Abstract

Small current modulations in a subset of DIII-D radio-frequency (RF) coils were used in RMP ELM suppression experiments at ITER-like conditions (shape, collisionality, RMP spectrum) to control position of the divertor particle and heat fluxes. Qualitative agreement was found between measured HFS particle flux striations and modeling of magnetic footprints with and without ideal MHD plasma response. However, the radial separation of the particle flux striations is 4–7x larger than the modeled splitting. Striations are generally not observed in the divertor heat flux to the inner strike point in RMP ELM suppression at ITER-like conditions; this is understood to be due to an increase in the volumetric carbon radiation in the inner divertor which washes out the striations in the heat flux to the inner strike point. This suggests that radiative divertor operations in ITER may avoid striated peaks. The divertor target plate heat flux can be reduced 2–3x, with a 60% radiated power fraction, by establishing a narrow radiating mantle between \(0.95 \leq \Psi_R \leq 1\). This is achieved without the loss of ELM suppression by using neon or argon injection into the main chamber. These radiating mantle discharges show for the first time that ELM suppression can be maintained over a wide range of pedestal collisionalities \((0.1 < \nu^* < 1.1)\) with only a modest increase in the line average electron density.
1. INTRODUCTION

Edge-Localized Modes (ELMs) present one of the most demanding control issues in high confinement ITER H-mode plasmas. ELMs can be stabilized by applying small 3D Resonant Magnetic Perturbation (RMP) fields [1]. ELM suppression in ITER will be achieved using RMP from ELM control coils [2]. Application of 3D perturbation fields is expected to cause splitting of the separatrix resulting in formation of spiral lobes on the divertor surface with field lines connected to the inner regions of the plasma and deliver high heat and particle fluxes to additional regions formed by non-axisymmetric lobes on the divertor surface [3]. ITER and future burning tokamaks with tight divertor baffling will have to mitigate heat and particle fluxes while maintaining ELM suppression in H-mode. Different methods have been proposed for reduction and control of the heat and particle fluxes. Two such methods described in this work are (i) control of the lobes, sometimes referred to as the divertor footprints, with small changes in ELM coil currents, and (ii) main chamber intermediate-Z impurity gas injection to enhance the radiated power fraction. The latter is consistent with ITER plans to puff intermediate-Z impurities (i.e. N or Ne). It is estimated that ITER will need to dissipate up to 70% of total power via radiation to avoid damages to plasma facing components [4].

Modeling of magnetic footprints in ITER during standard operating scenarios with RMP ELM suppression [5] showed that it was possible to reduce these heat and particle fluxes by rigid rotation of n=3 or n=4 fields while maintaining good H-mode plasma conditions. Unfortunately, such rigid rotation of the perturbation fields in ITER may not be possible due to mechanical stress limitation on the ITER ELM coils and divertor tiles.

Independently, another experiment and modeling [6] demonstrated that divertor footprints were affected by the toroidal spectrum of the perturbation field from the DIII-D I-coils when individual I-coils were turned on and off. Combination of these two results suggested that it might be possible to use small changes in a subset of ELM control coils to sweep the location of the non-axisymmetric magnetic footprints across the divertor surface while maintaining ELM suppression without exceeding the mechanical stress limits on the ELM coils and divertor tiles in ITER.

2. SMALL PERTURBATIONS IN A SUBSET OF ELM COILS FOR CONTROL OF NON-AXISYMMETRIC FLUXES TO DIVERTOR

A set of experiments was performed on DIII-D to determine whether or not small current changes in subsets of DIII-D I-coils [7] could smooth the heat and particle fluxes to the divertor by slowly sweeping the radial position and toroidal structure of the magnetic footprint. The configuration of the coils in DIII-D includes two rows of six I-coils located inside the vessel connected to a limited number of power supplies. In the experiments described here, the I-coils were connected in quartets to 3 power supplies, as shown in Fig. 1. This wiring configuration allowed us to perform simultaneous ramps of current amplitudes in several I-coils, i.e. IU30 quartet ramp changed current amplitudes in coils IU30, IU90, IU210, and IL270 (indicated by blue color in Fig. 1) while keeping the current amplitudes in other coils at a nominal value.

The experiments were performed in ITER similar shape (ISS) plasmas [8] with low pedestal densities \( n_{\text{ped}} < 2.5 \times 10^{19} \, \text{m}^{-3} \) and low electron pedestal collisionalities \( \nu_e^* < 0.2 \) [9]. Long periods of ELM suppression were obtained to allow for different I-coil quartet coil ramps. A ramp in a particular I-coil quartet resulted in slow
decrease of the applied \( n=3 \) field component and increase in the \( n=1 \) toroidal sideband component. These perturbation fields from the I-coils interfered (sometimes constructively, sometimes destructively) with the DIII-D intrinsic error fields and with the \( n=1 \) EF correction field from the DIII-D C-coil. Good H-mode confinement was maintained throughout the discharge with \( H_{\text{b,2}}=1.2 \). Peeling-balloonning mode analysis done after the experiment confirmed that the discharges were stable to these edge modes in the ELM suppressed phase during I-coil quartet ramps. The additional \( n=1 \) field often led to early termination of the discharges due to core mode locking.

DIII-D is equipped with several diagnostics for estimation of heat and particle fluxes to the divertor plates. The locations and lines of view for these diagnostics are shown in Figure 2. The diagnostics relevant to this experiment include an infrared TV camera operated in line scan mode at 60°, visible imaging of molecular deuterium emission using a fast camera at 75° imaging the lower divertor from above over a 50° toroidal span of view [10], lower divertor Langmuir probes located at 180°, and a Tangential TV camera imaging CII emission from 240° with a field of view toward 200° toroidal angle in DIII-D. There is a large toroidal region (between 240° and 60°) that is not covered by diagnostics capable of measuring heat and particle fluxes. This region is labeled as “dark side” in Figure 2.

Visible-range and infrared cameras provided high temporal and spatial resolution of the inner strike point. The view of the outer strike point in this ISS plasma shape was largely obstructed by the “nose” of the outer shield. The visible range camera imaging was done using a filter passing molecular D\(_2\) emission from the Fulcher band in the range of 594-610 nm. This provided spatially localized emission near the target plate and allowed resolution of the strikepoint structures [10].

Figure 3 shows the temporal evolution of the D\(_2\) emission as measured by the Visible-range camera on the high field side at the toroidal angle of 75°. The emission intensity is plotted as a function of the wall coordinate \( S \)-coord, which is the distance along the limiter starting from the inboard midplane and increasing in the counterclockwise direction. The location of the inner strike point, shown by the red dashed line, is controlled by the Plasma Control System (PCS).

As the amplitude of the current in the IU30 quartet was ramped down, the changes in the applied perturbation spectrum caused changes in the location of the divertor footprints. Figure 4 shows the calculated location of the magnetic footprints near the ISP as predicted by TRIP3DGPU vacuum modeling (neglecting plasma response) and by ideal plasma response (VMEC with “virtual casing” extension outside of the last closed flux surface [11]). It is often assumed that the location and the shape of the magnetic footprints is determined by the non-resonant kink response while the internal structure of the lobes is defined by the resonant fields in the plasma. As shown in Figure 4 (a-b), the ideal plasma response has significant impact on the toroidal position and structure of the magnetic footprints including in the region near 60° imaged by the visible and infrared cameras.

Figure 4 (c) shows the locations and widths of the calculated lobes at the toroidal location of the particle flux measurements as predicted by both codes and when uniformly stretched (scaling factors of 4 – 5.25) in the \( S \)-coord. direction to match the location of the far-most lobe. This scaling ratio was found to be constant at different toroidal locations, i.e. at 75° (Visible-range camera) and at 150° (DiMES TV, 431nm CD-band filter). Both imaging diagnostics were measuring in the visible part of the emission spectrum (D\(_2\) and CD band filters respectively). As predicted by modeling, the radial separation of the lobes was found to be larger at the toroidal location of DiMES TV.

FIG. 4. Magnetic footprint at 3300ms in 166439 predicted by field line tracing neglecting plasma response (a) and with ideal plasma response (b). Toroidal locations of IRTV and visible-range camera are indicated by dashed lines and shaded rectangles. (c) D\(_2\) photon flux and heat flux radial profiles at ISP at \( \phi=60° \). Position and widths of the magnetic footprint lobes at \( \phi=60° \) are shown as calculated and when stretched in \( S \) direction to match the measured D\(_2\) profile peaks.
This direct comparison in Fig. 4c of high-resolution magnetic footprint modeling results with the high spatial and temporal resolution particle flux measurements (red) shows that both models (pink, blue) capture similar structures of the divertor footprints qualitatively (although VMEC does not represent relative lobe spacing well), but not at all quantitatively, with far smaller striation displacements predicted by the simulations. The shortfall in vacuum model prediction of the size of the separation between the lobes was previously reported in [12]. The lobe separation in the VMEC modeling with ideal plasma response is of the same magnitude as in the vacuum modeling. The origin of this underprediction is not yet fully understood. As figure 4 suggests, the ideal plasma response alone may not be sufficient in this case and resistive plasma response should be included [13]. It is possible that resistive response (i.e. field penetration to drive locked islands) may play a role in explaining these differences; currently, there is an active effort to study the effect of the resistive linear and nonlinear plasma response on the magnetic footprint geometry using the M3D-C1, JOREK and M3D MHD codes to explore this and other hypotheses.

In DIII-D, strike point splitting in the divertor recycling and particle flux is routinely observed during RMP operation. Typically, the observed splitting is consistent with the toroidal mode number n of the applied perturbation (usually n = 3 in DIII-D RMP ELM suppression experiments). Similar splitting in the heat flux profile would have serious consequences for heat flux handling during RMP ELM suppression operation in ITER. However, in most RMP ELM suppression cases at ITER collisionalities, there is little impact of these particle flux lobes on the measured divertor heat flux (Fig. 6). In order to understand why some DIII-D ISS discharges have splitting in the heat flux profile, while the others don’t show distinct splitting, we analyzed discharge 147170 from an RMP ELM suppression experiment in ISS plasmas where clear presence of one or more divertor footprint lobes in the measured heat flux was in direct correlation with the magnitude of the heat flux at the main strike point as shown in Figure 5(a).

The heat flux profile in ISS RMP ELM suppressed discharge 147170 evolves from multi-lobe with high amplitude of peak heat flux at the ISP to single lobe broad profile with lower amplitude of peak heat flux at the ISP while plasma parameters (shape, q95, βn, input power and torque etc.) remained constant. At the same time, the total radiation in the lower divertor region is increasing (Fig. 5b), leading to smoothing in the measured heat flux profile and loss of clear lobe structure. This increase in the volumetric radiation in the lower divertor is confirmed by the increase in the CIII radiation in the lower inner divertor leg (Fig. 5c, d), and is due to the change in the divertor conditions associated with the slow rise in plasma density. This increased radiation reduces net conducted heat flux to the inner strike, and thus the peak heat flux to below approximately 2 MW/m², with no distinct lobes seen in the heat flux convected and conducted to the divertor target plate (Fig. 5a). This threshold value is similar to the peak heat flux amplitude in the discussed I-coil quartet experiments discussed above (Fig. 4e) where no splitting in the heat flux was observed. This observation suggests that once the peak heat flux power drops below a critical level the resulting divertor heat flux distribution no longer provides a clear indication of the conducted heat flux flowing into individual divertor footprint lobes due to the large contribution of the heat flux to the divertor surface from the omni-directional radiated power (Fig. 5b), which fills in the valleys in the heat flux between the lobes. This washing out of the divertor heat flux footprint lobes is often more prevalent at the high-
field side strike point due to an enhancement in the radiated power in that region compared to the low-field side. A reduction of the conducted heat flux into the divertor due to an increase in the upstream radiated power also contributes to this washout effect of the heat flux in divertor footprints. One of the ways to increase the upstream radiated power in the plasma edge is by impurity injection as has been previously proposed for RMP ELM suppressed discharges [1]. This approach, when implemented in ITER, may be beneficial for controlling the divertor heat flux.

These results in DIII-D plasmas with ITER-like conditions suggest that when ITER operates with intermediate-Z impurity puffs and radiating plasma edge divertor conditions, it may not have to deal with non-axisymmetric heat flux to the divertor, and the main focus may be on a reduction of the heat flux to the main strike point, and this may be achieved by intermediate-Z (neon or argon) injections in the plasma while maintaining RMP ELM suppression in H-mode plasma as described below.

3. COMPATIBILITY OF IMPURITY GAS INJECTION AND RMP ELM SUPPRESSION

In these experiments, impurity gas injection into the RMP ELM suppressed phase of the discharge was used to study compatibility of RMP ELM suppression in ITER-like conditions with an impurity radiation-enhanced boundary. Two medium-Z gases, neon and argon, were puffed into the main chamber form the upper divertor. The location of the puffs is different from previous experiments on plasma detachment during application of RMP fields, where impurities were injected directly into the lower divertor region [13]. A series of similar discharges was done with varying levels of impurity injections. The injections were approximately 500 ms long starting at 3000 ms during the 1-coil quartet ramps when the amplitude of the IU30 coil quartet current reached 2.5kA. The discharge evolution and heat flux deposition in a “no puff” reference case was compared to 0.8, 1.8 and 2.8 Torr-L neon injections and to 3.2 and 5.5 Torr-L argon injections. The levels of neon and argon after the shots were checked to ensure that no significant amounts of these impurities were left in the plasma at the start of consecutive discharges.

In all cases with neon and argon injections, impurity penetration into the plasma (top of the pedestal region and inside) was observed approximately 500 ms after the start of injection (Fig. 6a). Neon and Argon concentrations in the plasma were measured by multiple diagnostics including an EUV survey spectrometer (SPRED), visual bremsstrahlung (VB), and charge exchange recombination (CER) spectroscopy tuned to neon and argon emission spectra on multiple channels. The VB chords on DIII-D look through a 30 Å wide filter centered at 5230 Å. In post-processing of the shot with Ne injection and modulated 30° neutral beam, the data was processed to exclude times with intersecting beams, due to the neon charge exchange. Careful analysis of the VB and CER data for impurity concentrations showed that the Z_{eff} at the top of the pedestal increased to Z_{eff}=6.5 with Ne impurity injection, and to Z_{eff}=12.5 with Ar.

In all cases with gas impurity injections, the concentration of impurities slowly increased in all regions of the plasma reaching maximum levels approximately 800-900ms after the start of the puff. The measured argon

![Graph](image-url)
profiles were hollow until the end of the shots. This is in contrast to the neon which had a flat profile early, and peaking profile later.

The strongest effect of the gas injection on the heat flux to the divertor was observed during the argon puffs. Figure 7 shows the calculated perpendicular heat flux from divertor floor Langmuir probe measurements. In the reference case (Fig. 7a), the maximum value of heat flux reaches amplitudes of 3 MW/m² at just outside of the location of the inner strike point (indicated by red line). Again, as shown in the IR camera measurements in the previous section, the heat flux measured by Langmuir probes has only one broad peak in the profile without any additional lobe structures. Moderate level of Ar injection at 3.5 Torr-L (Fig 7b) leads to strong heat flux reduction at 3400ms (900ms after the start of Ar puff). Even stronger and earlier (600ms after the start of Ar puff) reduction of heat flux to the divertor is seen with 5.5 Torr-L of argon in Fig 7c.

As shown in Fig. 6b, 5.5 Torr-liter of argon injection led to formation of a stable radiating mantle in the edge region of the plasma (0.95 ≤ Ψ ≤ 1). This happened along with the 4x increase in the core radiated power and an increase in radiated power fraction to 60%. Heat flux profiles measured by the IR camera before (red) and after (blue) the formation of the radiating mantle are shown in Fig 6c and 6d for inner and outer strike point regions of the divertor. Part of the outer strike point region is not visible to the IR camera and is indicated by grey shading. These profiles show significant reduction in the deposited heat flux at both strike points.

The reduction of the heat flux to the divertor is also accompanied by a strong increase in molecular D₂ emission from the divertor region adjacent to the inner strike point as measured by the fast visible-range camera diagnostics. At the same time, the tangential TV camera shows relocation of the carbon CII emission from the region near the surface next to the inner strike point to the region away from the surface in the private flux region. Often, these conditions appear to correspond to partial detachment of the plasma. It is difficult to characterize these plasmas as detached due to the complexity of the detachment and lack of data essential for this analysis. Nevertheless, it is very important that we were able to reduce the heat flux at the strike point location by a factor of two while maintaining good plasma confinement and RMP ELM suppression throughout the discharges with impurity injections.

During these periods of RMP ELM suppression and reduced heat fluxes, the plasma density was controlled by the PCS. In most of these discharges, the density was held constant throughout the RMP phase. In the discharge with the highest level of argon puff (5.5 Torr-L), the density very slowly increases from 2x10¹⁹ to 3x10¹⁹ m⁻³ during the length of the discharge. The rate of the density increase was rather small and did not indicate ELM-free regime (where ELM suppression is achieved by fast uncontrolled increase of plasma density).

It is also important to note that by varying the type/gas and the amount of the impurity injection we were able to achieve RMP ELM suppression over a range of electron pedestal collisionalities while maintaining constant pedestal electron density. Previously, scans of collisionality effect on RMP ELM suppression were performed on DIII-D via density scans as the two plasma parameters are coupled for fixed T_e and Z_\text{eff} values. In these scans, the RMP ELM suppression was typically achieved either in low density and low collisionality regime [15], or in high density and high collisionality regime [16]. In previous experiments, RMP ELM suppression was not achieved in...
the range of medium collisionalities and densities as shown by open red triangles (RMP ELM suppression at low density/collisionality), green squares (ELM mitigation) and black circles (ELMing phase) in Fig. 8. In the present experiment, RMP ELM suppression was obtained over a range of pedestal electron collisionalities from 0.15 to above 1.0. This is shown by red closed triangles in Fig 8 which represent the data from the shot with 5.5 Torr-L of Ar injection described above. One should be careful interpreting this result because the RMP ELM suppression in this (and other impurity puff cases) was achieved at low electron pedestal collisionalities first, and then the collisionality was increased without loss of ELM suppression. It is not yet clear if RMP ELM suppression may be achieved at these collisionalities directly. Nevertheless, this result is very important as it is decoupling the threshold in density/collisionality for RMP ELM suppression and is extending the range for RMP ELM suppression in collisionality parameter space.

4. CONCLUSIONS

Small current modulations in a subset of DIII-D I-coils were used to control position of the divertor particle fluxes. Modeling of divertor magnetic field line footprints with and without ideal MHD plasma response confirms the qualitative features of the measured HFS particle flux striations. Although both the vacuum and ideal plasma response models mostly reproduce the overall structure of the footprints in the particle flux (the number of striations and the toroidal phase of the striations – although ideal-MHD response does this less well), neither model reproduces the observed splitting quantitatively. Further studies are underway to explore additional effects such as resistive response and penetration if fields to drive island formation.

The lack of clear lobe structures in the divertor heat flux to the inner strike point, a robust feature of RMP ELM suppression operation at ITER-like conditions, is now understood to be due to an increase in the volumetric carbon radiation in the inner divertor as the peak heat flux drops to < 2 MW/m². Due to the lack of diagnostic access to the outer divertor, it is not known if this same process occurs at the outer strike point. Striated peaks in the HFS measured heat flux are observed when the near-surface radiated power is low (∼< 2 MW/m²) but as this local radiation source increases these peaks are reduced and the striations are washed out. This suggests that radiative divertor operations in ITER may well result in low target plate heat flux distributions that do not have such striated peaks.

Experiments in which neon and argon were injected into the main chamber during ELM suppression demonstrated that the divertor target plate heat flux can be reduced by a factor of ~2 by establishing a narrow radiating mantle at the plasma edge without the loss of ELM suppression. It has also been shown for the first time in these experiments that ELM suppression can be maintained while the pedestal collisionality is continuously increased by an order of magnitude from 0.1 with only a modest increase in the line average electron density. This is highly encouraging for the prospect of achieving RMP-ELM suppression with radiative mantles.

ACKNOWLEDGEMENTS

This material is based upon work supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences, using the DIII-D National Fusion Facility, a DOE Office of Science user facility, under Awards DE-FG02-07ER54917, DE-FG02-05ER54809, DE-FC02-04ER54698, DE-SC0012706, DE-AC52-07NA27344, DE-NA0003525, and DE-AC04-94AL85000. DIII-D data shown in this paper can be obtained in digital format by following the links at https://fusion.gat.com/global/D3D_DMP. Disclaimer: This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof. The authors would like to thank Dr. J. Herfindahl and Dr. C. Chrysal for their help with VB and CER diagnostics analysis. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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