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Computational Support and Model Development
for Edge-Plasma and Energy Transport Simulations:
Progress Report for 1989

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The focus of the work was on two dimensional modelling of scrape-off layer (SOL) and divertor plasmas in the JET tokamak, with the objective of guiding the ITER divertor design, particularly in the area of fundamental edge plasma data. Additionally, a diffusion model for neutral atom transport in edge plasmas was developed and tested.

I) Summary of Simulation Results for JET

The B2 (Braams) two dimensional edge plasma code was used to analyze boundary plasma conditions in the JET tokamak for a specific documented H-mode shot, number 15796. The poloidal magnetic field obtained from the equilibrium magnetic flux profile for this shot was used to develop a 2D numerical mesh for a complete single null geometry with inboard and outboard divertor plates. A reasonable approximation to the field profile was achieved by the analytic conformal-mapping mesh generation scheme inherent in the code. This method of mesh generation, however, results in disconnected inboard and outboard divertor zones for plasma transport, i.e., an artificial reflecting boundary is introduced between the two zones. Furthermore, use of an orthogonal system forces the boundaries to be everywhere perpendicular to poloidal field lines which yields an incorrect curvature of the divertor plates relative to the experimental configuration.

A range of empirical transport coefficients for transport perpendicular to field lines (D , $X_{e\perp}$ and V) were investigated (with strictly classical transport along field lines) and predictions of density, temperature and flow velocity profiles were compared with available experimental results [1]. The latter were available only at the location of a reciprocating probe [1] - no data was available for parameters at the divertor target. Given the scatter in the experimental data, the study focused on estimating transport coefficients that best fit observed e-folding lengths in the H-mode shot.

In summary, $D_{\perp} \lesssim 0.5 \text{ m}^2/\text{s}$, $X_{e\perp} \sim 1 \text{ m}^2/\text{s}$ and an inward radial convective velocity $\bar{V}_{\perp} \sim -20 - -40 \text{ m/s}$ gave good match of e-folding lengths to measured values at the probe. For $D_{\perp} \sim 0.33 - 0.5 \text{ m}^2/\text{s}$, $X_{e\perp} = 1 \text{ m}^2/\text{s}$ and $V_{\perp} \sim -20 - -30 \text{ m/s}$, the density e-folding length was in the range $\lambda_n = 10-13 \text{ mm}$, while the electron temperature e-folding length was fairly constant at 40 mm, both in good agreement

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with experiment. The corresponding density at the separatrix was in the range $n = 5 - 9 \times 10^{18} \text{ m}^{-3}$ and the electron temperature $T_e = 58 - 70 \text{ eV}$. Measured values [1] were $n \sim 5 \times 10^{18} \text{ m}^{-3}$ and $T_e \sim 55 \text{ eV}$. For zero, inward pinch velocity the match was extremely poor, with $\lambda_n \sim 50 \text{ mm}$. Numerical results were observed to be generally sensitive to the magnitude of $V_\perp$.

Experimental values of density and temperature at the divertor plate were not available but the simulation gave somewhat unexpected results. In all the cases considered with non-zero pinch velocity (including those that gave good agreement of e-folding lengths with experiment) the inboard divertor plasma was less dense and hotter than the outboard, contrary to observations in most tokamaks. The peak temperature at the inboard target was approximately 60 eV while that at the outboard target was near 40 eV. The power deposited on the inner plates invariably exceeded that on the outer plates, with a ratio $P_{\text{outer}}/P_{\text{inner}} \sim 0.79-0.89$ for the matrix of H-mode transport coefficients investigated. Interestingly, for $V_\perp = 0$ the power ratio increased to between 1.15-1.25 which is more in line with observations. However, as noted above, the profiles at the probe location did not match well with experiment in this case. The power sharing was observed to steadily favor the inboard with increasing inward pinch velocity.

All simulations were satisfactorily converged, with respect to both global balance and local grid values, and thus the above anomaly cannot be explained by lack of numerical convergence. A plausible explanation is that the effect is a manifestation of different SOL areas on the inboard and outboard sections. The result is that a strong enough inward pinch can convect energy back into the confined plasma and this occurs preferentially in the outboard where the SOL is narrower. This is consistent with the observed numerical scaling of power sharing ratio with magnitude of the inward drift velocity.

This clearly demonstrates the dangers of imposing an arbitrary drift velocity that is not based on a sound physical model. Although modelling appears to indicate that there is a need for an inward convective component to the flux, in order to reproduce the steep gradients in density and temperature observed during H-mode, simplistic or straight forward generalizations of the transport model can result in apparent inconsistencies such as encountered in this study. A consistent model of inward convective flow can be constructed on the basis of local phenomena, such as pressure gradient and $\mathbf{E} \times \mathbf{B}$ drifts, as well as possibly grad- $\mathbf{B}$ and curvature drifts. These can be readily incorporated into a 2D transport code.

In conclusion, it is important that predictions based on ad hoc convective flow models be used with considerable caution in ITER, especially as the divertor design depends critically on the heat load distribution on the plates.

II) Summary of Neutral Atom Transport Modelling

Starting with the Boltzmann equation for the neutral particle distribution function, an approximate 3D diffusion equation for the neutral atom density was obtained, based on a low order spherical harmonics-Hermite polynomial expansion of the distribution function. The model was successfully tested in 1D with fixed plasma parameters, as well as in the quasi-2D plasma transport code, ODESSA. In 2D, the model may be used to supplant the simple neutral transport model currently employed in the B2 code. However, as the neutral and plasma transport must be performed on separate meshes (to avoid restricting neutrals to following field lines), the numerical implementation is likely to be a complex task. For fixed plasma parameters and a simpler 2D Cartesian geometry, though, the model can be readily implemented and useful results obtained for the charge exchange flux incident on the target and first wall, ionization sources, CX energy losses etc.

Acknowledgement

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Reference

- [1] L. de Kock et al, "The Plasma Boundary in JET," JET Report, JET-P(88)78, p. 89-100, 1988. November 2, 1989

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