1. Purpose of Experiments

The goal of these experiments is to explore the physics that determines the radial width of the heat flux ‘foot print’ on C-Mod’s divertor target plates and to map out its dependences on global plasma parameters (plasma current, magnetic field, density, input power) as well as appropriate non-dimensional plasma physics parameters (collisionality, $\alpha_{MHD}$, normalized ionization mean-free paths, …). The relevant physics is expected to include three key ingredients, which are addressed experimentally by this MP: (1) cross-field transport, setting the ‘upstream’ boundary layer profiles of electron temperature and density near the last-closed flux surface (i.e., near the outer midplane), (2) the mapping of heat flux along open magnetic field lines from the ‘midplane’ to the divertor volume, (3) and dissipation of a portion of this heat flux (via radiation and neutral interactions) prior to its arrival onto the target plates. These experiments are in support of a FY10 DoE Joint Facilities Milestone†, “…measure the divertor heat flux profiles and plasma characteristics in the tokamak scrape-off layer in multiple devices to investigate the underlying thermal transport processes…” The present MP focuses on L-mode plasmas, which span a much larger machine parameter space (field, current, density, power) than H-modes. A separate (and more abbreviated) MP will be developed to extend these investigations to H-mode regimes.

2. Background

Physics-based plasma transport models that can accurately simulate the heat-flux power widths observed in the tokamak boundary are lacking at the present time. Yet this quantity is of fundamental importance for ITER and most critically important for DEMO, a reactor similar to ITER but with ~4 times the power exhaust. Divertor target ‘footprints’ for ITER are estimated on the basis of empirical scalings from existing machines [1], leading to an effective ‘power-width’ ($\lambda_p$) mapped to the outer midplane of $\lambda_p \sim 5$ mm. Yet, the reliability of this estimate is uncertain. Temperature gradient length measurements at the outer midplane suggest that $\lambda_p$ scales with major radius and not
much else [2]. In contrast, scalings from divertor heat-flux measurements [3] do not indicate a major radius scaling for $\lambda_p$ and appear to vary with other parameters ($P_{\text{Sol}}, B_\phi, q_{95}$). This seemingly ambiguous situation calls for a dedicated experimental program to sort out the physical processes that are at play and to develop a clearer understanding of the key scaling parameters – at least in an empirical sense, if not theoretically. Recognizing the importance of this plasma science area, C-Mod, DIII-D and NSTX will focus experiments in FY10 towards characterizing the divertor ‘footprint’ and its connection to conditions ‘upstream’ in the boundary and core plasmas†.

A number of factors are expected to contribute to the width of the heat flux footprint that is observed on the divertor target plate. The most important of these is the cross-field plasma transport in the vicinity of the last-closed flux surface (LCFS), i.e., the physics that sets the width of the heat flux channel upstream from the divertor target along open field lines. Experiments clearly indicate that heat and particle losses through the LCFS occur predominantly on the low-field side, near the outer midplane – plasma flow and parallel heat conduction carry the particles/heat to other parts of the boundary layer. Thus, a primary area of focus must be on the width of the power channel at the outer midplane, its empirical scaling with local and global parameters, and the associated behavior of the plasma turbulence near the LCFS.

Previous experiments in C-Mod have uncovered clear correlations between pressure gradients near the last-closed flux surface (LCFS), the plasma current in the discharge, and the local collisionality [4, 5]. Expressed in terms of dimensionless variables, pressure gradients normalized to the poloidal beta gradient (i.e., the MHD ballooning parameter, $\alpha_{\text{MHD}}$) are found to be invariant in L-mode plasmas over a wide range of plasma currents, magnetic fields and plasma densities when plotted versus a normalized plasma collisionality, $\Lambda = (\lambda_{ei}/R)^{1/2}/q_{95}$ (see Fig. 1). Here, $\lambda_{ei}$ is the local electron-ion mean free path, $R$ is major radius and $q_{95}$ is the rotational transform at the 95% flux surface. This behavior makes contact with results from first-principles simulations of electromagnetic fluid drift turbulence [6-10], which identifies these dimensionless expressions as governing the character and strength of the turbulence/transport. Plasma flow shear also appears to play a role, with stronger shearing rates correlated experimentally with higher values of $\alpha_{\text{MHD}}$ [11]. H-mode plasmas tend to exhibit a similar behavior – peak gradients in the H-mode pedestal tend towards a fixed value of $\alpha_{\text{MHD}}$ for a fixed plasma collisionality [12]. Thus the boundary layer plasma appears to exhibit a ‘critical gradient’ behavior near the LCFS – it is ‘clamped’ at a ~fixed value of $\alpha_{\text{MHD}}$, dependent on collisionality (principally) and also affected by flow shear. In light of these observations, perhaps it is not surprising that the dominant scaling for the upstream $\lambda_p$ across tokamaks is major radius, as uncovered by Kallenbach [2]. But what about the scaling of the heat flux footprint in the divertor? In the past, C-Mod was not equipped to make systematic heat-flux profile measurements on the divertor targets as the upstream plasma conditions and their empirical dependences were being mapped out. Now with the newly installed divertor heat flux instrumentation, such scaling studies can be pursued – a principle goal of this MP. Moreover, with the recent improvements to the gas-puff turbulence imaging systems (outer midplane, X-point, inner...
midplane [13]) and installation of scanning Langmuir-Mach probes on both the low and high-field scrape-off layers [14], we are in a unique position to track the behavior of plasma turbulence, including shear-layer behavior and plasma flows, as the heat flux profiles upstream and in the divertor are varied.

Fig. 1 - Electron pressure gradients and $\alpha_{MHD}$ versus normalized collisionality, evaluated 1 mm outside the separatrix in LSN discharges (left) and USN discharges (right) with normal magnetic field direction. Data points represent average values from a number of probe scans; error bars indicate typical $\pm 1$ standard deviation in data sample; smooth curves are spline fits to the data. Despite wide range of plasma currents and densities, the data map to the same characteristic curves in this dimensionless parameter space. Correlated with the higher pressure gradients, LSN discharges (left) exhibit a higher level of poloidal flow shear near the LCFS [11].

Having measured the upstream heat-channel width, the next question is: Does the heat simply map along magnetic flux tubes onto the target plate, i.e. $q_{||}/B \sim$ constant, under conditions when volumetric dissipation in the divertor (radiation, charge exchange) is negligible? Yet, along the way magnetic flux tubes experience significant magnetic shear, becoming elongated in the minor radial direction, squeezed in the poloidal direction, as well as tilted. Turbulence imaging shows that plasma fluctuations (‘blobs’, quasi-coherent structures) obey this magnetic mapping, as expected theoretically [15]. Turbulence imaging also shows that field-aligned plasma energy pulses (i.e., blobs) appear intermittently in the scrape-off layer with a variety of spatial scales. As these entities are stretched in the radial direction and compressed in the poloidal direction near
the X-point, is the heat able to conduct/advect along the short distance poloidally to adjacent, colder flux-tubes, thereby breaking the often-assumed flux-tube mapping?

Next, as the heat arrives in the divertor volume, what role does volumetric dissipation play in affecting the shape of the heat-flux profile and the resultant heat-flux footprint width? Certainly, as the divertor plasma becomes detached (which happens first near the strike point), we expect that the heat flux density at this location will fall preferentially. As a result, the footprint would appear wider than it is upstream, but with reduced magnitude. Perhaps this effect can account for the differences between the upstream and divertor heat flux width scalings noted above. Experiments must be designed to systematically vary and track the divertor plasma conditions (collisionality, radiation fraction, degree of detachment, etc.) as potentially important scaling parameters. In addition, some basic power accounting needs to be performed as a cross-check in these experiments: Can we account for were all the input power (Ohmic + RF) goes? This is important because we need to know if our measurements are quantitatively consistent, tracking the dominant power loss channels as parameters are varied.

Finally, what is the relationship between the heat flux footprint on the divertor surface, as measured from IR and thermal sensors, compared to that same quantity computed from local plasma densities and temperatures measured by embedded Langmuir probes? Measurements of the effective plasma-sheath heat transmission factor at the divertor plate in a number of tokamaks have been reported to vary from 2 to 30 (compared to the theoretical value of ~7), depending on plasma conditions [16, 17]. The discrepancy is very significant since this parameter is a fundamental quantity from basic plasma-sheath analysis. Does the shallow incidence angle of the magnetic field with respect to the divertor target play a role here? Ion temperature? Neutral effects? We have the potential opportunity to address such issues in these experiments.

3. Approach

With the above goals in mind, a new set of divertor heat-flux instrumentation was installed during C-Mod’s FY09 maintenance period. This includes a suite of embedded heat-flux sensor probes (tile thermocouples, calorimeters, surface thermocouples) combined with an improved IR thermography system (ElectroPhysics Titanium 550M camera). In addition, a new divertor bolometer system was installed to assess radiative contributions to the divertor power balance (see Fig. 2).
The instrumented tiles on the outer divertor consist of two columns of ‘ramped tiles’, which are tilted in the toroidal direction by ~2 degrees relative to standard tiles (Fig.3). This ensures that the tile surfaces will not be shadowed toroidally by misalignments in adjacent tiles. It also increases the thermal load to the local tile surface, improving signal-to-noise for IR and sensor-based diagnostics.

Fig.3 – View of J-port divertor ‘ramped tiles’ and map of embedded sensors.
A new ElectroPhysics Titanium 550M camera has been installed to view the J-port divertor module via the A-port top IR periscope. This system is able to record movies of IR emission, monitoring the heat-flux footprint as it evolves during a plasma. An example snapshot is shown in Fig. 4. The camera has demonstrated sufficient resolution, even without the need to deploy a x4 zoom capability, to resolve ~1 mm scale features on the outer divertor surface (Fig.5). Since the camera views the tile sensors directly, in-situ cross calibrations can be performed on a shot-to-shot basis, allowing the images of IR emission to be translated directly into tile surface temperature measurements.

Fig. 4 – IR view of J outer divertor ‘ramped tiles’ from A-top periscope during a C-Mod plasma.

Fig. 5 – Map of ramped tile sensors (left) and standard-resolution IR camera view (right).
At a different toroidal location (F-port module), the outer divertor is instrumented with an array of 10 Langmuir probes, with similar poloidal spacing as the thermal sensors. These probes continuously record poloidal profiles of plasma density and electron temperature at the divertor surface (~ 5 ms per sample). Taken together with the embedded thermal sensors and IR images, these data will allow plasma-sheath heat-flux transmission factors to be inferred. In essence, the thermal measurements allow us to ‘calibrate’ the heat flux profiles and the corresponding estimates of total power received by the outer divertor that were inferred previously by the Langmuir probe system.

These new instruments, combined with our present knowledge of the near scrape-off layer behavior (Fig. 1), call for a simple and straightforward experimental approach – simply map out profiles, heat flux footprints and turbulence behavior over the wide range of parameters accessible to C-Mod, making contact with results from the previous edge plasma ‘phase-space’ studies, MP#409 [18]. The machine parameter space that we hope to cover is shown below in Figs. 6 and 7. This plan revisits the plasma parameters covered previously in MP#409, except that it sticks to lower single-null topologies and optimizes the outer divertor strikepoint location (with a slow sweep) to best utilize the divertor heat-flux instrumentation.

![Fig.6 - Proposed combinations of plasma current and toroidal field to be investigated. Density scans will be performed at each condition. Ohmic and Ohmic plus ICRF power scans will be done at 5.4 tesla in forward and reversed field directions.](image-url)
At first we will concentrate on ohmic discharges. These are known to be less noisy for the embedded sensors and, since we are not tied to the ICRF heating scenario, we can readily explore a wide range of magnetic fields. Additionally, these shots are less likely to cause damage to the scanning probes and avoid the complexities of RF-sheath rectification effects. Therefore, the ohmic-only parameter space should be fully explored first. Then, focusing on 5.4 tesla discharges, we will investigate the power-dependence of the L-mode heat flux width scalings by turning up the ICRF power while staying below
the L-H threshold. Finally, by reversing the magnetic field (5.4 tesla), we can explore the highest range of ICRF input power for L-mode discharges. As an aside, it would be interesting to see if the divertor heat-flux widths are affected by the changed magnetic field direction. But if they are, we may need to take more time to investigate this effect before pushing on with the highest ICRF power cases. Anyway, since there is a risk of damaging (i.e., melting) the embedded sensors, ramped tiles and scanning Langmuir probes in the highest-performance L-modes, this part of the experimental investigation should be performed last.

4. Resources

4.1 Machine and Plasma Parameters

- Toroidal Fields: 2.7, 4.0, 5.4, 8.0 tesla
- Plasma Currents: 0.4, 0.54, 0.8, 1.1, 1.2 MA
- Working Gas Species: D₂
- Densities: Density scans, NL04 ~ 0.2 to 2.4x10²⁰m⁻²
- Boronization: Low radiated power fractions are required, potentially requiring boronization. However, do not want to run this MP right after a boronization because of the non-conducting boron layer that tends to form on divertor Langmuir probes. This MP should therefore be executed at least one run day or more after a boronization (but when the radiated power fraction remains low).

Equilibrium configuration:

Standard lower single-null equilibria will be run (1 second flat-top, ending at 1.5 seconds), with the outer divertor strike-point sweeping (as it naturally does) across surface thermocouple sensor #5. This sensor should enter the private flux zone just before 1.5 seconds. A good example is shot# 1090806020, although the strike point could start lower on the divertor plate and sweep up the plate faster.

4.2 Auxiliary Systems

- ICRF Power, pulse length, phasing:
  - Normal Field Direction, 5.4 Tesla: 0.5 to 1.5 MW heating from D+E antennas; RF on from 0.5 to 1.5 seconds.
  - Reversed Field Direction, 5.4 Tesla: up to 4 MW heating from D+E+J; RF pulse length to be determined by ramped-tile surface temperature response.
- LHCD Power, pulse length, phasing: None
- Pellet Injection (species): None
- Impurity blow-off injection: None
- Diagnostic Neutral Beam: May piggy-back
- Special gas puffing: D₂ puffs from NINJA as required
- Cryopump: None

August 20, 2009
Non-axisymmetric Coils: Standard lock-mode compensation
Other:

4.3 Diagnostics

- Plasma profiles from edge Thomson and plasma + flow profiles from scanning Mach probes (ASP, FSP, WASP), reaching to the separatrix such that the phase velocity shear layer is recorded on scanning probes.
- Divertor heat flux ‘footprints’ as inferred from embedded sensors, IR camera and divertor Langmuir probes.
- Radiated power measurements from core and divertor bolometer arrays.
- SOL turbulence measurements from camera and fast-diode based gas-puff imaging (outer midplane, inner midplane, near x-point region).
- Divertor Langmuir probes operated with 0.5 MHz sampling rate to capture Isat fluctuations during I-V sweep.
- Time-averaged ionization profiles on the outer SOL inferred from Lyman-alpha diode arrays ($L_{\alpha}$).

5. Experimental Plan

5.1 Run sequence Plan

Table 1 below lists a set of target plasma parameters that would fulfill the experimental run plan. 10 run days would be required: 4 days with ohmic heating along, 3 days with ohmic + ICRF in normal field direction and 3 days with ohmic + ICRF and reversed field. Clearly, this aggressive plan, if fully executed, will span the FY09 and FY10 run campaigns. Research priority planning from the 2009 C-Mod Ideas Forum ([http://www.psfc.mit.edu/research/alcator/program/ideas2009/index.html](http://www.psfc.mit.edu/research/alcator/program/ideas2009/index.html)) has resulted in ‘at least 4 runs in FY09’ allocated to this MP (Edge/Divertor Group). The set of four ohmic-only run days in Table 1 would be the best place to start. It would also be good to have completed one or two ohmic + ICRF run days by the time of the November APS meeting, since this meeting affords an opportunity to coordinate research activities with NSTX and DIII-D.

All discharges would be programmed to achieve flat-top values of current and toroidal field at the indicated parameters. In low-field discharges (4 and 2.7 tesla), the start of this nominal ‘flat-top’ may be delayed in order to ramp-down the toroidal field to the target toroidal field. However, this will compromise the time available for the strike-point sweep. Therefore, it is desirable to use the newly-developed low-field startup capability to minimize the need for a toroidal field ramp-down [19].

For ohmic-only cases, discharge density would be varied most often, followed by plasma current and toroidal field. For ICRF heated cases (5.4 tesla), RF power would be varied most often, followed by discharge density and plasma current. Because the ICRF can not operate when the midplane neutral pressure is too high, some of the higher density data points in the scan will be dropped.
5.2 Shot sequence plan

We desire to set up steady-state discharge conditions for the entire duration of the flat-top so that the divertor heat-flux ‘footprint’ sweeps across the embedded thermal sensors and Langmuir probes without changing its shape. Scanning Langmuir-Mach probes would be targeted to reach the separatrix on three separate scans. Therefore, the gaps must be controlled and constant throughout the shot. At mid-to-lower plasma densities, the NINJA puffs required for the GPI system will perturb the plasma density, spoiling the constancy of the upstream conditions during the strike-point sweep. So we will need to repeat shots with and without NINJA puffs for these cases. On the first shot at each condition, we will target the probes and puff NINJA for GPI. On the repeat shot, we will optimize the probe targeting and turn NINJA off, as necessary.

**Ohmic**

A typical shot sequence for a 5.4 tesla, 0.8 MA ohmic-only run would proceed as follows:

Dial up 5.4 tesla, 0.8 MA discharge, start at lowest density, NL04 ~ 0.4x10^{20} m\(^{-2}\)
Optimize gaps and probe targeting – 2 shots
NL04 ~ 0.4 – 2 shots (first with NINJA, second without)
NL04 ~ 0.6 – 2 shots
NL04 ~ 0.8 – 2 shots
....
NL04 ~ 1.6 – 2 shots
~16 shots total (~1/2 run)
Change plasma current and repeat...

**Ohmic+ICRF**

A typical shot sequence for a 5.4 tesla (normal direction), 0.8 MA, ohmic+ICRF run would proceed as follows:

Dial up 5.4 tesla, 0.8 MA discharge, start at lowest density, NL04 ~ 0.4x10^20 m^-2
Optimize gaps and probe targeting – 2 shots
NL04: ~ 0.4
  RF: 0 MW – 2 shots (first with NINJA, second without)
  RF: 0.6 MW – 2 shots
  RF: 1.2 MW – 2 shots
NL04: ~ 0.6
...
NL04: ~ 0.8
...
NL04: ~ 1.0
...
NL04 ~ 1.2
Neutral pressure may be too high for ICRF above NL04 ~ 1.2
~32 shots total, ~1 run
  Change day, change plasma current and repeat...

For reversed-field discharges, RF power can stepped up in 1.5 MW increments.

6. **Anticipated Results**

Data collected by this MP will provide critical information on the scaling of the heat-flux footprints in C-Mod and the underlying physics that controls it. This activity is in direct support of a FY10 project milestone† and will likely support several physics publications in FY10, including preliminary presentations of the results (three posters) at the November 2009 APS meeting.

7. **References**

† Proposed US DoE Joint Facilities Milestone for FY10: “Conduct experiments on major fusion facilities to improve understanding of the heat transport in the tokamak scrape-off layer (SOL) plasma, strengthening the basis for projecting divertor conditions in ITER. In FY2010, FES will measure the divertor heat flux profiles and plasma characteristics in the tokamak scrape-off layer in multiple devices to investigate the underlying thermal transport processes. The unique characteristics of C-Mod, DIII-D, and NSTX will enable collection of data over a broad range of SOL and divertor parameters (e.g., collisionality, beta, parallel heat flux, and divertor geometry).
Coordinated experiments using common analysis methods will generate a data set that will be compared with theory and simulation.


