Large RF field amplitudes in the SOL and far-field RF sheaths: a proposed mechanism for the anomalous loss of RF power to the SOL of NSTX

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We propose a new model for the anomalous loss of high-harmonic fast-wave (HHFW) heating power to the scrape-off layer (SOL) of the National Spherical Torus eXperiment (NSTX). A significant fraction, up to 60%, of the coupled HHFW power can be lost along scrape-off layer field lines [1], creating bright spirals of heat deposition on the upper and lower divertor regions [2,3]. The spiral heat flux can be relatively intense; up to 2 MW/m^2 of heat flux has been measured in the far divertor region for 2 MW of applied HHFW power [2]. It is important to determine the underlying mechanism because, with 20 MW of ICRF power planned for ITER, a similar loss of ICRF power may erode the divertor and produce unacceptable impurity levels. We hypothesize that the SOL losses are caused by a two-step process. First, the radiofrequency (RF) field amplitude becomes quite high in the SOL when, as suggested by both experiments [1,2] and modeling [4, 5], the right-hand fast-wave cutoff layer is positioned too close to the HHFW antenna. Second, these RF fields setup far-field RF sheaths on the divertor tiles and drive an enhanced heat flux into the divertor [6]. We present results from a cylindrical cold-plasma model, which allows an in-depth study of the condition for substantial wave propagation in the peripheral plasma. Experimental evidence for RF rectification is presented, and our analysis suggests that they could produce additional heat fluxes consistent with infrared camera measurements of the HHFW heat flux within the spirals. This suggests that the SOL losses can be minimized, and heating efficiency maximized, by controlling the right-hand cutoff layer position through tailoring of SOL density and antenna phasing.

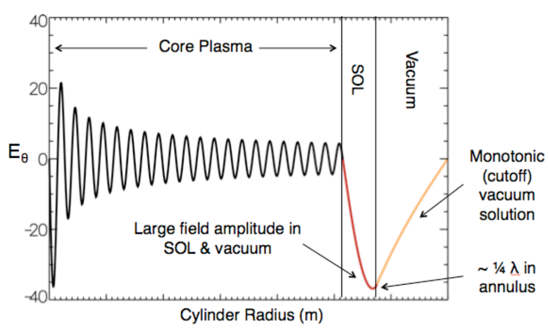

Figure 1. The cylindrical cold-plasma model demonstrates a special class of modes with large RF fields in the SOL. Here, we plot the radial profile of the poloidal RF electric field for the m=0 (azimuthally symmetric) case with a core density of 5 x 10^{19} m^{-3} and SOL density of 2 x 10^{18} m^{-3}. Similar modes exist for other poloidal mode numbers. This mode conducts over half of its wave power in the edge plasma. An antenna that selectively excites this mode will couple significant power to the edge plasma, similar to observations on NSTX.

We use a cylindrical cold-plasma model to demonstrate a class of modes that conduct a significant fraction of their wave power in the peripheral plasma [7] (Fig 1). A build-up of fast-wave power propagating in the SOL was originally hypothesized based on experimental observations [1], and simulations using the full-wave code AORSA [4] also show enhanced RF fields in the SOL as the right-hand cutoff is moved beyond the antenna [5]. However, interpretation of the AORSA results has not been straight forward [8], so the cylindrical model was developed to see if the main AORSA results could be reproduced and understood in a simplified geometry. The modes with large RF field amplitudes in the SOL appear when roughly a quarter radial wavelength fits into the SOL (Fig. 1). These modes are also local maxima of loading resistance and thus readily couple to an antenna. If an antenna selectively excites these modes, then a large portion of the total injected wave power is predicted to propagate in the edge, establishing the conditions for far-field RF sheaths in the divertor. In this paper, we quantify the wave power flowing in the edge as a function of key parameters, such a SOL density, and compare the results against experiments and calculations from the AORSA code.

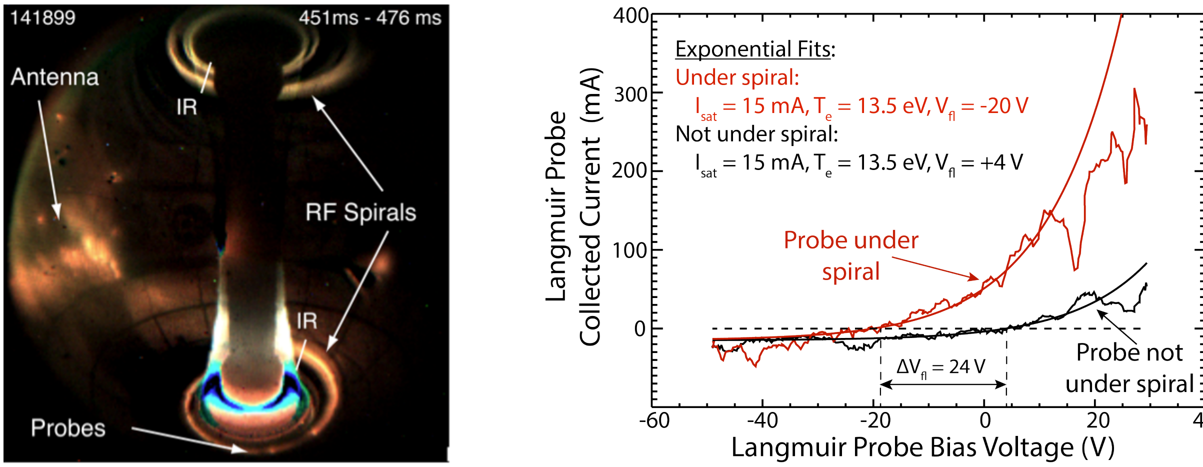

FIGURE 2. LEFT: Spirals on the upper and lower divertor. B_T = 4.5 kG, Ip = 1.0 MA (magnetic pitch in the SOL ~ 39.6°), P_{RF} = 1.3 MW, P_{NB} = 2 MW, and deuterium. RIGHT: Evidence of RF rectification is found in the negative shift in floating potential of a Langmuir probe located under the spiral (red curve) compared to one nearby that is not under the spiral (black curve) without appreciable change in electron temperature. The shift in floating potential is $\Delta V_{fl} \approx 33.5$ V, which equates to an RF potential of $V_{RF} \approx 76$ V. Characteristics are averaged over six consecutive 1 ms sweeps starting at 0.4515 s during the RF pulse.

An RF potential at a plasma-surface interface will setup an RF sheath, and we show that such RF sheaths can drive additional heat flux into the surface. Langmuir probe characteristics show signs of RF rectified voltages but also of rectified electron currents at vessel potential [6] (Fig. 2). First principle calculations that include the rectified currents suggest that these sheaths can drive a substantial heat flux to the surface: for one case studied, the predicted heat flux increases from 0.10 MW/m^2 to 0.21 MW/m^2 [6]. This predication is based on probe measurements from a weak portion of the spiral (Fig. 2, left), and the heat flux is expected to greatly increase at other portions of the spiral. Precise comparison between the computed heat flux and IR thermography cannot yet be made due to toroidal separation of the diagnostics (Fig. 2, left), but projections from IR camera measurements indicate that the predicted heat flux based on RF rectification is within a factor of three of the measurements. Thus, the RF-rectification hypothesis is not inconsistent with the data, and better measurements, to be taken shortly on NSTX-Upgrade, will test this hypothesis. This will be key to obtaining efficient fast-wave heating on NSTX-Upgrade and for predicting when such SOL losses could become an issue for high-power long-pulse ICRF heating on fusion devices, such as ITER.

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[1] J. C. Hosea *et al*., *Phys. Plasmas* **15** (2008) 056104.
[2] J. C. Hosea *et al*., *AIP Conf. Proc*. **1187** (2009) 105.
[3] R. J. Perkins *et al*., *Phys. Rev. Lett*. **109** (2012) 045001.
[4] D. L. Green *et al., Phys. Rev. Lett*. **107** (2011) 145001.
[5] N. Bertelli *et al*., *Nucl. Fusion* **54** (2014) 083004.
[6] R. J. Perkins *et al*., *Phys. Plasma* **22** (2015) 042506.
[7] R. J. Perkins *et al*., *41th EPS Conference on Plasma Physics* (2015) P-1.011.
[8] N. Bertelli *et al*., *Nucl. Fusion* **56** (2016) 016019.