by more than an order of magnitude. The molecular densities were $n_{DD} = 1.9 \times 10^{12} \text{ cm}^{-3}$, $n_{DT} = 4.0 \times 10^{12} \text{ cm}^{-3}$, and $n_{TT} = 1.6 \times 10^{12} \text{ cm}^{-3}$, for a total of $1.5 \times 10^{13} \text{ nucleons/cm}^3$.

The charge-exchange power and neutral-impact erosion were restricted mostly to the vicinity of the limiter. Table 2 lists these quantities along the limiter. As this table indicates, both the largest power flux (6 W/cm^2) and the largest erosion rates (3 cm/yr) occur near the point at which the plasma rests on the limiter. These quantities are significant only within 20 cm or so of this point, and, in general, they are less than the corresponding plasma quantities by at least an order of magnitude.

III. CONCLUSIONS

We have presented results which describe the behavior of the plasma and neutrals in the vicinity of the bottom pump limiter of TPCX. These calculations indicate that a moderately cool (5 - 100 eV) and moderately dense ($n_d + n_t \approx 10^{13} \text{ cm}^{-3}$) plasma can be formed in the scrapeoff. The neutral density in the plenum can build up to the neighborhood of 10^{13} cm^{-3} . The charge-exchange power deposition and erosion are localized near the point at which the core plasma touches the limiter, and there they are far outweighed by the plasma power flux and erosion.

Finally, it should be noted that the scrape-off model employed in WHIST is only one-dimensional, and that parallel transport is modeled by an effective parallel loss time $\tau(\phi)$ given by the connection length divided by the sound speed (with sheath effects also taken into account in the electron energy loss). The ionization and charge-exchange sources calculated by DEGAS have therefore been averaged over flux surfaces before being transmitted to WHIST. In reality, of course, gradients in the plasma parameters will be established along scrape-off field lines, and the effect will be greatest along those field lines passing through the region of greatest recycling, i.e., those intercepting the bottom face of the limiter. Although the scrape-off model used here cannot be expected to be reliable along those field lines, it might be reasonably satisfactory along field lines intercepting only the top face. In any case, a more realistic model would involve a two-dimensional calculation of the plasma density, temperature, and flow velocity, analogous to that of Ref. [6] for poloidal divertors. There are plans to add such a model in the near future.

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TABLE 1. Plasma parameters and power balance.

T_e	9.0 keV
$T_e(0)$	21.0 keV
T_i	9.7 keV
$T_i(0)$	23.1 keV
$\bar{n}_D$	$5.7 \times 10^{13} \text{ cm}^{-3}$
$\bar{n}_T$	$5.7 \times 10^{13} \text{ cm}^{-3}$
$\bar{n}_\alpha$	$4.3 \times 10^{12} \text{ cm}^{-3}$
$\bar{n}_I$	$2.1 \times 10^{10} \text{ cm}^{-3}$
$Z_{eff}(0)$	2.5
$\beta_t(\text{thermal})$	0.057
$\beta_t(\text{total})$	0.063

Fusion heating (e)	37.4 MW
Fusion heating (i)	11.9
Ohmic heating	1.4
	<hr/>
	50.7

Electron transport	18.7 MW
Radiation (Bremsstrahlung, cyclotron)	6.9
Impurity line radiation	17.2
Ionization, dissociation	1.9
Ion transport	5.9
Charge exchange	0.1
	<hr/>
	50.7

TABLE 2. Charge-exchange power and neutral erosion rates along the wall. Erosion rates due to ions are also listed for reference. Segments are numbered as in Fig. 1.

SEGMENT	POWER(w/cm^2)	EROSION RATE (cm/yr)	
		Neutrals	Ions
1	0.3	--	0.1
2	0.5	--	0.9
3	0.6	--	1.8
4	1.7	0.36	3.8
5	1.0	0.03	4.9
6	0.22	0.04	--
7	0.13	0.08	--
8	2.1	0.53	32.
9	4.8	2.0	40.
10	4.7	2.5	41.
11	6.0	3.4	50.
12	5.3	2.2	22.
13	3.9	2.0	21.
14	2.7	1.4	21.
15	1.7	0.87	20.
16	1.3	0.63	16.
17	2.5	0.62	15.
18	2.2	0.40	8.8
19	1.3	0.35	4.4
20	0.4	--	0.7
21	0.7	--	0.1

FIGURE CAPTIONS

FIG. 1. Geometry of the calculations. Only the bottom half of the tokamak is shown, since negligibly few neutrals penetrate to the top. The flux surfaces within the plasma proper are as calculated by WHIST. At the edge of the plasma proper, the elongation and triangularity parameters are, respectively, 1.6 and 0.285. The plasma minor radius at the top is 131 cm. Sample wall segments along the limiter are numbered, beginning at the right-hand part of the lower face and preceeding clockwise to the right-hand part of the upper face. The leading edge is segment 8.

FIG. 2. Plasma profiles in the edge region.

FIG. 3. Representative contours of the neutral D density (cm^{-3}). Numbers give the exponent.

FIG. 4. Representative contours of the neutral DD density.

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Fig. 1

Fig. 2

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Fig. 3

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Fig. 4

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