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During
ELMing H-Mode Plasma**

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I. Introduction

In this report we present measurements of the divertor heat flux in DIII-D for ELMing H-mode and radiative divertor conditions. In previous work we have examined heat flux profiles in lower single-null diverted plasmas and measured the scaling of the peak heat flux with plasma current and beam power. One problem with those results was our lack of good power accounting ($P_{\text{meas}} \leq 0.7 P_{\text{heat}}$). This situation has been improved to better than 80-90% accountability with the installation of new bolometer arrays, Fig. 1, and the operation of the entire complement of 5 Infra-red (IR) TV cameras using the DAPS (Digitizing Automated Processing System) video processing system for rapid inter-shot data analysis.

We also have expanded the scope of our measurements to include a wider variety of plasma shapes (e.g., double-null divertors (DND), long and short single-null divertors (SND), and inside-limited plasmas), as well as more diverse discharge conditions (e.g., detached divertor plasmas with deuterium and neon injection). Double-null discharges are of particular interest because that shape has proven to yield the highest confinement (VH-mode) and beta of all DIII-D plasmas, so any future divertor modifications for DIII-D will have to support DND operation. In addition, the proposed TPX tokamak is being designed for double-null operation, and information on the magnitude and distribution of divertor heat flux is needed to support the engineering effort on that project. So far, we have measured the DND power sharing at the target plates (in/out and up/down) and made preliminary tests of heat flux reduction by gas injection.

After describing the diagnostic and analysis techniques in Section II, we focus on the issue of power balance (Section III). Obtaining good power balance is important to our understanding of divertor operation, since knowing how much power is lost by radiation vs. how much is conducted to the target plates defines the boundary conditions used by 2-d predictive models which attempt to relate main plasma parameters to divertor target conditions. More fundamentally, power

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balance measurements have helped us identify the underlying physical processes (e.g., impurity radiation, ion-neutral collisions) which are responsible for heat and particle flux reductions observed in our radiative divertor experiments. This will help us extrapolate our work to ITER and TPX, devices which will rely on this divertor concept for high power operation.

Detailed descriptions of the first-wall heat-flux distribution follow our discussion of power balance. Section IV considers the effect of divertor geometry in terms of X-point height, inner-wall gap, and up-down balance in the double null configuration. Heat flux profiles during radiative divertor operation form the subject of Section V. In Section VI we show toroidal asymmetry in the lower divertor, using the new camera view. Our results are summarized in Section VII.

One point we wish to make here is that we have consciously decided to focus only on quasi-stationary ELMing H-mode discharges. This mode is likely to appear in future large tokamaks and is characterized by good energy confinement, low impurity content, and acceptable helium transport. While we have routinely measured the divertor heat flux profiles for VH-mode plasmas, we have not tried to tabulate the data in any systematic way because of the transient nature of the VH-mode. So far, VH-mode discharges exhibit a monotonic rise in line-average density and stored energy unless the heating power is steadily reduced. Thus, the scaling of divertor conditions with main plasma parameters is time-dependent and its value for model validation and machine design is unclear.

II. Diagnostic Technique

The first-wall heat flux in DIII-D is inferred from the temporal behavior of the measured surface temperature of the graphite armor tiles. The surface temperatures are measured using video cameras detecting 3-5 μm or 8-12 μm infrared blackbody radiation. The cameras view the inner wall of the tokamak through vacuum windows of zinc selenide, using anti-reflection coated germanium focusing lenses outside the vacuum. We now have a total of five IR TVs operating on DIII-D: two cameras viewing 20° segments of the upper and lower divertor targets, two viewing the center post, and one more viewing a second 20° segment of the lower divertor spaced 105° toroidally from the other lower divertor camera. Since these cameras provide 2-d images, tile edges or other areas which receive non-uniform heating can be identified and avoided when constructing heat flux profiles.

The IR image data is recorded at 60 pictures/sec during the discharge using standard VHS format videocassette recorders which are controlled automatically by the DAPS microcomputer system. After each plasma pulse, the DAPS sequentially replays and digitizes 300 to 600 video images (representing about 80 Mbytes) from each camera. From these image data, the DAPS extracts surface temperature profiles (0.5 msec time average at 16.7 msec intervals) across the region of interest and combines them into a 2-d array (space, time) which is then sent to the DIII-D computer system for archival storage.

Absolute calibration of the IR TV camera systems on a regular basis is essential to making quantitative heat flux measurements. We calibrate the system using the DAPS to record camera images at five minute intervals during bakeout of the DIII-D vessel, when the temperature, as measured from thermocouples inside the vessel, rises from 30 to 350°C. In this way, the effect of the signal processing electronics, window and optics losses, and surface emissivity of the carbon tiles can be quantified. Calibration to higher temperatures (up to 1000°C) is accomplished using side-lab blackbody sources.

The IRTV data provides only tile surface temperature. To convert to heat flux, we use a 1-d semi-infinite solid approximation to model the thermal response of the tiles to external surface heating¹. This neglects heat diffusion across the face of the tiles or any temperature rise on the back side of the tiles. We have thus minimized the processing time needed to obtain the heat flux in order to allow data from more shots to be analyzed. Comparison with more careful 2-d simulation of the thermal response shows that using a 1-d approximation is accurate to within 10-20% for heating pulses lasting less than 2 sec. Total power onto the first wall is calculated by assuming toroidal symmetry, and extrapolated from measurements at one toroidal location. In the near future, we plan to automate the calculation of heat flux from the raw digitized data, to make this information routinely available shortly after the shot.

III. Power Balance

Power balance in DIII-D has improved from the 50-70% level to 80-90%. Roughly half the improvement originates from increased estimates of the radiative losses as determined from the new bolometer array. Previously, only a single array of detectors viewed the plasma from a port at the midplane, and it was assumed that the radiated power was uniformly distributed along the line of sight of any single detector. Now we allow for a non-uniform distribution and combine data from two arrays, one located above, and the other below the midplane.

Improvement in power accountability has also come with new calibrations of the IR systems, using the technique described in the introduction. Previous calibration data was obtained over a period of years on each of the four cameras used in the power balance measurement (Upper divertor, upper and lower center post, and lower divertor). This is the first time all the IR systems have been simultaneously re-calibrated, insuring that no calibration is out of date.

We are making modifications to the cameras to allow more automation of the calibration process, thus reducing the human error factor and increasing accuracy. This will also allow us to make calibrations more frequent and more nearly routine. A computer will be used to record and digitize the calibration data described in Section I, recording data for each camera scale at set time intervals. These tasks were previously executed manually, requiring precise settings of camera gain every 5 minutes for six camera scales, over a period of four to six hours. The new system will allow one or two people to calibrate all cameras simultaneously, rather than requiring multiple bake sessions of the tokamak to get all cameras calibrated. This will reduce personnel time expended by both GA and LLNL, at the same time insuring currently valid calibration data. An existing computer will control existing cameras, so very little new equipment is being bought.

Power accountability in the range of 80-90 percent is now observed over a wide range of power. Fig. 2 shows good global power balance during discharges ranging in power from 8 to 19 MW. The various loss mechanisms are stacked one upon the other, with the lowest circular dots representing heat flux to the outboard lower strike point. The squares are the bolometer-detected radiation at the outer divertor leg, the diamonds are the heat flux to the inboard strike point, the triangles are the radiation from the inner divertor leg, the topmost circular dots are the core radiation, and the x's are the inner wall heat flux.^{2,3}

IV. Effect of Magnetic Geometry on the Heat Flux Distribution

The peak heat flux on any surface in the tokamak is a sensitive function of the magnetic geometry. For example, in using the BNDY database, we previously found⁴ that the peak divertor heat flux decreased linearly with the distance between the magnetic separatrix and the center post for SND plasmas, as shown in Fig. 3. Here we plot the normalized peak heat flux ($P_{\text{norm}} = P_{\text{div}}/I_p/P_{\text{beam}} \cdot \delta_{\text{div}}/\delta_{\text{mp}}$) to the lower divertor as a function of the gap (cm). The observed variation is consistent with power being scraped off on the center post when the gap is small. However, it was hard to understand this since the effect was seen when the gap was several times the scrape-off-layer thickness.

With the new poloidal coverage by IRTV, we were interested in exploring this effect in more detail. As a further investigation, we examined a discharge which began center-wall limited and moved outward to form a lower single-null diverted plasma. The gap between the inner wall and the separatrix as a function of time in the discharge is shown in Fig. 4(a). The measured routes for power flow out of the plasma are detailed⁵ in Fig. 4(b). Initially, as expected, a large amount of power flowed to the inner wall. As the inner gap increased, the majority of the heat flux began flowing to the divertor.

Due to the poor confinement of the limiter discharge, the energy content of the plasma began to rise when the inner gap increased. Some of the input power went into a large positive dW/dt (increase in stored energy) during that time. This partially accounts for the improvement in power accountability as the plasma approaches an equilibrium diverted discharge later in the shot. This is seen by comparing the total measured output power from the plasma with the input power (upper solid line).

There is still some mystery as to why the inner gap continues to affect the plasma radiation and heat flux., even when the gap is more than two scrape-off layer thicknesses. It is possible that even at that gap, sputtering from the inner wall affects the Z_{eff} and radiative losses from the confined plasma.

The peak divertor heat flux is also a strong function of the X-point height. Understanding this scaling is important for validation of edge-plasma models such as UEDGE or b2.5. The scaling of peak heat flux in the divertor with plasma current, toroidal field, and input power to the plasma, has been presented elsewhere.^{4,8} A deep divertor with long field-line connection lengths reduces parallel heat conduction and lowers the electron temperature at the target plates. It also allows for more radial diffusion into the private region, which further lowers the plasma temperature. On the other hand, a deep divertor usually has less magnetic flux expansion, so that the peak heat flux is higher.

Here we show the scaling with X-point height. The dependence of heat flux on X-point height is relevant to the design choices for the future divertor in DIII-D, and the TPX divertor design.

The peak heat flux onto the divertor plates varies with different X-point heights because of the changing magnetic geometry and connection length. To investigate this, X-point height scan

experiments were carried out in single and double-null diverted plasma. The value of L_{Poloidal} was varied from 2 to 32 cm in different single-null plasma discharges, and from 10 to 20 cm in double-null plasmas. Beam power of 4.8 or 6.8 MW was used for the single-null shots discussed here. Divertor heat flux was measured at the fixed toroidal location of an IR camera. We analyzed the peak heat flux at the outer strike point, where electron temperature and heat flux were highest.

Note in Fig. 5 that the peak outer strike point heat flux increases as the x-point is raised.⁷ This is due to the compression of the magnetic flux surfaces with distance from the x-point (see Fig. 6). However, when the ratio of midplane magnetic field to magnetic field at the divertor is used to map the peak heat flux to the tokamak midplane, the mapped heat flux decreases with increasing x-point height.

This indicates that there must have been increased spreading of the heat flux profile (beyond the value given by the flux mapping), or there was additional loss in the Scrape-Off Layer (SOL) as the x-point was raised. The total power reaching the divertor plate at the outer strike point remained constant, showing that no additional power was lost in the scrape-off layer. This shows that the power was diffusing across the magnetic field, e.g. into the private flux region. This is similar to the effect shown for JET.⁸

Double-null operation causes some of the heat flux to be deposited on the ceiling tiles of the upper divertor (Fig. 7). The heat flux profile on the upper divertor is often ragged, rather than two neatly defined strike points.⁹ This may be due to magnetic field errors. There is an N=1 coil on the tokamak which may contribute to such field errors.

Experiments have been done to investigate the balance of power between upper and lower divertors in the double-null configuration (ELMing H-mode with no radiative divertor). The magnetic configuration was varied continuously between two states during a shot.

The first state was essentially lower single-null and the second was nearly upper single-null. This was achieved by having the separatrix associated with one null occur on a flux surface which mapped at the midplane to smaller minor radius than the separatrix of the other null. The divertor which was magnetically connected to the inner flux surfaces intercepted most of the heat before the energy could be transported across field lines to the outer flux surfaces connected to the other divertor.

The intermediate states were obtained by decreasing the distance between the two separatrices. This distance between the separatrices will be referred to as the DRSEP parameter. The value of DRSEP can be held constant by the control system to within 2 mm (Fig. 8), according to the EFIT code output.⁹ (Note also in this figure, the more than 5:1 out/in heat flux asymmetry in the lower divertor.)

For the magnetically balanced case, the two nulls lay on the same flux surface. However, this did not correspond to balanced heat flux between the two divertors (Fig. 9). This is possibly due to the 1-cm uncertainty in calculating the location of a separatrix. This calculation was done using the EFIT computer code, with data from magnetic pickup loops. The offset in the heat flux balance compared to the magnetic balance might also be due to drifts having a preferred direction.

Note that the transition between all heat flux to the upper divertor and all to the lower is smooth and controllable, according to this data. This bodes well for future attempts to share power loading equally between upper and lower divertors.

V. Radiative divertor Experiments

The peak heat flux in DIII-D at the divertor plate is less than 700 W/cm², which causes only slow erosion of the tiles. However, in ITER, the heat density would be many times greater. In response to the request for ITER-related R&D, we have investigated heat flux reduction by gas injection. Deuterium, argon, and neon all have been used to lower the peak divertor heat flux in DIII-D. Previously, we have presented results from the D₂ puffing experiments¹⁰. In those experiments it was thought that the deuterium puff increased the divertor density and radiative losses, thus lowering the heat flux. However, the main plasma density increased along with the divertor density. Impurity puffing may provide a means of increasing divertor and edge plasma radiation without producing a large change in core density. In this case, the issue is impurity retention and keeping Z_{eff} in the core sufficiently low.

Recent experiments has used neon puffing from the private flux region to reduce the peak heat flux at the outer strike point by more than a factor of five. A good example is shown in Fig. 10, in which 1.3 T-liters of neon was injected during the ELMing H-mode phase of a SND discharge. This neon puff represents about 5% of the total electron content of the main plasma. The sharp reduction in peak heat flux occurs within 200 msec of the gas puff. At the same time, there is an increase in neon content in the main plasma.

The likely mechanism for the strong reduction in heat flux is line radiation by neon in the edge plasma. The change in total radiation power just matches the reduction in divertor heat flux. Fig.

11 shows the radial emissivity profile determined by inversion of the bolometer data. We can reproduce this profile using the MIST impurity transport code along with measured temperature and density profiles to calculate the radiative loss produced by a uniform 3% concentration of neon. This level of neon should increase Z_{eff} to nearly three, which agrees with visible Bremsstrahlung data. This concentration of neon represents about half the total injected into the vessel, showing that there can be significant impurity penetration in our present open divertor configuration.

VI. Toroidal Asymmetries

The second IR TV camera which we installed allows us to look for the existence of toroidal asymmetries. The ASDEX team reported asymmetries as large as 2:1 during LH heating experiments using target plate calorimetry. We inferred that such asymmetries much sometimes exist in DIII-D because of the observation of bifurcated divertor heat flux profiles.

So far, we have observed significant toroidal asymmetries only after Locked Modes have developed; without them, the heat flux profiles look remarkably similar at two toroidal locations. Figure 12 shows a typical example of such bifurcated profiles¹¹. The radial heat flux profiles were measured at 105° and 165° toroidal angles: that at 165° is clearly bifurcated when compared to the one at 105°. At 165° the two peaks are displaced by about 10cm. In this case, the locked mode was triggered by biasing the N=1 coil to increase the vertical field errors above the threshold value. Similar behavior is seen with naturally occurring locked modes as well. It is important to note that, even though more total power is going to the divertor region where the profile is bifurcated, the peak values appear similar. In cases where the mode was slowly rotating before it locked, we found that the peak flux was modulated by only about 20%.

The implication of these observations on divertor design are both good and bad. On the good side, the absolute maximum peak heat flux does not appear to be strongly modulated by the locked modes. On the other hand, the very broad and complicated profile produces heating in unexpected places. Thus, the leading edges of narrow divertor slots proposed for future high power tokamaks may receive far more power than they will be designed to handle. Moving them further away from the separatrix to reduce such heating may reduce their intended effectiveness for isolating gas from the main plasma. Clearly, we need to better understand how to control these asymmetries.

VII. Summary

By means of infrared and bolometer measurements, we have established power balance in DIII-D plasma at the 80-90% level. This is an improvement, due to increased bolometer coverage,

expanded poloidal IRTV coverage of the first wall, and new calibrations. This accountability holds over a wide range of input powers.

The control of up-down heat flux balance in double-null plasma has been explored, and found to vary smoothly with the DRSEP parameter. The magnetically balanced plasma (upper and lower null on the same flux surface) did not correspond exactly to equal heat load, but required approximately a 1-cm displacement in favor of the upper divertor to achieve heat balance. The DRSEP parameter was found to be controlled to within 2 mm by the DIII-D control system.

The radiative divertor is found to reduce the heat flux to the divertor plates by more than a factor of 5, when neon is puffed into the plasma.

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Figure Captions

Fig. 1. Viewing chords of the DIII-D bolometers (left), and current IRTV coverage of the first wall as of 1993 (Right).

Fig. 2. Global power balance as a function of power. The different loss mechanisms are stacked vertically to obtain total power loss at each level of input power. The lowest circular dots represent heat flux to the outboard lower strike point. The squares are the bolometer-detected radiation at the outer divertor leg, the diamonds are the heat flux to the inboard strike point, the triangles are the radiation from the inner divertor leg, the topmost circular dots are the core radiation, and the x's are the inner wall heat flux.

Fig. 3. The normalized peak divertor heat flux increases linearly with the inner wall gap.

Fig. 4. A) Inner gap was varied with time during a discharge. B) total power into the plasma during that same shot, is compared with power out of the plasma, as measured by IRTV and bolometers. Some of the missing power at the beginning of the plot went into stored energy in the plasma.

Fig. 5. Peak heat flux at the outer divertor target (top) and flux-mapped to the midplane (bottom), vs. x-point height. The injected neutral-beam power was either 4.8 MW (solid line) or 6.8 MW (dotted lines). Also shown are the power reaching the outer divertor target plate as a fraction of beam power, and the radiation power loss in the divertor as a fraction of the beam power.

Fig. 6. EFIT for a high X-point. This discharge is an extreme of the x-point height scan. The magnetic flux surfaces at the divertor are closer together with increasing distance from the x-point.

Fig. 7. Heat flux profiles in the upper divertor (dotted lines) and lower divertor (solid lines)

Fig. 8. Double-null diverted plasma with DR_{SEP} held nearly constant. Shown are line-averaged density, stored energy, DR_{SEP} (distance between lower and upper separatrices), and peak power to the lower inner, lower outer, and upper outer strike points.

Fig. 9 Up-Down heat flux balance vs. DR_{SEP} . The difference in upper and lower peak heat flux is divided by the sum of upper and lower peak heat flux.

Fig. 10. Divertor heat flux reduction by neon puffing. Neon was puffed for only 20 mS, but continued to recycle into the plasma.

Fig. 11. Bolometer measurements of neon radiation (top); MIST code modeling of neon charge state distribution and neon line radiation (bottom). Both the modeling and the measurement show neon radiating mainly at the edge of the plasma.

Fig. 12. Heat flux profiles in the lower divertor, separated toroidally by 105 degrees. The bifurcation of heat flux at 165 degrees is quite striking, but is not visible at 60 degrees.

New Diagnostics for Power Balance

- New Bolometer Arrays for Improved Coverage
- DAPS for Intershot Processing of IRTV Data

Figure 1

Power Loss Fractions Remain Constant Over a Wide Range of Injected Powers

Figure 2

Normalized Peak Power vs. Inner Wall Gap

Figure 3

Figure 4

POWER BALANCE AND VARIATION OF PEAK HEAT FLUX SUGGEST DIFFUSION INTO PRIVATE REGION IS IMPORTANT

Figure 5

EFIT for a High X-Point

Figure 6

HEAT FLUX PROFILES FOR DN PLASMAS

Figure 7

80045

TPX-like Double Null Divertor. $I_p=1.2\text{MA}$. $B_t=2.1\text{T}$

Figure 8

MEASURED HEAT FLUX PEAKS IN LOWER DIVERTOR FOR STANDARD DN CONFIGURATION

- A double-null ELMing H-mode plasma was displaced up and down; the resulting peak heat flux on the divertor targets was measured using infrared TV cameras.
- In this configuration, the ion grad-B drift direction is towards the lower divertor.

Figure 9

Divertor Heat Flux Reduction is Sustained with Neon Injection

Figure 10

Neon Radiates in the Edge Plasma

Figure 11

Figure 12

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