Asymmetric Halo Current Rotation In Post-disruption Plasmas

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Submitted in partial fulfillment of the requirements for the degree of Doctor of Philosophy in the Graduate School of Arts and Sciences

COLUMBIA UNIVERSITY

2023
ABSTRACT

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Halo currents (HCs) in post-disruption plasmas can be large enough to exert significant electromagnetic loads on structures surrounding the plasma. These currents have axisymmetric and non-axisymmetric components, both of which pose threats to the vacuum vessel and other components. However, the non-axisymmetric forces can rotate, amplifying the displacements they cause when the rotation is close to the structures’ resonant frequencies. A new physically motivated scaling law has been developed that describes the rotation frequencies of these HCs and has been validated against measurements on HBT-EP, Alcator C-Mod, and other tokamaks [1, 2]. This scaling law can describe the time-evolution of the asymmetric HC rotation throughout disruptions on HBT-EP as well as the time-averaged rotation on C-Mod. The scaling law can also be modified to include the edge safety factor at the onset of rotation ($q_{\text{onset}}$), which significantly improves its validity when applied to machines like C-Mod, where $q_{\text{onset}}$ changes frequently. The $q_{\text{onset}}$ dependence is explained by the relationship between the poloidal structure of the HC asymmetries and the MHD instabilities that drive them, which has been observed experimentally for the first time using a novel set of current sensing limiter tiles installed on HBT-EP. The $1/a^2$ and $q_{\text{onset}}$-dependence of the rotation suggest that the HCs predominantly rotate poloidally. This remains consistent with the toroidal rotation observed on HBT-EP and other tokamaks through the “Barber Pole Illusion” and the direction of rotation’s dependence on the direction of $I_p$. This scaling law is used to make projections for next generation tokamaks like ITER and SPARC, which predicts that rotation will be resonant on ITER. However, resonant effects can still be avoided if the duration of the disruption is kept short enough to prevent two rotations from being completed [3].
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Acknowledgments

No matter how much effort one puts into their work, none of it would be possible without others around to support them. My time at the Plasma Lab has been a long six years, but thanks to the friends I’ve made here and the colleagues I’ve worked with, the time has felt short and my work has been fruitful. I would first like to thank my advisor and mentor Michael Mauel, who has shaped my perspective on science in many ways. His immense enthusiasm to not only investigate physics, but to excite those around him is one of the reasons I decided to come to Columbia in the first place. His mentorship has given me a more realistic perspective of the scientific process, as well as the confidence to feel comfortable as a scientist myself. He has also taught me to value the most critical skill required to be a good scientist, the ability to communicate my thoughts well to an audience. You can solve the mysteries of the universe, but no one will listen if they can’t follow what you’re saying. I would also like to thank Gerald Navratil, who has often acted as a co-advisor for me as well. Gerry was the progenitor of my main thesis topic and has provided invaluable insight towards my investigation.

I’m grateful to Jeffrey Levesque, who in many ways has been just as much a mentor as Mike and Gerry. Jeff is by far the one I’ve bounced the most ideas off of in my time here, and he has always been receptive and willing to add in his perspective. Keeping a tokamak running smoothly is no easy feat, and I doubt anyone would deny that HBT is still operating thanks to his dedication and expertise. I would also like to thank Jim Andrello for keeping HBT from literally falling apart. It’s been hard maintaining HBT with only one engineer, but I’m grateful that Jim has been able to handle it for as long as I’ve been at the lab.

I would also like to thank my fellow graduate students, who were instrumental both in developing my knowledge of physics and making my days as a grad student more enjoyable. Ian Stewart has been with me since I first came to Columbia, and the “debates” we have had over a number of
physics topics are some of the best learning experiences I’ve ever had. I hope that wherever I go in the future there will be someone who can challenge my understanding of what I known in the same way. Todd Elder has given me as much advice as some of my mentors. Anytime I thought I had a new result, Todd would be there to help either validate my thoughts or provide much needed constructive criticism. I am also grateful for the company he’s provided at the lab, being around more often than anyone else. I’m thankful to Rian Chandra for helping me shore up my knowledge in some of my weaker subjects and for taking over the social responsibilities in the department; carrying on a torch passed down through the Plasma Lab. I would also like to thank the rest of the HBT crew: Yumou (William) Wei, Boting Li, David Arnold, and Nigel DaSilva, for being around and helping me carryout experiments and install new diagnostics.

I’m also thankful for Mel Abler and Zane Martin for taking me in when I first arrived and showing me how much fun New York can be. A thesis is long experience, and the downtime is just as crucial to finishing as the work itself.

I would also like to thank Alex Tinguely for helping me with my work on C-Mod. Alex’s enthusiasm made the whole collaboration process seamless, and I’m excited to continue working with him in the future.

I also feel it necessary to thank Montserrat Fernandez-Pinkley (Montse). The bureaucracy of graduate school is a nightmare all on its own, and I’m grateful for the time you’ve spent making it easier for all of us.

Lastly, I would like to thank my family for supporting me throughout this entire process. It’s been tough living so far away, but you’ve been there for me non-the-less.

This work was funded by the United States Department of Energy Grant DE-FG02-86ER53222 and was supported by the HBT-EP team.
Chapter 1

Introduction

The development of fusion energy reactors is now approaching the point where net energy can be produced. However, next generation tokamaks like ITER and SPARC will have to operate at higher currents than any modern device in order to achieve high performances. As a result, their disruptions will be accompanied by stronger electromagnetic (EM) forces on their walls and other components. On existing lower-current devices, these forces are not strong enough to cause significant damage, however higher-current machines are more vulnerable, and therefore there is a need to determine what exactly these forces will be. This thesis helps to understand the extent of these forces through the development of a physically motivated scaling law for the rotation of halo currents (HCs), which are partially responsible for the EM loads. It is found that the HCs, and subsequently the displacement resulting from their loads, will likely rotate on ITER at frequencies that are resonant with its walls, potentially amplifying the destructive effects.

This chapter introduces many of the topics relevant to characterizing halo current forces and their rotation. Section 1.1 discusses fusion energy and the requirements to produce net energy. Section 1.2 introduces the concept of the tokamak, which was developed to meet these requirements. Section 1.3 details disruptions on tokamaks and how they can damage the device. Sections 1.4 and 1.5 discuss what is known about the sources of electromagnetic loads (HCs) as well as how they rotate. Lastly, section 1.6 outlines the remainder of the thesis.
1.1 Fusion Energy

The development of fusion energy has many motivations: there is a need to reduce the emission of carbon into the atmosphere and temper climate change, to enhance energy security and reduce the reliance on oil and natural gas, to rely primarily on an abundant fuel source that replenishes itself faster than it is consumed [4], and to provide an energy source that can scale with the energies needs of a growing civilization. Fusion energy has the potential to satisfy all these needs. The roots of fusion energy date back to the early 20th century, when fusion was first discovered to be the primary mechanism responsible for the energy produced by the Sun; the celestial body in our solar system that provides all of the energy we use today, albeit, indirectly. Tracking the source of solar, wind, and even fossil fuel power inevitably leads back to the sun, but only after (at least) several steps of inefficient losses of energy. Fusion energy avoids relying on these relatively inefficient energy sources and instead aims to harness the energy generated by fusion directly, by effectively creating a man-made star here on earth, hence the name Stellarator (star-machine) given to first fusion device conceptualized in the United States.

Fusion produces more energy per unit mass of fuel (specific energy), than any other existing fuel source. The specific energy (SE) associated with fusion is about a factor of 100 times larger than that of fission and an even greater 1,000,000 times larger than that from any chemical reactions (i.e. coal, fossil fuels, natural gas). The large disparity between fusion and chemical SEs is a result of the difference in scales between the two types of reactions. Chemical reactions occur on the (small energy density) molecular scale while fusion reactions occur on the (high energy density) nuclear scale. Fission also exists on the nuclear scale, but has a smaller SE than fusion. The large SE associated with fusion makes it one of the most efficient known sources of energy and allows for the energy generation to scale well into the future.

Both fusion and fission release energy by taking advantage of the relationship between energy and the rest mass of an atom. Unlike the properties of mass on a macroscopic scale, the rest mass of an atom is not equivalent to the summed rest masses of its component parts (protons, neutrons, and electrons). Instead, the rest mass is always lighter than that of its constituents, and it is the variance in exactly how much lighter it is (called the binding energy) between different nuclei that allows fusion and fission reactions to release energy. If one atom were to transmute into another through some reaction, the new atom would have a different binding energy (BE) and the reaction would
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![Diagram of fusion reaction](image)

Figure 1.1: Cartoon of the fusion of a Deuterium and Tritium nuclei to create a high energy Helium nuclei and neutron.

Fusion and fission are the two ways in which atoms can be transmuted. Fusion is when two particles fuse together to generate one or more new particle(s) (see Figure 1.1) and fission is when one atom splits into multiple new particles. Both reactions can either capture or release energy depending on the reactants (original particles) and products (new particles). Figure 1.2 illustrates the circumstances under which both reactions capture or release energy. The curve shown is the BE associated with the number of nucleons (protons+neutrons) and is notably non-monotonic with an inflection about Iron (Fe). Fusion generally increases the number of nucleons and fission decreases it, denoted by the solid and dashed arrows respectively. From this curve, it can be seen that fusion will release energy if it fuses particles into products with up to 56 nucleons and fission will release energy if it splits atoms to products down to the same amount. The reason for why fusion has a much higher SE can also be seen in Figure 1.2. The slope of the curve for atoms lighter than Iron is significantly greater than it is for those that are heavier, implying that the difference in BE between the reactants and products is much larger for fusion than it is for fission.

Although fusion has the capability to generate more energy than fission (and far more than chemical), fusion requires more demanding conditions to occur. The fusion of two nuclei requires them to get close enough to each other such that the strong force can overcome the electric repulsion between the two positively charged nuclei. The electric field imposes a potential barrier that
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Figure 1.2: Binding energy curve [5] as a function of number of nucleons. Fusion paths along this curve are denoted by solid arrows and fission paths by dashed arrows. Paths that release energy are in red, while paths that capture it are in blue.

separates the two nuclei and requires quantum tunneling (a statistically improbable event) to pass through. In order for the two nuclei to get close enough such that quantum tunneling can take place at a significant rate, the kinetic energy associated with either of the nuclei needs to be very large. The average kinetic energy in the system, or temperature, then needs to be hot enough or fusion reactions will scarcely occur. This imposes the first constraint for fusion to occur; the system needs to be very hot (high temperature, $T$). This is not the only constraint. The system could be very hot, resulting in a high frequency at which fusion reactions could occur, but if that frequency is short compared to how long the heat actually stays in the system, then most particles will not retain the kinetic energy they need long enough to fuse before leaving the system. This imposes the second constraint for fusion to occur; the confinement time of the heat needs to be large compared to fusion frequency (high energy confinement time, $\tau_E$). The final barrier is that there needs to be enough particles in the system. The fusion rate within the system will be faster the more particles there are in it. This imposes the final constraint for fusion to occur; the density of particles in the system needs to be large enough (high density, $n$). Together, these three constraints form what is known as the Fusion Triple Product\(^1\), $nT\tau_E$ [10], which needs to be large enough such that the fusion reactions in the system are generating more energy than is being lost from the system.

\(^1\)The exact derivation of the Fusion Triple Product is more complex than the qualitative explanation provided here.
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Achieving a large enough triple product in practice is a daunting challenge. The minimum temperature alone needs to be in the range of keV \((10,000,000 \ °C)\), which is around 1,000 times hotter than the surface of the Sun and comparable to the temperature at its core. Additionally, this heat needs to be confined for a significant amount of time, which contrasts the nature of hot systems to quickly lose particles. The Sun somewhat cheats to meet this criterion, as the mechanism it uses to retain particles is its own gravitational field. This mechanism is not practical to reproduce, so instead fusion energy efforts have been devoted to finding alternative methods of achieving high sustained triple products, like through the use of magnetic confinement, as discussed in the following section.

1.2 The Tokamak

The closest approach to putting fusion energy on the electrical grid is through using a device called a Tokamak, initially developed in the mid-20th century in the USSR. The device makes use of magnetic fields in order to confine heat and particles and belongs to a class of devices that exploit magnetic confinement. Other types magnetic confinement devices include stellerators, magnetic mirrors, and pinches among other things. The presence of magnetic fields confines the fusion fuel because the high temperatures require it to be in a plasma-state. A plasma is a ionized gas where the electrons are stripped from the ions and both can move freely independent of each other. The freedom of the ions and electrons, as charged particles, allow them to be influenced by electromagnetic fields, even if the system itself is approximately neutral. Through the Lorentz force, the magnetic field traps charged particles in orbits that gyrate around magnetic field lines. If the radius of these gyro-orbits is significantly smaller than the size of the system (via a strong enough magnetic field), then the ionized fusion fuel will remain trapped within the system, leading to long confinement times.

Tokamaks exploit the confinement properties of a plasma in a magnetic field and eliminate the loss of particles in the direction parallel to the field lines (FLs) by wrapping them into toroidal rings such that they bite their own tails, as illustrated in Figure 1.3(a), but they also twist in the azimuthal (poloidal) direction like that shown in Figure 1.3(b). Without the twist, the ions and electrons would drift across the field lines, accumulating on the top/bottom of the vessel and significantly reducing the confinement time. The degree of “twistiness” in the field lines is characterized by the safety factor \(q\), the number of toroidal transits a FL makes as it completes a single poloidal transit. The
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Figure 1.3: Closed loop magnetic field line structures around a toroidal plasma (purple). (a) Purely toroidal field line that bites its own tail after a single revolution. (b) Twisted field line that bites its own tail after making three complete revolutions ($q = 3$).

The lowest order dynamics in a tokamak are governed by its field lines and as a result their exact geometry significantly effects everything from confinement to stability. In a tokamak, the FL-geometry is generated in two parts. First, the toroidal (and stronger) component is generated using a toroidal array of toroidal field (TF) coils, like those shown in Figure 1.4. Second, the poloidal (and weaker) component is generated via a toroidal current that exists within the plasma itself. The toroidal current in the plasma is induced through transformer action between the plasma and a solenoid located (co-axially) at the center of the torus (see Figure 1.4).

The poloidal field generated by the plasma current ($I_p$) will vary significantly within the plasma, as the amount of current enclosed within a surface in the poloidal plane (and consequently the poloidal field at the edge of that surface) will vary as the surface is expanded in the direction of the minor radius. As a result, the safety factor will also vary within the plasma. Contours of constant
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Figure 1.4: Cartoon of the fundamental sources of magnetic fields that generate the field lines in a tokamak; the TF coil current, the OH coil current, and the plasma current ($I_p$).

Figure 1.5: Cartoon of nested flux surfaces in a circular large aspect ratio tokamak.


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safety factor within the plasma (like those seen in Figure 1.5) are referred to as flux surfaces and are also representative of contours of constant pressure \((P)\) as well as the enclosed toroidal/poloidal magnetic flux \((\psi)\), hence the name. These contours extend toroidally around the machine creating a surface in 3D-space. The nested nature of these surfaces makes them convenient choices for a pseudo-radial coordinate that can be defined with respect to either the toroidally \((\psi_\phi)\) or poloidally \((\psi_\theta)\) enclosed flux. The safety factor at each surface can be defined with respect these fluxes through the equation

\[
q = \frac{\partial \psi_\phi}{\partial \psi_\theta}
\]

which in the specific case of a circular cross-section tokamak with high aspect ratio can be approximated as

\[
q_{circ} \approx \frac{B_\phi r}{B_\theta R}
\]

These surfaces are also the surfaces over which individual field lines are confined to. This implies that a particle gyrating about a field line on this surface will, to lowest order, be confined to it. As result, the confinement capabilities of a tokamak come from the existence of its flux surfaces, and if they were to be disturbed or destroyed then the confinement could be degraded or even completely lost.

One of the most significant challenges to tokamak fusion is the latter case, where confinement is completely lost. These events are difficult to control once they occur, as the plasma will begin to dissipate on timescales much shorter than that at which a control system can typically act on. When these events are unrecoverable, they are referred to as disruptions. This thesis investigates the dynamics of these disruptions.

1.3 Disruptions

A disruption is the abrupt termination of a plasma due to loss of control and confinement. Through a cascade of events, many of the plasma’s closed flux surfaces are destroyed, the confinement of heat and particles is lost, and most of the plasma’s stored energy is released. Disruptions occur on fast timescales (on the order of \(1 \to 10\) ms on current devices) relative to the equilibrium, which makes
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them difficult to respond to in real-time with a plasma control system (PCS). These events are generally considered harmful, and a significant research effort is underway to either prevent them from occurring or mitigating their deleterious effects in the event that they occur.

Most of the work done in this thesis addresses the dynamics of disruptions and how they may induce damaging currents in the structures surrounding tokamaks. This section discusses the basic anatomy of a disruption, their causes, and the dangers they pose to the tokamak device.

1.3.1 The Anatomy of a Disruption

A disruption can be broken down into three primary events: (i) the thermal quench (TQ), (ii) the current quench (CQ), and (iii) the motion of the plasma into the wall. The first two capture the expulsion of the plasma’s stored energy while the third captures the changes in plasma geometry. Each of these are a consequence of the loss of confinement and there is some variance in the order in which they can occur in.

The thermal quench is the event over which most of the plasma’s stored thermal energy is lost and is often the fastest (and initiating) event in the disruption. This event occurs directly in response to the loss of confinement as heat is transferred convectively (or initially radiatively) out of the plasma faster than it can be replenished. The timescales associated with this process are fast relative to the other events because they are associated with the fast thermal transport along the field lines [6]. This is facilitated by the destruction of the flux surfaces, as the field lines are no longer confined to a single coherent surface and a single field line can instead travel from the core to the edge of the plasma. Predicting an approximate duration of the TQ for a given disruption is difficult to do however, and is still an open area of research [7, 8]. In response to the fast transport of heat, the temperature (stored thermal energy of the plasma) drops considerably. The pressure in the plasma drops with the temperature as well, but the current does not immediately drop and the density often doesn’t see as drastic of a change either [9]. With the increased transport provided by the destruction of the flux surfaces, the temperature and current profiles also become broader (flatten). However, the stored magnetic energy within the plasma is often conserved on the timescale of the TQ [11], and as a result the plasma current increases temporarily as the current profile broadens consistently with the equation

\[ W_B \sim \frac{1}{2} LI^2 \]  

(1.3)
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Figure 1.6: Example of a thermal quench at the onset of a disruption on JET. Reproducing Figure 4 from Ref [9], a relatively early review paper detailing disruptions. The TQ in this example is illustrated by the drop in the average electron temperature ($T_{e0}$) and is divided into two stages; an initial partial-TQ (stage-1) and a final full-TQ (stage-2). The spike in current associated with the broadening of the current profile follows the stage-2 TQ.

where $W_B$ is the magnetic energy stored within the plasma current and $L$ is the self-inductance of the plasma. An example of this occurring in response to the TQ for a disruption on JET [9] is shown in Figure 1.6. As can be seen for this disruption, the profile broadening (and current spike) does not necessarily occur simultaneously with the TQ, although they are often close. Beyond the TQ, however, the destruction of the flux surfaces is not necessarily permanent and/or complete. The confinement of runaway electrons during some disruptions is evidence that some semblance of confinement is still present following the TQ [12]. Whether this confinement is the result of the flux
1.3. DISRUPTIONS

Figure 1.7: Disruption characteristics on C-Mod and HBT-EP. Disruption curves of (a) $I_p$, (b) centroid position ($Z_p$ for C-Mod and $R_p$ for HBT-EP), (c) cross-section area, (d) $q_{95}$, and (e) sample HC signals. C-Mod curves are in black while HBT-EP profiles are in red (and dashed). Signal units are normalized for comparison between machines except in the case of $q_{95}$, and time-axis is normalized to the current quench time (defined separately for the two disruptions). (f) LCFS of the C-Mod plasma at the three times shown in the temporal profiles, denoted by the vertical lines with corresponding colors.

surfaces reforming or just not being completely destroyed in the first place remains to be seen.

The current quench is the event over which most of the plasma’s stored magnetic energy (in the form of the plasma current) is lost and is often one of the slowest events in the disruption. The onset of the CQ occurs in response to the cooling of the plasma during the TQ. As the plasma cools, it becomes significantly more resistant ($\eta \propto T_e^{-3/2}$) and the loop voltage ($V_L$) generated by the Ohmic transformer action can no longer sustain the pre-disruption (flattop) current. As a result, the current begins to drop like that shown in Figure 1.7(a) for example disruptions on HBT-EP and Alcator C-Mod. However, the self-inductance of the plasma prevents the current from dropping linearly with (and as fast as) the resistance. So instead the current drops on a timescale set by the $L/R$-time of the plasma, described by

$$
\tau_{L/R} = \frac{L}{2\pi R_0 \eta(T_e)/S} \quad \text{(1.4)}
$$
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where $L$ is the self-inductance of the plasma, $R_0$ is the major radius, $\eta(T_e)$ is the temperature-dependent resistivity, and $S$ is the plasma poloidal cross-sectional area. This timescale is often long compared to other timescales associated with the disruption and therefore the disruption duration is defined by the duration of the current quench (CQ-time). This timescale is typically on the order of $1 \rightarrow 10$ ms and can vary from machine-to-machine and even shot-to-shot. This variation is dominantly a result of variances in the plasma cross-section, however variance in the temperature can have an effect as well. As the current drops, the changing poloidal flux can drive a toroidal current in the vacuum vessel and lead to electromagnetic loads on the vessel. These are discussed in more detail in Section 1.3.3.

The motion of the plasma into the wall is an event associated with the loss of radial or vertical force balance. This can begin either before the TQ and cause the disruption, or after the TQ in response to the drastic changes in plasma parameters that accompany it [7]. The force balance that is broken and the direction that the plasma moves in typically depends on whether the plasma is circular or elongated. Circular plasmas (like HBT-EP’s) have a circular poloidal cross-section and are prone to horizontal instabilities. When they disrupt, the hoop force associated with the plasma pressure drops and the $J \times B$ force associated with the vertical field overcomes it, causing the plasma to move horizontally. Elongated plasmas (like Alcator C-Mod’s) have a shaped (non-circular) poloidal cross-section. This shaping is facilitated via toroidal coils (called shaping coils) that run current above and below the plasma. The first force balance to break in elongated plasmas is usually that between the forces on the plasma from the top and bottom shaping coils, causing the plasma to move vertically, like what is shown in Figure 1.7(e). Elongated plasmas are always vertically unstable and require constant feedback from a PCS to maintain equilibrium [13, 14]. If the break in vertical force balance occurs prior to the TQ, this is referred to as a Vertical Displacement Event (VDE). If the break does not occur prior to the TQ, it will inevitably occur following it, as the quenching of the plasma parameters will provide the necessary kick.

Regardless of the direction the plasma moves in, it will always move into the wall, becoming limited if it wasn’t already. The timescale on which the plasma moves into the wall is dominantly set by the magnetic diffusion time of the wall ($\tau_{wall}$) [12] and the average plasma-wall separation, as the motion of the plasma into the wall will induce currents in the vessel that then act to repel the plasma. This motion into the wall will then shrink the poloidal cross-section of the plasma as shown in Figure 1.7(e). The edge safety factor of the plasma $q_{edge}$ at this time will then increase or decrease
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Figure 1.8: Illustration of how overlapping instabilities generates stochastic field line regions. (a) Cartoon of the deformations to the flux surfaces from an island-like (green) mode and a kink-like (blue) mode. Stochastic region is generated (red) where the two sufficiently overlap. (b) Poincare plot of two overlapping islands chains viewed as a poloidal cross-section generated from Ref [16].

(typically decreasing) depending on the relative changes in $I_p$ and $S$, approximately consistent with equation (1.2). The timescale of the change in $I_p$ is set by the CQ-time, which itself depends on the $T_e$ and $S$. As a result, the relative changes in $I_p$ and $S$ will primarily be a function of the post-TQ $T_e$, the pre-TQ $S$, and $\tau_{\text{wall}}$.

1.3.2 The Causes of a Disruption

The lead up to a disruption is often characterized by a complicated cascading of events, each of which either directly or indirectly leading to the loss of confinement associated with the onset of the disruption. While the destruction of the flux surfaces is always the inevitable result that completely dissipates the thermal energy in the plasma, it is not necessarily what causes the disruption in the first place. The destruction of the flux surfaces is facilitated via the growth of magneto-hydrodynamic (MHD) instabilities. The destabilization of these instabilities can either be the initial cause of the disruption or can be the result of other instabilities; most often either radiative and vertical instabilities. A detailed report on the most frequent causes of disruptions can be found in Ref [15].

MHD instabilities are inevitably what destroy the flux surfaces. These are the instabilities associated with the conductive and fluid-like nature of the plasma, and they draw primarily on the
free energy stored within the current and pressure gradients. These instabilities manifest either as kink-like (blue in Figure 1.8(a)) or island-like (green in Figures 1.8(a) and 1.8(b)) perturbations to the flux surfaces. When multiple perturbations co-exist and sufficiently overlap [17], this generates the stochasticity (red in Figures 1.8(a) and 1.8(b)) in the flux surfaces that is representative of their destruction [12]. MHD instabilities usually draw most of their energy from gradients in the current and pressure, and as such their onsets are typically associated surpassing limits in current and pressure respectively. The current limits manifest more interpretably in limits on the safety factor \( q \), which is approximately inversely proportional to the current. This puts lower limits on \( q \) as opposed to upper limits on \( I_p \), which explains why higher values of \( q \) are more stable, hence the name “safety factor”. The most notable limit on \( q \) is the \( q_{\text{edge}} = 2 \) limit [18], where the edge \( q \) dropping below 2 will always result in the growth of a \( m/n = 2/1 \) kink mode that leads to the TQ.

The pressure limits are more extrapolable when normalized to the magnetic pressure as

\[
\beta = \frac{P}{B^2/2\pi}
\]  

which is also a more convenient definition as the maximum achievable \( \beta \)-limit illustrates how the maximum fusion power generated by a reactor scales like \( B^4 \). There are many different limits in \( \beta \) that beyond which some instability begins to grow, but the two most notable are the Neoclassical Tearing Mode (NTM) [19, 20] \( \beta \)-limit and the Troyon [21] \( \beta \)-limit. The former is a softer limit as the growth of NTMs won’t necessarily disrupt the plasma, however the latter is a hard limit, as surpassing it will lead to the growth of pealing-ballooning modes that disrupt the plasma. MHD instabilities can also become destabilized via error fields. Under nominal conditions, these error fields are screened out by the plasma, preventing them from penetrating to a flux surface inside the plasma where they might perturb the surface and make it easier for the instability to grow. However, sufficient screening requires a large enough plasma density \( n_e \) and/or rotation, and as a result there are minimum density limits that exist as well.

**Radiative instabilities** (radiative collapses or density limits) can be an indirect cause of the TQ. They are the instabilities associated with power imbalance in the plasma. For a plasma to maintain thermal equilibrium, the power input into the plasma needs to match the power output. Most of the power is lost from the plasma either through conductive, convective, and radiative means. If any of these loss mechanisms begin to outpace the power input, then the plasma will start to cool.
Figure 1.9: Illustration of the vertical forces acting on the plasma from the divertor coils for a double null plasma on MAST [24]. The divertor coils are highlighted in yellow and the green arrows represent the average directions (and magnitudes) of the associated $J \times B$ forces.

down until they match again. For conductive and convective losses, this is a stable process, however for radiation this not necessarily the case. The line-radiation from low-Z impurities increases with lower temperatures, so as the plasma cools down, it is possible for it to radiate more [15]. This can lead to a sustained cooling of the plasma (a partial TQ) that can quickly drop the plasma pressure over time. The colder edge plasma will cool faster than the core, and so a large pressure gradient will eventually build up until an MHD mode becomes unstable and subsequently destroys the flux surfaces, leading to the full TQ. These instabilities can trigger when the radiated power ($P_{\text{rad}}$) becomes larger than the input power ($P_{\text{in}}$), so they become a concern if the plasma density or impurity concentration in the plasma become too large [22, 23].

*Vertical instabilities* (VDEs) are another indirect cause of the TQ, and belong to a general class of positional instabilities (briefly discussed already in Section 1.3.1). They are the most common manifestation and are specific to elongated plasmas. These instabilities are associated with the force
1.3. DISRUPTIONS

balance between the $J \times B$ forces that the shaping coils apply on the plasma. The shaping coils exist
to stretch and pull on the plasma vertically, as shown in Figure 1.9. Nominally, the forces from the
two coils cancel out so the plasma stays centered, but the equilibrium is always unstable and requires
constant feedback. As a result, if the plasma is spontaneously displaced far enough towards one of
the coils such that the PCS cannot respond in time, it will continue to be pulled towards that coil.
As the plasma moves towards the coil, it will eventually hit the divertor tiles (the wall) and begin
to shrink in size. In doing so, the shrinking plasma size will drop $q_{edge}$ until it reaches an unstable
value (often $q_{edge} = 2$), triggering an MHD instability, destroying the flux surfaces, and starting the
TQ.

1.3.3 The Dangers of a Disruption

Disruptions are a concern because they interfere with consistent plasma operation and also because
they may damage the device itself. When a plasma disruptions, it ejects its stored energy, which
is redistributed to the walls that surround it. This transfer of energy from the plasma to walls can
involve mega-Joules of energy being deposited onto the walls in as little as hundreds of microseconds,
which can damage the device. The ultimate goal of studying disruptions is to prevent these
potentially dangerous events from occurring, and failing that, to mitigate the damaging effects. The
most concerning effects come from (i) the heat deposition onto the walls, (ii) relativistic runaway
electrons striking the walls, and (iii) the electro-magnetic loads shaking the walls. The latter of
these being the focus of this thesis.

Heat deposition is the primary mechanism through which the plasma’s stored thermal energy is
transferred to the walls. In modern high-performance tokamaks the stored thermal energy within a
plasma can reach over 1 MJ, most of which needs to be distributed to the wall over the short time-
period of the TQ. These heat loads, when large enough, can melt the plasma-facing components
[9]; redistributing impurities throughout the vessel, introducing toroidal asymmetries to the walls,
and impeding the performance of subsequent shots. In un-mitigated cases, most of this heat is
deposited onto the wall locally via convection (where the plasma physically makes contact with the
wall), which introduces far larger heat loads than if the heat was distributed globally. Attempts
to mitigate these effects involve the injection of impurities with the intention that they will radiate
most of the stored energy evenly throughout the vessel.

Runaway electrons (REs) are one of the two primary mechanisms through which the plasma’s
stored magnetic energy is transferred to the walls. Modern high-performance tokamaks operate with plasma currents as high as several mega-Amps and self-inductances on the order of micro-Henrys, implying that the stored magnetic energy \( (W_B = 1/2LI^2) \) that needs to be dissipated is on the order of several hundreds of kilo-Joules. REs can dissipate a significant fraction of this energy locally with spot sizes on the order of 10 cm \([25–27]\) over intermittent spikes on the order of 10 \( \mu s \) long that persist for several milliseconds \([25]\). Runaway electrons are a group electrons that are accelerated by the inductive drive of the CQ beyond velocities that collisions can normally impede. These REs reach relativistic speeds before synchrotron radiation limits further acceleration, and can carry up to as much 50\% the pre-disruption current \([25]\). The REs, originally born within the plasma, eventually leave as a localize beam of current that strikes some region of the wall, potentially melting the strike-zone. These beam strikes can damage plasma-facing actuators/diagnostics and potentially pierce through the first wall. Attempts to mitigate these effects involve methods of deconfining the runaway beam so that it exits the plasma and strikes the wall before building up a sufficiently large current.

Electro-magnetic (EM) loads are the second mechanism responsible for transferring the rest\(^2\) of the stored magnetic energy to the walls. In contrast to REs, which transfer the energy purely convectively through a beam of electrons, EM loads transfer the energy partially convectively and partially inductively by transferring large currents into the walls. While in the walls, these currents apply \( J \times B \) forces in the presence of the large magnetic fields. These currents can then displace the current-carrying structures surround the plasma, possibly damaging them. Attempts to mitigate these effects involve active tailoring of the CQ-time so as to minimize the inductive current drive simultaneously with the contact between the plasma and the wall. This thesis focuses on the possible amplification of these EM loads via the resonance effects associated with asymmetries in the loads. The following section discusses the EM loads and the currents that apply them in more detail.

1.4 Electro-magnetic Loads and Halo Currents

This section further details the electro-magnetic loads present during disruptions and the currents that facilitate them. The currents appear in the walls via one of two sources: eddy currents or halo currents. Eddy currents are those that are driven in the wall inductively and halo currents are those

\(^2\)Some magnetic energy is actually converted Ohmically to thermal energy, but this is a relatively small amount.
that are transferred through direct contact between the plasma and the wall. A cartoon visualizing the inherent differences between the two is shown in Figure 1.10. Both of these currents sources also exist prior to the disruption, although they are significantly weaker in the absence of the driving mechanisms present during the disruption. The first two subsections discuss the sources of eddy and halo currents respectively, while the final subsection discusses the role these currents play in applying EM loads.

### 1.4.1 Eddy Currents

Eddy currents (eddies) are currents driven in the wall inductively in response to changing magnetic flux. This change in the magnetic flux can be associated with the global motion of current (global plasma motion), the local motion of current (local plasma motion), or changes in the magnitude of the plasma current. When they respond to the motion of current, the kinetic energy associated
1.4. ELECTRO-MAGNETIC LOADS AND HALO CURRENTS

with that motion is partially transferred to the eddies, which then shape themselves and act to repel the plasma and impede its motion. When they respond to the changes in the magnitude of the current, the magnetic energy associated with that current is partially transferred to the eddies, which then shape themselves and act to minimize the loss of total current between the plasma-wall system. During the disruption, the plasma both moves and quenches its current, so both eddy responses are present. However, the eddy drive associated with the CQ is often far stronger and dominates the eddy response in the wall [28]. In order to counter the drop in plasma current (which is mostly in the toroidal direction), strong co-$I_p$ toroidal eddy currents are formed with toroidal symmetry throughout the vessel. The magnitudes of these eddies will depend on the magnitude of the pre-disruption plasma current and the CQ-rate. The larger the current and faster the current quench, the stronger the $\dot{B}$ drive, and subsequently the stronger the eddies. Eddy currents can have poloidal and asymmetric components associated with them due to changes in the plasma’s diamagnetism, asymmetries in the plasma motion, asymmetries in the machine geometry, and the presence of rotating/growing 3D MHD instabilities, but the magnitudes of the rotating components of these currents are often significantly weaker, considering the far weaker current drive relative to that of the CQ.

1.4.2 Halo Currents

Halo currents (HCs) are currents driven in the wall via direct contact between the plasma edge and the plasma-facing components (PFCs) of the wall. The theory for them is less developed than for their eddy current counterparts, as the physics involved considers both the open-field line region of the plasma (a poorly understood region on its own) and the complicated plasma sheath physics associated with plasma-wall contact. In contrast, eddy current physics is relatively well understood and can be modeled approximately using coupled circuit equations between the plasma and the wall [29]. The disparity in knowledge between the two is partially due to the differences in complexity, although another significant factor is that eddy current physics also plays a large role in flattop plasma operation, while the same has yet to be shown for halo currents (despite efforts to show otherwise [30, 64]). A lot of work has gone into developing the underlying physics behind halo currents based on both experiments and theory.

The classical definition for halo current is a current whose path involves both the plasma and the wall (although a zoo of more subtle definitions has seen some development to differentiate between
1.4. ELECTRO-MAGNETIC LOADS AND HALO CURRENTS

Figure 1.11: Reconstruction of the plasma (pink) and its flux surfaces (blue) for a sample disruption on C-Mod that illustrates the poloidal cross-section of the halo-like paths of the HCs (orange).

different types of plasma-wall currents [31, 32]). The name halo current is a reference to the halo-like way in which the current encloses a poloidal cross-section of the plasma, like that shown in Figure 1.11. *Scraper-off layer current* (SOLC) is another name they go by, although it is more appropriate to describe halo currents as a subset of SOLCs instead. SOLCs are more specifically currents in the scrape-off layer (open-field line region of the plasma), but the open field-line nature of the region implies that these currents will end up passing through the wall as well.

The paths that the HCs take has been investigated intensely [33, 34, 37, 56, 58], as the distribution of currents in the walls will effect both their magnitude (based on the resistance of the path), their direction (to be crossed with the magnetic field), and the locality of their forces. These paths will depend on their medium; the plasma or wall. While in the plasma, the HCs are mostly confined to run parallel to the field-lines (see Figure 1.10). This is because of the low plasma pressure present in the SOL. The pressure gradient force is too small to support any perpendicular current and so the current runs parallel to the field lines. Inside the wall the paths of the currents can get significantly more complicated. In some cases, like on Alcator C-Mod [37], the vessel acts as a poloidal short
1.4. ELECTRO-MAGNETIC LOADS AND HALO CURRENTS

and currents run primarily poloidally, like in Figure 1.10. However, in other cases like on HBT-EP where the vessel and PFC geometries are more complicated, the paths within the walls can be a daunting challenge to diagnose. The sometimes awkward paths of currents in the walls can also influence their structure at the PFCs when rotating asymmetries are involved, as the path through the wall that the current wants to take may be dependent on the phase of the asymmetry [12].

The halo current drives during the disruption can be divided into symmetric and asymmetric drives. The symmetric component of the HCs has been observed and characterized on many tokamaks [28, 28, 34, 35, 45, 54], with theory developed to explain it [33, 56, 58]. These HCs come from a combination of the CQ, current profile broadening, and enhanced plasma wall contact. As the plasma current quenches, the self-inductance of the plasma will redistribute the remaining current more uniformly throughout the plasma, broadening the plasma current density \( J_p \). This broadening of \( J_p \) will introduce more current to the very edges of the plasma. When the last closed flux surface (LCFS) of plasma then comes in contact with the wall (if wasn’t already), then a low resistance (compared to the SOL) path for the current to travel from the plasma to the wall is formed, and the larger edge current associated with the broadened \( J_p \)-profile (from the \( \partial I_p / \partial t \) drive) can move into the walls. When the motion of the plasma into the wall is toroidally symmetric, this produces symmetric HCs in the walls. This interpretation of the symmetric HC drive is supported by the observed correlation of peak HC magnitudes with the pre-disruption current [7, 28, 37, 45, 52] and the plasma becoming limited [37]. The temporal correlation specifically to \( \partial I_p / \partial t \) is shown more explicitly in Sections 5.1.2 and 5.2 for HBT-EP and Alcator C-Mod respectively.

Asymmetric HCs have been observed just as frequently as their symmetric counterparts [37, 38, 45, 46, 48–50, 53]. The possible asymmetric HC drives are more numerous than that for the symmetric HC drive, and can be subdivided into static and rotating asymmetries. The static asymmetries are sourced externally from the plasma, while the rotating asymmetries are born from the plasma itself (the only source of rotation). The static asymmetries include machine geometry and error field asymmetries, both of which effectively perturb the symmetric drive by introducing asymmetries in the plasma-wall contact. The machine geometry asymmetries apply this perturbation via the wall side and the error field asymmetries from the plasma side (as the error fields bend the plasma). The rotating HC asymmetries (RHCAs) appear to be related to MHD instabilities, which are sometimes referred to as asymmetric VDEs (AVDEs) because of how they perturb the vertical motion of the plasma. A connection between the two has been recognized since the 1990s [36, 37], their temporal
correlation has been observed previously [41, 47, 48, 53], and Section 5.1.1 shows their spatial correlation for the first time. RHCAs are also observed to grow after $q_{\text{edge}}$ falls below some threshold amount (often $q_{\text{edge}} = 2$ [47, 48] but sometimes $q_{\text{edge}} = 1$ [48] or $3$ [2]), which is consistent with the destabilization of an edge MHD mode. The physical mechanism responsible for the asymmetric currents they introduce has been an open topic of debate for the last decade [12, 31, 32, 57, 59, 61–63, 66]. Most theories fall into the camp that the enhanced PWC brought on by the perturbed plasma edge is siphoning either current from the walls into the plasma or vice versa. The two most common examples of explanations are the ATEC [62] and WTKM [32] models. The *Asymmetric Toroidal Eddy Current* (ATEC) model predicts that the enhanced PWC is causing the eddy currents to short from the walls through the plasma in an asymmetric manner. This would effectively lead to an asymmetry in the eddies that is facilitated via HCs; a notable characteristic of which being that the asymmetries would be purely unipolar. The *Wall Touching Kink Mode* (WTKM) model contrasts this by predicting that the enhanced PWC is driving more current into the wall from the plasma, and that this exchanged current is associated with the current sheet corresponding to the MHD mode driving the perturbation. The current entering the wall can then be positive or negative depending on the phase of the perturbation, suggesting that the asymmetry is bipolar. Both models are consistent with the phase relationship between observed HC perturbations and MHD mode phases [32, 48, 53], which makes them difficult to distinguish experimentally. It is also possible that the effects of both mechanisms are present and comparable to each other.

The rotation of the RHCAs is another open area of discussion [32, 73, 110]. Attempts to understand the physics of the rotation through theoretical means have been proposed [12], but have not been experimentally verified. To date, only an empirical prediction for the rotation frequency of asymmetric HCs has been proposed using observations from multiple mid-to-large scale tokamaks [3]. While the existing projections are accurate enough to describe the rotation on the tokamaks included in that database, it under-predicts the frequencies of the fast rotating HCs observed on HBT-EP by more than an order of magnitude. Further details on the existing scaling law are provided in Section 1.5 and the development of a new physically motivated scaling law that captures the rotation on all machines observing RHCAs is provided in Chapter 6.
1.4. ELECTRO-MAGNETIC LOADS AND HALO CURRENTS

![Goldilocks Curve Diagram](image)

Figure 1.12: Cartoon of the Goldilocks curve often used to describe the trade-off between eddy and HC loads as a function of the CQ-time $\tau_{CQ}$. The blue portion of the curve is dominated by eddy forces, while the green portion by HC forces. The red horizontal line represents some critical structural threshold and the yellow shaded region represents the window in $\tau_{CQ}$ in which the loads are kept below this threshold.

1.4.3 Electromagnetic Loads

Coexisting with the eddy and halo currents within the walls of a tokamak are the stray components of the magnetic fields associated with confining the plasma. These magnetic fields apply $\mathbf{J} \times \mathbf{B}$ forces on the structures carrying the currents. In existing high performance tokamaks, the forces can reach as high as several MN [48, 67]. Next generation devices, however, will have much stronger currents and fields, and are expected to have several tens of MN of EM forces instead [48, 68, 69].

For eddy and halo currents, most of these forces come from different components of the current crossed with the magnetic field. Eddy currents, being primarily toroidal, produce the strongest forces when crossed with the poloidal magnetic fields; so their smaller poloidal component is often ignored [70]. In contrast, halo current forces, having both poloidal and toroidal components, are strongest when crossed with the toroidal field. As a result it is their poloidal components that receive the most attention.

The more dangerous of the eddy and halo current forces depends on the CQ-time; their comparison often represented through what is called the Goldilocks Curve [71], illustrated in Figure 1.12. The shorter the CQ, the larger $\dot{B}$ and stronger the inductive eddy drive. The inductive drive of the
1.4. ELECTRO-MAGNETIC LOADS AND HALO CURRENTS

![Figure 1.13: Cartoon illustrating the differences in net forces between symmetric and asymmetric forces. The toroidal distribution of forces on HBT-EP are shown above the schematics visualizing the components of the forces. (a) Symmetric forces. (b) Asymmetric forces.](image)

Symmetric HCs also increases, but the contact between the plasma and the wall is relatively poor until the plasma can move close enough to the wall. As a result, the HC forces tend to get worse with longer CQ times instead because a larger fraction of the current in the plasma edge can be shared the better the plasma-wall contact is. This inverse relationship with the CQ-time ($\tau_{CQ}$) between eddy and HC forces creates this trade-off between the two. The “Goldilocks” aspect of this curve refers to window of CQ-times within which the sum of the eddy and HC loads falls below the structural threshold of the tokamak vacuum vessel and other components (the yellow region in Figure 1.12). The duration of the CQ is desired to be in this window, and the active control of the CQ-time to enforce this is one of the pillars of disruption mitigation. For example, calculations for the minimum and maximum $\tau_{CQ}$’s on ITER bound the desired value to be between $50\,\text{ms} < \tau_{CQ} < 150\,\text{ms}$ [72].

The components of these currents/forces can be further subdivided into their symmetric and asymmetric components. These two components can have very different effects on the vessel loads, as shown in Figure 1.13. The symmetric components of the loads can only apply net vertical forces on the vessel. The forces associated with both the eddy and halo currents locally act in the direction...
1.5. **ROTATION OF HALO CURRENT ASYMMETRIES**

of the minor radius, as shown in Figure 1.13(a), and the toroidal symmetry of these forces results in the horizontal components of these forces canceling out around the machine. All that is left then is the vertical components of the forces, which fail to cancel out as well because of the poloidal asymmetries of the forces. In contrast to this, the toroidally asymmetric components of the loads can exert net vertical, horizontal, and tilt forces on the vessel. As illustrated in Figure 1.13(b), the introduction of the toroidal asymmetries prevents the net cancellation of the horizontal components of the forces, allowing net horizontal and tilt forces to exist. The existences of these extra net forces then applies additional loads on the vessel, making them more dangerous relative to their symmetric counterparts. In addition, the displacements caused by the asymmetric loads are susceptible to resonance in the presence of rotation, which can further amplify the damage. That is why this thesis focuses heavily on the rotation of these asymmetric forces, so as to be able to predict when resonance in them may occur.

When it comes to rotating asymmetric loads, halo currents tend to have the strongest drive relative to their symmetric components. When HC asymmetries are present, they are often comparable in size, or even larger than their symmetric components [1, 37, 47, 53]. The severity of the asymmetries is often characterized using what is known as the *Toroidal Peaking Factor* (TPF), which is defined as the absolute maximum of the toroidally resolved HC measurements normalized to their toroidally averaged value (e.g. a sinusoidal asymmetry with an amplitude equivalent to the mean would have a TPF of 2). Typical values for TPFs of various machines are consistently reported to be on the order of 2 [1, 36, 37, 47, 52]. In this thesis, a focus is placed on the asymmetries and rotation of halo currents specifically.

### 1.5 Rotation of Halo Current Asymmetries

In the previous section, the concerns around the resonant amplification of asymmetric loads motivated the need to understand rotation during the disruption. This section details the existing scaling law for the rotation of RHCAs and how it under-predicts the rotation observed on HBT-EP, and develops the physics basis for the new scaling introduced in Chapter 6.
1.5. ROTATION OF HALO CURRENT ASYMMETRIES

Figure 1.14: Empirical scaling of the average halo current rotation frequency using the Myers scaling. The original version of this Figure is taken from Figure 12 in Myers et al [3]. The horizontal axis is the predicted rotation frequency made using the Myers scaling and the vertical axis is the observed rotation frequency. The regions outside of the logarithmic $3\sigma$ envelope are shaded in gray. The projected region for ITER is shown in yellow, where the range is a result of the uncertainty in predicting ITER’s CQ-time. The location of HBT-EP in this Figure has been added above the original window.

1.5.1 The Myers Scaling Law

The intrinsic rotation of the plasma has been a difficult (and phenomenological) topic considering the complexity of plasmas [106]. During the disruption, most of the auxiliary forms of momentum input into the plasma are removed (or suppressed), and as a result the rotation during the disruption is mostly intrinsic. This makes first principle predictions of the disrupting plasma rotation difficult to develop, so instead predictions based on empirical scaling laws has been the norm. The plasma rotation referred to here specifically refers to the rotation of MHD instabilities (and subsequently RHCAs), which tends to be tied more to the motion of electrons than to that of the bulk plasma.
The previous scaling law was developed in Myers et al [3], and is an empirical scaling law based on the observations of RHCAs on five different tokamaks of varying sizes and field strengths. A least-squares regression in logarithmic-space found that the disruption averaged HC rotation frequency ($\langle f_{\text{rot}} \rangle$) depended on the pre-disruption major radius ($R$) and HC rotation duration ($t_{\text{rot}}$) as

$$\langle f_{\text{rot}} \rangle = C_f R^{-1} (t_{\text{rot}}/R)^{-1/2}$$  \hfill (1.6)

where $C_f$ ($\approx 25$) is a linear regression parameter, the exponential regression parameters for $R$ and ($t_{\text{rot}}/R$) are approximately $-1$ and $-1/2$ respectively, and the units are $\langle f_{\text{rot}} \rangle$ [Hz], $R$ [m], and $t_{\text{rot}}$ [s]. In the same paper that derived equation (1.6), an empirical scaling for $t_{\text{rot}}$ was also developed that approximately gave $t_{\text{rot}} \approx 0.6 \tau_{CQ}$. This scaling law described the rotation observed on the five tokamaks included in the database (C-Mod, NSTX, AUG, DIII-D, and JET) relatively well (shown in Figure 1.14), narrowing down the logarithmic 3$\sigma$ envelope to just under an order of magnitude.

The use of this scaling law, however, is relatively narrow (compared to the poloidal drift scaling introduced in Chapter 6). It makes use of only pre-disruption and disruption averaged parameters, and therefore can only predict the average $\langle f_{\text{rot}} \rangle$ for a given disruption.

This Myers scaling was also found to under-predict significantly the rotation observed on HBT-EP, a small low-field tokamak. Using $\tau_{CQ} \sim 1$ ms and $R \sim 1$ m, the Myers scaling law predicts a rotation frequency on the order of 1 kHz for HBT-EP, but this underestimates the observed $20 \rightarrow 60$ kHz by between $1 \rightarrow 2$ orders of magnitude. The under-prediction for HBT-EP can be seen to scale relative to the predictions for other machines in Figure 1.14. HBT-EP does not lie very far away in the parameter-space of the scaling law either, with a scale-factor ($R^{-1} (\tau_{CQ}/R)^{-1/2}$) similar to that of C-Mod. This severe under-prediction provided the motivation to develop a new scaling law that could incorporate the rotation observed on HBT-EP, in addition to that observed on the five tokamaks in the Myers database.

### 1.5.2 The Physics Basis for the New Scaling Law

The new scaling law considers that the rotation of the RHCAs approximately depends only on motion perpendicular to the magnetic field. This is a result of the connection between the rotation of RHCAs and the rotation of the MHD instabilities driving them. The MHD instabilities, and the magnetic topology associated with them, rotate with the electron and ion fluids. When taking into account the
1.6. THESIS OUTLINE

geometry of the MHD mode structures, the flow of the ions/electrons of interest can be specified. The MHD instabilities have helical structures to them and therefore have directions of symmetry parallel to said helical structures. Resonances, however, are independent of motion parallel to directions of symmetry. Therefore, the rotation of interest is limited to the motion of ions/electrons perpendicular to the direction of symmetry of the MHD instability. This interpretation can be simplified further by considering how the direction of symmetry is set. The most unstable MHD instabilities, and the ones that are present during disruptions, are those that have wavenumbers (\(k\)) that are parallel to the magnetic field at their resonant surfaces, i.e. \(k \cdot B = 0\). This implies that the symmetry direction of the helical mode will match that of the field lines at the rational surface, and will be close to that of the field lines just outside of it. This suggests that the motion of the ions/electrons perpendicular to the field lines is a good approximation to their motion perpendicular to the symmetry direction of the MHD instability; which is convenient for simplifying the interpretation of the rotation of interest.

The motion of the ions and electrons perpendicular to the magnetic field is partially determined by their drift velocities

\[
\mathbf{u}_{\perp,\text{drift}} = \frac{\mathbf{E} \times \mathbf{B}}{B^2} \pm \frac{\nabla P \times \mathbf{B}}{en_e B^2}
\]  

where the first term on the right-hand-side is the \(E\times B\) drift and the second is the diamagnetic drift. The simplification made in Section 6.1 to approximate the rotation of the RHCAs assumes that the drift terms dominate the perpendicular motion of the ions/electrons, and that either the \(E\times B\) drift is insignificant or it takes on a simpler form that is proportional to the diamagnetic drift.

1.6 Thesis Outline

The purpose of this thesis is to address the need to understand the rotation of RHCAs so as to avoid the resonance of EM stresses during disruptions on tokamaks. This need is addressed through the development of a physically motivated scaling law for the rotation frequency of RHCAs that is shown here to be valid over several orders of magnitude in frequency as well as for varying machine sizes and field strengths. A detailed investigation of the RHCAs on the HBT-EP and Alcator C-Mod tokamaks is provided, along with a comparison of their observations to those on other devices.
1.6. THESIS OUTLINE

This new poloidal drift scaling law is used to make predictions for the rotation of RHCAs on next-generation devices like ITER and SPARC, and finds that resonant rotation is at least likely on ITER.

This thesis is organized in a way that it (i) details the machines that are thoroughly investigated, (ii) discusses the disruption and halo current physics observed on these machines, contextualizing the existence of their RHCAs, and then (iii) motivates, introduces, and validates the new scaling law. The following chapters are organized as follows:

Chapter 2 - Overview of HBT-EP: This chapter discusses the HBT-EP plasmas, vacuum vessel structure, diagnostics, and actuators. A specific focus will be placed on the details that are relevant for investigating the disruption and halo current dynamics.

Chapter 3 - Overview of Alcator C-Mod: This chapter addresses the same topics as that in Chapter 2, but instead details the Alcator C-Mod tokamak instead of HBT-EP.

Chapter 4 - Characterization of Disruptions: This chapter characterizes the disruptions on HBT-EP and C-Mod to help identify the relevant physics involved, so as to better predict the characteristics of disruptions in the future.

Chapter 5 - Characterization of Halo Currents: This chapter characterizes the halo currents on HBT-EP and C-Mod. Information pertaining to the magnitudes of these currents, their distribution around the vessel (asymmetries), and their response to the presence of MHD instabilities helps to contextualize the rotation of their asymmetries and the danger it poses.

Chapter 6 - New Scaling Law for the Rotation Frequency of Asymmetric Halo Currents: This chapter introduces the new poloidal drift scaling law and validates it across devices. The validity of the scaling also helps to inform the direction of rotation during the disruption.

Chapter 7 - Conclusion: This chapter concludes the paper; summarizing its results and proposing future projects to continue investigating the topic of halo currents and disruption rotation.
Chapter 2

Overview of HBT-EP

The *High Beta Tokamak - Extended Pulse* (HBT-EP) machine is a compact, research-scale tokamak located at Columbia University. It is circular, limited, and has a large aspect ratio \((R/a \approx 6)\), which makes it useful for studying tokamak physics in the absence of many of the higher order effects present on higher performance machines that can make it difficult to verify basic plasma physics. Historically, HBT-EP has primarily been used to study MHD instabilities and their control, but more recently it’s branched out to study a broader range of physics like H-mode/turbulence [75], scrape-off layer currents (SOLCs) [1], and disruptions [1, 76]. HBT-EP is useful for studying disruption physics, despite being relatively low performance and cold, since it’s post-TQ temperatures appear to be similar to those observed in disruptions on other higher performance devices. The rotation of the plasma’s edge on HBT-EP is also easily characterized through several redundant measures, including an abundant array of magnetics, current sensors, and fast-camera imaging. A recently installed set of segmented current-sensing limiter tiles also allows for first-time detailed measurements of the poloidal structure of halo current asymmetries during the flattop phase of the plasma, which provide insight into the disruption HC structures.

In this chapter we discuss the HBT-EP plasmas, vacuum vessel (VV) structure, diagnostics, and actuators. The finalized SOLC sensor tiles were designed and installed as part of this thesis, while the prototypes and other diagnostics/actuators were already present on HBT-EP. A detailed discussion of disruptions on HBT-EP will be saved for a later chapter, so section 2.1 will focus more on the general characteristics, which will provide the context for interpreting flattop SOLC physics and making comparisons between HBT-EP and other machines. Section 2.2 details the VV
2.1. HBT-EP PLASMAS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plasma Current $I_p$</td>
<td>$10 - 20$ kA</td>
</tr>
<tr>
<td>Toroidal Field $B_T$</td>
<td>$0.34$ T</td>
</tr>
<tr>
<td>Major Radius $R$</td>
<td>$92$ cm</td>
</tr>
<tr>
<td>Minor Radius $a$</td>
<td>$15$ cm</td>
</tr>
<tr>
<td>Temperature $&lt;T_e&gt;$</td>
<td>$50 - 150$ eV</td>
</tr>
<tr>
<td>Density $&lt;n_e&gt;$</td>
<td>$1 - 3 \times 10^{19}$ m$^{-3}$</td>
</tr>
<tr>
<td>Edge Safety Factor $q_a$</td>
<td>$2 - 4$</td>
</tr>
<tr>
<td>Pulse-LENGTH</td>
<td>$O(10)$ ms</td>
</tr>
</tbody>
</table>

Table 2.1: Ranges of global plasmas parameters for HBT-EP.

structure of HBT-EP, which plays an important role in understanding the possible current paths between the plasma and the walls. Section 2.3 will discuss the SOLC tile and jumper cable HC sensors on HBT-EP, with a focus on the design of the SOLC tiles. Section 2.4 will then discuss the other diagnostic capabilities and actuators relevant to studying disruptions and halo currents, these primarily being the magnetics, fast-cameras, and control coils.

2.1 HBT-EP Plasmas

HBT-EP has historically specialized in studying MHD stability and as a result its plasmas are optimized for studying MHD stability instead of high performance. To achieve this end, HBT-EP operates a low edge safety factor ($q_a$), high poloidal beta ($\beta_p \equiv \frac{P}{B_T/2\mu_0}$), and high Greenwald density fraction ($n/n_G$) while maintaining low performance (cold plasmas with low toroidal magnetic field). Ranges for these parameters as well as other relevant plasma parameters on HBT-EP are provided in Table 2.1. The experiments performed for this thesis made use of deuterium plasmas that were purely Ohmically heated.

The most relevant of these parameters to halo currents and their rotation are the low plasma current, low toroidal field, and small poloidal cross-sectional area. The low plasma current leads to relatively small HC magnitudes and subsequently small EM loads on the VV. The low field and compact size are found to be important for setting the fast post-TQ HC rotation frequency, as will be shown in Chapter 6. The low edge safety factor is also likely responsible for the continuous MHD activity during the disruption, but this will be discussed more in Chapters 5 and 6.

HBT-EP plasmas are also short-lived, with typical plasma lifetimes from startup to disruption
2.2. HBT-EP VESSEL STRUCTURE

Figure 2.1: CAD of the HBT-EP vacuum vessel. Yellow spool pieces are the insulating quartz pieces. Red tiles are the scrape-off layer current tile limiters, lasting on the order of 10 ms. This is currently a limitation of the Ohmic heating and vertical field coils, which are designed to peak in strength around 10 ms following plasma startup, but efforts are currently being made to introduce an active-feedback plasma control system (PCS) that may extend the pulse-length to be purely limited by the available Ohmic flux.

2.2 HBT-EP Vessel Structure

The structure of the vacuum vessel and how the plasma makes contact with it will constrain the paths of halo currents between the two. The VV itself is segmented into ten sections as shown in Figure 2.1. Adjacent sections can either be electrically connected or insulated from each other through a stainless-steel or quartz spool piece respectively. There are four insulating spool pieces (yellow in Figure 2.1) around the machine which separate the VV vessel into two separate grounding schemes, the North and West grounds. This is illustrated in Figure 2.2. The North and West grounds are connected below the machine, such that the current leaving one ground must return through the other. Three of the ten sections can be left floating if the insulation across the quartz pieces are maintained (sections 4, 5, and 10), however, this insulation can be shorted in order to ground these otherwise floating sections to one of the two grounding schemes. A separate third ground (the South ground) is used to provide a ground for many of the diagnostics and can also be used to provide a
2.2. HBT-EP VESSEL STRUCTURE

Figure 2.2: Cartoon top-down view of HBT-EP’s vacuum vessel. The direction of $I_p$ and $B_T$ are also provided for clarity.

ground for one of the otherwise floating section.

Inside the vacuum vessel lies the movable shells, shown in black in Figure 2.3 and also visible in Figure 2.1. Each section has two separate shells that cover the low-field side of the plasma and lie nominally 1 cm away from the surface when fully inserted. These shells can move out radially at $\pm45^\circ$ angles with respect to the outboard midplane, as indicated by the blue arrows in Figure 2.3, to a distance of up to about 5 cm from their fully inserted positions. Following the installation of the new shunt scrape-off layer tiles, sections 2, 6, and 10 can only be inserted to 2 cm from the surface of the plasma. Feedback sensors are also located on the plasma side of the shells (as well as a poloidal array of magnetic sensors in sections 3 and 8), which stick in about 2/3 cm from the plasma-side surface of the shells and can serve as significant conducting paths between the plasma and the shells when the shells are inserted close enough to the plasma.

HBT-EP has three primary stainless-steel limiters that can set the last closed flux surface (LCFS) of the plasma and exchange the highest density of current between the plasma and the walls. These include the blade limiters, the SOLC tiles, and the high-field side (HFS) flanges. These surfaces act as poloidal blade limiters and the poloidal locations of each are illustrated in Figure 2.4. The blade limiters and SOLC tiles limit the plasma on the LFS while the HFS flanges limit it on the HFS. Which of these is setting the LCFS will depend both on the location of the plasma as well as
the positions of the LFS limiters. If the radial position of the current centroid of the plasma \( R_p \) is \( > 90 \) cm, then the plasma will be limited on the LFS by either the blade limiters and/or SOLC tiles, and if \( R_p < 90 \) cm, then it will be limited on the HFS by the HFS flanges. Both the blade limiters and the SOLC tiles can be moved to adjust the LFS limiting surfaces. The SOLC tiles are attached to the shells and can therefore be moved with them, while the blade limiters can be moved independently of the rest of the vessel. Both the blade limiters and the SOLC tiles can limit the plasma at the same time. The toroidal locations of each of these limiters are shown in Figure 2.2. There are a total of two blade limiters, six SOLC tile poloidal arrays, and 20 HFS flanges (two per section). Half of the SOLC tile arrays use the prototype design while the other half use the finalized design. The prototype design has the a separation of 1 cm between the plasma-side surface of the shell and the plasma-side surface of the tile, while the finalized design has a 2 cm separation instead.

2.3 Halo Current Sensors

HBT-EP is equipped with five distinct sets of halo current sensors that can measure the current passing between the plasma and the vessel. These include the scrape-off layer current sensor tiles, the jumper cable Rogowski coils, the \( I_p \) Rogowski coil, the poloidal arrays of Mirnov sensors, and the ground current Rogowski coils. The SOLC sensor tiles, jumper coils, and ground current Rogowski coils act as direct measurements of the halo current while the \( I_p \) coil, and Mirnov sensors act as indirect
Figure 2.4: CAD drawings illustrating the three possible limiting surfaces on HBT-EP. In both figures the plasma is being limited on the LFS. The HFS flange limiters are shown in blue. (a) Blade limiters in brown. (b) Prototype low-res SOLC tiles in red. (c) Photo viewing section 2 SOLC tiles and HFS flange limiters in the co-$I_p$ direction.

Figure 2.5: Locations of some of the HC sensors on HBT-EP. Only two of the SOLC tiles and one of the jumper coils are visible from this view. Section 4 of the vessel is labeled so that the view can be compared to Figure 2.2.

measurements. The locations of the $I_p$ coil, poloidal Mirnov sensors, and some of the jumper coils and SOLC tiles around the tokamak are shown in Figure 2.5. The ground current Rogowskis are wrapped around the chamber grounding paths shown in Figure 2.2. This section discusses these diagnostics in detail and clarifies what currents they measure.
2.3. **HALO CURRENT SENSORS**

![Diagram of Path of Halo Currents]

Figure 2.6: Path of the halo currents that are measured by the SOLC tile sensors. The separation between the plasma and the shell shown here is accurate for the finalized SOLC tile design.

![Designs for Two SOLC Tile Iterations]

Figure 2.7: Designs for the two SOLC tile iterations. (a) Prototype design that uses a Rogowski coil to measure current. (b) Finalized design that uses a shunt to measure current.

### 2.3.1 SOLC Sensor Tiles

The scrape-off layer current (SOLC) sensor tiles are a set of poloidal arrays of segmented limiter tiles that can limit the plasma on the top, bottom, and outboard midplane of the plasma. Their locations relative to the vessel and the plasma are shown in Figure 2.4(b), and their toroidal distribution is shown in Figure 2.2. A photo of the section 2 tiles can also be found in Figure 2.4(c). These sensors measure the current traveling directly between the plasma edge and the wall through these limiter tiles and are segmented to provide poloidally resolved measurements of these exchanged currents. The tiles are attached to their local shells, providing a path between the plasma and the shells as described in Figure 2.6. The polarities of the sensor measurements are defined such that a positive signal corresponds to electrons moving from the plasma into the shells, consistent with the polarities often used for Langmuir probes.

These diagnostics make use of two different types of current sensors, shunts and Rogowski coils. Shunts measure the current by directly measuring the voltage drop across both ends of the shunt.
2.3. **HALO CURRENT SENSORS**

material. In the absence of a strong frequency response (verified below), Ohm’s law implies that the voltage drop and the current are linear with each other

\[ V(t) = I(t)R \]  \hspace{1cm} (2.1)

so the resistance that maps the two can just be calibrated using a known current. Rogowski coils can measure the current passing through their axis of symmetry by measuring the changes in induced flux inside the coil caused by the currents. The shape of the coil allows for only the flux from the axial current to make a net contribution to the induced voltage, rejecting flux contributions from external sources. The resulting voltage across the coil is then given by

\[ V(t) = -\frac{\mu_0 AN}{L} \frac{dI(t)}{dt} \]

where \(A\) is the area of a single loop, \(N\) is the number of turns, \(L\) is the circumference of the loop, \(I\) is the total current enclosed in the loop, and \(\mu_0\) is the permeability of free space. The current enclosed can then be measured by integrating the voltage.

\[ I(t) = -\frac{L}{\mu_0 AN} \int V(t) dt \]  \hspace{1cm} (2.2)

The conversion factor from integrated voltage to current can be approximated using the designed values for \(A\), \(N\), and \(L\), but this conversion factor is more reliably calibrated by passing a known current through the coil.

There are two separate sets of SOLC tile designs. The first (and earlier) design is the prototype design and the second (and later) design is the finalized design. The primary differences between the two are the sensors used to measure the current and separation between the plasma and the shells. Figure 2.7 illustrates these differences. The initial prototype design used a Rogowski coil to measure the current while the finalized design used a shunt. The Rogowski coils were easier to design and construct, so they were used in the prototype design, but they were found to have two major flaws that warranted the switch to the shunt design with the following upgrade. The Rogowski coils were susceptible to both significant amounts of magnetic pickup as well as the magnetization effects of the materials close to the Rogowskis (the metal spacer, screws, and clamps were all found to be magnetic). The magnetic pickup dominated the equilibrium signals of the sensors and the
2.3. **HALO CURRENT SENSORS**

Figure 2.8: Frequency response of the section 10-3 shunt SOLC tile. (a) Resistance response (plot is zero-suppressed). (b) Phase lag response.

magnetization effects then made it difficult to remove the pickup using a simple mutual coupling (+eddy currents) model. However, this pickup primarily effected the equilibrium signal, leaving the fluctuations in current to be relatively reliable.

The shunt design improved upon the Rogowski by removing the magnetic pickup all-together, however, the difficulty in designing the shunts compared to the Rogowskis required minimizing the frequency response of the shunt to acceptable levels, i.e. the self-inductance and skin-depth effects. This was resolved by using stainless-steel 3mm shim-stock as the shunt material, similar to that used in the HC sensors on NSTX [46]. The thin nature of this material reduces the $L/R$-time to under 2 $\mu$s and makes the skin-depth 10 times greater than the material thickness even at 1 MHz. The acceptable frequency response was verified through both in-vessel and ex-vessel calibration as well. The ex-vessel calibration for sensor 10-3 is shown in Figure 2.8, which shows that the frequency response measured through a discrete frequency scan sees less than a 2% change in resistance from DC to 80 kHz (the highest expected frequencies during the disruption) and less than a 4$^\circ$ change in phase. The drop in resistance seen in Figure 2.8(a) is believed to be a result of the frequency response of the Pearson Rogowski used to calibrate the resistance, with an increase in the effective resistance of the Pearson leading to an apparent drop in the resistance in the shunt.

The resistance of the shunts was designed to be about 25 $m\Omega$s, so that the digitizers would be able to resolve currents as small as 100 mA and as large as 500 A without the need for the signal to be amplified. The actual resistance for each of the shunts ended up falling somewhere within a factor of 2 of this desired value, with construction variance leading to these errors. This low shunt
2.3. HALO CURRENT SENSORS

resistance also wouldn’t interfere significantly with the paths the currents would take into the walls. This is because the sheath impedance is likely much larger than the impedance of the shunts. The sheath impedance for a classic electrode is given by [77]

\[ R_{\text{sheath}} = \frac{dV}{dI} = \frac{(T_e/e)}{I_{\text{sat}}} \]

which gives a minimum resistance set by

\[ R_{\text{sheath}} \geq \frac{(T_e/e)}{I_{\text{i,sat}}} \]

where \( V \) is the voltage drop across the sheath, \( I \) is the current passing through the sheath, \( T_e \) is the electron temperature in the sheath, \( e \) is the electric charge, and \( I_{i,sat} \) is the ion saturation current. The temperature in the scrape-off layer (SOL) region of the plasma on HBT-EP is measured to be on the order of \( \sim 10 \) eV, so an \( I_{i,sat} > 500 \) A would be required for the minimum resistance to become comparable to shunt resistance. \( I_{i,sat} \) for the SOLC tiles during flattop operation is believed to be far less than this. Dedicated \( I_{i,sat} \) experiments for the tiles were difficult to perform due to power supply limitations, but approximating \( I_{sat} \) using its classical definition for the largest area tiles assuming typical SOL parameters (SOL width \( \lambda_{\text{SOL}} \sim 1/2 \) cm, \( T_e \sim 10 \) eV, and \( n_e \sim 1/2e^{19} \) m\(^{-3}\)) gives

\[ I_{\text{sat}} = An_eC_s e \sim O(40 \) A\)

where \( A \) is the collection area on the tile (both sides), \( n_e \) is the average electron density in the near SOL, and \( C_s \) is the average sound speed in the near SOL. This is consistent with the order of magnitude for the currents measured in the tiles (\( \sim O(10 \) A\)). Additionally, the 500 A requirement assumes that the actual current passing through the tile isn’t comparable, or more likely close, to \( I_{i,sat} \). Based on this, the resistance of the shunts weren’t expected to significantly effect current paths during flattop operation. Whether or not this effect was significant during disruptions (where currents can get up to 100’s of amps in the jumper cables) was not heavily investigated, considering that the plasmas on HBT-EP disrupt inboard, away from the tiles.

The other significant difference between the two SOLC tile designs was the distance between the plasma and the shells when the shells were fully inserted. The finalized SOLC tile designs have the shells further away from the plasma by 1 cm. The purpose of this was to prevent the feedback
2.3. HALO CURRENT SENSORS

(FB) sensors and shells from conducting significant amounts of currents from the plasma themselves, which could allow for un-diagnosed current paths into the wall. Experiments using the prototype tiles suggested that the SOL conducting width ($\lambda_{SOL}$) was less than 1 cm, so adding an additional centimeter to the separation between the plasma and the wall was expected to significantly reduce the un-diagnosed current collected by the shells. This did not end up being the case however, and will be further discussed in Chapter 5.

Each of the tile designs required ex-vessel circuits before being digitized. The prototype Rogowski design signal is sent to a variable gain amplifier circuit before being digitized and is integrated in post-processing to back out the current. The resistance of the finalized shunt designs were chosen to generate voltages that can be directly measured by the digitizers and therefore they did not require an amplifier. However, the shunt signals are first fed through isolation transformers to break the ground between the shunts (which are electrically connected to the vessel) and the digitizers.

Both the prototype and finalized SOLC designs made use of high-resolution (HR) and low-resolution (LR) poloidal arrays, photos of which are shown in Figure 2.9 for the finalized design. The difference between these resolutions is the number and poloidal extent of each tile. The LR arrays had four tiles spanning a little over 180° poloidally, with each sensor individually spanning $\sim 54°$, and the HR arrays had 8 tiles spanning a similar total poloidal range, but with each individual sensor instead spanning only $17.72°$. The advantage of the HR arrays was that they could measure HC asymmetries with higher poloidal resolution, making them capable of resolving higher-order
2.3. HALO CURRENT SENSORS

Figure 2.10: Photo of jumper 9.5 located between sections 9 and 10.

poloidal asymmetries like $m > 3$. The advantage of the LR arrays was that they could collect more current, allowing them to better resolve smaller fluctuations than the LR arrays. Two HR arrays and one LR array were installed for the prototype design, while one HR array and two LR arrays were installed for the finalized design, for a total of six diagnosed sections (the locations of which are provided in Figure 2.2).

2.3.2 Jumper Rogowski Coils

The jumper Rogowski coils are a set of Rogowski coils wrapped around jumper cables that can be used to electrically connect otherwise insulated vessel sections. The locations of these jumper cables for the experiments performed for this thesis are shown in Figure 2.2, with the jumper cable connecting sections 9 and 10 being referred to as jumper 9.5 and the cable connecting sections 3 and 4 being jumper 3.5. A photo of jumper 9.5 is shown in Figure 2.10.

These jumper coils measure the current passing between the adjacent vessel sections, and because they act as the only connections between these vessel sections they are a measure of the total toroidal current in the wall at that toroidal location. Additionally, the currents they measure are known to
be strictly halo currents, i.e. currents passed between the plasma and the wall. This is because the jumper cables provide the only electrical vessel connection for one of the adjacent vessels, implying that any current entering/exiting that section via the jumper cable must necessarily exit/enter that section via the plasma.

The existence of the jumper cables significantly effects the current paths around the machine, and their poloidal locations around the vessel as well as their resistances can potentially effect the current paths as well. However, their locations and resistances do not effect the observed rotation frequency of the HCs, and as a result their locations were not varied to investigate the effects. All experiments contributing to this thesis had the jumper cables placed on the outboard midplane of the vessel. The resistance of the jumpers could be varied by adding resistors in series with the jumper; up to about 10 Ω before the measured current fell below measurable values.

**Calculating the HC rotation frequency**

The currents measured by the jumper coils are used to characterize the rotation frequency of the rotating HC asymmetries during disruptions HBT-EP. There are up to two jumpers attached at a time, but they are not close enough to $90^\circ$ apart to be treated as a $sin$ and $cos$ pair. Instead, the rotation frequency is measured by applying a Hilbert transform to the Jumper 9.5 measurement. The Hilbert transform is a signal processing technique that has been applied on HBT-EP in the past to identify the amplitude and phase of a signal [78]. The transform takes in a real-valued signal and returns the expression necessary to convert said signal into its analytic representation, i.e. $x(t) \rightarrow x(t) + iH(x(t))$ where $H(x)$ is the Hilbert transform of $x(t)$. Assuming $x(t)$ can be represented by a single wave, the analytic representation generated by the Hilbert transform can then be used to calculate the amplitude and phase using

$$A(t) = \sqrt{x^2(t) + H^2(x(t))}$$

$$\theta(t) = \arctan\left(\frac{H(x(t))}{x(t)}\right)$$

(2.3)  

(2.4)

The frequency can then be calculated in Hz as $f(t) = \frac{d}{dt}(\theta(t))$. Prior to applying the Hilbert transform to jumper 9.5, a high-pass filter above 5 kHz is first applied. This is done to remove the background signal associated with the CQ, and is necessary considering that interpreting the amplitude and frequency returned by the Hilbert transform requires the signal to be represented by
2.3. HALO CURRENT SENSORS

Figure 2.11: Steps taken to calculate the rotation frequency of the rotating HC asymmetries for the disruption of shot 110428 on HBT-EP. Time is shifted to start with the current-spike initiating the disruption. The red regions denote the period where the decay rate of the background current becomes comparable to the high-pass frequency. (a) Jumper 9.5 halo current. (b) high-pass filter of jumper 9.5 signal above 5 kHz. The orange curve is the amplitude returned by the Hilbert transform. (c) The frequency calculated using the Hilbert transform of the filtered jumper current.

A single wave at each time-point. An example of the steps taken to calculate the rotation frequency for shot 110428 on HBT-EP is provided in Figure 2.11.

The use of the Hilbert transform has limitations though. The two most significant here are that (i) an error is introduced when the change in the background signal is fast enough to bypass the high-pass filter and that (ii) it doesn’t provide information of the direction of rotation. When the change in the background signal is comparable to the high-pass filter frequency, the background signal is no longer completely removed and the “amplitude” and “frequency” output by the Hilbert transform are no longer easily interpretable. This effect is often seen towards the very end of the disruption when the CQ slows down and the current drive in the jumpers quickly decays (the red period in Figure 2.11(a)). When this decay rate is comparable to the 5 kHz high-pass frequency, some of the background current remains following the filter (the red period in 2.11(b)), artificially slowing down the frequency as interpreted by the Hilbert transform. This effect could be avoided by raising the high-pass frequency, but if the high-pass frequency becomes comparable to the rotation frequency of the RHCA early on in the CQ, then it will interfere with the early frequency measurement. The
minimum RHCA frequency measured during the CQ is close to $\sim 10$ kHz, so 5 kHz was decided as the high-pass frequency.

2.3.3 $I_p$ Rogowski Coil and Poloidal Mirnov Arrays

The $I_p$-Rogowski coil and poloidal Mirnov arrays can act as indirect measures of the toroidal HC at their respective toroidal locations. They do this by measuring changes in the current enclosed within them. The $I_p$-Rogowski is a large Rogowski coil surrounding the quartz spool separating sections 9 and 10. The spool piece conducts no current, so the total current enclosed by the $I_p$ Rogowski is therefore necessarily coming purely from the plasma itself (hence why it’s called the $I_p$-Rogowski). However, even though it is only measuring current in the plasma, changes in the toroidal HC can be observed based on fluctuations in $I_p$. There can be no toroidal asymmetries in the total toroidal current (sum of plasma and wall) because of current conservation and the fact that the only current drives are inductive and symmetric. Therefore, the fluctuations in the measured plasma current can only come from two sources, either fluctuations in the Ohmic current drive or changes in the currents exchanged between the plasma and the wall. High-frequency fluctuations in the Ohmic drive (likely due to coupling to MHD instabilities) are often too small to significantly effect the total poloidal flux, as a result, high-frequency (MHD instability range) fluctuations in $I_p$ therefore indicate changes in the exchange of current between the plasma and the wall.

Fluctuations in $I_p$ being consistent with changes in the toroidal HC can also been seen experimentally as well. The $I_p$-Rogowski is toroidally co-located, but lies inside of, jumper 9.5, implying that the sum of $I_p$ and the jumper 9.5 current gives the total toroidal current. Figure 2.12(b) shows that comparisons of the fluctuations in $I_p$ (black) and jumper 9.5 (blue) are comparable and out of phase with each other by $180^\circ$ (the signal plotted is $-1\times$ the jumper current) when the jumper is connected, implying that an increase in current in one path is accompanied by a similar decrease in current in the other, consistent with the interpretation that the current being exchanged is changing. Figure 2.12(b) also shows the case where jumper 9.5 is disconnected (red), such that the $I_p$-Rogowski is now measuring the total toroidal current on its own, and in this case there is a clear suppression of the fluctuations in $I_p$, consistent with current no longer being exchanged between the plasma and the wall. Figure 2.12(c) shows that the magnitude of the MHD instabilities are not suppressed in the disconnected jumper case, implying that the suppression of the $I_p$ fluctuations can’t be a direct result of the suppression of MHD instabilities. The change in exchanged HC can only be
2.3. HALO CURRENT SENSORS

Figure 2.12: Comparison of two disruptions where jumper 9.5 is connected (110459 - black/blue) and disconnected (red). The time-bases are normalized to each disruption’s respect CQ time, as defined by the extrapolated time from the 80% – 20% $I_p$ drop. (a) Plasma current normalized to $I_p$ at the current-spike and the jumper 9.5 current as a percent of $I_{p,0}$. (b) Plasma current and $-1 \times$ jumper 9.5 fluctuations (high-pass filtered above 10kHz). (c) Mode amplitude as measured by the TA array.

Figure 2.12: Comparison of two disruptions where jumper 9.5 is connected (110459 - black/blue) and disconnected (red). The time-bases are normalized to each disruption’s respect CQ time, as defined by the extrapolated time from the 80% – 20% $I_p$ drop. (a) Plasma current normalized to $I_p$ at the current-spike and the jumper 9.5 current as a percent of $I_{p,0}$. (b) Plasma current and $-1 \times$ jumper 9.5 fluctuations (high-pass filtered above 10kHz). (c) Mode amplitude as measured by the TA array.

measured by the $I_p$-Rogowski during disruptions. This is because only disruptions provide large enough fluctuations in $I_p$ to be reliably resolved by the $I_p$-Rogowski.

The poloidal Mirnov arrays can indirectly measure the fluctuations in halo current the same way the $I_p$-Rogowski does. The poloidal Mirnov arrays are poloidal arrays of 32 sensors each that span the entire poloidal range of the plasma. They sit just outside of the plasma ($\sim 1/3 - 1$ cm from the plasma surface when the plasma is limited on the side closest to the sensors) and measure $\dot{B}$ in both the radial and poloidal directions. The sum of each entire poloidal array can be used to emulate a Rogowski coil when using the poloidal $\dot{B}$ measurements, and as a result can be used to measure the enclosed current the same way the $I_p$-Rogowski can. However, there are two primary difference between the capabilities of the $I_p$-Rogowski and the poloidal Mirnov arrays; (i) the Mirnov "Rogowskis" are more prone to the influence of eddy currents (because of local difference in eddies between the sensors), and (ii) the currents in walls surrounding the Mirnov arrays are un-diagnosed.
2.4. OTHER RELEVANT DIAGNOSTICS AND ACTUATORS

These two differences make the poloidal Mirnov arrays less useful than the $I_p$-Rogowski.

2.3.4 Ground Current Rogowski Coils

The ground current Rogowski coils are a pair of Pearson Ammeters that have the grounding paths to the North and West racks fed through them. These current sensors diagnose all of the current leaving the North and West grounded vessel sections, and when none of the nominally floating vessel sections are grounded to the South rack ground, they measured all of the current leaving the vacuum vessel. The North and West rack grounds only have a single connection below the machine, so the current leaving through one grounding path is the same as that entering the other. This has been verified experimentally.

The currents measured by these Rogowskis are also necessarily currents that have passed between the plasma and the walls, i.e. halo currents. This is because the North and West rack grounds are connected only at a single location, which restricts any currents passing between the two through this junction to complete their circuits through the plasma. As a result, these ground current Rogowskis measure the total halo current being exchanged between the North and West rack grounded vessel sections through the plasma, the magnitude of which is often comparable to that of the total current measured by the SOLC tiles.

2.4 Other Relevant Diagnostics and Actuators

HBT-EP is equipped with many other diagnostics and actuators, and in this section those that are most relevant to investigating halo currents and disruption physics are detailed. These include (i) the magnetic sensors used to diagnose the characteristics of MHD instabilities, (ii) the $cos(\theta)$-Rogowski used to measure the plasma’s major radius, (iii) the fast-cameras used to observe contact between the plasma and the wall, and (iv) the magnetic control coils (CCs) used to influence the SOL plasma and plasma-wall contact.

2.4.1 Magnetic Sensors

HBT-EP has an extensive array of Magnetic Mirnov sensors measuring $\dot{B}$ in the poloidal and radial directions around the machine. There are two poloidal arrays and five toroidal arrays. All together, there are a total of 134 distinct sensor locations. Their locations can be seen in Figure 2.13.
2.4. OTHER RELEVANT DIAGNOSTICS AND ACTUATORS

The two poloidal arrays (PAs) have already been discussed in Section 2.3.3. Each PA set has 32 sensors that make complete poloidal coverage of the plasma. The two PA sets are toroidal separated by 180°, which makes them useful to characterizing odd \( n \)-number characteristics. The primary use of these sensors, besides their involvement in equilibrium reconstructions, is to diagnose the poloidal structure of MHD instabilities located at the edge of the plasma. The high poloidal resolution allows for resolving high-order poloidal structures, up to \( m = 16 \) at the theoretical max.

The five toroidal arrays are separated into a set of four feedback (FB) arrays and one toroidal array (TA). The FB arrays are located on the plasma facing sides of the shells, as shown in Figure 2.3, and the TA set is located on the HFS midplane. Each FB array has a single sensor in each section while the TA set has three sensors located in each section. Both toroidal array sets are primarily used (outside of equilibrium reconstructions) for diagnosing the toroidal structure of edge MHD instabilities and tracking their phase and amplitude. Which array is best to use depends on the location of the plasma. If the plasma is limited on the HFS (like during disruptions) the TA signal will be stronger, but if the plasma is limited on the LFS the FB arrays will be better. The TA array is best at most simply tracking the phase and amplitude of MHD instabilities. This is because the three sensor locations per section allow there to be exactly 90° separated sensors that can be used to calculate the phase and amplitude.

Figure 2.13: Locations of magnetic sensors on HBT-EP.
2.4. OTHER RELEVANT DIAGNOSTICS AND ACTUATORS

2.4.2 \( \cos(\theta) \)-Rogowskis

The \( \cos(\theta) \)-Rogowski is the diagnostic used to measure the major radius of the plasma, or more specifically, the major radius of the plasma’s current centroid \( R_c \). For the purpose of calculating the plasma’s poloidal cross-sectional area \( S \), the current centroid measured by this diagnostic is often assumed to be close to the plasma’s geometric centroid \( R_{geo} \). This is because HBT-EP’s low pressure and high-aspect ratio lead to a small Shafranov-shift (~1 cm during flattop) and a small difference between the magnetic-axis and current centroid [79]. Additionally, this assumption becomes more appropriate during the disruption. Following the thermal quench the poloidal beta \( (\beta_p) \) can drop by more than a factor of 2x and the aspect ratio can double, leading to an even smaller difference between \( R_c \) and \( R_{geo} \).

The \( \cos(\theta) \)-Rogowski itself is a large Rogowski coil wound poloidally around the quartz spool piece between sections 9 and 10 (co-located with the \( I_p \)-Rogowski). The difference between a \( \cos(\theta) \)-Rogowski and a normal Rogowski, which allows it to identify \( R_c \), is the poloidal variation in its winding structure. The density of turns in a normal Rogowski is poloidally uniform, but the density in a \( \cos(\theta) \)-Rogowski has the structure \( n_{\text{turns}} \propto \cos(\theta + \phi) \), where \( \theta \) is the poloidal location in the frame of the Rogowski and \( \phi \) sets the phase offset. For \( \phi = 0 \) and assuming deviation of \( R_c \) from the center of the coil \( (\Delta R) \) is small compared to the radius of the coil \( (a_R) \), the integration of the flux measured by the coil will be approximately linearly proportional to \( \Delta R I_p \). Normalizing the Rogowski signal \( (S_{\cos}) \) to \( I_p \) and calibrating the pre-factors will then provide a measurement of \( R_c \).

The linearity of the relationship between the \( S_{\cos} \) and \( \Delta R \) remains accurate as long as \( \Delta R \ll a_R \), but there are periods during the disruption where this assumption is no longer valid. The error associated with the non-linearity that arises starts to become significant as \( R_c \) drops below 83 cm \( (a_{\text{plasma}} \approx 8 \text{ cm}) \). As a result, measurements of \( R_c \) and the plasma’s cross-section \( S \) are considered unreliable below this threshold.

2.4.3 Fast Cameras

HBT-EP has two fast cameras (Phantom S710) that can view the plasma through either of the viewing-ports in section 3 or 10. Optical fiber bundles mounted near the port capture visible light and send it to a nearby camera to be recorded. A 2:1 demagnification is applied between the fiber optics and the cameras to increase the light intensity at the cost of poorer image resolution. For the
2.4. OTHER RELEVANT DIAGNOSTICS AND ACTUATORS

Figure 2.14: Fast camera view between sections 8 and 9. (a) 3D view with an approximation of the window in red. (b) Actual signal recorded by the camera for shot 111622 illustrating the objects in view. The signal used is the BD mode 0 taken between 4.2-4.5ms. The view is horizontally inverted because of the use of a mirror to capture the light.

Experiments discussed in this thesis, the camera(s) had a frame rate of 100 kfps, capturing 96x256 pixels (vertical × horizontal) over a ~16 cm vertical span centered roughly about the midplane.

Most of the visible light observed by the cameras on HBT-EP comes from line radiation at the edge of the plasma, with the dominant source being $D_\alpha$ emission [80]. The intensity of line radiation is proportional to electron density ($n_e$) times the neutral density ($n_n$), so the light intensity measured by the camera is a good qualitative indicator of the plasma density. Additionally, when the plasma makes contact with the wall, processes like recombination and sputtering will locally increase the neutral density and subsequently locally enhance the intensity of line radiation. As a result, regions of higher intensity along the limiting surfaces of the plasma are likely indicative of stronger plasma-wall contact. This makes the camera useful for identifying regions of stronger plasma-wall contact.

The views of these cameras can be adjusted to focus on areas within the midplane region of the plasma. In the experiments conducted for this thesis only a single camera was used and was pointing in the counter-$I_p$ direction from the section 10 viewing-port as shown in Figure 2.14(a). The view was focused on the HFS midplane so as to resolve changes in plasma-wall contact on the HFS limiting flange between sections 8 and 9 (on the section 9 side). The actual view as recorded by the camera is shown in Figure 2.14(b). A mirror is used to reflect the image into the fiber optics, so the resulting image is horizontally inverted from what it should be. Operating the camera with this view requires completely retracting the shells in section 10. As a result the SOLC tiles in section 10 measure relatively little current in these cases.
2.4. OTHER RELEVANT DIAGNOSTICS AND ACTUATORS

There are a total 40 saddle-loop coils (control coils or CCs) on HBT-EP that can be used to actively influence the plasma’s edge. They are located on the vessel-side of the shells, inside the main chamber. Their locations are shown in Figure 2.15. Each section has 4 poloidal locations for coils, and they are positioned such that they simplify applying $m = 3$ resonant perturbations by forcing each adjacent coil to be roughly 180° out of phase with its neighbor when using that pattern. Each location is equipped with three coils of different sizes (small, medium, and large). The experiments performed here make use of only the large set of coils, as they cover a larger toroidal span of the plasma. The large coils each have approximately 12 turns and can drive a max of $\sim 40$ A for a total of $\sim 480$ A. This translates to fields as strong as $\sim 40$ G on the surface of the plasma, which are more than strong enough to interact with $\sim 5 – 10$ G long-wavelength MHD instabilities at the plasma edge.

Windows have been bored out of the shells matching the shapes of these coils so as to reduce the magnetic field penetration time through the shells to the plasma. Sheets of shim-stock are placed over these windows to prevent plasma from reaching behind the shells, but they are too thin and resistive to introduce significant penetration time effects.

The CCs can apply fields with any number of patterns, including dominantly vertical fields or dominantly horizontal fields, but they are most useful for their ability to apply helical fields that are resonant with MHD instabilities at the edge of the plasma. Fields with this applied pattern are referred to as Resonant Magnetic Perturbations (RMPs). For plasma control purposes, RMPs can
2.4. OTHER RELEVANT DIAGNOSTICS AND ACTUATORS

be used to stabilize edge MHD instabilities [81, 82], but for this thesis, they are used to introduce controlled edge perturbations so as to investigate how halo currents respond to them.
Chapter 3

Overview of Alcator C-Mod

The Alcator C-Mod tokamak was a compact high-field conventional tokamak located at the Massachusetts Institute of Technology. C-Mod was the only high-field compact tokamak of its time, allowing it to break records in volume averaged plasma pressure [83] and contribute significantly to many areas of fusion science, including disruptions. Disruption prediction [88], disruption mitigation [43, 84–87], disruption modeling [89, 90], runaway electron [91], and halo current [37, 54] studies have all been done in the past. This made C-Mod well equipped studying HC rotation physics. However, C-Mod finished operation in 2016, so only a database analysis could be performed for this thesis.

In this chapter we discuss the C-Mod experiment. Section 3.1 provides a description of the C-Mod plasmas. Once again focusing on general characteristics and saving the discussion of disruptions for Chapter 4. Section 3.2 reviews the vacuum vessel and regions of strongest plasma-wall contact on C-Mod. Section 3.3 details the halo current diagnostics used to measure HC rotation. Sections 3.4 and 3.5 respectively discuss the filament reconstruction and $q_{\text{edge}}$ calculation used to characterize the plasma edge during the disruption. Finally, Section 3.6 details C-Mod’s impurity injection capabilities, which can be used to influence the plasma disruption.

3.1 C-Mod Plasmas

C-Mod’s plasmas were compact, high-field, and high performance, operating with a plasma current of up to 2 MA, an edge safety factor between 2 and 8, and with both shaped and diverted plasmas.
A list of relevant global parameters for C-Mod is provided in Table 3.1. C-Mod boasts a large range of possible toroidal fields, making it useful for investigating the toroidal field dependence of the HC rotation scaling law that will be introduced in Chapter 6.

The plasma can be operated either in a double- or single-null configuration, with the x-point(s) of the separatrix lying above and/or below the plasma, but typically operated in the lower single-null configuration. A sample plasma cross-section featuring the flux surface contours is provided in Figure 3.1. The diverted nature of C-Mod makes the physics covering the plasma-wall contact different from that on the limited HBT-EP tokamak, with the highest density exchange of halo current occurring far from the main the plasma. While diverted, the LCFS was set by the separatrix, as opposed to a limiting surface like it is on HBT-EP.

The plasmas consisted of Deuterium as the main species, but also with a minority species of either Hydrogen for low-to-moderate field operation or Helium-3 for high field operation. The primary heating source was Ion Cyclotron Radio Frequency (ICRF) heating. The current drive was sourced both Ohmically and through Lower Hybrid Current Drive (LHCD).

### 3.2 Vacuum Vessel

The C-Mod vacuum vessel was designed to house a diverted plasma that typically operates in lower-single null configuration. To that end, C-Mod is equipped with a dedicated lower divertor, the HFS edge of which was removed in the early 2000’s. The poloidal cross-section of C-Mod’s vacuum vessel
Figure 3.1: Poloidal cross-section of C-Mod featuring flux surface contours during flattop operation for shot 950214038. The surfaces were generated using the filament equilibrium reconstruction detailed in Section 3.4. The thick curves are the flux contours of the LCFS. The red crosses denote the x-points of the separatrix. The plasma is in a lower-single null configuration.

is shown in Figure 3.2(a). The vessel itself is segmented into 10 symmetric sections, each of which are electrically connected to each other. The divertor itself is as a result toroidally symmetric. The diverted nature of C-Mod leads to very different characteristics of plasma-wall contact compared to that on HBT-EP. On C-Mod, the primary regions of plasma-wall contact (PWC) are toroidally continuous, unlike HBT-EP where the discrete poloidal blade limiters lead to significant toroidal asymmetries in PWC. Similar to HBT-EP, however, all plasma facing wall components are made of metal, with the divertor and wall tiles being made of Molybdenum.

### 3.3 Halo Current Sensors

Alcator C-Mod has had several different sets of halo current sensors placed throughout the device to measure currents in different locations. These include four sets of sensors in the lower divertor, one set at the top of the vessel, and one set on the inboard side of the vessel. The abundance of
3.3. HALO CURRENT SENSORS

Figure 3.2: Poloidal cross-section of C-Mod. (a) Vacuum vessel. The region in red was the HFS region of the divertor that was later removed. (b) Halo current sensor locations, as reported in Refs [37, 54]. The lower full Rogowski and the partial Rogowski loop are roughly co-located.

Sensors in the lower divertor is due to the fact that the plasma tends to move vertically down over the course of a disruption, localizing the largest exchange of current to that region.

These sensor sets make use of three current measurement techniques: Rogowski coils, shunts, and Langmuir probes. Descriptions of how Rogowski coils and shunts are used to measure currents are already provided in Chapter 2.3.1. The probes can be used as true Langmuir probes, sweeping $I-V$ characteristics to get local electron temperatures, densities, and potentials, however they are used here instead as halo current sensors by grounding the probes to the vessel and measuring the currents along those paths. Some of the Rogowski coils are also segmented into partial Rogowski coils. Partial Rogowski coils are Rogowski coils that are only wound partially in the azimuthal direction. An example is shown in red in Figure 3.3. The benefit of using partial Rogowski is that a complete azimuthal set of them can be used to characterize the azimuthal distribution of currents passing through their axis, like that shown in Figure 3.3. Partial Rogowski on their own are essentially flux loops, and in the case of a completed loop, each individual partial Rogowski will measure a stronger or weaker flux depending on the distribution of current passing through the loop. The measured relative flux strengths then provide information on what that current distribution is. The magnitude of the current distribution can also be determined by treating the sum of the signals
3.3. HALO CURRENT SENSORS

Figure 3.3: Cartoon of partial Rogovskis forming a complete loop. An individual partial Rogowski coil is highlighted in red. This example considers a bored out center such that current only travels axially at the edge of the ring. The sizes of the blue dots denote the relative strengths of the currents passing through that region and the sizes of the orange dots denote the relative magnitudes of the voltages measured.

Figure 3.4: Toroidal view of halo current sensors on C-Mod. The first three sub-figures are reproduced from Ref [37] and the last is an image from Ref [54]. (a) Full Rogowski coils. (b) Toroidal array of partial Rogowskis. (c) Vertical array of partial Rogowskis. (d) Rail probes.

from each partial Rogowski as the signal one would get from a full Rogowski coil and calibrating that measurement to a known current. The linear relationship between the integrated-voltage and current in a Rogowski coil then allows for the same volts-seconds→amps conversion applied to the full Rogowski signal to also be applied to the partial Rogowski signals as well.

Of the six sets of halo current sensors on C-Mod, one is a set of shunts, one is a set of probes, two are full Rogowskis, and two are sets of partial Rogowskis. The poloidal cross-section of their locations is shown in Figure 3.2(b). The shunts lie between the outboard divertor tiles and the rest
3.3. HALO CURRENT SENSORS

of the vacuum vessel, with each section having a single shunt. They provide the only conducting path between the two and therefore measure all of the current leaving the plasma and entering the vessel on the outboard side. The probes replace some of the outboard divertor tiles (shown in Figure 3.4(d)) and are only located in a single section. They are toroidally elongated and the array runs poloidally along the divertor. The probe array measures the halo currents entering the outboard divertor with high radial resolution. The two full Rogowskis are located at the top and bottom of the vessel and run toroidally around the machine (see Figure 3.4(a)). These Rogowskis measure the total halo current passing poloidally between the HFS and LFS of the machine (see Figure 3.5(a)). The currents measured are also restricted to only halo currents (as opposed to including eddy currents), because they measure only the poloidal component of the current while the eddies are predominantly toroidal due to the symmetry of the CQ. Only one of the Rogowskis measures significant current at a time though. This is because the plasma moves to towards one side of the wall as it disrupts, leading to a larger exchange of current on that side of the vessel as the current paths will be shorter. One of the sets of partial Rogowskis is a set of four sensors stacked vertically around the inboard vessel as shown in Figure 3.4(c). These partial Rogowskis extend toroidally only one section (36°) and are separated vertically from each other by 10 cm. The purpose of the sensors was to determine the SOLC width and distribution as current enters the vessel on the HFS. The final set of sensors is the complete loop of partial Rogowskis roughly co-located with the lower full Rogowski. These are a set of ten Rogowskis (one per section) that form a complete loop as shown in Figure 3.4(b). These partial Rogowskis measure the same currents as the lower full Rogowski, except they are also capable of toroidally resolving the distribution of poloidal currents. For a more detailed description of the Rogowski and shunt diagnostics see Granetz et al 37, and for a more detailed description of the the probes see Tinguely et al 54.

The primary halo current sensors of interest for this thesis were the toroidal array of partial Rogowskis. This was for two reasons: (i) the location of the partial Rogowskis allows for a continuous measurement of the rotating HC asymmetries throughout the disruption, and (ii) the toroidal resolution allows for the identification of the toroidal mode number and rotation direction while also simplifying the calculation of the rotation frequency (see Section 3.3.1). The only HC sensors that measure halo currents throughout (downward moving) disruptions are the Rogowskis located in the lower divertor. The probes and inboard partial Rogowskis can transiently measure halo currents, but their measurements are sensitive to the plasma position. Most of the halo current is exchanged
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Figure 3.5: Paths of the poloidal halo current components with respect to the lower full Rogowski and toroidal partial Rogowski array. The yellow dot denotes the locations of HC sensor arrays, the orange arrows are the current paths, and the green lines are the points of entry from the plasma into the wall. (a) Current path prior to the removal of the HFS divertor corner. (b) Current path following the removal of the HFS divertor corner (in red).

near the surface of the wall limiting the plasma [54], and the limiting surface changes as the plasma moves during the disruption. For the probes, this means that significant currents will only be measured as the LCFS passes by the probes, and for the inboard partial Rogowskis, significant currents will only be measured if the Rogowski is below the LCFS (plus $\lambda_{SOL}$). The timescale on which the plasma moves with respect to the spatial extent of the probes is comparable to the rotation period of the asymmetric HCs on C-Mod, so as a result the probes and inboard partial Rogowskis don’t consistently measure significant current for at least one period of rotation.

The focus on the toroidal array of partial Rogowskis does, however, limit the scope of the database available. The removal of the HFS corner of the divertor in the early 2000s led to a significant change in the halo current paths on C-Mod, as illustrated in Figure 3.5(b). As a result, most of the halo currents no longer passed through the partial Rogowskis following this change. The useful database for studying HC rotation was therefore narrowed down to the campaigns performed between 1995 (when the partial Rogowskis were installed) and the early 2000s.
3.3. HALO CURRENT SENSORS

3.3.1 Calculating the HC rotation frequency

The rotation frequency of the rotating HC asymmetries was calculated using the toroidal array of partial Rogowskis. Automated calculation methods similar to that used on HBT-EP were attempted using various techniques. These methods included performing a Hilbert Transform, a Biorthogonal Decomposition, and fitting the spatial structure to a sinusoid (like that done in Myers et al [3]). However, most of these methods were susceptible to errors associated with the short duration of the rotating asymmetry. As will be discussed more in Chapter 4, very few complete rotations ($\leq 2$) are usually made during disruptions on C-Mod and the growth/decay rates of the modes are often comparable to the rotation frequencies. This makes it difficult for automated methods like the Hilbert transform and Biorthogonal Decomposition to identify the appropriate instantaneous phase needed to characterize the frequency, often leading to an underestimation of the frequency around the onset and termination of the mode. When the onset and termination periods of the mode then make up a significant portion of the total rotation time, as they do on C-Mod, the automated analysis becomes unreliable for a majority of the rotating period.

The issue with the Myers method (fitting the instantaneous spatial structure to a sinusoid) was twofold. First, automatically identifying the rotating asymmetry period within the CQ can be difficult when a large static asymmetry precedes or follows it. Myers et al [3] introduced a frequency threshold for identifying the rotating period, but we found that choosing an adequate frequency threshold that was applicable to a majority of the database was difficult. Second, the Myers method assumes that there is no significant static asymmetry existing simultaneously with the rotating one, which is often not necessarily the case. This automated method could possibly be refined to resolve these issues, but that remains outside of the scope of this work.

To avoid the underestimation of the rotation frequency associated with automated methods, manual "by-eye" methods were used instead. The choice to use manual methods restricted the rotation scaling analysis to focusing on only the average rotation frequencies ($<f_{\text{rot}}>$). The specific manual method used depended on how many complete rotations were made during a given disruption.

If two or more complete rotations were made, then a frequency was calculated by tracking the times between peaks on a single partial Rogowski sensor to get a period. Using the equation

$$<f_{\text{rot}}>=\frac{1}{2}(\tau_1^{-1}+\tau_2^{-1})$$

(3.1)
3.3. HALO CURRENT SENSORS

Figure 3.6: Toroidal array of partial Rogowski halo current signals on C-Mod illustrating the frequency calculation for each case. The toroidal positions of the sensors are given in degrees, with values increasing in the counter-$I_p$ direction. The HC signals here are unitless and the time-base is referenced relative to the time when $I_p$ reached 80% its pre-disruptive value. Red corresponds to the analysis using the initial phase and green corresponds to that for the complimentary phase. The sensor at 144° was not working for these shots. (a) Example disruption where more than two complete rotations are made. (b) Example disruption where less than two complete rotations are made.

where $\tau_1$ and $\tau_2$ are the periods of the first two complete rotations. Only the first two complete rotations were included to avoid complications introduced by the slowing down that is occasionally seen to occur for cases when many more than 2 complete rotations are made. Figure 3.6(a) shows an example of this occurring following the 1 ms mark. The sensor used to calculate the frequency was that in which the rotating asymmetry first appeared. The calculation was also performed for an additional sensor in order to establish an error, prioritizing the use of a second sensor that was located the furthest from the first. An example calculation using this method is shown in Figure 3.6(a), which gives $<f_{rot}> = 7.0 \pm 0.1$ kHz.

If less than two complete rotations were made, then a frequency was calculated by tracking peaks as they moved toroidally across partial Rogowski sensors to get a phase difference over time. Using the equation

$$<f_{rot}> = \frac{\Delta \theta}{2\pi \tau}$$

(3.2)
3.4. FILAMENT EQUILIBRIUM RECONSTRUCTION

Figure 3.7: Flux surface output of the filament reconstruction for the disruption of shot 950214038 at a single time-point. The plasma is limited at this time.

where $\Delta \theta$ is the toroidal separation of two sensors and $\tau$ is the time it takes for a peak to travel from one sensor to the other. We prioritized tracking the peak as it moved from it’s initial toroidal position to the location closest to $180^\circ$ away. Another additional calculation was then performed using another pair of sensors (ideally as close to $90^\circ$ out of phase with the other sensor pair) in order to establish an error. An example calculation using this method is shown in Figure 3.6(b), which gives $<f_{\text{rot}}> = 1.3 \pm 0.1$ kHz.

3.4 Filament Equilibrium Reconstruction

Due to the inherent transient nature of disruptions, measurements of common plasma parameters through conventional equilibrium reconstructions (like EFIT [92]) are often not reliably available. So in order to characterize needed parameters like the plasma cross-sectional area ($S$), vertical position ($Z_p$), and edge safety factor ($q_{\text{edge}}$), a filament-based equilibrium reconstruction code was used instead, which was developed at MIT for the express purpose of characterizing disruptions on C-Mod. This reconstruction approximates the plasma as an array of toroidal current filaments and
3.5. \( Q_{\text{EDGE}} \) APPROXIMATION

uses magnetic measurements and coil currents to constrain the magnitude of the currents in each plasma filament. The resulting plasma filaments and coil currents are used to construct poloidal flux surfaces \((\psi_p)\). The minimum of \(|\psi_p - \psi_{p,\text{mag-axis}}|\) on the walls relative to \(|\psi_p - \psi_{p,\text{mag-axis}}|\) at the separatrix determines if the plasma is limited or diverted and establishes the plasma’s last closed flux surface (LCFS). The poloidal cross-sectional area is then calculated from the LCFS using the Shoelace method and the vertical position is defined as the filaments’ current centroid. An example of the flux surfaces output by this reconstruction for the disruption of shot 950214038 is shown in Figure 3.7.

There are two primary limitations to the use of this filament reconstruction. The first is that it lacks internal plasma information like the current and pressure profiles, and the second is that it becomes less reliable the smaller the plasma poloidal cross-section gets. The lack of internal profiles implies that the internal flux surfaces output by the reconstruction are unreliable. Fortunately, the plasma boundary and vertical position as we define it are not influenced by the internal flux surfaces, so this limitation does not significantly effect those measurements. However, what this limitation does effect is the measurement of the safety factor inside of the LCFS, which will be will be addressed in the next section. The reconstruction also becomes less reliable as \(S\) gets smaller because the number of spatially-fixed current filaments inside the plasma becomes smaller. This effect becomes significant as the diameter of the plasma approaches the spatial separation of the filaments, such that only one or two filaments remain inside of the LCFS. At that point, changes in the magnitude of the current in those filaments won’t significantly effect the shape of the plasma and subsequently the measured area artificially stagnates. A hard lower limit of \(Z_p > -0.38 \text{ m}\) was placed on the plasma’s vertical position to avoid situations where only a single filament inside the plasma, and a soft limit of \(Z_p > -0.33 \text{ m}\) was placed to avoid situations where only two filaments were inside instead.

3.5 \( q_{\text{edge}} \) Approximation

The edge safety factor during the disruption \((q_{\text{edge}})\) was calculated using the \(\psi_p\) 2D grid output by the filament reconstruction. This was done using a script written by R.A. Tinguely that solves the 3D coupled ordinary differential equations for \(dp/ds = \hat{B}\) (where \(p\) is the position vector and \(s\) is the parameterization of motion) to track the trajectory of a field-line starting at an arbitrary point.
Figure 3.8: $q_{edge}$ time-evolution for the disruption of shot 950315029 on C-Mod using the $q_{95}$ and $q_{LCFS}$ definitions. $q_{LCFS}$ is only calculated for the limited period of the disruption. The periods where the filament reconstruction only include one or two filaments inside the plasma (the plasma-size thresholds) are also provided.

somewhere on a flux surface of interest. $q$ is then calculated as the number of toroidal transits made after completing a single poloidal transit.

The relevant $q_{edge}$ for characterizing MHD instabilities during the disruption was chosen to be $q_{95}$, where $q_{95}$ is $q$ at the flux surface where $\psi_{95} = 0.95(\psi_{LCFS} - \psi_{mag-axis})$ and $\psi_{mag-axis}$ is $\psi_p$ at the magnetic axis. This choice was made because $q_{LCFS}$ is not well defined when the LCFS is set by the separatrix (i.e. when the plasma is diverted) and switching from $q_{95}$ to $q_{LCFS}$ as the plasma transitions from diverted to limited over the course of the disruption would introduce unnecessary complexity. There were concerns that the lack of internal measurements included in the equilibrium reconstruction made the calculation of $q_{95}$ (an internal quantity) uncertain, but comparisons between $q_{95}$ and $q_{LCFS}$ during limited periods of the disruptions showed that the two often agreed with each other to within 2%. This is exemplified in Figure 3.8. Differences as large as 10% are observed, but only during periods where the plasma is too small to pass the plasma-size thresholds specified in the previous section.
3.6 Impurity Injection

Alcator C-Mod was lastly equipped with a multitude of actuators capable of introducing impurities into the plasma. The presence of these impurities can have many beneficial effects, from divertor detachment [93, 94], improved core transport [95], wall conditioning, and disruption mitigation [37]. The latter of these is where the impurity injection’s use in studying halo currents comes into play. The use of impurity injection preceding the CQ to reduce the CQ-time is a common approach to mitigate the deleterious effects of halo currents and radiate a larger fraction of the plasma’s stored thermal energy. The introduction of impurities into the plasma on C-Mod was facilitated via both pellet injection and gas puffing. The following subsections detail the implementation of these two injection schemes on C-Mod.

3.6.1 Impurity Pellet Injector

Pellet injection on C-Mod was done using the Impurity Pellet Injector system. This pellet injector served a number of purposes including acting as part of a current density diagnostic, conditioning the first wall with lithium or boron, improving transport properties through the injection of lithium [95], and studying the effects of high-Z “killer pellets” to mitigate disruptions.

The system injects small high speed ($\sim O(1 \text{ km/s})$) pellets into the center of the plasma. The small size of the pellets ($\sim O(1/2 \text{ mm}^3)$) was set by the “rule of thumb”, $N_{\text{pellet}} \approx N_{\text{plasma}}$ ($N_e$ is the total electron inventory), which was required for the pellets to penetrate to the core of the plasma [95]. Two different types of room-temperature pellets were used, lithium pellets and High-Z doped polypropylene pellets. The lithium pellets were fabricated by cutting out cylinders from lithium metal ribbons. The polypropylene pellets were doped with either silver or gold powders. A high-pressure gas gun would then be used to accelerate the pellets through guide tubes that direct the pellets towards the magnetic axis of the plasma. The number and frequency of injections was pre-programmed before every shot.

3.6.2 Gas Injection

Gas injection on C-Mod could be done through a number of different systems. These include: (i) the Pulsed Gas Valves, (ii) the NINJA system, and (iii) the Laser Ablation Technique. Each system has its own set of advantages and disadvantages. Investigating the dependence of disruption mitigation
3.6. IMPURITY INJECTION

on the system used falls outside the scope of this work, so there will be no focus placed on any one system in particular. The impurities of interest introduced by these systems were the noble gasses Neon, Argon, and Xenon. The number and frequency of injections was pre-programmed before every shot.

The Pulsed Gas Valves [96] used piezo-electric gas valves to release impurities into the chamber from one of five locations around the machine. Gas was released by opening the valve and allowing the pressure differential between the vacuum and the plenum holding the gas to eject the gas. The amount of gas released could be moderated by controlling the amount of time the valve was open and the pressure in the holding plenum. The peizo-electric valves improved the conduction of gas and allowed for this system to have the fastest response time. However, these gas valves were located at the outboard ends of their local ports and therefore injected gas far from the plasma.

The NINJA system [96] was a divertor gas injection system capable of injecting impurities at multiple (up to 28) poloidal locations around the plasma. This system provided the most freedom in terms of poloidal locations to inject impurities, but it also required long tubing to guide the gas, resulting in relatively slow response times.

The Laser Ablation Technique [96] used the laser ablation of a thin film to inject impurities. Located at the outboard end of one of the ports, this technique was capable of puffing in impurity gases with finite energy (up to 10 eV). While the pellet injector could also inject impurities with finite energy, the Laser Ablation Technique had the advantage that it did not disturb the plasma as significantly as the pellet injector does.
Chapter 4

Characterization of Disruptions

When tokamak plasmas disrupt, they go through a series of events eventually leading to their termination. These result in rapid transport of energy to the wall and rapid plasma cooling. A confined plasma for fusion energy production can decrease in temperature from several keV to 10’s of eV during the thermal quench (TQ) and the resulting plasma’s high resistance then causes dissipation of its stored magnetic energy during the current quench (CQ). Variance among machines and even from shot-to-shot can lead to differences in the specifics of these characteristics during disruptions. The characterization of disruptions helps to identify the relevant physics involved, so that we may better predict the characteristics of disruptions in the future. In the context of asymmetric HC rotation, the diagnosis of global parameters (like poloidal plasma cross-section, toroidal field, temperature, and MHD stability) is necessary to characterize the rotation frequencies and adequately predict them in the future. In this chapter we characterize the disruptions on both HBT-EP and Alcator C-Mod.

4.1 Disruptions on HBT-EP

Disruptions occur with every discharge on HBT-EP because it was designed to generate fast-growing MHD instabilities over short timescales with little concern for the avoidance of the inevitable disruptions that would likely follow their growth. Even when pre-programmed to operate in more stable (high-$q_a$, low-$n_e$) regimes, the ability of the vertical field coils to maintain the necessary field strength to keep the plasma centered will degrade in time (due to the lack of an active plasma control sys-
4.1. DISRUPTIONS ON HBT-EP

tem), eventually leading to the plasma falling inboard and shrinking in size until \( q_a \) drops below an unstable value, destabilizing an instability and disrupting the plasma. The ubiquity of disruptions on HBT-EP in addition to HBT-EP’s high shot rep-rate (\( \sim O(50 \text{ shots/day}) \)) leave HBT-EP with an abundance of disruptions to characterize.

This section characterizes the disruptions on HBT-EP as follows. Subsection 4.1.1 describes the general sequence of events associated with disruptions on HBT-EP. Subsection 4.1.2 discusses multi-staged disruptions and how they can complicate the characterization of the disruption. Subsection 4.1.3 characterizes the temporal shape and duration of the CQ, introducing the slow and fast phases of the current quench and characterizing the conditions at which the transition between the two occurs. Subsection 4.1.4 calculates the post-TQ electron temperature \( T_e \). And lastly, subsection 4.1.5 characterizes the MHD instabilities present during the disruption.

4.1.1 Sequence of events

Disruptions on HBT-EP typically have the following characteristics illustrated in Figure 4.1. A disruption is triggered with the onset of one or more large rotating \( n = 1 \) MHD mode(s) and a sudden increase in visible radiation. The MHD mode characteristics change in response to the disruption, but they continue to be present throughout. A current-spike occurs (0 ms in Figure 4.1) and is immediately followed by the current quench (\( \tau_{CQ} \sim 1 \text{ ms} \)). Simultaneous with the current-spike, the plasma major radius moves inboard in response to the loss of horizontal stability as the pressure-driven component of the hoop force is depleted via the thermal quench. The vertical position of the plasma stays relatively centered as the major radius continues to move further inward, eventually falling below 0.90 m and causing the minor radius to decrease as the plasma limits on the high-field side. The plasma size shrinks faster than the current quenches, and as a result the edge safety factor drops over the course of the CQ, but remains above 1 throughout. As the plasma current drops below around 20% its pre-disruptive value, the rotating MHD instabilities present throughout the disruption begin to disappear along with the currents in the walls. The timescale of the entire disruption is fast compared to the rate of change of the toroidal field (\( B_T \)), and as such \( B_T \) remains relatively fixed throughout the disruption period.
4.1. DISRUPTIONS ON HBT-EP

Figure 4.1: Plasma signals during a typical disruption evolution on HBT-EP for shot 110428: (a) plasma current, (b) minor radius, (c) major radius (current centroid), (d) vertical position (current centroid) measured using top (07P) and bottom (26P) PA sensors from both PA1 (black) and PA2 (red), (e) edge circular safety factor, (f) a poloidal magnetic field sensor (filtered above 5 kHz), (g) contour of poloidal field perturbations in the toroidal direction. The green dashed line is the time of the current spike, defined as the ‘disruption event’. The orange dashed lines represent the $t_{80}$ and $t_{20}$ times, where Ip has fallen to 80% and 20% of its pre-disruptive values. The red dashed line is the time where the analysis is cutoff, due to the major radius dropping too low to be reliably measured.

4.1.2 Multi-staged disruptions

Many disruptions on HBT-EP go through multiple stages in addition to simple TQ→CQ picture outlined in Section 1.3.1. Some of the more common stages are the slow and fast CQ decay phases, discussed in more detail in Section 4.1.3, but there are others that can complicate the characterization of disruptions. The most common additional stages on HBT-EP are the minor-disruption/beta-collapse precursors and multiple thermal quenches. Beta-collapse precursors are often observed...
4.1. DISRUPTIONS ON HBT-EP

Figure 4.2: Examples of off-normal/multi-staged disruptions on HBT-EP. The blue, orange, and green vertical lines indicate the disruption onset, $t_{\text{crit}}$, and $t_{20}$ respectively. (a) Plasma current and major radius for shot 113658. The sharp drops in $R_p$ accompanied by small $I_p$-spikes are indicative of beta-collapses or strong outer wall contact occurring. (b) Plasma current and conductivity temperature for shot 114147. The spike in $T_{e,\text{conduct}}$ at the current-spike is an artifact of the temperature calculation.

when operating at low $B_T$ and/or $I_p$, like the shot shown in Figure 4.2(a), but these conditions are not a requirement for them to happen. However, as can be seen from Figure 4.2(a), beta-collapses can complicate assigning the onset of the disruption and make it more difficult to interpret their evolution. It will also be shown in Section 4.1.4 that multiple TQs may be responsible in part for some of the slow-to-fast CQ decay transitions. Some disruptions don’t even appear to have any distinct TQ quenches throughout their duration, like in shot 114147 (Figure 4.2(b)), although these cases tend to appear accompanied by smaller pre-disruption temperatures.

To avoid many of the complications associated with these multi-staged processes, large-database analyses of HBT-EP disruptions, like those discussed in Chapter 6, ignore the earlier stages of the disruption and focus only on the fast CQ period. This period of the disruption is often the one of most interest when concerning currents, as the inductive current drive is the strongest, and therefore it is the most relevant period of the disruption to focus on.

4.1.3 CQ shape and CQ time

The temporal waveform of the current quench and its duration is of much concern because it effects the strength and time-evolution of the eddy and halo currents in the walls. The faster the CQ the stronger the eddy current drive in the walls, but at the same time the slower the CQ the stronger the halo current drive to the wall. Additionally, the speed of the CQ can change as the plasma
4.1. DISRUPTIONS ON HBT-EP

![Current quench curve for shot 113026 on HBT-EP exemplifying the reverse “S”-shape structure of the CQ. The time-base is shifted to start at the current-spike. The orange vertical line is the critical time at which the CQ transitions from a slow linear decay to a fast linear decay, and the green vertical line is when $I_p$ reaches 20% its pre-disruptive value ($t_{20}$), which approximately corresponds to the transition from the fast linear decay to the short exponential decay.](image)

Figure 4.3: Current quench curve for shot 113026 on HBT-EP exemplifying the reverse “S”-shape structure of the CQ. The time-base is shifted to start at the current-spike. The orange vertical line is the critical time at which the CQ transitions from a slow linear decay to a fast linear decay, and the green vertical line is when $I_p$ reaches 20% its pre-disruptive value ($t_{20}$), which approximately corresponds to the transition from the fast linear decay to the short exponential decay.

current drops from $\sim 100\% \rightarrow <10\%$ its pre-disruptive value. Other tokamaks report varying shapes for their current quenches, with some machines even seeing variance between shots on a single device [42, 44, 45, 97–99]. The simple resistive dissipation picture often used to describe the CQ would suggest an exponential decay shape, and is sometimes observed [45, 97, 99]. However, other waveforms are reported as well, including linear decays [42, 44, 97–100], reverse “S”-shaped decays [42, 44, 45, 101], and even combinations of the previously mentioned shapes [98, 99]. The exact cause of the variance in shapes is not well understood, but correlations between certain shapes and other disruption characteristics have been reported for individual machines [42, 44, 97, 98]. These correlations, however, are not always consistent from machine-to-machine. For instance, JET [42], NSTX [44], and JT-60U [97] report that faster CQs are better characterized by an exponential decay while slower CQs are more linear, but in contrast to this J-TEXT [98] found that their faster CQs were best described by a linear decay. The presence of a runaway electron plateau has also been found to influence the shape, with runaway disruptions having a more exponential decay [42]. Some disruptions are also found to best be characterized by a sequential combination of linear and/or exponential decays. HL-2A [99] reports that their slower disruptions have two phases, a slow exponential decay followed by a fast linear decay. J-TEXT, JET, and NSTX also report similar 2-phased shapes in some of their slower disruptions, with the shapes of the 2-phases sometimes changing from machine-to-machine. This 2-phased structure is also what describes the shapes of CQs
4.1. DISRUPTIONS ON HBT-EP

On HBT-EP, HBT-EP’s CQs tend to start with a slow more-linear decay that eventually transitions to a sharper linear decay. That is until the plasma current reaches 20% its pre-disruptive value, at which time a short exponential decay brings the current down from \(\sim 20\% \rightarrow 0\%\) its pre-disruptive value. This slow-linear \(\rightarrow\) fast-linear \(\rightarrow\) short exponential decay shape (exemplified in Figure 4.3) is similar to the reverse “S”-shaped CQs reported on other machines like JET, NSTX, ASDEX Upgrade [45], C-Mod (discussed in the next section) and DIII-D [101], and has been expected to describe disruption on ITER [7].

In addition to the shape of the CQ curve, its duration is also of interest since it influences the strength of eddy and halo currents. The faster the CQ the stronger the inductive drive of the eddy currents and the longer the CQ the stronger the halo currents are as the plasma makes better contact with the wall. A collective effort has been made to determine a scaling law for the CQ-time (\(\tau_{CQ}\)), but the varying CQ shapes complicates the defining of a universal definition for \(\tau_{CQ}\). The previous definition uses a linear decay time, i.e. \(\tau_{CQ} = (t_{Y\%} - t_{X\%})/(X\% - Y\%)\), where \(t_{X\%}\) is the time at which \(I_p\) reaches \(X\%\) its pre-disruptive value. The ITPA Disruption Database [102], a multi-machine database collected to develop scaling laws to predict disruptions on ITER, uses 80\%–20\% to define the CQ-time, but many machines have also employed other linear decay definitions, such as 100\%–40\% [42], 100\%–30\% [103], and 90\%–10\% [98, 100]. Each definition normalizes the CQ-time based on the choice of \(X\%\) and \(Y\%\), making their definitions extrapolable to each other. The variance in the choices of \(X\%\) and \(Y\%\) come from the variance in CQ-shape. A definition with a high starting percent (\(\geq 90\%\)) is often chosen when the CQ-shape usually resembles a single linear decay, a high ending percent (\(\geq 30\%\)) is often used when it resembles a single exponential decay, and a low starting percent (\(\leq 80\%\)) is often used when two phases are involved. In the case of a two-phased CQ, the timescale of the faster decay is the most relevant to EM loads. So the lower starting percent prevents the slower decay from diluting the measurement of the relevant CQ-time.

HBT-EP is often two-phased, and as a result the 80\%–20\% definition characterizes its disruptions relatively well. However, the best definition to use can vary from disruption-to-disruption. In light of this, we have developed a method for identifying the most appropriate definition for each individual shot. We have found that this new dynamic definition both resolves several issues associated with using a fixed definition and also helps to inform when to expect the transition from the slow linear phase to the fast linear phase to occur on HBT-EP.
4.1. DISRUPTIONS ON HBT-EP

Figure 4.4: Examples clarifying that the fixed definition for $\tau_{CQ}$ can lead to poor linear fits of the CQ. The orange line describes the linear decay that using the 80%-20% definition would fit. The vertical red line denotes $t_{80}$ and the vertical purple line is $t_{20}$. (a) Example of a good fit. (b) Example of a poorer fit. (c) Example of a very bad fit.

We define the dynamic CQ-time as

$$\tau_{CQ} = \frac{t_{20} - t_{crit}}{crit\% - 0.2} \quad (4.1)$$

where $t_{crit}$ is the critical time at which the CQ transitions from a slow linear decay to a fast linear decay and $crit\%$ is $I_p$ at that time normalized to the pre-disruption $I_p$. This dynamic definition prevents the slow linear decay from diluting $\tau_{CQ}$ on any given shot, when the fixed definition would inevitably still include some portion of the slow decay for some disruptions (as illustrated in Figure 4.4). The critical time is calculated for each disruption by rotating the $I_p(t)$ curve in the $I_p$-$t$ plane as shown in Figure 4.5. The angle of rotation is made such that the points of the current-spike and $t_{20}$ are level in the new horizontal axis. The critical time is then chosen as the maximum point in the rotated frame (Figure 4.5(b)). This method works well as long as the two decay phases are monotonic and the first decay phase is slower than the second. The advantage of this method is that it makes very little assumptions about the shape of the decay phases, and is robust to both linear
4.1. DISRUPTIONS ON HBT-EP

Figure 4.5: Example of how the critical time is determined for the disruption of shot 113026. The blue dot marks the time of the current-spike, green is $t_{20}$, and orange is the critical time. (a) Plasma current highlighting the period from the current-spike to $t_{20}$. The dashed red line is provided to illustrate rotation. (b) The rotation of $I_p(t)$ (and the dashed red line) in the $I_p$-$t$ plane.

and exponential decays. The disadvantage, however, is that it fails to work when the current-spike is large compared to the drop in $I_p$ during the slow decay phase. This case is rare on HBT-EP, but we will later see that it is significant on C-Mod.

One major advantage of using the dynamic $\tau_{CQ}$ over using a fixed definition is that it clarifies the relationship between the normalized $\tau_{CQ}$ and plasma current density ($J_p$). This relationship is important because it’s often used to characterize the inter-machine scaling of the CQ-time [7, 42, 44, 100, 103]. The reason for this is that $\tau_{CQ}$ normalized to the poloidal plasma cross-sectional area ($S$) is expected to be proportional to the plasma resistivity ($\eta_p$), which is thought to be set by the Ohmic heating of the plasma caused by $J_p$ (see Subsection 4.1.4 for more detail). When using the fixed $\tau_{CQ}$ definition, most machines report a large spread in $\tau_{CQ}/S$ vs $J_p$, similar to what is seen on HBT-EP when using the fixed 80%–20% definition (shown in Figure 4.6(a)). A trend in the minimum of this spread is then used to scale the minimum CQ-time, which can be useful for predicting the fastest the CQ can be on a given device. This trend is often treated as fixed minimum of $\tau_{CQ}/S$ (independent of $J_p$) [7, 44, 103], but some machines [42, 100], including HBT-EP, observe a clear linear trend between $\min(\tau_{CQ}/S)$ and $J_p$. On HBT-EP, this lower bound scaling is roughly described by $\min(\tau_{CQ}/S) \sim 25 \text{[ms/MA]} \times J_p$, which is similar to the 5 ms/MA reported on JET [42] and the approximately 20 ms/MA seen on ADITYA [100]. Previously, only one other machine, ADITYA [100], has reported a linear trend between $\tau_{CQ}/S$ and $J_p$ without a large spread in the
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Figure 4.6: Normalized CQ-time vs current density on HBT-EP, scanned over 8000 shots. The left-half plots are scatter plots and the right-half plots are 2D histograms with a logarithmic colorbar. (a) The 80%–20% definition for $\tau_{CQ}$ is used along with $S$ and $J_p$ at $t_{80}$. The line $\tau_{CQ}/S = 25 \text{ [ms/MA]} \times J_p$ roughly sets a lower bound on the scatter. (b) The dynamic definition for $\tau_{CQ}$ is used along with $S$ and $J_p$ at $t_{crit}$.

data, although an explanation for why this is the case was not given.

The use of the new dynamic CQ-time definition provides a clear low-spread relationship between $\tau_{CQ}/S_{crit}$ and $J_{p,crit}$ and potentially helps to explain why ADITYA saw a low-spread relationship while other machines did not. This is illustrated in figure 4.7(a), which shows the $\chi^2$ value of the linear fit of the CQ using the 80%–20% definition for all 8000 shots included in the database scan. In it we see that the $\chi^2$ value increases significantly with distance from the lower bound of the scatter plot. This shows that the spread in data is largely attributed to error associated with the fixed definition of $\tau_{CQ}$, and if the cases with too high a $\chi^2$ value are filtered out (above some threshold) then a lower spread relationship is revealed, shown in Figure 4.7(b). Using the dynamic CQ-time definition, the spread in the relationship can be reduced directly by avoiding large $\chi^2$ errors, as shown in Figure 4.6(b).

The remaining spread left in the data could be attributed to many possible factors. One possible cause is the variance in impurity content during the disruption. The physical motivation for the relationship between $\tau_{CQ}/S$ and $J_p$ is grounded in how the post-TQ $T_e$ can influence the $L/R$ decay-time of the plasma current, with power balance setting $T_e$ based on the Ohmic heating $J_p$. However, the other side of the power balance equation is likely dominated by impurity radiation, so the impurity content may also influence $\tau_{CQ}/S$ through its effect on $T_e$. Another possible cause is the use of a linear decay time to characterize an inherently exponential $L/R$ decay-time.
4.1. DISRUPTIONS ON HBT-EP

Figure 4.7: Figures demonstrating the effects of poor 80%–20% fits. (a) $\tau_{CQ}/S$ vs $J_p$ using the 80%–20% definition, with the colorbar denoting the $\chi^2$ for each disruption. (b) $\tau_{CQ}/S$ vs $J_p$ using the 80%–20% definition, bit filtering out all disruptions where $\chi^2 > 0.005$.

It’s also worth noting that for the analyses on HBT-EP shown here, we used $J_p$ and $S$ at the start of the fast linear decay phase (80% or crit%) instead of at the onset of the disruption, which has been used to characterize other devices. The use of disruption onset values is common because these values are more readily available and better for making projections for future machines, however, they are less appropriate to use than their phase-transition counterparts. Using $J_p$ and $S$ at the onset of the disruption on HBT-EP still gives the same trends shown above, but using their values at the start of the fast decay phase reduces the spread even more.

This new dynamic $\tau_{CQ}$ definition also allows us to better investigate what causes the transition from the slow to fast phase to occur. Two common trends were found to describe the onset of the phase transition, these being that (i) $I_p - \frac{I_p}{I_{p,0}} \geq 60\%$ (shown in Figure 4.8(a)) and (ii) $a_{crit} = f(I_{p, crit})$ (shown in Figure 4.8(b)), where $a_{crit}$ is the minor radius of the plasma at the critical time. The narrow range of the $I_p - \frac{I_p}{I_{p,0}}$ helps to explain why a fixed $\tau_{CQ}$ definition works well enough for some machines. Only disruptions with $I_p - \frac{I_p}{I_{p,0}}$ below the chosen $X\%$ will have significant error. On HBT-EP however, Figure 4.8(a) shows that a significant fraction of the disruptions fall below the 80% line, consistent with the large spread in $\tau_{CQ}/S$. A fixed starting percent closer to 60% instead might then lead to a significant reduction in spread without the need to use a dynamic definition. The roughly linear relationship between $a_{crit}$ and $I_{p, crit}$ is possibly more indicative of the cause of the transition, and suggest that the transition occurs at a specific minor radius that
4.1. DISRUPTIONS ON HBT-EP

Figure 4.8: 2D histograms of global parameters associated with the critical point transition on HBT-EP over 8000 shots. Color-scales are logarithmic, and a minimum bin count of 3 shots was enforced. (a) $I_{p,crit}$ vs $I_p$. The dashed lines correspond to $I_p - frac = 100\% / 80\% / 60\%$ for black, blue, and orange respectively. (b) $a_{crit}$ vs $I_{p,crit}$.

depends on the plasma current. However, the exact explanation for this is still unclear.

4.1.4 Post-TQ $T_e$

As noted in the previous section with the relationship between $\tau_{CQ}/S$ and $J_p$, the post-TQ temperature is often of great interest as it sets the $L/R$-time of the plasma current. Most machines report ubiquitous post-TQ temperatures on the order of $\sim 10-50$ eV [42, 97, 100, 101, 103, 104], and similar temperatures are observed on HBT-EP as well. The post-TQ $T_e$'s on HBT-EP have been measured through two distinct methods. The first makes use of the measurement of the CQ-time ($\tau_{CQ}$) and the second makes use of the loop voltage measurement ($V_{loop}$) during the CQ. Both methods more specifically measure the conductivity temperature ($T_{e,conduct}$), which is an approximation of the average $<T_e>$ assuming the conductivity of the plasma is dominantly set by classical collisions. However, the interest in the post-TQ $T_e$ stems from its effect on the conductivity, so in that context these measurements are appropriate.

The $\tau_{CQ}$ method has been used to characterize the post-TQ $T_e$ on many machines [42, 100, 103, 104]. It is the most common method because of the simplicity of the calculation and the single scalar $T_e$ it assigns to each disruption. This method uses the measurement of $\tau_{CQ}$ to approximate the $L/R$-time of the plasma and assumes that the resistivity is set by the classical Spitzer resistivity
4.1. DISRUPTIONS ON HBT-EP

[105] to relate $\tau_{CQ}$ and $T_e$. The self-inductance of a torus is given by

$$L = \mu_0 R_0 \left\{ \ln\left(\frac{8R_0}{a}\right) - 2 + \frac{l_i}{2} \right\}$$ (4.2)

where $R_0/a$ is the aspect ratio and $l_i$ is the normalized self-inductance. The Spitzer resistivity is given by [42, 100]

$$\eta_S = (5.2e-5) Z_{eff} \ln(\Lambda)/T_e^{3/2}$$ (4.3)

where $Z_{eff}$ is the effective ionization state of the plasma, $\ln(\Lambda)$ is the Coulomb Logarithm, and $T_e$ is given in eV. The $L/R$-time is then

$$\tau_{L/R} = \frac{L}{2\pi R_0 \eta_S / S}$$ (4.4)

where $S$ is the poloidal cross-section of the plasma. So $T_e$ [eV] can then be calculated as

$$T_e = \left( \frac{(3.3e-4) Z_{eff} \ln(\Lambda)}{\mu_0 S \left\{ \ln\left(\frac{8R_0}{a}\right) - 2 + \frac{l_i}{2} \right\} \tau_{L/R}} \right)^{2/3}$$ (4.5)

Measurements of $Z_{eff}$, $\ln(\Lambda)$, and $l_i$ are not available for HBT-EP disruptions, but these variables often don’t change significantly, so assumptions for them are made. The Coulomb Logarithm is weakly varying characteristic of the plasma that we approximate as $\ln(\Lambda) \approx 11$. The term $\{\ln(8R_0/a) - 2 + l_i/2\}$ only weakly depends on $l_i$, which is of order 0.5, so we assume $l_i \sim 0.5$ (a flat current profile). The $Z_{eff}$ is a more machine dependent quantity, and is approximated as $Z_{eff} \sim 1.4$ for HBT-EP. The exponential decay time $\tau_{L/R}$ is then approximated by the linear decay time $\tau_{CQ}$, allowing for a measurement of $T_e$ to be made. Example calculations of $T_e$ using $\tau_{CQ}$ are shown in Figure 4.9.

The $V_{loop}$ method is less often used, but provides a time-resolved measurement of $T_e$. It assumes that the plasma current evolution is well described by the circuit equation

$$V_{\text{plasma}} = R_S I_p - L \frac{\partial I_p}{\partial t}$$ (4.6)

where $V_{\text{plasma}}$ is the plasma’s toroidal loop voltage and $R_S$ is the Spitzer resistance. Using equation
4.1. DISRUPTIONS ON HBT-EP

(4.2) to describe $L$, $\eta_S$ can be written in terms of $V_{\text{plasma}}$.

$$\eta_S = \left( V_{\text{plasma}} + L \frac{\partial I_p}{\partial t} \right) \frac{S}{2\pi R_0}$$

(4.7)

$\eta_S$ is calculated assuming once again that $l_i \approx 0.5$ and $Z_{\text{eff}} \approx 1.4$, and can then be used to calculate $T_e$ by inverting equation 4.3. The complication that often comes with this method, however, is in the precise measurement of $V_{\text{plasma}}$. Most machines employ a loop voltage monitor ($V_{\text{loop}}$) to approximate $V_{\text{plasma}}$, but this approximation becomes invalid during the disruption. The difference between $V_{\text{plasma}}$ and $V_{\text{loop}}$ is a result of the loop voltage monitor being placed some finite distance away from the plasma. On HBT-EP the loop voltage monitor is located on the inboard midplane just outside of the vessel. This results in differences between the two when there is a significant change in flux between the two loops. There is always a difference in flux between two which is caused by the poloidal field coils and the poloidal field generated by the plasma, but the change in this flux difference is often small during flattop, when $I_p$ doesn’t change significantly. During the disruption, however, the current quenches and this difference in flux changes significantly, resulting in a significant difference between the enclosed fluxes of the two loops. Fortunately, for a circular plasma like HBT-EP this flux difference can be approximated and used to convert the measured $V_{\text{loop}}$ to the needed $V_{\text{plasma}}$. The difference in flux can be approximated by

$$\Delta \Phi = \int_0^{2\pi} \int_{R_{\text{loop}}}^{R_{\text{edge}}} B_\theta(r) dR d\phi \approx 2\pi R_0 (R_0 - R_{\text{loop}}) \frac{\mu_0 I_p}{2\pi a}$$

(4.8)

where $R_{\text{edge}} = R_0 - a$ is the inboard edge of the plasma ($V_{\text{loop}}$ is on the inboard side of the plasma), and we have approximated that $\int_0^{2\pi} \int_{R_{\text{loop}}}^{R_{\text{edge}}} dR d\phi \approx 2\pi R_0 (R_0 - R_{\text{loop}})$, $B_\theta(R) \approx B_\theta(a)$, and that $B_\theta$ can be approximated as a straight wire. This quantity is calculable from measurements of $I_p$, $R_0$, and $a$, so the conversion from $V_{\text{loop}}$ to $V_{\text{plasma}}$ is then given by

$$V_{\text{plasma}} = V_{\text{loop}} + \frac{(\partial \Delta \Phi)}{\partial t}$$

(4.9)

With this measurement of $V_{\text{plasma}}$, equations (4.3) and (4.7) can then be used to dynamically measure $T_e$. Example calculations of $T_e$ using $V_{\text{loop}}$ are shown in Figure 4.9 as well.

Figure 4.9 shows that both the $\tau_{\text{CQ}}$ and $V_{\text{loop}}$ methods agree well with each other during the fast phase of the CQ. The stronger agreement during the fast phase is expected considering that
4.1. DISRUPTIONS ON HBT-EP

Figure 4.9: Calculations of the post-TQ $T_e$ for three disruptions on HBT-EP. Each figure shows $I_p$, $V_{\text{plasma}}$ (toroidal loop voltage in the plasma), $T_e$ calculated using $V_{\text{loop}} (T_e(V_{\text{loop}})$, $T_e$ calculated using $\tau_{\text{CQ}}$ (black dashed line), $T_e$ calculated using $\tau_{\text{slow}}$ (dashed red line), the normalized $L/R$-time calculated using $V_{\text{loop}}$, and the normalized CQ-time (dashed black line). The vertical lines denote the current-spike (blue), critical time (orange), and $t_{20}$ (green). (a) Shot 110428. (b) Shot 113026. $T_{e,\text{slow}}$ is outside the window at $\sim 42\text{eV}$. (c) Shot 110414.

$\tau_{\text{CQ}}$ is defined with respect to the fast phase. During the fast phase, $T_e$ is on the order of 10 eV, which is consistent with the temperatures observed during disruptions on other tokamaks [42, 97, 100, 101, 103, 104]. The vast difference between the pre-TQ temperatures on other machines and that on HBT-EP highlights the ubiquity of the post-TQ temperature.

The advantage of the $V_{\text{loop}}$ method over the $\tau_{\text{CQ}}$ method is illustrated in Figure 4.9, where we see that $T_e$ appears to decrease over the course of the disruption. In addition, it illuminates a potential difference in $T_e$ between the slow and fast CQ phases. Figure 4.9(a) shows a drop in $T_e$ that is mostly constant over the CQ, but there are some cases like shot 113026 (Figure 4.9(b)) which see a sharp drop in $T_e$ at the critical point correlated with a small $I_p$-spike, indicative of a second TQ. In cases such as these, it is possible that the second TQ is the cause of the transition, where the change in the $L/R$-decay time can be explained by a corresponding change in the temperature. Figure 4.9(c) is a good example of this, where the temperature during the slow decay region calculated using the $V_{\text{loop}}$ method is also consistent with that calculated using the slow decay phases’ CQ-time ($\tau_{\text{slow}} = (t_{\text{crit}} - t_{\text{disrupt}})/(1 - \text{crit}%)$), implying that the change in the quench rate between phases is consistent with the change in temperature. However, the slow phase of the disruption in Figure 4.9(b) actually sees a disagreement between $T_e(V_{\text{loop}})$ and $T_e(\tau_{\text{slow}})$, suggesting that the slow decay for this disruption may not be well described by an $L/R$-decay and that some other effect is involved in setting the decay time during the slow decay phase.

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4.1. DISRUPTIONS ON HBT-EP

Figure 4.10: Disruption magnetic fluctuations for shot 110423. The blue, orange, and green vertical lines represent the current-spike, critical time, and $t_{20}$ respectively. Each time-slice in the contour plots is RMS normalized along the spatial coordinate so as to normalize the colorbar. (a) Plasma current. (b) Edge circular safety factor. (c) Poloidal magnetic field fluctuations (>5 kHz) measured by PA1-01. (c) Contour plot of poloidal magnetic field fluctuations measured by the PA1 sensors which are spatially varying in the poloidal direction. (e) Contour plot of poloidal magnetic field fluctuations measured by the TA sensors which are spatially varying in the toroidal direction on the HFS midplane.

The analyses of the post-TQ $T_e$ on HBT-EP shows that it is similar to that reported on other machines. This has been verified by two distinct methods for calculating the post-TQ $T_e$. It is shown that it is possible that sharp changes in $T_e$ during the CQ (indicative of a second TQ) may explain the transition from a slow to fast CQ decay. However, disagreements between the two $T_e$ measurement methods during the slow phase of some disruptions suggests the existence of other significant mechanisms, like reheating of the plasma during the slow phase.

4.1.5 MHD instabilities

Disruptions on HBT-EP are unique in that large rotating long-wavelength MHD instabilities are present in nearly all discharges and persist throughout most of the CQ. In addition, it is common
4.1. DISRUPTIONS ON HBT-EP

![Image of BD analyses for three periods of MHD activity shown in Figure 4.10; displayed at a fixed toroidal angle, with the poloidal angle varying from the HFS to the LFS. The energy spectrum as well as the first four spatial modes of each analysis is provided. The spatial modes for the dominant mode are given by $m_1$ and those for the subdominant mode are given by $m_2$. (a) BD of the 2/1 dominated pre-disruptive period from -300 → -100 ns. (b) BD of the 3/1 + 5/2 mix early CQ period from 100 → 300 ns. The BD of the TA sensors providing the toroidal spatial structure are also provided in $n_1$ and $n_2$. (c) BD of the 3/1 dominated later CQ period from 400 → 600 ns.](image)

for multiple modes to be present simultaneously. Most other tokamak disruptions only observe single transient modes that become destabilized over the course of the disruption as $q_{\text{edge}}$ drops to an unstable value. HBT-EP, however, is always unstable to an MHD instability immediately following the initiating TQ. The dominant characteristics of these instabilities will change over time though, as $q_a$ drops from its initial value to somewhere between 1 and 2. This is illustrated in Figure 4.10(d), where we have plotted contours of $\delta B_p(t, \theta)$ as measured by the PA1 magnetics for the disruption of shot 110423. This visualization of the MHD perturbation to the edge magnetic field is useful for qualitatively characterizing the poloidal periodicity ($m$) of the MHD instability at the plasma edge, which can be approximated as the number of peaks present at a single time-slice. The toroidal periodicity of the MHD mode ($n$) can also be inferred using the toroidal array (TA) of magnetic sensors.

Figures 4.10(d-e) show the evolution of the dominant $m/n$ MHD structure over the course of the disruption. In them we can see that in this specific shot, the plasma is disrupted by some internal 2/1 mode ($q_a > 2$ suggest that the mode is internal), but following the TQ the structure changes to
become some mix of a $3/1 + m\ell/n\ell$ mode. The mixing of $m$-numbers is apparent in the contour plot as the stripes become less coherent, but we can still qualitatively make out the dominant component of the mixed modes (the $3/1$). This observation is also supported by the Biorthogonal Decomposition of the poloidal array in this region of time (Figure 4.11b), which more explicitly shows the dominance of the $3/1$ mode in this case, but also reveals that the mode it’s being mixed with is a $5/2$. The relative mixing of these two modes compared to the pre-disruption dominance of the $2/1$ can be seen in the energy spectra of the BD modes (Figure 4.11). The energy spectrum of the pre-disruptive $2/1$ mode shows more than an order of magnitude separation between the dominant $2/1$ mode and the other subdominant modes. In contrast, the energy spectrum of the mixed $3/1 + 5/2$ modes shows less than an order of magnitude separation.

As the CQ progresses, $q_a$ continues to drop and the structure of the edge MHD instabilities changes. In this case, the $5/2$ mode decays until the $3/1$ mode becomes dominant with little mixing (seen as an increase in separation between the dominant and subdominant mode in the energy spectrum). Then a $2/1$ starts growing and mixes with it. As the plasma moves further inboard and farther away from the LFS set of PA1 sensors, the BD analysis becomes less useful at identifying the mode numbers, and the qualitative contour plots become the best indicator. It’s possible that at some point the $2/1$ mode becomes dominant, but by the end of the disruption the plasma is so far from the LFS sensors that neither the contour plot nor the BD analysis are capable of discerning the dominant mode number(s). While the specific mode numbers and relative mixing of modes varies from disruption-to-disruption on HBT-EP, the overall change in dominant mode structure as $q_a$ decreases and occasional mixing of modes is typical of most HBT-EP disruptions.

The magnitude of these perturbations also increases in time, with a significant increase in strength following the TQ and another increase often occurring around the transition from a slow CQ decay to a fast one.

The nature of these modes appears to be that of external kink/RWM modes. This is based both on the fact that $q_a$ is below $m/n$ during the periods that each of these modes are present. $q_a$ is below 2.5 from the beginning of the disruption, consistent with the $3/1$ and $5/2$ surfaces being located outside of the plasma, and the $2/1$ component of the perturbation does not appear to grow until after $q_a$ falls below 2. The dominance of kink/RWM modes during the disruption is consistent with the profile broadened picture of the disruption. The current-spike at the onset of most disruptions is associated with the flattening of the current profile, suggesting that plasma is more susceptible
4.1. DISRUPTIONS ON HBT-EP

Figure 4.12: Fast camera BD spatial modes windowed over the 4.2→4.5 ms period of the disruption in shot 111622. Focusing on the HFS limiter between sections 8 and 9, as viewed from the section 10 port. (a) The background mode 0. Highlights the objects in view of the camera. (b) RMS of the mode pair reconstructed signal. The red horizontal line denotes the midplane on the HFS.

to external modes than internal ones. This increased susceptibility to external modes following the current-spike can be seen in shot 110423, as the pre-disruptive internal 2/1 mode disappears and is replaced by the 3/1 + 5/2 mix following the current-spike. Not all MHD modes during the disruption are necessarily external, however. Some cases have been observed where $q_a$ transiently remains above 3 following the TQ, accompanied by a 3/1 perturbation, implying that the mode at the very beginning of the disruption was instead internal.

These modes always rotate in the electron-diamagnetic drift direction, which is down/up on the HFS/LFS for poloidal motion and counter-Ip for toroidal motion. This is true for both the pre-disruptive and disruption modes. These modes rotate on the order of 20→60 kHz, with the frequency increasing over the course of the CQ, but this will be discussed in more detail in Chapter 6.
4.1.6 Fast-camera tracking of MHD modes

Fast camera imaging has also been used to correlate the rotation of disruption MHD instabilities with regions of high plasma wall contact during the disruption. A camera facing the HFS limiter between sections 8 and 9 (as described in Section 2.4.3) views changes in plasma-wall contact during the disruption. The regions of strongest changes in emission (indicative of changes in PWC) are localized to the plasma-side edge of the HFS limiter, shown as the white region in Figure 4.12(a). To characterize the changes in emission ($\delta \epsilon$) during the disruption, a BD analysis is performed to remove the background emission modes (0 and 1) and isolates the first mode pair of emission, the temporal components of which correlate with magnetic field fluctuations due to the MHD instabilities (Figure 4.13). The BD windows are kept relatively short ($\sim 300$ µs) to avoid including significant changes in the MHD characteristics, like the changes in dominant mode structures discussed previously. The video is then reconstructed using only the first mode pair, and the RMS of the signal in each reconstructed pixel is used to characterize the strength of the emission fluctuations. Figure 4.12(b) shows the RMS of the emission at each pixel taken over the period of strongest PWC during the disruption of shot 111622 (4.2 → 4.5 ms), and shows that the emission fluctuations are strongest on the HFS limiter. We also see that the emission fluctuations on the HFS limiter are slightly stronger below the midplane.

Figure 4.14 shows the spatial structure of the first BD mode pair. In it the poloidal variation in emission fluctuations associated with perturbations from the MHD instability can be seen. A single peak and trough pair can be seen on the flange at a given time. The spatial structure of the emission on the HFS flange also matches that of the HFS PA2 sensors roughly co-located with it. This was verified by mapping the pixels associated with the HFS limiters' plasma-facing side to the poloidal positions of HFS PA2 sensors just toroidally beyond the limiter. The spatial location of each pixel on the HFS limiter is approximated given the dimensions ($\Delta x, \Delta y$) = (3.8 mm, 1.65 mm) for each pixel and that $y = 40$ in Figure 4.12(a) roughly corresponds to the HFS midplane. These dimensions were estimated using the known radial width of the HFS limiter and the vertical separation between the TA sensors and the bolts holding them in place (both in view of the camera) for the horizontal and vertical directions respectively. These dimension were calibrating using HFS wall dimensions, which make them applicable for approximating positions on the HFS wall, but no where else. Given the spatial positions of each pixel on the HFS limiter, they can then be mapped to the poloidal plane.
4.1. DISRUPTIONS ON HBT-EP

Figure 4.13: Correlation between the BD temporal component of the emission fluctuations and the magnetic fluctuations (PA1-01). (a) Temporal signal of the $\delta B_p$ sensor PA1-01 (black) and temporal component of the fluctuating emission's BD mode (red) over the BD window. (b) Cross-correlation of the two signals.

Figure 4.14: Spatial structure of the first BD mode pair from the fast-camera emission. Yellow corresponds to regions of stronger emission while blue corresponds to weaker emission. (a) Mode 2 - the higher energy mode of the pair. (b) Mode 3 - the lower energy mode of the pair.
4.1. DISRUPTIONS ON HBT-EP

Figure 4.15: Spatial structure comparison of PA2 magnetic fluctuations and HFS limiter emission. The phases of the \( \cos \) and \( \sin \) BD modes for both the emission and magnetics are artificially fixed using appropriate linear recombination of the mode pairs. This was done to ease comparisons of their structures. (a) Poloidal locations of the HFS PA2 sensors shown in red. The LFS PA2 sensors are in view of the camera. (b) \( \cos \) BD spatial modes for camera emission (black) and PA2 sensors (red). (c) \( \sin \) BD spatial modes.

of the PA2 array. There are a total of 5 PA2 sensors at the poloidal positions in view of the camera, PA2-31 → PA2-03, illustrated in Figure 4.15(a). The emission at each poloidal position is then averaged over the radial direction between the white curves shown in Figure 4.12(a) to get a single emission signal per poloidal position. The white curves outline the plasma-facing side of the HFS limiter and were generated manually by varying the radius and center of each curve to approximately match the curvature of the limiter. The poloidal structure of the magnetic fluctuations measured by the PA2 sensors in view and the limiter emission at those poloidal positions can then be compared using a BD analysis to show that they match well (Figures 4.15(b-c)).

The phase relationship between magnetic and emission fluctuations has also been investigated. Figure 4.16(a) shows the five PA2 magnetic signals (\( \delta B_p \)) in view and plots alongside them the radially averaged emission signals at their poloidal positions that were used in the previous BD analysis. It is qualitatively clear from these comparisons that two signals are out of phase, with \( \delta B_p \) lagging behind the emission signal. This observation is quantitatively supported by the Fourier phase difference analysis given in Figure 4.16(b), which shows that phase difference at each sensor is around 60°. This 60° phase lag between \( \delta B_p \) and the emission (which should be indicative of stronger plasma-wall contact) is unexpected, given that radial perturbation of the plasma outwards should be strongest when \( -\delta B_p \) is strongest, implying a phase difference of 180°. This 60° differ-
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Figure 4.16: Phase difference between PA2 $\delta B_p$ measurements and emission measurements poloidally local to those sensors. (a) Temporal signals of $\delta B_p$ and $\delta \epsilon$ at the five HFS PA2 sensor locations. (b) Phase difference between $\delta B_p$ and $\delta \epsilon$. The blue dashed line is the average phase difference between the sensors.

ence observation may then suggest that the regions of strongest emission are not also regions where the plasma is perturbed furthest into the wall. Another possible explanation is that line-of-sight integration effects are shifting the apparent phase of the emission at the limiter surface. A more rigorous reconstruction may need to be done to determine the significance of this effect. The physical relationship between the observed emission fluctuations and MHD perturbations to plasma is therefore still unknown, however it is clear that the two correlate very well in time and space with each other, albeit with an unexpected phase difference.

4.2 Disruptions on C-Mod

Unlike HBT-EP, not every plasma discharge on Alcator C-Mod ends in a disruption. The C-Mod plasma discharge was not designed to destabilize large MHD instabilities on very fast time-scales and C-Mod was equipped with an active plasma control system (PCS) that worked to maintain the plasma equilibrium. However, pushing the boundaries of plasma physics tends to involve operating in new regimes and technical difficulties can lead to undesired effects on the plasma. As a result, disruptions were still common.

This section characterizes the disruptions on Alcator C-Mod similarly to the characterization on HBT-EP, but also introduces a couple more topics not discussed in the previous section. Subsection 4.2.1 describes the general sequence of events associated with disruptions on C-Mod. Subsection
4.2. DISRUPTIONS ON C-MOD

Figure 4.17: Disruption characteristics on C-Mod. (a) Disruption curves of $I_p$, (b) current centroid vertical position $Z_p$, (c) cross-section area $S$, (d) $q_{95}$, and (e) total HC signal. The time-axis is normalized to the current quench time. (f) LCFS of the C-Mod plasma at the three times shown in (a-e), denoted by the vertical lines with corresponding colors.

4.2.2 discusses the CQ characteristics such as CQ-shape and CQ-time. Subsection 4.2.3 is new, and details mitigated-like disruptions on C-Mod. And lastly, Subsection 4.2.4 describes MHD instabilities during disruptions on C-Mod.

**4.2.1 Sequence of Events**

Disruptions on Alcator C-Mod share many of the same characteristics as those on HBT-EP. Figure 4.17 provides waveforms for a sample disruption on C-Mod. The disruptions can be initiated by a multitude of causes, including MHD instabilities, vertical displacement events (VDEs), and radiative collapses. The former two being the most common. A disruption is considered initiated by a VDE if the vertical position of the plasma’s current centroid is below 2 cm at the onset. The disruptions have two stages, first a thermal quench ($\tau_{TQ} \sim 100 \mu s$) and then a current quench ($\tau_{CQ} \sim 2$ ms). Following the TQ, the plasma loses vertical stability (if wasn’t already lost prior to the disruption) and begins to move vertically towards one of its divertors (typically the lower divertor). As the plasma moves towards the divertor it shrinks in size and eventually transitions from being diverted to limited. The shrinking of the plasma cross-section often occurs faster than the current quenches,
resulting in \( q_{95} \) decreasing. As the current quenches, large HCs up to 25% the pre-disruptive plasma current are driven in the walls, but this will be discussed more in Chapter 5. The decrease in \( q_{95} \) can lead to the destabilization of MHD instabilities. However, not every disruption that has \( q_{95} \) drop to low values sees the growth of a rotating mode, which suggests a sensitivity to some other stability parameters of the plasma that have not yet been identified.

### 4.2.2 CQ Shape and CQ time

The shape of C-Mod’s CQs, as can been seen from Figure 4.17(a), often takes on either the reversed “S”-shaped decay or a purely linear decay. The current-spikes at the onset of the disruption are often much larger than they are on HBT-EP. This leads to issues in the critical time finding algorithm used on HBT-EP, and therefore makes it more difficult to assign an appropriate fast CQ decay period. To approximate a CQ-time, the 80%–20% period was used. Figure 4.18 shows the normalized CQ decay time vs the current density at \( t_{80} \) for the disruptions included in the analysis discussed in Chapter 6 (\( D_{RHCA} \)), and in it we see that there is once again a trend in the data suggesting that \( \tau_{CQ}/S \) increases with \( J_p \). The lower bound of the scatter does not show as clear of a trend as it does on HBT-EP, but this may be attributed to using a fixed definition for \( \tau_{CQ} \). An approximate linear lower bound is provided by the line \( \tau_{CQ}/S = 2.5 \text{ [ms/MA]} \times J_p \). This is an order of magnitude

<table>
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<tr>
<th>( \tau_{CQ}/S_{80} ) [( \mu s/cm^2 )]</th>
<th>( J_{p,80} ) [A/cm^2]</th>
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<tbody>
<tr>
<td>2.5 ms/MA</td>
<td>0</td>
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4.2. DISRUPTIONS ON C-MOD

smaller than HBT-EP’s 25 ms/MA, but is close to the \( \sim 5 \) ms/MA observed on JET [42]. Figure 4.18 also differentiates between un-mitigated and mitigated-like (see Subsection 4.2.3) disruptions, and shows that on average the mitigated-like disruptions do appear to have shorter normalized CQ times.

4.2.3 Mitigated Disruptions

The mitigation of disruptions refers to the suppression of the deleterious effects during disruptions. One common form of mitigation is the injection of impurities into the plasma, so that they radiate a larger fraction of the plasmas stored thermal energy during/prior to the TQ, lessening the heat loads on the plasma-facing components and reducing the CQ time to suppress the drive of halo currents. The potential significance of excess impurities in the plasma during the disruption therefore highlights the need to account for their presence when characterizing the disruption. However, very few intentionally mitigated (or “killer pellet”) experiments were performed on C-Mod in the database analyzed. Instead it was far more common for excess impurities to be introduced coincidentally, in cases referred to as “mitigated-like” disruptions. In these cases, impurities were injected into the plasma for reasons other than mitigation and were not intended to disrupt the plasma. However, when too much was injected in, a disruption soon followed, with many of these impurities possibly remaining throughout the disruptive event. We call these cases “mitigated-like”, and there are many cases of this occurring \( (O(30) \in D_{RHCA}) \) for both low and high-Z impurities on C-Mod.

The most common low-Z impurity was Lithium (Li), while the most common high-Z impurities were Neon (Ne) and Argon (Ar). The characterization of impurities in disruptions on C-Mod was categorized discretely based on the injected impurities: (i) no extra impurities (un-mitigated), (ii) significant amounts of low-Z impurities (low-Z mitigated), and (iii) significant amounts of high-Z injected impurities (high-Z mitigated). The extent to which these mitigated-like disruptions actually experienced significant changes as a result of these impurities is difficult to verify, as there rarely existed un-mitigated reference disruptions to compare them to. The investigation of mitigated-like disruptions is therefore a preliminary analyses, and the inclusion of intentionally mitigated disruptions should be done in the future.
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Figure 4.19: Signature of a rotating MHD instability observed in the toroidal HC sensor array for the disruption of shot 950119017 on C-Mod. HC sensors are plotted vertically on top of each other. The orange and green vertical lines denote $t_{80}$ and $t_{20}$ respectively. The time-base is normalized to $\tau_{CQ}$. The purple vertical line denotes the onset of the instability and the red arrows help guide rotation of the asymmetry.

4.2.4 MHD Destabilization

When $q_{95}$ drops low enough on C-Mod, it’s possible for an MHD instability to become destabilized. These MHD instabilities are not observable in the magnetics on C-Mod, as the Mirnov sensors either are too far from the plasma during the CQ or have too slow a sampling frequency. However, their effects are observable on the HC sensors, like that shown in Figure 4.19. The toroidal array of HC sensors typically indicate an $n = 1$ structure to these modes, and they make anywhere between $<1 \rightarrow 7$ complete rotations, with most cases completing fewer than two complete rotations. Of the cases where at least one complete rotation is made, the rotation frequency is observed to range between $1 \rightarrow 8$ kHz. Toroidal rotation is always in the counter-Ip direction, even when the direction of Ip is reversed (which appears as a flip in the direction of rotation in the lab-frame). More on the rotation of these instabilities will be discussed in Chapter 6.

One major distinction between the MHD characteristics on C-Mod and those on HBT-EP is that the MHD modes are not always present throughout the disruption, and instead only potentially become destabilized at some onset time. This not only severely restricts the number of complete
rotations that can be made by the mode over a single CQ, but it also suggests that only a single mode is usually present at a time, in contrast the multi-mode behavior often observed on HBT-EP. This can simplify the identification of the poloidal structure of the perturbed plasma edge.

Without magnetics to provide the full spatial structure of these modes, it is not possible to say for certain what type of MHD instabilities these are. However, observations on other machines, including HBT-EP, suggest that the destabilized modes are external kink/RWM modes. This suggestion is supported by the distribution of $q_{95}$ values at the onset of the modes shown in Figure 4.20. Here we see that the distribution lies between $1 < q_{95(\text{onset})} < 3$ and is two peaked with a separation at $q_{95} = 2$. These two characteristics suggest that $q_{95}$ crossing either the $q = 3$ or $q = 2$ surfaces is significant in terms of the destabilization of the MHD mode, which is consistent with the external kink/RWM picture. The bounding of the $q_{95}$ distribution between 1 and 3 is similar to what is observed for most of HBT-EP’s disruptions, and in the context of kink/RWM modes suggest the poloidal structure of these modes is either $m = 2$ or $m = 3$. This differs from the observations on JET which see a mode grow after $q_{95}$ falls below 1, resulting in an $m = 1$ structure instead [48].
Chapter 5

Characterization of Halo Currents

Before developing the scaling law used to predict HC rotation (given the global parameters developed in the previous chapter), it’s first necessary to characterize the halo currents themselves. Information pertaining to the magnitudes of these currents, their distribution around the vessel (asymmetries), and their response to the presence of MHD instabilities helps to contextualize the rotation of their asymmetries and the danger it poses. Most notably, the identification of the halo current 2D $(\theta, \phi)$ structure and rotation as matching that of the MHD instability that drives it helps to appropriately characterize their disruption rotation frequency and direction respectively. This chapter characterizes the halo currents observed on both HBT-EP (a limited device) and Alcator C-Mod (a diverted device).

5.1 Halo Currents on HBT-EP

HBT-EP is a limited tokamak, and as such the exchange of halo currents between the plasma and the wall is primarily made through its limiting surfaces: the low-field side (LFS) scrape-off layer current (SOLC) tiles, the blade limiters, and the high-field side (HFS) flanges. All of which are documented in detail in Chapter 2. Of these surfaces only the SOLC tiles are diagnosed (measure current), where they make a direct measurement of the current exchanged between the plasma and the wall at the interface. During the disruption, however, the plasma moves away from the SOLC tiles and into the HFS flanges, eliminating the tiles as a means of measuring the halo current, and leaving the jumper Rogowskis as the best measure of HC.
5.1. HALO CURRENTS ON HBT-EP

The path of the current on a limited device like HBT-EP is complicated severely by the asymmetric nature of its limiting surfaces and insulating breaks, and as a result investigating the explicit paths of the currents is left outside of the scope of this thesis. Instead we focus on the exchange of current at the plasma-wall interface during flattop (post-startup and pre-disruption regime) and the global current characteristics during the disruption.

This section is divided into two parts. The first characterizes the HCs during the flattop phase of operation, where a focus is played on the exchange of HC at the plasma-wall interface and its 2D \((\theta, \phi)\) structure. In this subsection the spatial structure of the RHCAs is shown to match that of the MHD instability driving it. The second characterizes the HCs during the disruption, where a focus is placed on the toroidal HC flowing through the vessel.

5.1.1 Flattop Phase

Equilibrium currents

The \textit{flattop} phase is a bit of a misnomer on HBT-EP, as the plasma current is often still increasing throughout it’s duration. This ramping up of the plasma current may have a direct effect on the evolution of HCs. However this is not clearly seen in the SOLC tile sensors, as the magnitudes of HCs tend to decrease in time. Figure 5.1 shows the SOLC measurements for a shot that exemplifies the typical behavior of flattop HCs on HBT-EP. It shows that the typical magnitude of the currents exchanged at each tile are relatively small (compared to \(I_p\)), on the order of \(2 \rightarrow 10\) A, which is \(< 0.1\%\) the total plasma current. The magnitude of the current at each tile also isn’t significantly different between the high-res (smaller) and low-res (larger) tiles. The two sizes of tiles tend to have similar average currents, however the low-res (LR) tiles tend to have a larger maximum current. The total HC exchanged as measured by the tiles at any given time is on the order of 30 A \(< 0.3\%I_p\), as shown in Figure 5.2. Here the total current entering and leaving the tiles is defined as

\[
I_{total,\pm}(t) = \sum_{i=1}^{N} S_{\pm,i}(t) * I_{tile,i}(t)
\]

where \(S_{\pm,i}(t)\) is 1 if sign of the tile signal at time \(t\) is \(\pm\) and 0 otherwise, \(N\) is the number of tiles, and \(\pm\) corresponds to current leaving the tile and \(-\) corresponds to current entering. The discrepancy between the current entering and leaving the tiles implies that some of the exchanged HC is un-
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Figure 5.1: Finalized SOLC tile measurements for shot 109559 on HBT-EP. Each column corresponds to a different vessel section of tiles and descending rows correspond to the descending positions of tiles from the top (∼+90°) to the bottom (∼−90°). Sections 2 and 10 have low-res (LR) tile resolution and section 6 has high-res (HR) tile resolution. The name for each tile is given by SOLC$X$-$Y$ where $X$ is the section number and $Y$ is given in the legend. The outboard midplane is shown as the horizontal green line. The areas beneath positive values of current are colored red and the areas beneath negative values are colored blue to illustrate the top-down asymmetry in polarities.

diagnosed. The larger of the two is the minimum amount of current circulating between the plasma and the wall. Another measurement of the minimum circulating HC is the ground current (detailed in Chapter 2.3.4), which measures the HC traveling from one insulated side of the machine to other. The ground current also suggests a similar global HC of $O(30 \, A)$. The observation of a smaller amount of positive current entering the tiles than leaving it is possibly consistent with the effects of $I_{sat}$, where the larger positive current leaving the tile is only allowed by the exchange of current in un-diagnosed regions.

These currents tend to be strongest during the first $1 \to 2 \, ms$ of flattop and then reduce in magnitude over time, often eventually stagnating. This characteristic can be found in both the individual tiles and the total current. While the magnitude tends to drop in time, a reversal in the individual tile polarity usually doesn’t occur, at least in the absence of fluctuations and changes in the plasma’s vertical position. The larger current early on in the plasma lifetime is possibly a result of a broader current profile at that time, allowing for a larger density of current to exist in
Figure 5.2: Total HC circulating throughout HBT-EP on shot 109599. The black and red curves denote the total current leaving and entering the walls through the tiles respectively, and the blue curves are the North and West rack ground current measurements.

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the plasma edge and SOL region, that is until the current profile begins peaking.

The polarity of the equilibrium (non-fluctuating) currents is generally top-down asymmetric, with the top tiles emitting positive current to the plasma while the bottom tiles collect it. This reversal in polarity consistently occurs at the outboard midplane, but the inversion point has been seen to move down under the influence of an applied horizontal field pushing the vertical position of the plasma down. The motion of this inversion point with vertical position is shown in Figure 5.3. Feed-forward vertical control is turned on at 2 ms, and with enough strength it reverses the polarity of the current in the section 2 lower-midplane tile SOLC10-2. The polarity inversion point being roughly the point of strongest contact with the plasma is reminiscent of the inboard-outboard asymmetry in divertor tiles (when the plasma is limited), but further investigation of this observation is a topic for future work.

The toroidal asymmetry of the equilibrium currents is sensitive to the limiter and shell configurations. Figure 5.4 compares the section 2 (LR) SOLC tile measurements to those in section 10 (LR) for two sample shots that illustrate the differences in toroidal symmetry between two different limiter/shell configurations. The comparison on the left shows the case where all the limiters (finalized SOLC tiles, prototype SOLC tiles, and blade limiters) and most shells (all but section 5’s) were
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Figure 5.3: Halo currents measured by SOLC10-2 for a scan of the vertical position of the plasma. The black curve (110426) has no feed-forward vertical control (the plasma is roughly centered), the red curve (110250) has weak feed-forward vertical control (the plasma is roughly 0.5 cm down from its usual position), and the blue curve (110253) has strong feed-forward vertical control (the plasma is roughly 1 cm down from its usual position). The green curve is the time trace of the control coils applying a roughly horizontal field to push the plasma down.

Figure 5.4: Toroidal asymmetries in equilibrium HCs on HBT-EP for different limiter/shell configurations. Section 2 tile signals are in black while section 10 tile signals are in red. Each row corresponds to a descending (vertically) finalized SOLC tile sensor. (a) All limiters and all shells (except section 5) fully inserted. (b) Only the section 2, 6, and 8 shells and their corresponding limiters are inserted.
5.1. HALO CURRENTS ON HBT-EP

Figure 5.5: Non-sinusoidal characteristics of flattop HC fluctuations on HBT-EP for shot 110246. (a-red) Raw time-trace of the HCs measured by SOLC2-4 for the period between 3.8-4.6 ms. (a-black) Time-trace of the HCs measured by SOLC2-4 high-pass filtered above 3 kHz. (b) PDF of the windowed HPF signal.

inserted, while the comparison on the right shows the case where only the section 2, 6, and 8 shells were inserted and their corresponding finalized SOLC tiles were the only limiters. This comparison shows that the halo currents are relatively toroidally symmetric (among similar sized tiles) when all the shells/limiters are inserted and that the toroidal symmetry breaks down as shells and limiters are retracted.

Low frequency fluctuations (MHD response)

The low frequency ($O(10 \text{ kHz})$) components of the fluctuating halo currents are not only comparable in magnitude to their equilibrium counterparts, but they’re often much stronger when present. This can be seen in Figure 5.1 in the later half of the shot where the fluctuations are larger enough relative to the equilibrium that the polarity of the currents fluctuates back-and-forth between positive and negative. In contrast to the large fluctuations in the individual tiles, the fluctuations in the total current entering/leaving the walls (Figure 5.2) aren’t as large when compared to their equilibrium, suggesting that the fluctuations are relatively unipolar. Closer inspection of these low frequency fluctuations also illuminates non-sinusoidal characteristics to them. This is exemplified in Figure 5.5, where a window of the halo current fluctuations in SOLC2-4 is enlarged and a PDF (Proba-
5.1. HALO CURRENTS ON HBT-EP

Figure 5.6: Correlation between SOLC 2-4 and FB 2-4 (its local FB sensor) during shot 109085. (a-black) Time-traces of the SOLC 2-4 signal. (a-red) Time-traces of the FB 2-4 signal. (b) Cross-correlation between the two signals taken over the period bounded by the blue vertical lines in (a). The absolute max cross-correlation is 0.71.

bility Distribution Function) of the signal within the window is taken. The PDF has non-Gaussian characteristics to it which quantify the deviation from sinusoidal and clarify what these deviations are. The skewness of the distribution is relatively high at 0.87 and implies that the fluctuations are relatively unipolar (uni-directional). This can be seen in the high-pass filtered (HPF) signal, where fluctuations appear to be strong in one direction and relatively weak in the other. The kurtosis of the distribution is also very low at 0.05 and implies that very little time is spent at the tail ends of the distribution. This can be seen in the HPF signal, where the fluctuations appear more “spikey” in nature, as they spend less time at their peak values. The cause of the unipolarity remains an open question and lies outside of the scope of this work. However, $I_{sat}$ does not appear to be the direct cause, as the weaker phases of the fluctuations still remain in the electron-collection regime. Not all of the SOLC tile sensors exhibit these strong non-Gaussian features either. They tend to be more frequently present in the electron-collecting top tiles.

These same low frequency HC fluctuations correlate strongly with the local perturbations caused MHD instabilities. This can be seen in the high cross-correlation ($\approx 0.71$) between the HC fluctuations and local $\delta B_p$ fluctuations shown in Figure 5.6. This local correlation is observed in all SOLC tile measurements and is also observed globally via the use of Biorthogonal Decomposition (BD) on the poloidal SOLC tile and magnetic arrays. Figure 5.7 shows the temporal and spatial components of the first mode pairs for the section 6 SOLC tiles and the PA1 array during shot
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Figure 5.7: BD analysis of the PA1 magnetic and SOLC6 HC poloidal arrays taken over the period 1.7-2.5 ms for shot 109085. The spatial components for both analyses were re-generated from a linear combination of their originals such that the phase of the cos component at the outboard midplane is 0° and the phase of the sin component is 90°. The blue traces correspond the analysis of the PA1 sensors and the orange ones correspond to the analysis of the SOLC6 sensors. (a) Temporal cos components. (b) Spatial cos components. (c) Spatial sin components.

Figure 5.8: Contour (stripey) plots with poloidal resolution of fluctuations in time for shot 109085 (the same shot and period used in Figure 5.7). (a) Contour of the PA1 sensors. The horizontal white lines correspond the poloidal extent of the HC array. (b) Contour of the SOLC6 sensors.

109085. The phases of the cos and sin components have been fixed via linear recombination of the original pairings to align the phases of the HC and $B_p$ spatial structures at the outboard midplane (0°). This was done to allow for easier comparison of the spatial components. A comparison of the temporal components verifies the correlation already observed in the comparison of HC fluctuations.
and their local $B_p$ perturbations, and further solidifies their global relationship through an improved max cross-correlation of 0.87.

Comparison of the spatial structures shows that the HC fluctuations match the magnetic perturbations not only in time, but also in space. The low frequency HC fluctuations take on similar spatial structures to that of the MHD mode, with $m_{HC} \sim m_{MHD}$ (assuming $m \sim ka$). This can also be seen qualitatively in the contour (stripey) plots of the PA1 and SOLC section 6 tile measurements provided in Figure 5.8. The region between the white dashed lines in the PA1 contour correspond to the same spatial region of the SOLC6 contour, and it is apparent that the structures are qualitatively similar.

The physics responsible for the response of the halo currents to the perturbations caused by MHD instabilities is still an open area of research. Two of the more popular theories attribute the HC response to a stronger exchange of current either from the plasma into the wall (WTKM [32]) or a from the wall into the plasma (ATEC\(^1\) [62]), both of which are facilitated by the edge plasma perturbing further into the wall (which to lowest order is described by the $\delta B_p$ perturbations). However, experimentally distinguishing these two fluctuating current sources is found to be difficult in practice, as they can both result in similar phase relationships between the HC and $B_p$ fluctuations. Additionally, the expected phase relationship for the WTKM explanation can be complicated significantly depending on the structure of the MHD instability and the SOL characteristics.

**Phase difference between MHD and HC fluctuations**

The observed phase relationship between the HC and local $B_p$ fluctuations is found to be complicated as well. This is exemplified in Figure 5.9 which shows the change in the phase difference ($\alpha = \phi_{HC} - \phi_{B_p}$) between HC fluctuations and their local $\delta B_p$ as a function of time. It shows that early on in the discharge the HC and $B_p$ fluctuations are significantly out of phase, with the magnetic signal lagging behind the HCs. Then later on (by about 3 ms) the two fluctuations become relatively in-phase. This evolving phase difference phenomena is present throughout HBT-EP discharges with strong enough MHD activity to elicit significant HC responses.

The evolution of this phase difference appears to be tied to the evolution of the $q_a$-profile, hints of which can be seen in Figure 5.9 as the phase difference relatively stagnates after $q_a$ drops below

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\(^{1}\)The ATEC theory is not really considered a realistic mechanism during the flattop phase, as it lacks the strong toroidal eddy current drive of the CQ, but it’s referenced here as an example of HCs moving from the wall into the plasma.
5.1. HALO CURRENTS ON HBT-EP

Figure 5.9: Phase lag between HC (SOLC4-3 signal) and magnetic (FB4-3 signal) fluctuations for shot 102010 on HBT-EP. Both signals are high-pass filtered (above 3 kHz) to compare their fluctuations. (a) $q_a$ time-trace. (b-black) HC signals from sensor SOLC4-3. (b-red) magnetic signals from sensor FB4-3. (c) Zoom in of the HC and magnetic signals between 2-3 ms.

Figure 5.10: Phase difference between HC and local $B_p$ fluctuations ($\alpha$) vs $q_a$ for 55 shots on HBT-EP. Each time-point represents the cross-correlation over a 200 $\mu$s window that passed a 0.5 max cross-correlation threshold. The analysis for SOLC4-3 and SOLC1-6 (and their respective local magnetic sensors) are shown in black and red respectively.
3. A more thorough investigation of the effect of $q_a$ can be seen in Figure 5.10, which looks at the phase difference as a function of $q_a$ for over 55 shots on HBT-EP. Each data point represents the cross-correlation calculation of the phase difference taken over a 200 µs window (2-3 periods of oscillation) that is rolled over the first 4 ms of flattop to generate 20 points per shot. Points with a max cross-correlation below 0.5 are filtered out to remove contributions from weak MHD periods and/or disruptions. Figure 5.10 shows that the phase difference consistently changes as a function of $q_a$ and can be classified into three regions of $q_a$-space; ($R_1$) $\alpha$ is sporadic for $q_a < 2.5$, ($R_2$) $\alpha$ is a slowly changing function of $q_a$ for $2.5 < q_a < 3.0$, and ($R_3$) $\alpha$ is a fast changing function of $q_a$ for $q_a > 3.0$.

The observed change in $\alpha$ as a function of $q_a$ is too steep to result from the off-local positions of the $B_p$ sensors that naturally introduce the phase difference

$$\alpha_{natural} = m\Delta\theta - n\Delta\phi$$

where $\Delta\theta$ and $\Delta\phi$ are the separation in lab-space of the HC and $B_p$ sensors and $m/n$ are the poloidal and toroidal mode numbers respectively. The changes in $q_a$ typically change $m$ as opposed to $n$, so the slope of $\alpha_{natural}$ vs $q_a$ is roughly $\Delta\theta$, which is only $1.3^\circ$ for the comparison of SOLC4-3 and FB4-3. This is significantly smaller than the $\sim 160^\circ$ observed in $R2$ and the $\sim 600^\circ$ observed in $R3$. Figure 5.10 also shows the $\alpha(q_a)$-profiles for both SOLC4-3 and SOLC1-6, which are toroidally separated by $108^\circ$ but are roughly poloidally co-located. The two profiles overlap quite well, suggesting that this phenomenological lag is toroidally symmetric. Comparisons of HC measurements at the same toroidal location but different poloidal locations also suggest that a small poloidally asymmetry in this phenomenon is present. The explanation for this relationship should investigated further, but is out of the scope of this thesis.

**RMP response**

In a similar manner to how they respond to MHD, the HCs also respond to resonant magnetic perturbations (RMPs). An RMP is the application of a 3D external magnetic field that has a $m/n$ helical structure that resonates with the plasma. The magnetic structure of the plasma edge then perturbs itself into a similar shape if the coupling between the RMP and corresponding $m/n$ MHD instability is strong. They have been used here to study the effects of perturbing the plasma edge
5.1. HALO CURRENTS ON HBT-EP

Figure 5.11: Section 2 halo current responses to a static 3/1 RMP compass scan with $3.5 < q_a < 4.0$. (a) Time-traces of the positive responses (red), negative responses (blue), and the reference case (black). The static RMPs are turned on at 2 ms, denoted by the vertical green line. (b) Polar contour plots of the responses $(HC(\phi_{RMP}) - HC_{ref})$ for each sensor.

on the halo currents in a controlled environment. The RMPs have been kept static to simplify their effect on the currents and a scan of RMP phases (called a compass scan) was performed.

RMPs can have both strong and weak effects on HCs, depending on the phase of the static RMP and the location of the sensor. This is illustrated in Figure 5.11, where the response of the HCs varies with the phase of the static RMP and the sensor’s location. The static RMP is turned on at 2 ms in Figure 5.11(a) and is left on beyond the end of the time-window. The reference shot without a static RMP applied is shown in black, whereas RMP phases that elicited a positive current response are shown in red and a negative response in blue. It can be seen from this coloring that tiles above the midplane tend to collect more electron current in response to an RMP while tiles below the midplane tend to emit it.
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Figure 5.12: Contour of SOLC tile current (normalized to LCFS value) plotted against the radial position of the plasma-facing edge of the tile (relative to the LCFS) and time. Scanned over 5 shots/radial positions in increments of 0.5 cm from 0 → 2 cm. The $\lambda_{J,SOL}$ time-trace is given in red. Only the time window up to 3 ms is included, as the LCFS signal in SOLC10-3 drops too close to zero to continue the analysis after this time. (a) Contour of SOLC10-3 tile current. (b) Contour of SOLC10-4 tile current.

Figure 5.11(b) visualizes these responses via a polar contour-plot representation, where the $\theta$-axis corresponds to the RMP phase, the $r$-axis is time (bounded by the bias window), and the color is the HC response ($HC(\phi_{RMP}) - HC_{ref}$). The contour plotting function extrapolates between applied RMP phases to smooth out the images. Using this representation illustrates more clearly the non-linearity in the response of the HCs to the different phases of the static RMP. The polarity of the response in most cases does not flip with a phase flip of the RMP, and instead each tile has a tendency to respond to the RMP with a specific polarity. This unipolar-like tendency is often strongest in the top and bottom-most tiles, where 80 + % of RMP phases lead to the same response polarity. The response polarities of the two midplane tiles are also notably inverted from each other, which could be related to the top-down asymmetry noted previously in the equilibrium polarities, but could also be a result of $m/n = 3/1$ nature of the RMP applied (for $m = 3$, poloidally adjacent LR SOLC tiles have a natural MHD phase difference of 180°). An explanation for these unipolar-like RMP responses is a topic for future investigation.

SOL width

The decay length of current density ($\lambda_{J,SOL}$) in the SOL has been estimated by scanning the positions of the SOLC tile diagnosed shells one at a time, such that the tiles attached integrate current...
collected over regions further and further out from the LCFS. The decay length of the integrated current collected is the same as that of the current density in the SOL if the decay is exponential, therefore the integrated current collected by the tiles is a good proxy for approximating $\lambda_{J,SOL}$. These scans retracted the current collecting tiles over 2 cm away from the LCFS. Retracting out further would put the plasma-facing edge of the tiles behind that of the shells, and the shells would then become the primary collector of current beyond that radial position.

Figure 5.12 shows the results of an example scan for the section 10 top shell (and correspondingly SOLC tiles 10-3/10-4). The contours plot the radial position of the plasma-facing edge of the tile against time, with the colorbar corresponding to the current collected by the tile normalized the current collected at the LCFS position. The measurement at each radial position (1/2 cm increments) corresponds to a different shot taken with the shell at a new position, and the contouring function interpolates between these positions. Variations in the MHD response phase are also present from shot to shot, requiring that it be filtered out from the HC signals using a LPF. The decay length is calculated at each time-point as the radial position at which the collected current drops to $e^{-1}I_{LCFS}$ (interpolating between radial measurements).

The scan in Figure 5.12 shows these decay lengths as the red curves, which can be seen to range between $\lambda_{J,SOL} \sim 0.5 \rightarrow 1.5$ cm, depending on the time and tile. The tile-dependence suggest some poloidally asymmetry in $\lambda_{J,SOL}$, which can be significant considering there are periods where $\frac{\lambda_{J,SOL}(SOLC10-4)}{\lambda_{J,SOL}(SOLC10-3)} \sim O(3)$. Some poloidal asymmetry is expected, considering that the tiles do not move uniformly away from the LCFS as the shell is retracted (top/bottom tiles move away further for the same motion of the shell), but a factor of 3 difference may be too large to be accounted for by this effect. The time-dependence in SOLC10-4 is transient ($<0.5$ ms) and the cause is unclear seeing as no other transients are noticeable at that time in other diagnostics.

**Tracking current paths**

The finalized SOLC tiles are unable to account for all of the current exchanged between the plasma and the wall, suggesting that there are a significant number of undiagnosed current paths. Evidence of these undiagnosed paths is seen from many sources, the first of which being the total currents entering and leaving the tiles (Figure 5.1) differing by a factor of 2 in some cases. Other sources include Figures 5.13 and 5.14, which compare the electron currents collected in each section to other vessel current measurements and track the collection of bias current respectively.
Figure 5.13: Comparisons of currents from shot 109559 that should match assuming that the SOLC tiles diagnose the exchange of all current between the plasma and the walls. (a) Net current collected by the section 10 tiles plotted against the current passing through jumper 9.5. (b) The net current collected by the section 2 tiles plotted against the current moving from the vessel to the North rack ground. (c) The sum of the current collected by the section 6 tiles and jumper 9.5 plotted against the current moving from the vessel to the West rack ground.

Figure 5.13 looks at the comparison of measured currents at three locations. The first, Figure 5.13(a), compares the net current collected by the section 10 tiles to the current passing through jumper 9.5. If the section 10 tiles truly diagnosed the exchange of all current in that section, then the two would be the same, which is close to what is seen. Figures 5.13(b) and 5.13(c), however, see differences between currents that should be the same. The section 2 tiles are nominally the only limiters in the sections grounded to the North rack and as a result the net current collected by those tiles should be the same as the current moving from the shells to the North rack ground, but there appears to be a significant difference. Similarly, all of the current running from the shells to the West rack ground should be recovered by the section 6 tiles and the current coming from section 10 (jumper 9.5). However, not only is there a large difference between the two, but they even have opposite polarities.

Figure 5.14 partially tracks the paths of currents introduced into the plasma via biasing of the section 10 vessel relative to section 9. Nominally these currents either pass through jumper 9.5 to reach section 9 or leave through the section 10 tiles into the plasma to be collected elsewhere by the other SOLC tiles (before eventually returning to section 9). Figure 5.14 shows that for the latter
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Figure 5.14: Bias current moving from section 10 into the plasma for shot 113062. The blue trace is the total bias current leaving section 10 through the plasma, the red trace is the total bias current emitted by the tiles in section 10, and the black trace is the total bias current recovered by the tiles in other sections \((\sum_{i \neq \text{sec6}} I_{\text{SOLC},i}(\text{bias}) - I_{\text{SOLC},i}(\text{ref}))\).

path not all the current leaving the section travels through the tiles and that not all of the current introduced into the plasma is recovered by SOLC tiles in other sections. The blue trace is the total bias current leaving section 10 through the plasma and the red trace is the total bias current emitted by the tiles in section 10. Comparing the two shows that about a third of the bias current moves into the plasma through paths outside of the tiles. The black trace is the total bias current recovered by the tiles in other sections \((\sum_{i \neq \text{sec6}} I_{\text{SOLC},i}(\text{bias}) - I_{\text{SOLC},i}(\text{ref}))\), which is often less than the total current emitted by the section 10 tiles and smaller by a factor of 2 than the total current emitted from section 10 into the plasma.

Some of the possible undiagnosed paths responsible for these discrepancies include current collected by the HFS limiters, the PA/FB sensor shielding boxes, and the shells (especially considering the deformations of the shells from their nominal positions). Changes to the finalized SOLC tile design to increase the distance between the plasma and the shells by 1 cm were made in an attempt to avoid this issue, but it appears that this change was not enough to suppress the passing of currents through undiagnosed surfaces. Pushing the shells even further back may not be enough to resolve this issue in general either, as it is possible that changing the plasma-wall separation may be influencing the width of the SOL as well. One possible method for diagnosing these previously undi-
agnosed paths could be to install current sensors on the armatures connecting the shells to the rest of the vacuum vessel. This would allow for all the current collected by the shells and transferred to the vessel to be measured. A prototype Rogowski coil was installed around the armature of the top and bottom shells in section 2 so as to get estimates of the discrepancy of current measured between the SOLC tiles in section 2 and that traveling through the shell armature, however a permanent digitization scheme was never established for the diagnostic and a dedicated experiment was never performed. The completion of these experiments and the installation of a complete set of armature sensors remains a topic of future work.

5.1.2 Disruption Phase

As the plasma disrupts, it moves inboard and becomes limited by the HFS limiter flanges. The SOLC tiles are then no longer the primary limiting surface and therefore the magnitude of the current exchanged between them and the plasma drops significantly. This makes the SOLC tiles a poor diagnostic for studying disruption HCs, and the next best option becomes the jumper Rogowskis. The jumper Rogowskis, however, are a powerful HC diagnostic during the disruption, as they can diagnose the total equilibrium (non-fluctuating) current moving toroidally in the wall. A set of high-field side SOLC tiles are currently being installed on HBT-EP that may better diagnose disruption currents, but use of them remains outside the scope of this thesis.

Equilibrium currents

The jumper Rogowskis measure an an order of magnitude increase in the magnitude of the HCs as the plasma transitions from the flattop state to the disruption. This is illustrated in Figure 5.15(c), where both the jumper 9.5 and 3.5 currents grows to be as large as 2% the pre-disruption plasma current ($\sim 400 \text{ A}$), over an order of magnitude larger than the 0.05% observed just prior to the disruption. The magnitude of the equilibrium currents increases in time until the plasma current reaches about 20% it’s pre-disruptive value, where it then quickly falls to zero. The evolution of the equilibrium component of the HCs matches well with the evolution of the plasma current decay ($-\partial I_p/\partial t$), as shown in Figure 5.15(b). This is consistent with the inductive current drive (of the CQ) in the SOL-region. A result of this relationship is that the strongest HCs are localized in time to the periods of fastest current decay, i.e. the fast CQ decay phase of the disruption.

There are small toroidal asymmetries in the toroidal wall HCs, as the equilibrium current in
jumper 3.5 is consistently smaller than that in jumper 9.5 (see Figure 5.15(c)). The explanation for this is likely that the total wall HC is set by the relative resistances of the parallel toroidally co-located jumper and plasma paths. If the resistance of the jumper paths relative to the plasma paths is different between the two jumpers, then this would result in differences in the current drawn from the edge by the jumper paths.

An attempt to thoroughly investigate the relative resistances of the jumper paths to the plasma paths was made by introducing a known bias current into the plasma-wall system and tracking the distribution of current. The section 10 vessel was biased with respect to section 9 while a jumper cable continued to electrically connect the two. The current traveling through the jumper was measured by its Rogowski, the current passing through the plasma from section 10 to 9 via the short toroidal direction was measured via changes in the \( I_p \)-Rogowski, and the remaining current passing through the plasma via the long toroidal direction was implied through the difference between the total bias current and the jumper/\( I_p \)-Rogowski currents. The relative resistances between the jumper paths
and the two plasma paths could then be inferred from the relative measured currents. It was found that about 75% of the bias current was passing through the jumper, suggesting that the resistance of that path was smaller, but comparable to that through the plasma. However, this experiment had a flaw in that it assumes that the HC drive is sourced in the walls, which is significant as it then assumes that the resistances associated with the plasma-wall interface contribute to the resistance of the plasma paths. The resistances associated with this interface can significantly effect the resistances of either paths, depending on which it contributes to. The equilibrium HC, however, is expected to be strongly associated with current driven in the plasma edge by the CQ, and therefore the assumption of a wall-sourced HC drive in this experiment was invalid.

A more successful investigation was a qualitative one that looked at the relative currents passing in parallel between the plasma and jumper paths. This was done by disconnecting one of the jumpers and seeing how it effects the HC in the other. The results of this are shown in Figure 5.16. Case (a) shows the measurements of the two jumpers with and without jumper 3.5 connected, and case
5.1. HALO CURRENTS ON HBT-EP

(b) shows the same with and without jumper 9.5 connected. In both cases, the disconnection of one jumper does not significantly effect the magnitude of the current in the other. This implies that of the parallel paths (jumper path vs plasma path) most of the plasma edge current continues to run through the plasma, as opposed to the jumpers, and that therefore the jumper path is significantly more resistive. The high resistance of the jumper paths is consistent with the possibly high impedance associated with the sheaths that the edge currents must pass through both as the exit and re-enter the plasma.

Rotating HC asymmetries (MHD response)

The low frequency fluctuating components of the halo currents measured by the jumpers are often as large in magnitude as their equilibrium counterparts \( (O(400 \text{ A})) \), but their size diminishes as the CQ progresses (opposite to the evolution of the equilibrium currents). In jumper 9.5 the magnitudes of these fluctuations usually are not large enough for the polarity of the current to change, but in jumper 3.5 (where the equilibrium current is smaller) a reverse in the polarity occurs early on in the CQ. This suggests that at least some component of the fluctuating current drive is in the counter-\( I_p \) direction, which implies that mechanisms like fluctuations in the relative resistances of the plasma and jumper paths can’t explain the observations alone.

The frequency of these fluctuations is on the order of \( 20 \rightarrow 60 \text{ kHz} \) and increases as the CQ progresses (an explanation for this will be given in Chapter 6). Similarly to the pre-disruption fluctuations, these correlate relatively well with the MHD fluctuations in the magnetics, to a max cross-correlation of 0.7 between the jumper 3.5 fluctuations and the most in phase TA sensor TA2-2 (see Figure 5.17). Comparisons of the fluctuations in jumper 9.5 to those in emission at the HFS limiters (observed by the fast camera) also give a relatively strong cross-correlation of 0.6. The correlation with the MHD perturbations suggests that these fluctuations are the result of plasma rotation and that they rotate with the MHD perturbations. Given these characteristics, we refer to this component of the HCs as the Rotating Halo Current Asymmetries (RHCAs). The direction of rotation can’t be identified directly using the HC sensors, since there are only two toroidally separated jumper Rogowskis available, but it can be inferred from the rotation direction of the magnetic perturbations to be in the electron diamagnetic drift direction. The presence of only two HC sensors also makes it difficult to identify any toroidal peaking factors (TPF), but the comparable sizes of the equilibrium and MHD response currents suggests TPFs on the order of 2.
5.2. HALO CURRENTS ON C-MOD

The structure of halo currents on Alcator C-Mod, being a diverted tokamak, is quite different than it is on HBT-EP. One of the most significant differences is that the structure changes as the plasma transitions from a diverted to a limited regime. When the plasma is still diverted, the bulk of the HC exchanged is between SOL and the divertor plates, and does not directly involve the main plasma. As the LCFS of the plasma hits the wall (likely the divertor plates themselves), the plasma then becomes limited and a direct exchange of HC between the plasma edge and the walls can occur. In either case (diverted or limited), for a symmetric vertical displacement of the plasma (and in the absence of 3D effects like MHD instabilities, error fields, or geometry errors) the exchange of HC is toroidally symmetric. Additionally, comparisons of the currents measured entering the divertor plates to those passing through the toroidal array of segmented Rogowskis (which only measure poloidal currents) show identical structure and phase, implying that HCs are dominantly flowing in the poloidal direction through the vessel rather than in the toroidal direction [37]. The poloidal direction is primarily through the shorter path between the inboard and outboard contact points of the plasma, i.e. the through the bottom of the vessel for downward plasma displacements and the top of the vessel for upward plasma displacements (as verified by the disparity in current measured
5.2. HALO CURRENTS ON C-MOD

Figure 5.18: Halo current characteristics on C-Mod for shot 950119017. Disruption curves of (a) $I_p$, (b-black) total poloidal HC normalized to pre-disruption plasma current, $I_{p,0}$, and (b-red) $-\partial I_p/\partial t$ normalized to $I_{p,0}$ and the CQ time for that disruption, $\tau_{CQ}$. The time-axis is shifted to start at the current-spike and is normalized $\tau_{CQ}$. (c) HC signals for 9 of the 10 toroidal partial Rogowski sensors stacked vertically in order of their toroidal positions. Red arrows were drawn on top to follow the growth/rotation of the asymmetry and the vertical dashed purple line is the location in time of the first peak. The time-axis is shifted to start when $I_p$ reaches 80% its pre-disruption value (vertical orange line) and is also normalized to $\tau_{CQ}$ by the full Rogowskis on the top and bottom of the vessel [37]). This suggests that the general paths of the currents resemble something close to (i) entering one side of the wall from the plasma, (ii) moving poloidally to the other side through the shortest path, and (iii) re-entering the plasma through the wall on the other side (see Figure 3.5).

The halo current diagnostics on C-Mod were designed to study disruptions specifically, and as a result they aren’t especially useful for investigating flattop HCs (the gains are too low). Therefore, this section focuses on characterization of HCs during the disruption. Some of the information provided here has already been reported in Ref [37].

**Symmetric currents**

The magnitude of the total poloidal current passing beneath the divertor (as measured by the full Rogowski) can be as large as 25% the pre-disruption plasma current, and peaks concurrently with the peak in the CQ-rate ($-\partial I_p/\partial t$). In fact, the temporal structure of the total HC often resembles the $-\partial I_p/\partial t$-profile very well, as seen in Figure 5.18(b). This is once again consistent with the inductive
5.2. HALO CURRENTS ON C-MOD

Figure 5.19: Distribution of toroidal peaking factors (TPF) among the C-Mod RHCA database (69 disruptions). The TPF is averaged over the first period of rotation following the onset in each disruption.

current drive of the CQ at the plasma edge and suggests that the strongest HC's will be present during the fast CQ period of the disruption. For the normal direction of $I_p$ (clockwise direction when viewed from above) and a downward vertical motion of the plasma, positive current flows from plasma into the inboard divertor plates and re-enters the plasma via the outboard plates, as illustrated in Figure 3.5. The direction of current reverses when the direction of $I_p$ (and simultaneously $B_T$) reverse. Additionally, the direction also reverses with the direction of plasma motion, but this reversal only refers the the poloidal direction, not whether the current enters on the inboard or outboard side.

Rotating HC asymmetries (MHD response)

The rotating HC asymmetries (as seen in the toroidal HC array in Figure 5.18(c)) can be as large in magnitude as their symmetric counterparts when present. These asymmetries do not always appear with every disruption like they do on HBT-EP however, and when they are present they appear only transiently as $q_{95}$ drops below some critical value. This can be seen in Figure 5.18(c) as the onset of the RHCAs (the vertical purple line) does not occur until 20% of the way through the 80% → 20% decay. The toroidal array of HC sensors typically indicate an $n = 1$ structure for these asymmetries, but $n = 2$ structures are seen on occasion. The rotating nature of the asymmetric
HCs suggest that they must be driven in response to the presence of an MHD instability, as the plasma is the only source of rotation and the MHD instabilities are required to generate these long wavelength asymmetries. A direct comparison to the magnetics to further verify the presence of MHD instabilities is not possible, however, since the sampling rate of the magnetics close enough to the plasma is too slow relative to the periods of oscillation. The distribution of TPFs during the first period of rotation for these RHCAs is shown in Figure 5.19. They range from $1.25 \to 2.5$, and average at a little under 2.

The RHCAs make anywhere between $<1 \to 7$ complete rotations, with most cases completing fewer than two complete rotations. Of the cases where at least one complete rotation is made, the rotation frequency is observed to range between $<1 \to 8$ kHz. Toroidal rotation is always in the counter-Ip direction, even when the direction of Ip is reversed (which appears as a flip in the direction of rotation in the lab-frame). The rotation frequency also often does not change significantly as the disruption evolves, and in some cases slows down like that shown in Figure 5.18(c).

Only a minority of the disruption on C-Mod observe noticeable RHCAs. A scan of disruptions on C-Mod from 1995-1997 only found 69 shots (of almost 2100 disruptions) where RHCAs were clearly recognizable and made at least one complete rotation.

### 5.3 Comparing Halo Currents on HBT-EP and C-Mod

HBT-EP and Alcator C-Mod, being limited and diverted tokamaks respectively, share many similarities and differences in their HC characteristics. Despite their differences, a lot of the physics is fundamentally the same. This section is partitioned into first a discussion of the differences in HCs between these two machines, and then second a discussion of their similarities.

#### 5.3.1 Differences

Many of the the differences do not specifically come from the fact that HBT-EP is a limited device, but instead are a result of the limiter configuration that HBT-EP uses. Limited tokamaks that make use of toroidal limiters (located at a single poloidal position and wrap toroidally around the machine) can approximate plasmas limiting on divertors with respect to their toroidal symmetry. HBT-EP, however, uses a toroidal array of poloidal limiters, which themselves introduce a $n \approx 20$ toroidal asymmetry that complicates the paths the HCs can take relative to diverted cases. In addition to
5.3. COMPARING HALO CURRENTS ON HBT-EP AND C-MOD

this toroidal asymmetry, HBT-EP vessel sections are insulated from each other at some locations, introducing other asymmetries as well. These asymmetries complicate significantly the paths of HCs on HBT-EP relative to diverted devices like C-Mod.

HBT-EP and C-Mod also differ in the directions of HC that they measure. C-Mod was found to have primarily poloidally directed currents, with the toroidal components making up only a negligible amount of the current [37]. In contrast to this, the only HCs that can be measured on HBT-EP are the toroidal components at specific locations. However, this does not necessarily imply that the HCs on HBT-EP aren’t dominantly poloidal, because poloidal HC measurements aren’t available. Considering that the magnitudes of the HCs measured on HBT-EP are an order of magnitude smaller than those on C-Mod relative to their respective pre-disruption plasma currents, it’s possible that this difference is the result of most of the HC flowing undiagnosed in the poloidal direction on HBT-EP.

The biggest measurable differences between the HCs on HBT-EP and C-Mod come from the characteristics of their RHCAs, specifically with respect to (i) the frequency of their presence, (ii) their onset, and (iii) their evolution.

RHCAs are present on every single disruption on HBT-EP, however they are more infrequent on C-Mod. This gives HBT-EP a larger database of RHCAs to analyze and makes the automation of the database analysis simpler. An additional algorithm had to be developed for the C-Mod analysis just to find disruptions with RHCAs in the first place. The reason for this discrepancy may just be that C-Mod’s disrupting plasmas are more stable to MHD instabilities or often have growth rates that are longer than the CQ-time, but further investigation is required.

The onset of RHCAs on HBT-EP is always immediately following the TQ while on C-Mod it takes some finite amount of time for the MHD mode to destabilize. This delay in C-Mod onset may be because C-Mod operates at a higher pre-disruption $q_{95}$ than HBT-EP ($q_{95}(C-Mod) \sim 5 \rightarrow 6$ vs $q_{95}(HBT-EP) \sim 3$), which makes it take longer for $q_{95}$ to decrease to an unstable value like 2 (see the $q_{95}$ profile in Figure 4.17). This is corroborated with the observation that rotating asymmetries become destabilized around $q_{95} \sim 2$ on C-Mod (shown in Figure 4.20). The rotation onset time is therefore not necessarily universal, and depends on both $q_{95}$ at the onset of the disruption and the rate at which $q_{95}$ decreases.

The RHCAs on HBT-EP persist throughout most of the disruption and their rotation frequency always increases with time. This contrasts with the evolution of RHCAs on C-Mod, which only
transiently exist and often stagnate in time with respect to their rotation frequencies. The transient existences on C-Mod may be a result of the disrupting plasmas stabilizing faster on C-Mod than they do on HBT-EP, but the stagnation in time is unexpected considering the scaling law for the rotation frequency that will be covered in Chapter 6. The transient existence makes the C-Mod RHCAs more difficult to characterize, but otherwise has beneficial effects with respect to damage to the machine. The shorter the duration, the fewer complete rotations are made, and therefore there is a reduced risk of resonant effects. The stagnation of rotation, however, could have either a detrimental or beneficial effect depending on the context. If the rotation frequency starts off close to the resonant frequency, the ramp up in rotation frequency (if fast compared to the rotation frequency itself) can move the rotation away from the resonant frequency before more than a few complete rotations are made. On the other hand, it’s possible for the ramp up to move the rotation frequency closer to the resonant frequency, introducing resonant effects when there otherwise may have been none.

5.3.2 Similarities

HBT-EP and C-Mod have many similarities with respect to their HCs. The main two being (i) their response to the CQ and (ii) their response to MHD instabilities. HCs on both machines see similar temporal structures to their CQ decay rates \(-\partial I_p/\partial t\). This is because the dominant symmetric HC drive, the inductive current driven at the edge of the plasma/wall by the CQ decay, is the same between the two. They both also see a similar response to the presence of MHD instabilities, where RHCAs are produced that match the MHD modes in frequency and spatial structure (seen poloidally on HBT-EP and toroidally on C-Mod). The TPFs are also similar between the two, with values around 2.

Another significant similarity in the context of this thesis is that the rotation frequencies of the RHCAs on both HBT-EP and C-Mod can be described by the scaling law detailed in Chapter 6, but this will be discussed more then.

One last possible similarity is with respect to the polarities of the HCs on either side of where the plasma makes contact with the wall. On C-Mod, and other diverted tokamaks, the polarities of the HCs on either side of the divertor are opposite, suggesting that current is exiting the plasma on one side and re-entering on the other. A similar observation is seen in the top-down asymmetry of the equilibrium HCs on HBT-EP, where positive currents tended to be emitted by the plasma on the bottom and collected on the top. Whether or not two characteristics are actually a result of the
5.3. COMPARING HALO CURRENTS ON HBT-EP AND C-MOD

same mechanism is, however, a topic for future investigation.
Chapter 6

New Scaling Law for the Rotation Frequency of Asymmetric Halo Currents

The rotation of the rotating halo current asymmetries (RHCAs), diagnosed in the previous chapter, potentially poses a significant threat to the safety of any tokamak device that disrupts. When the rotation is resonant with the plasma’s surrounding structures, the displacement of these structures in response to the EM loads can be amplified significantly, possibly damaging the machine. The potential danger introduced by resonant rotation warrants the need for it to be avoided, and therefore the development of a scaling law to predict the rotation frequency is needed. Making these predictions requires knowing what plasma parameters lead to specific frequencies. The two previous chapters characterizing the disruption and halo currents provide this information, and this chapter introduces a new physically motivated scaling law that clarifies the relationship between the two. It was discussed in Section 1.5.1 that the Myers scaling law under-predicted the rotation observed on HBT-EP by a couple orders of magnitude. This new scaling law resolves this issue and can capture the rotation observed on HBT-EP in addition to that on other tokamaks via the assumption of drift-dominated rotation.

This chapter is structured as follows: Section 6.1 introduces the new scaling law and the poloidal drift physics that motivates it. Section 6.2 validates the scaling law through its application to three
6.1. **THE NEW POLOIDAL DRIFT SCALING LAW**

mostly separate databases; (i) a Multi-Machine database, (ii) the HBT-EP database, and (iii) the Alcator C-Mod database. Section 6.3 compares two parameterizations of the scaling law, showing that the more accessible pre-disruption parameterization is an adequate substitution for the more accurate post-disruption parameterization. Section 6.4 investigates the sensitivity of the scaling law to VDEs and mitigation, and finds that neither influence it significantly. Section 6.5 provides evidence consistent with the dominance of poloidal rotation during the disruption. And finally, Section 6.6 discusses projections of rotation frequencies for next-generation tokamaks like ITER and SPARC, and suggests that resonant rotation is feasible at least in ITER, but that resonant amplification may yet still be avoided.

**6.1 The New Poloidal Drift Scaling Law**

This new scaling law assumes that despite the complexity of the post-disruption plasma, the rotation is dominated by some gradient driven drift. Drifts have the form

\[ u = \frac{F \times B}{|B|^2} \]

and the gradient driven subset assumes that \( F = \nabla G \). Some common gradient driven drifts include the diamagnetic drift (\( F = -\nabla P \)) and in some cases the \( E \times B \) drift (when \( F = E \propto \nabla P \)), both of which in some way use the pressure gradient as the driving force. Other non-pressure gradient drifts are also present, but would not contribute to the following scaling. The post-TQ plasma temperature is relatively ubiquitous, which makes pressure driven drifts similar to within a factor of the temperature’s radial scale length for most machines. This new scaling law constrains \( F \propto \nabla T_e \), i.e.

\[ u \nabla T_e \propto \frac{\nabla T_e \times B}{|B|^2} \quad (6.1) \]

The rotation frequency of a helical asymmetry like RHCAs at the plasma edge given \( u \nabla T_e \) is

\[ f_{\text{rot}} = \frac{m}{2\pi a} u_\theta \pm \frac{n}{2\pi R} u_\phi \quad (6.2) \]
6.1. THE NEW POLOIDAL DRIFT SCALING LAW

where \( a \) and \( R \) are the minor and major radius of the plasma respectively, \( m \) and \( n \) are the poloidal and toroidal periodicities of the helical asymmetry, and the sign relating the poloidal and toroidal components depends on the directions of the two components relative to that of the helicity of the asymmetry. Substituting equation (6.1) into equation (6.2) (assuming neoclassical poloidal damping \cite{114} is insignificant) then reduces to

\[
f_{\text{rot}} \propto \frac{m(\nabla_r T_e)}{B_T a} (1 \pm (\epsilon/q_a)^2)
\]  

(6.3)

where \( q_a = \frac{B_B a}{B_B R} \) is the circular safety factor, \( \epsilon \) is the inverse aspect ratio, and we have assumed \( m/n \sim q_a \). The first term in parentheses on the RHS of equation (6.3) is the poloidal contribution to the rotation frequency while the second term is the toroidal contribution. A comparison of the two shows that the toroidal contribution is smaller by a factor of \((q_a/\epsilon)^2\), which for most conventional tokamaks ranges from \(\sim 30 \rightarrow 100\), suggesting that the helical rotation frequency of a drift is dominantly contributed to by its poloidal component. Based on this dominance, the rotation frequency can be approximated as

\[
f_{\text{rot}} \propto \frac{m(\nabla_r T_e)}{B_T a}
\]  

(6.4)

This scaling law was initially motivated to compare rotation over orders of magnitudes between machines, so the dependence on \( m \) which varies like \( \Delta m/m \sim O(1) \) for long wavelength MHD instabilities can also be removed for simplicity. The last simplification then comes from the ubiquity of the post-TQ \( T_e \) between tokamaks, regardless of size or field strength. This ubiquity in the face of extreme differences in devices was shown in Chapter 4 where the post-TQ \( T_e \) approximated for HBT-EP using multiple distinct methods was similar to that found on other devices. The reasoning for the ubiquitous post-TQ \( T_e \) is expected to be a result of atomic processes during the disruption. The low temperatures during the CQ facilitate stronger radiation and ionization rates which can limit how hot the plasma can get. The ubiquitous post-TQ \( T_e \) can be exploited to simplify the \( \nabla_r T_e \) dependence in equation (6.4) such that

\[
\nabla_r T_e \propto L_T^{-1} \propto a^{-1}
\]
6.1. THE NEW POLOIDAL DRIFT SCALING LAW

where $L_{Te}$ is the radial gradient length scale of $T_e$ at the edge of the plasma. The weak sensitivity to $m$ and the reduction of $\nabla_r T_e$ then finally reduces the equation for $f_{\text{rot}}$ to that of the new poloidal drift scaling law

$$f_{\text{rot}} = A \frac{1}{B_T a^2}$$

(6.5)

where $A$ is some constant that varies weakly with other machine parameters. For rotation dominated by the electron diamagnetic drift, this scaling constant would be approximately $A_{\omega^*} \sim 40 \text{ Hz} \text{ Tm}^2$ (assuming $T_e \sim 10 \text{ eV}$, $m \sim 2.5$, and $L_{Te} \sim 0.1 a$). This poloidal drift scaling law is shown in Section 6.2.1 to be capable of describing the rotation frequency of RHCAs on both HBT-EP and the tokamaks included in the Myers database.

The version of the drift scaling law in equation (6.5) also assumed a circular cross-section when converting from a drift velocity in equation (6.1) to a rotation frequency in equation (6.2), but this assumption can be relaxed to allow for an arbitrary shape. Integrating the drift velocity ($u = u(a(\theta))$) over the LCFS generalizes the drift scaling law to be

$$f_{\text{rot}} = A \frac{1}{B_T (S/\pi)}$$

(6.6)

where $S$ is the poloidal cross-section of the plasma and is normalized to $\pi$ to be consistent with the scaling law in equation (6.5).

The final possible modification to the scaling law discussed in this work is the re-introduction of the dependence on the poloidal periodicity ($m$). This dependence is ignored in equation (6.6) as the changes in $m$ between machines is expected to be small relative to changes in $B_T$ and $S$, but the changes in $m$ can become significant when comparing differences between disruptions on a single machine. For these comparisons, there are two ways that the $m$-dependence can be re-introduced. The first way is that the scale factor ($\frac{1}{B_T (S/\pi)}$ in equation (6.6)) can be modified to include $m$, i.e. $\frac{1}{B_T (S/\pi)} \rightarrow \frac{m}{B_T (S/\pi)}$. However, this method changes the definition of the scaling pre-factor $A \rightarrow A^m$, making comparisons between $A$ and $A^m$ less meaningful. Most machines do not have access to measurements of $m$ during the disruption and therefore the capability to have meaningful comparisons to definitions without $m$ is useful. The second way instead tries to keep comparisons of the scaling pre-factor between definitions with and without the $m$-dependence meaningful. It does
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Figure 6.1: Difference factor between Myers et al’s scale-factor and the drift scale-factor, plotted for the machines considered in Myers et al’s database (black), as well as HBT-EP (blue), and ITER and SPARC (red).

this by dividing $f_{\text{rot}}$ by $m$, but normalizing $m$ to the database averaged $<m>$ (in this context, the database is that used to fit $A$). This makes it appropriate to compare the $m$-dependent $A^*$ to the $m$-independent $A$, since $A$ has the dependence on $m$ absorbed into it and the inclusion of $<m>$ reintroduces the average of that dependence into $A^*$. This version of the scaling law is given by

$$\frac{f_{\text{rot}}}{m/<m>} = A^* \frac{1}{B_T(S/\pi)}$$

(6.7)

6.1.1 Reconciling Differences Between the Myers and Drift Scaling Laws

The most notable difference between the drift scaling and the Myers scaling is the significance of the toroidal field, $B_T$. Myers et al [3] emphasize that their scaling was insensitive to the addition of the toroidal field as a fitting parameter, but this may be because the variance in $B_T$ among the machines in their database is approximately compensated for by comparable changes in other variables, namely $R^{1/2}/a$, that together characterize the difference between the Myers scaling and the drift scaling, $f_{\text{rot}}^{\text{Myers}}/f_{\text{rot}}^{\text{drift}} \approx B_T a/R^{1/2}$. This comparison uses the $t_{\text{rot}} \propto \tau_{CQ}$ scaling introduced in Myers et al [3] and assumes that $\tau_{CQ} \propto S \propto a^2$ based on the LR-decay of the CQ and a resistivity set by the Spitzer resistivity (which has been found to appropriate during the disruption [104]).

The approximate invariance of $B_T a/R^{1/2}$ in the Myers et al database is shown as black data
6.2. VALIDATION

points in Figure 6.1. This may lead to the toroidal field being absorbed into Myers’ scaling pre-factor, $A_{Myers}$ (called $C_f$ in Myers et al [3]). The addition of HBT-EP to the Myers et al database adds significant variance to the $B_{T_a}/R^{1/2}$ factor, and therefore highlights the role of the toroidal field. This is seen in the fact that the Myers et al scaling law under-predicts the rotation frequencies on HBT-EP by an order of magnitude, while the drift scaling does not, as will be shown in Section 6.2.1.

6.2 Validation

The validity of the drift scaling law given in equation (6.5) (all well as its alternative versions in equations (6.6) and (6.7)) has been tested on three mostly distinct databases. The first is a multi-machine database consisting of ranges of observed HC rotation frequencies and pre-disruption drift scale-factors ($\frac{1}{B_{T_a}}$) for each machine. The second is the HBT-EP disruption database, which consists of dynamic post-disruption measurements of $f_{rot}$ and corresponding scale-factors with temporal resolution. The last is a subset of the C-Mod disruption database that includes measurements from the toroidal array of partial Rogowski sensors, which consists of disruption averaged rotation frequencies and scale-factors measured at the onset of the RHCAs. Each of these database analyses find that the drift scaling accurately describes the observed rotation with scaling pre-factors ($A$’s) that are consistent between the databases.

6.2.1 Multi-Machine Scaling

A multi-machine comparison of post-TQ rotation frequencies and their corresponding pre-TQ scale-factors for disruptions on JET [48, 49], C-Mod [37], DIII-D [38], AUG [45], NSTX [47], KSTAR [52], COMPASS [28, 49], and HBT-EP [53] can be found in Figure 6.2(a). Toroidal fields and minor radii are approximated using typical pre-TQ plasma parameters for the given device, and an approximate disruption rotation frequency is estimated based on reported HC rotation frequencies. Equation (6.5) is used to define the drift scale-factor for this analysis. Table 6.1 provides a more detailed description of the sources for the data-points used in Figure 6.2.

A linear fit of the data shows that there is a clear correlation between rotation frequency and the proposed scale-factor over three orders of magnitude and eight devices of widely varying toroidal fields and sizes. The scaling constant ($A_{mm}$) is estimated to be about 292 HzTm$^2$. Even the fast
6.2. VALIDATION

Figure 6.2: (a) Typical disruption rotation frequencies across multiple machines, characterized by their corresponding approximate scale-factors ($\frac{1}{B_r a^2}$). See Table 6.1 for a description of each data-point. The black curve is a linear fit of the log-average frequencies for each machine. (b) Comparison of predicted vs observed rotation frequencies between the Myers (red) and drift (blue) scaling laws. The projection for ITER using the Myers scaling law is given by the yellow window.

* The colored spreads in JET and AUG data represent their carbon wall data-sets, while the black spreads (in addition to their colored spreads) includes their metal wall data-sets. Both datasets were included in the fit for these devices.

** A typical rotation frequency for SPARC is projected ($\sim 60$ Hz) using the linear fit, with nominal SPARC parameters [107] used to estimate its scale-factor.

*** A typical rotation frequency is projected for ITER ($\sim 10$ Hz), with nominal ITER parameters provided by Ref [108].

post-TQ rotation frequencies seen on HBT-EP, which do not fit the scaling law proposed by Myers et al [3], are in good agreement with this proposed scaling. The differences in observed vs predicted frequencies between the two scaling laws in shown in Figure 6.2(b). This is especially significant given that the characteristic scale-factors and rotation frequencies for HBT-EP are an order of magnitude larger than that of the next machine, which lends credibility to this scaling law’s ability to predict the rotation frequencies over several orders of magnitude.

6.2.2 HBT-EP Scaling

This section investigates the validity of the drift scaling law given in equation (6.5) on HBT-EP, measured using the methods described in Chapter 2. While Section 6.2.1 demonstrates the validity of this scaling law between various machines over several orders of magnitude in frequency, only pre-TQ parameters were used to define a scale-factor that characterizes a post-TQ frequency. Measurements of the HC rotation frequency and minor radius during individual disruptions on HBT-EP improve the
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<table>
<thead>
<tr>
<th>Machines</th>
<th>$&lt;f_{rot}&gt;$ [Hz]</th>
<th>$f_{rot}$ spread [Hz]</th>
<th>$&lt;B_T&gt;$ [T]</th>
<th>$B_T$ spread [T]</th>
<th>$a$ [m]</th>
<th>Scale-Factor [T$^{-1}$m$^{-2}$]</th>
</tr>
</thead>
<tbody>
<tr>
<td>HBT-EP$^a$</td>
<td>34,600</td>
<td>20,000 → 60,000</td>
<td>0.35</td>
<td>-</td>
<td>0.15</td>
<td>127.0</td>
</tr>
<tr>
<td>C-Mod$^b$</td>
<td>2,500</td>
<td>750 → 9,000</td>
<td>5</td>
<td>1 → 9</td>
<td>0.22</td>
<td>6.9</td>
</tr>
<tr>
<td>COMPASS$^c$</td>
<td>2,000</td>
<td>1000 → 4,000</td>
<td>1.5</td>
<td>0.9 → 2.1</td>
<td>0.23</td>
<td>13.8</td>
</tr>
<tr>
<td>KSTAR$^d$</td>
<td>1,220</td>
<td>600 → 2,500</td>
<td>1.75</td>
<td>1.7 → 1.8</td>
<td>0.5</td>
<td>2.3</td>
</tr>
<tr>
<td>NSTX$^c$</td>
<td>670</td>
<td>300 → 1,500</td>
<td>0.55</td>
<td>0.35 → 0.55</td>
<td>0.68</td>
<td>4.9</td>
</tr>
<tr>
<td>AUG$^{c,e}$</td>
<td>447</td>
<td>200 → 1000</td>
<td>2.5</td>
<td>-</td>
<td>0.6</td>
<td>1.1</td>
</tr>
<tr>
<td>DIII-D$^c$</td>
<td>400</td>
<td>200 → 800</td>
<td>2.0</td>
<td>1.5 → 2.1</td>
<td>0.67</td>
<td>1.3</td>
</tr>
<tr>
<td>JET$^{c,e}$</td>
<td>100</td>
<td>50 → 500</td>
<td>3</td>
<td>0.6 → 3.8</td>
<td>1.25</td>
<td>0.4</td>
</tr>
<tr>
<td>SPARC$^f$</td>
<td>60</td>
<td>-</td>
<td>12.2</td>
<td>-</td>
<td>0.57</td>
<td>0.25</td>
</tr>
<tr>
<td>ITER$^g$</td>
<td>10</td>
<td>-</td>
<td>5.3</td>
<td>-</td>
<td>2</td>
<td>0.05</td>
</tr>
</tbody>
</table>

Table 6.1: Reported post-disruption HC rotation frequencies and pre-disruption machine parameters used in the multi-machine scaling shown in Figure 6.2. An average rotation frequency ($<f_{rot}>$) and toroidal field ($<B_T>$) was chosen for each machine based on the average in log-space of the corresponding reported spreads. The characteristic scale-factor for each device is estimated using these average parameters.

$^a$ Average rotation frequency and spread seen on HBT-EP. There is no significant spread in the toroidal field for HBT-EP.

$^b$ Average rotation frequency and spread in data reported by Ref [3]. Toroidal field spread information reported by Refs [48] (JET), [37] (C-Mod), [38] (DIII-D), and [47] (NSTX). The frequency range for C-Mod has also been updated based on the observations provided in Chapter 5.

$^c$ Average rotation frequency and toroidal field, along with corresponding spread in data reported by Refs [28, 49].

$^d$ Average rotation frequency and toroidal field, along with corresponding spread in data reported by Ref [52].

$^e$ The JET and AUG data shown here consider observations during both carbon wall (JET-C/AUG-C) and metal wall operation (JET-ILW/AUG-W).

$^f$ Design parameters for SPARC provided by Ref [107]. Projected $<f_{rot}>$ based on fit to scaling law seen in Figure 6.2.

$^g$ Design parameters for ITER provided by Ref [108]. Projected $<f_{rot}>$ based on fit to scaling law seen in Figure 6.2.

validity of this scaling law by establishing a post-TQ scale-factor that characterizes a dynamic post-TQ rotation frequency. Disruptions are often characterized by a decrease in minor radius, and the drift scaling law reflects this in HBT-EP through an increase in HC rotation frequency throughout the evolution of the current quench. HBT-EP is particularly useful for this study, since it lacks many of the features common on larger devices which may require higher-order corrections. Namely, it lacks potential torques/drags driven by external effects like NBI and pellet/gas injection, and it has a relatively simple circular limited plasma cross-section, which helps to avoid any ambiguity in
The most notable characteristic of HBT-EP disruptions is that the rotation frequency (of both the HCs and the magnetics) increases quickly over the duration of the CQ (Figure 6.3(d)). This shows how the change in frequency correlates with the change in the instantaneously measured drift scale-factor. Figure 6.4 shows a 2D-histogram of instantaneous rotation frequencies vs drift scale-factors for the disruptions of 64 shots, sampled every 2 µs (diagnostic sampling frequency). The analyses were performed over the periods of the disruptions corresponding to the fall of plasma current from the 80% to 20% of its pre-disruption value (to be consistent with the existing convention used in Myers et al [3]), with the exception that data-points were only included when the major radius was above some threshold value while the measurement of the major radius was valid. This threshold major radius value corresponds to a minor radius about 0.08 m (about 57% the maximum pre-disruption minor radius), and this allowed for measurement of the dynamic drift scale-factor to range from 1 → 3 times its minimum pre-disruption value.
In Figure 6.4, we see a linear growth in frequency as the drift scale-factor increases (and the minor radius decreases). A linear least-squares fit was performed on the data, and the empirical constant, $A_{HBT}$, for the scaling law was estimated to be about $110 \text{ HzTm}^2$. This is similar to the $O(40) \text{ HzTm}^2$ consistent with an electron-diamagnetic dominated drift rotation, and suggests that $u_*$ is capable of explaining the observed rotation. The frequency predicted by this scaling law using the HBT-EP-specific $A_{HBT}$ constant and the measured scale-factor can be seen plotted against the measured rotation frequency in Figure 6.3(d) (blue), which shows good agreement over the course of the CQ, even prior to the $80\% \rightarrow 20\%$ window.

6.2.3 C-Mod Scaling

This section investigates the validity of the (shaped) drift scaling law given in equation (6.6) on C-Mod, measured using the methods described in Chapter 3. The investigation reveals a dependence on the poloidal MHD mode structure (as indicated by the edge safety factor). This Section is
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Figure 6.5: HC rotation frequencies plotted against $1/B_T(S/\pi)$ for both the disruption-averaged C-Mod data (blue dots) and the dynamic HBT-EP data (red histogram). The dashed black line is the linear fit of the HBT-EP data, and is projected down in $1/B_T(S/\pi)$ to where the C-Mod data exist.

The average rotation frequencies and corresponding scale factors ($1/B_T(S/\pi)$), specified in equation (6.6), are plotted for C-Mod in Figure 6.5 as the blue dots. For comparison, we have also included a projection of the linear fit of the HBT-EP database (from the previous Section) extrapolating to the region of $1/B_T(S/\pi)$-space for the C-Mod data. From this comparison, it is clear that the overall rotation observed on C-Mod is consistent with the HBT-EP scaling. The C-Mod scatter about this prediction is larger than that seen in the HBT-EP database, but the difference doesn’t exceed more
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than a factor of 5.

The main difference between the HBT-EP scaling and the C-Mod scaling shown in Figure 6.5 is that the C-Mod data do not show a clear linear trend between \(<f_{rot}>\) and \(1/B_T(S/\pi)\) when viewed in isolation like HBT-EP does. This linearity is recovered, however, when the poloidal structure of the MHD instability driving the RHCA is included in the scaling law. This is discussed in the next subsection.

Investigating the scaling law with an edge q modification

In the development of the drift scaling law in Section 6.1, a couple of parameters were absorbed into the scaling pre-factor \((A)\) because they were not expected to change significantly from machine-to-machine. One of these parameters is the poloidal structure of the MHD instability driving the RHCA \((m)\). The inclusion of \(m\) comes from the conversion of a poloidal velocity to an apparent poloidal rotation frequency when some periodicity is present \((f_{rot} = \frac{m}{q_{ext}} u_\theta)\), and it is not expected to change significantly from machine-to-machine because \(m\) is expected to be of order unity \((m \sim O(1))\) for all machines. However, while this effect may not be significant when making comparisons over large scales, it is possible that a noticeable effect could become significant when examining disruptions in the context of a single machine. This subsection will show that this effect is not only significant on C-Mod, but that it also explains the lack of a linear relationship between \(<f_{rot}>\) and \(1/B_T(S/\pi)\) in Figure 6.5.

In order to determine if differences in \(m\) are having a significant effect, it is first necessary to be able to characterize \(m\) for the asymmetries in a given disruption. During disruptions on C-Mod, the plasma moves far from most of the magnetic sensors, making them inadequate for use in identifying \(m\). This leaves \(q_{onset}\) as the best indicator of the poloidal structure of the mode.

Simulations of NSTX [109] and observations of \(m\) and \(q_{onset}\) in JET [48] and HBT-EP disruptions suggest that the MHD instabilities driving the RHCAs are external kink modes, which implies that \(q_{onset}\) can be a good indicator of \(m\) if \(n\) is also known. Ideal kink stability states that the most unstable mode for a given \(n\) will have a poloidal structure described by \(m = nq_{ext}\), where \(q_{ext}\) is the lowest integer value of \(q\) outside the LCFS. For \(n = 1\) (observed as the dominant toroidal structure in the HC measurements on C-Mod), this approximately maps \(q_{onset}\) to \(m\) through \(m = ceiling(q_{onset}) = \lceil q_{onset} \rceil\), which is used to determine \(m\) for the following analyses. The error associated with possibly misclassifying an \(m = 2\) mode as an \(m = 3\) mode for values of \(q_{onset}\) close
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Figure 6.6: Distribution of $q_{onset}$ on C-Mod.

to but just above 2 is minimal, considering that there are relatively few cases of $q_{onset}$ at these values (see Figure 6.6).

The distribution of $q_{onset}$ for each disruption case examined on C-Mod is shown in Figure 6.6. We see that $q_{onset}$ always remains above 1 and below 3, implying that the poloidal mode numbers are either $m = 2$ or $m = 3$. This is similar to what is seen for HBT-EP, which also has $q_{95} > 1$ throughout its CQs, but differs from the JET observations [48] which see a mode grow after $q_{95}$ falls below 1, resulting in an $m = 1$ magnetic structure instead.

The effect of $m$ on $<f_{rot}>$ is next exemplified by comparing two separate disruptions where the only significant parameter that changes between the two is their $m$-number. Two such disruptions are shown in Figure 6.7. Both disruptions are caused by vertical displacement events (VDEs), triggered by the excess injection of lithium, have similar CQ times (suggesting similar temperatures), the same $B_T$, and similar plasma cross-sections at the onset of their RHCAs. The latter two similarities suggest that the unmodified scaling law would predict the same rotation frequency. The main difference between the two disruptions is that $I_p$ is smaller in one case, resulting in a larger $q_{95}$ throughout the disruption and a larger $q_{onset}$. Based on $m = \lceil q_{onset} \rceil$, the $m$-number associated with the high-$q_{95}$ case is 3 and that of the low-$q_{95}$ case is 2.
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Figure 6.7: Comparison of plasma parameters for two disruptions on C-Mod that differ primarily in $I_p$ and $q_{95}$. The high-$q_{95}$ case is in red while the low-$q_{95}$ case is in black. Disruption curves for (a) plasma current $I_p$, (b) current centroid $Z_p$, (c) cross-sectional area $S$, and (d) edge safety factor $q_{95}$. The blue vertical line denotes the start of the CQ (the time-axis is also shifted to start here), and the dashed colored vertical lines are the RHCA onset times for both cases. (e) The LCFS of the plasma at the onset of the RHCA.

Figure 6.8(a) plots the HC signals for the two cases with the times shifted such that $t = 0$ is roughly the onset-time of the RHCA. Tracking the motion of the peaks between the two cases shows that the high-$q_{95}$ case rotates faster. The difference in frequencies ($\sim 1.8$ kHz vs $\sim 2.5$ kHz) is consistent with the expected factor of $\sim 1.5x$ associated with the difference between a dominant $m=2$ and $m=3$ periodicity. This is shown in Figure 6.8(b) by scaling the time-base of the high-$q_{95}$ disruption by a factor of 1.5, which results in the overlapping of the peaks in both cases.

We can now take into consideration the effect of the poloidal mode number by modifying the scaling law to normalize $<f_{rot}>$ to the $m$-number, as is done in equation (6.7). Figure 6.9 shows how this modification to the scaling improves the $<f_{rot}>$ vs $1/B_T(S/\pi)$ relationship on C-Mod. Figure 6.9(a) shows the isolated C-Mod data but continues to use the un-modified scaling, and colors each disruption based on whether $q_{onset}$ was below or above 2, indicating which cases are associated with a $m=2$ or $m=3$ structure respectively. A separation in $1/B_T(S/\pi)$-space between the two cases is clear, with $q_{onset}$ being larger for smaller values of $1/B_T(S/\pi)$. This is expected, considering that
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Figure 6.8: Comparison of the toroidal array of HC sensors for the high- (red) and low-\(q_{95}\) (black) disruptions. The legends correspond to the toroidal position of the sensor, and the direction of increasing toroidal angle corresponds to the counter-Ip direction. The time-axis is shifted such that \(t = 0\) corresponds roughly to the onset of the RHCA in each case and the colored vertical lines are the times of the current-spikes in each case. Some HC sensors where not working in either case. (a) No scaling of the time-axis. (b) The time-axis of the high-\(q_{95}\) case is scaled 1.5x.

\(q_{\text{onset}}\) and the scale-factor are roughly inversely related to each other through their dependencies on \(S\).

Figure 6.9(b) then shows the C-Mod data with the modified scaling given in equation (6.7) and shows that the trend between \(\frac{f_{\text{median}}}{m/\langle m \rangle}\) and \(1/B_T(S/\pi)\) is more linear. This increase in linearity can be seen quantitatively with the rise in the Pearson correlation coefficient by a factor of \(\sim 2.6\) (up to \(r_{\text{Pearson}} \sim 0.34\) in linear-space and \(\sim 0.44\) in log-space) from the un-modified to the modified version of the scaling law. A linear fit of the modified C-Mod data also gives a very similar scaling pre-factor \((A_{\text{CMOD}} \approx 116 \text{ HzTm}^2)\) to that found fitting the HBT-EP data \((A_{\text{HBT}} \approx 110 \text{ HzTm}^2)\). This is significant considering that the two databases were completely isolated and live in two very different regions of parameter-space.
Figure 6.9: C-Mod halo current rotation frequencies plotted against their scale-factors $1/B_T(S/\pi)$. Orange cases have $q_{95} > 2$ while blue cases have $q_{95} < 2$. The red dashed line is the HBT-EP scaling projected down to where this database exists. (a) Un-modified scaling. (b) Modified scaling using equation (6.7). The solid black line is the linear fit of the modified C-Mod frequencies.

6.3 Comparison of Parameterizations

Two different parameterizations for the scaling laws provided in equations (6.5) and (6.6) have been discussed so far: a post-TQ and a pre-TQ parameterization. They differ in their specific choices of $B_T$ and $S$, with the post-TQ parameterization using their values at the onset of the RHCA during the disruption and the pre-TQ parameterization using their respective machine’s nominal pre-disruption values. This section compares the use of the two parameterizations to show that the more accessible one (pre-TQ) can be used in place of the more accurate one (post-TQ). A small correction to the pre-TQ parameterization is also introduced based on the relationship between the two parameterizations.

The post-TQ parameterization was used in Figures 6.5 and 6.4 to characterize the rotation on an intra-machine basis for C-Mod and HBT-EP respectively. In contrast to this, the multi-machine analysis presented in Section 6.2.1 used the pre-TQ parameterization. For the multi-machine scaling, the post-TQ parameterization would have been more accurate, but the parameters required to
6.3. COMPARISON OF PARAMETERIZATIONS

Figure 6.10: Comparison of the two different scaling-law parameterizations. The black points use the pre-TQ parameterization, from various tokamaks described in Section 6.2.1, while the red points use the post-TQ one. The linear fit of the pre-TQ case is shown in black, as well as the projection to SPARC (cyan) and ITER (yellow) using it. The linear fit of the HBT-EP post-TQ case is shown in the dashed red line.

use it were not as readily available for all the tokamaks included in the scaling, so the pre-TQ parameterization was used instead.

A comparison of the two scaling parameterizations is shown in Figure 6.10. From this figure it can be seen that post-TQ parameterized scalings (red) are similar to the pre-TQ one (black), with the scaling pre-factors for the post-TQ parameterization ($A_{\text{post}}$) being smaller than that of the pre-TQ parameterization ($A_{\text{pre}}$) by about a factor of 3. This difference is expected considering how the typical plasma-size changes between the two choices. The disrupting plasma-cross section is going to be smaller than its nominal pre-disruption counterpart, so this shifts the scaling to higher values of $1/B_T(S/\pi)$ (to the right in Figure 6.10) and correspondingly drops $A_{\text{post}}$ relative to $A_{\text{pre}}$. The fact that the two parameterizations differ by a factor of about 3 then implies that the plasma cross-sectional area is about 3 times smaller on average during the disruption than it is pre-disruption, which is comparable to what has been observed on HBT-EP (see Figure 6.11). C-Mod, however, sees a larger and broader distribution of drops in the plasma cross-section (see Figure 6.12(a)), closer
6.3. COMPARISON OF PARAMETERIZATIONS

Figure 6.11: Distribution of the magnitude of the drop in poloidal cross-section \( \langle S_{\text{nominal}} / S \rangle \) on HBT-EP, where \( S_{\text{nominal}} \) is the nominal flattop cross-section for HBT-EP and \( S \) is the instantaneous cross-section during the disruption. Each count corresponds to an individual time-step (2 \( \mu s \)) within a disruption, each of which are recorded between the 80\% \( \rightarrow \) 20\% periods. The transparent bins correspond to size measurements where the major radius was below the critical threshold to be reliable. The average \( \langle S_{\text{nominal}} / S \rangle \) only considers bins above this threshold.

Figure 6.12: (a) Distribution of the magnitude of the drop in poloidal cross-section between the disruption and the onset of the RHCA \( S_{\text{disrupt}} / S_{\text{onset}} \) on C-Mod. (b) Distribution of \( q_{\text{disrupt}} \) immediately preceding the disruption. Both distributions only consider disruptions from the RHCA database.
to about 7 times on average, but this is likely attributed to the fact that $q_{95}$ at the start of the disruption is relatively large and broad as well (see Figure 6.12(b)) for the portion of the C-Mod database studied ($<q_{95,\text{disrupt}}(\text{C-Mod})> \sim 4.7$), implying that the plasma cross-section needs to shrink more in size to reach an unstable $q_{95} = 2$. This highlights the fact that using the pre-TQ parameterization requires a calibration factor for predicting $f_{rot}$ that is on the order of

$$C_{\text{pre}} \sim \frac{<q_{95,\text{disrupt}} >_{\text{machine}}}{<q_{95,\text{disrupt}} >_{\text{database}}}$$

where $<q_{95,\text{disrupt}} >_{\text{machine}}$ is the average $q_{95}$ at the onset of the disruption for a given machine and $<q_{95,\text{disrupt}} >_{\text{database}}$ is the $q_{95}$ at the onset of the disruption averaged over the entire multi-machine database. $C_{\text{pre}}$ will not often exceed a factor of 2, so this will not significantly affect the validity of the scaling law given in equations (6.5) and (6.6) when making comparisons over orders of magnitude. Additionally, next generation machines will be inclined to operate at low $q_{95}$ in order to efficiently use the magnetic fields, so a reduction in cross-sectional area during the period of interest in the disruption is expected to be closer to the $\sim 3$ times that’s characteristic of HBT-EP.

The consistency between the pre-TQ and post-TQ parameterizations suggests that the pre-TQ values are relatively accurate at predicting a typical HC rotation frequency. This is convenient considering that pre-TQ parameters are more readily predicted when designing next generation tokamaks, like ITER and SPARC, so a parameterization of the scaling law that uses those values can be confidently used in place of the more accurate post-TQ one.

### 6.4 Insensitivity to VDEs and Impurity Effects

The characteristics of a disruption can be influenced by its cause, so it is important to investigate if the categorization of the disruption has a significant effect on the $<f_{rot}>$ scaling. This section looks at two categorizations, the vertical position and the impurity content of the plasma at the onset of the TQ. This is because of how they affect the size of the disrupting plasma and the post-TQ temperature respectively. Only C-Mod data is considered here, as there are no VDE or mitigated disruptions on HBT-EP.

The vertical position of the plasma at the onset of the TQ depends on whether the plasma loses vertical stability before or after the TQ, with the disruptions of the former case being referred to as
vertical displacement events (VDEs). For VDEs, the plasma will have moved significantly away from its centered position by the start of the TQ. As a result, VDE disruptions will have smaller plasmas at the onset of the TQ. This is potentially significant considering that a smaller cross-sectional area at the onset of the TQ could lead to a larger cross-section at the onset of the RHCA (an explanation of this is given in Appendix A). This could influence the HC rotation scaling because a drop in the machine’s average location in $1/B_T(S/\pi)$-space would subsequently decrease its machine-averaged rotation frequency relative to its non-VDE disruptions.

The impurity content present in the plasma can affect the disruption in a number ways, but the most significant is its potential effect on the post-TQ temperature. One other parameters that was absorbed into the scaling coefficient ($A$) was a possible dependence on $T_e$, so significant changes in $T_e$ due to the presence of excessive impurities could influence $<f_{rot}>$. The most significant changes to the impurity content of the plasma during disruptions tend to come from the use of impurity injection, which is often used to mitigate the effects of heat deposition and EM loads. Considering the use of impurity injections, we categorize the disruption impurity content discretely as (i) no extra impurities (un-mitigated), (ii) significant amounts of low-Z injected impurities (low-Z mitigated), and (iii) significant amounts of high-Z injected impurities (high-Z mitigated). Very few intentionally mitigated (or “killer-pellet”) experiments were performed on the C-Mod database analyzed, so “mitigation-like” cases were investigated instead. In these cases, impurities were injected into the plasma for reasons other than mitigation and were not intended to disrupt the plasma. However, when too much was injected in, a disruption soon followed. We call these cases “mitigated-like”, and there are many cases of this occurring ($O(30)$) for both low and high-Z impurities on C-Mod.

Figure 6.13(a) shows the distribution of VDE vs non-VDE disruptions in $<f_{rot}>-1/B_T(S/\pi)$ space for all the disruptions included in Figure 6.9, and Figure 6.13(b) shows the distribution of un-mitigated, low-Z mitigated-like, and high-Z mitigated-like disruptions. From both, we find that neither of these differences between disruptions significantly affect the validity of the scaling. Additionally, VDE disruptions appear at similar $1/B_T(S/\pi)$ on average.

The fact that the VDE distribution is the same as that for non-VDEs implies that the size of the plasma at the onset of the RHCA does not change significantly between VDE and non-VDE disruptions. This is likely because $S$ often drops much faster over the disruption than $I_p$ does (between $4 \rightarrow 11$ times faster as shown in Figure 6.14), such that the change in $I_p$ has very little
6.4. INSENSITIVITY TO VDES AND IMPURITY EFFECTS

Figure 6.13: Distribution of disruption categorizations in $<f_{rot}>-1/B_T(S/\pi)$ space for all the disruptions included in Figure 6.9. (a) Distribution of TQ and VDE initiated disruptions. (b) Distribution of un-mitigated, low-Z mitigated-like, and high-Z mitigated-like disruptions. These disruption categorizations do not reveal any separation in rotation or $1/B_T(S/\pi)$ behavior.

Figure 6.14: Plasma cross-section shrink rate ($\gamma_S$) vs CQ-rate ($\gamma_{Ip}$) for disruptions on C-Mod included in the RHCA database. Each point represents a single disruption. The rates are approximated using the drops in size and $I_p$ from just before the disruption ($t_0$) to the onset of the RHCA. The shrink rate is approximated as an exponential decay and the CQ-rate is approximated as a linear decay. $\tau_{onset}$ is the time from the disruption to the onset of the RHCA. The lines of $\gamma_S = 4 \times \gamma_{Ip}$ and $\gamma_S = 11 \times \gamma_{Ip}$ are provided in blue and orange respectively.
6.5 Observations Consistent with Poloidal Rotation

Past reports of the rotation of the HCs during the disruption describe the rotation as toroidal [3, 12, 110, 111]. However, this new drift scaling law suggests that the rotation is dominantly poloidal instead. Evidence consistent with this can be seen directly in the dependence of $f_{\text{rot}}$ on the minor radius of the plasma (as opposed to the major radius) as well as the order of magnitude agreement between the observed rotation and the approximate electron diamagnetic drift pre-factor ($A_{\omega_e^*} \sim O(40 \text{ HzTm}^2)$), but there are other factors consistent with this as well, such as the dependence of $f_{\text{rot}}$ on $q_{\text{onset}}$ (shown in Section 6.2.3) and the dependence of the direction of $f_{\text{rot}}$ on the sign of $I_p$ (reported in Myers et al [3]).

The consistency with the direction of $I_p$ requires explanation. Myers et al [3] reports that the direction of observed toroidal rotation reverses with the direction of $I_p$ (and fixed $B_T$), with rotation consistently being in the counter-$I_p$ direction for all five machines included in their database. This
may at first appear inconsistent with equation (6.6), which has no explicit dependence on \( I_p \); however, the picture becomes clear when we consider why a poloidal rotation may be interpreted as toroidal. This conversion from a true poloidal rotation \( f_{\text{poloidal}} \) to an apparent toroidal rotation \( f^*_{\text{toroidal}} \) comes from the helicity of the rotating asymmetry, and is known as the “Barber Pole Illusion [112].”

A cartoon of how the Barber Pole Illusion manifests in this context is shown in Figure 6.15, and the comparison between Figures 6.15(a) and 6.15(b) illustrates how reversing the direction of \( I_p \) (while keeping \( B_T \) fixed) also reverses the direction of \( f^*_{\text{toroidal}} \) in the lab frame.

Considering that the Barber Pole Illusion introduces a dependence on the sign of \( I_p \), this apparent toroidal rotation direction can also include the effect of reversing the direction of the toroidal field, since the Barber Pole Illusion (BPI) conversion is dependent on \( B_T \) as well. However, the direction of \( f^*_{\text{toroidal}} \) is insensitive to the direction of \( B_T \), because the direction of the true \( f_{\text{poloidal}} \) reverses as a result of this change as well, i.e.

\[
\text{sign}(f^*_{\text{toroidal}}) = \text{sign}(f_{\text{poloidal}}) \times \text{BPI} \\
= \text{sign}(B_T) \times \text{sign}(B_T) / \text{sign}(I_p) \\
= \text{sign}(I_p)
\]

There have so far been no experimental reports of the effects of only reversing \( B_T \) on \( f^*_{\text{toroidal}} \), but reversing both \( I_p \) and \( B_T \) on C-Mod has been shown to reverse \( f^*_{\text{toroidal}} \) [37], consistent with this interpretation.

The direction of apparent toroidal rotation relative to that of \( I_p \) for this conversion was also consistent with the direction of poloidal rotation that was observed on HBT-EP. HBT-EP has both fast-camera imaging and magnetic measurements showing that the poloidal rotation is in the electron-diamagnetic drift direction, which is consistent with the Barber Pole Illusion converting it to an apparent counter-\( I_p \) toroidal rotation.

### 6.6 Projections for Next Generation Machines

The purpose of this drift scaling law is not only to provide insight into the physics governing rotation during the disruption, but it is also to predict the rotation frequencies on machines that have yet to be built. These predictions can be used to inform whether or not existing plans for a plasma’s
6.6. PROJECTIONS FOR NEXT GENERATION MACHINES

size and field strength will result in HC rotation that is resonant with the surrounding structures. The plasma parameters can then be adjusted in response to finding resonant rotation or stricter structural requirements can be placed based on the larger expected displacements.

Two of the more exciting tokamaks to be built in the near future are ITER [107] and SPARC [108]. The drift scaling law predicts rotation frequencies on the order of 10 Hz and 60 Hz respectively for the two, based on the scaling pre-factor fit from the multi-machine scaling \( A_{mn} \approx 292 \text{ HzTm}^2 \). The resonant frequencies of the surrounding structures on SPARC are still to be determined, but it has been reported that they are expected to be in the range of 3 \( \rightarrow \) 20 Hz for ITER [113]. The 10 Hz HC rotation projection for ITER then implies that resonant rotation is likely to occur. Additionally, this \( O(10 \text{ Hz}) \) prediction is narrowed down significantly from the 5 \( \rightarrow \) 500 Hz range predicted using the Myers scaling [3], and more confidently stresses the concern for ITER.

ITER’s size and field strength are no longer flexible to change, so the response to the concern of resonant rotation can only be to place restrictions on the structural integrity of ITER’s surrounding structures. Fortunately, significant resonance effects can still be avoided on ITER if \( t_{rot} \) (the rotation duration of the RHCAs) is short enough. This is because a minimum of two complete rotations are required for resonance effects to take place [3]. So if \( t_{rot} < 2/f_{rot} \), then no resonance will occur. For the fastest resonant frequencies of 20 Hz, this corresponds to an ideal \( t_{rot,ideal} < 100 \text{ ms} \). This desired \( t_{rot,ideal} \) can easily be achieved if the CQ-time (\( \tau_{CQ} \)) also meets this criteria, as \( t_{rot} \leq \tau_{CQ} \), and \( \tau_{CQ,ideal} \) conveniently lies within the existing desired CQ-times of 50 Hz < \( \tau_{CQ} < 150 \text{ Hz} \) for ITER (the lower bound is set by EM loads caused by eddies and the upper bound is set by EM loads caused by symmetric HCs [72]). Therefore, if \( \tau_{CQ} \) can be controlled well enough via mitigation methods (which are shown in Section 6.4 to have little effect on \( f_{rot} \)), then it’s possible to keep \( \tau_{CQ} \) below \( \tau_{CQ,ideal} \). \( \tau_{CQ,ideal} \) is also only a soft limit on the CQ-time based on the resonance criteria. This is because \( t_{rot} \) will likely be shorter than \( \tau_{CQ} \) (\( t_{rot} \sim 0.6\tau_{CQ} \) based on Myers et al’s scaling [3]) and there exists mechanisms that allow for \( f_{rot} \) to change on timescales comparable to the rotation period (i.e. the dynamic shrinking of \( S \) seen on HBT-EP and the anomalous deceleration seen on C-Mod). As a result, while the drift scaling law does predict resonant HC rotation frequencies for ITER, significant resonance effects may still yet be avoided.
Chapter 7

Conclusion

The work presented in this thesis addresses the concerns raised about the potential for resonant amplification of EM loads through the development of a new scaling law for the rotation frequency of rotating HC asymmetries (RHCAs). This scaling law is physically motivated as the manifestation of a poloidal drift and describes the rotation frequency as \( f_{\text{rot}} = A \frac{1}{B_T(s/\pi)} \) (or more accurately \( f_{\text{rot}} \frac{1}{m/<m>} = A^* \frac{1}{B_T(s/\pi)} \) if the periodicity, \( m \), of the RHCAs is available). This scaling law is validated over three semi-distinct databases that span several orders of magnitude in rotation frequencies and demonstrates the capability to describe the evolution of the rotation throughout the disruption for the first time. The scaling law is found to be relatively robust to the choices of \( B_T \) and \( S \) used to characterize the rotation, with the more easily accessible pre-TQ choices for parameterizations producing scalings that are consistent with the more accurate post-TQ choices. A small \( q \)-dependent correction factor does differentiate the accuracy of the two parameterizations, but it is small relative to the order of magnitude accuracy of the scaling law. The scaling pre-factors (\( A \)) for the two choices of parameterizations were fit using the three databases and are \( A_{\text{pre}} = 292 \text{ HzTm}^2 \) for the pre-TQ parameterization and about \( A_{\text{post}} \approx 110 \text{ HzTm}^2 \) for the post-TQ one. The order of magnitude for the post-TQ pre-factor is consistent with that expected for the electron diamagnetic drift motivation for of scaling law (\( A_{\Omega_e^*} \sim 40 \text{ HzTm}^2 \)).

The drift scaling law predicts that the rotation on ITER will likely be at resonant frequencies (\( \sim 10 \text{ Hz} \)), with a significantly higher degree of confidence compared to previous attempts to make the same prediction. However, while resonant rotation is likely to occur, resonant amplification can still be avoided on ITER through proper control of the CQ-time. For times shorter than about 100
ms it becomes difficult for even the fastest of resonant rotations to complete two or more rotations, and this limitation can be relaxed to over 150 ms (beyond the symmetric HC $\tau_{CQ}$-limitation) when considering that RHCAs often do not exist throughout the entire duration of the CQ. Projections for resonant rotation on SPARC are found to be on the order of 60 Hz, but the resonant frequencies of its vessel have yet to be reported.

In validating this new scaling law, a clearer understanding of the physics of disruptions has also been developed. The motivation for the drift scaling law, as well as other factors such as the dependence of the rotation on the minor radius, $q_{onset}$, and the direction of $I_p$ are all consistent with the interpretation that the RHCAs are rotating in the poloidal direction, and that they only appear to be rotating in the toroidal direction because of the helicity of the RHCA structure. The installation of the new LFS SOLC tiles on HBT-EP showed for the first time that the 3D spatial structure of the rotating HCs matched that of the MHD instability driving them. A discovery that was instrumental in implementing the $q_{onset}$-dependence in the drift scaling law, but also helps to clarify the relationship between RHCAs and MHD instabilities; especially when considered alongside the discovery of the spatial and temporal correlation of MHD perturbations with changes in PWC that were also shown for the first time on HBT-EP using fast-camera imaging of the plasma limiting on the wall during the disruption.

Lastly, a new dynamic definition for the CQ-time has been developed that is more consistent with the expected temperature dependence and helps to illuminate what causes many disruptions to transition from a slow current decay to a fast one. Instead of calculating $\tau_{CQ}$ as the time to drop from a fixed percentage of the pre-disruption plasma current ($I_{p,0}$) to about 20% $I_{p,0}$ (normalized by the total percent drop), a dynamic percentage is used instead for each disruption which depends on when the plasma transitions from the slow to fast current decay phases. This dynamic definition, when applied to disruptions on HBT-EP, significantly reduces the spread in the relationship between the normalized CQ-time ($\tau_{CQ}/S$) and the plasma current density ($J_p$), which is used to predict the CQ-time on next-generation tokamaks. Because such a large spread is often found to be present in this relationship for other machines which use the fixed $\tau_{CQ}$ definition, the minimum $\tau_{CQ}/S$ is historically the best characterization for the CQ-time available. The development of this new dynamic definition reduces the spread such that the distribution moves towards this minimum, and shows that this $\min(\tau_{CQ}/S)$ vs $J_p$ relationship is actually more representative of the true $\tau_{CQ}/S$ vs $J_p$ relationship than initially anticipated. The implementation of the dynamic CQ-time definition
also shows that the minor radius at which the plasma transitions from slow to fast CQ phases appears to depend on the plasma current at the transition, with the relationship being roughly linear between the two.

7.1 Future Work

This section proposes additional projects that can be done to extend the work discussed in this thesis. Some of the these projects can be continued on HBT-EP, while others involve data analysis on other machines. The projects are ordered based on their time commitment, from the most involved to the least.

1) **Thorough mitigation effects study:** This project involves a more thorough investigation of the effects of mitigation on disruption rotation. In the work presented here on the effects of mitigation on C-Mod, the analysis primarily used data from “mitigated-like” disruptions as opposed to intentionally mitigated disruptions. The difference is two-fold: (i) dedicated mitigation experiments are more likely to actually significantly effect the temperature (and subsequently effect the rotation), and (ii) most “mitigated-like” disruptions did not have good reference shots to compare too, which makes it difficult to determine whether or not the CQ was adequately affected. This project would focus on getting access to data from machines that have both performed dedicated mitigation experiments and are equipped to measure the rotation frequencies of the RHCAs during the disruption. This could involve dedicated mitigation experiments on HBT-EP, but would require additional work to guarantee mitigation. The simplest possible avenue for that direction would be to reconfigure the puff circuit to be able to puff in high-Z materials following startup, effectively using massive gas injection to achieve mitigation. Proper mitigation can be difficult to achieve though, so a thorough investigation of the mitigation capabilities of HBT-EP would need to be done in preparation. The other, possibly simpler option, would just be to get access to mitigated disruption data from other (non-HBT-EP, non-C-Mod) machines. With the data in hand, a more accurate study of the effect of mitigation on the disruption rotation frequency can be done.

2) **VDE effects study:** This project involves determining if VDEs can realistically ever have an effect on a machine’s average rotation frequency. It is shown in Appendix A that VDEs
7.1. FUTURE WORK

are capable of effecting a machine’s average rotation frequency, but Section 6.4 showed that this effect was not significant on Alcator C-Mod. However, this was explained by the fact that the plasma’s poloidal cross-sectional area shrunk \( \gamma_S \) significantly faster over the disruption than the plasma current \( \gamma_{I_p} \), such that the size of the plasma when an MHD mode becomes destabilized doesn’t change significantly (as long as the \( q_{\text{disrupt}} \) is the same). This project aims to determine if it is possible for \( \gamma_S \) and \( \gamma_{I_p} \) to be comparable, resulting in VDEs significantly effecting the machine’s average rotation frequency. This would entail getting access to data from several machines other than HBT-EP and C-Mod that have an abundance of VDE disruptions and the equipment to adequately measure the disruption rotation frequency. A similar analysis to that done on C-Mod in Section 6.4 could then be done over several machines to see how consistent the results are. A larger scope version of the project could involve investigating what sets the relative \( \gamma_S \) and \( \gamma_{I_p} \) rate, and determining if \( \gamma_{I_p} \) can ever be comparable to \( \gamma_S \).

3) Biased halo currents study: This project is a extension of an experiment originally performed on HBT-EP in an attempt to isolate the role of rotating 3D HC paths on HC fluctuations. The experiment would bias a chamber section and study how the the current paths from that chamber section into the plasma varied in the presence of rotating MHD perturbations. The total bias halo current introduced into the plasma would be kept fixed, so that any fluctuations in the distribution of the bias current would be only from the rotation of the MHD perturbations. The relative magnitude of the fluctuating bias currents compared to their non-fluctuating components would inform how significant an effect the rotation of the HC paths would have. A more detailed description of the experiment can be found in the thesis proposal paper and PowerPoint slides for A.R. Saperstein (located on the HBT-EP wiki). The issue with the original performance of this experiment was that the amplifiers used to drive the bias current were limited in the amount of current they could output, and as a result it was difficult to distinguish fluctuations in the bias currents (relative to the small intrinsic fluctuations) without moving the shells such that the PFCs were a couple centimeters beyond the LCFS. However, since the performance of this experiment, newer higher-power amplifiers have been setup to facilitate H-mode biasing experiments on HBT-EP. These amplifiers won’t be as limited in the amount of current they can output, and should hopefully be able to drive current up the \( I_{\text{sat}} \) limit in each of the SOLC tiles (\( \sim O(40) \) A/tile).
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Appendix A

VDE Effect on Plasma Size

This appendix section clarifies how a smaller plasma cross-section at the onset of the disruption may lead to a larger plasma cross-section at the onset of the RHCA. This characteristic follows from the constraint that the RHCA will nominally become unstable at some specific value of \( q_{95} \), which is dependent on both \( I_p \) and \( S \). The relationship is approximately

\[
q_{95} \propto \frac{S}{I_p} \quad \text{(A.1)}
\]

which suggests \( I_p \) and \( S \) have inverse effects on \( q_{95} \). Both \( S \) and \( I_p \) are monotonically decreasing functions over the course of the CQ, so whether or not \( q_{95} \) increases or decreases depends on which one is changing faster. In order for \( q_{95} \) to drop and eventually reach an unstable value (\( q_u \)), \( S \) needs to drop faster than \( I_p \). This is often what’s observed for many disruptions, including the ones discussed in this paper.

In the case where \( S \) drops significantly faster than \( I_p \), then the evolution of \( q_{95} \) is dominantly dependent on the change in \( S \). \( q_{95} \) will then approach an unstable value before \( I_p \) can change significantly. As a result, \( q_{95} \) would always reach the unstable value at the same value of \( S \) (if \( I_p \) starts at the same value), because its evolution doesn’t significantly depend on any other variables.

Now instead, consider the case where the rate at which \( I_p \) drops is smaller, but comparable, to the rate for \( S \). In this case, \( q_{95} \) will eventually reach that same unstable value, but it will take a longer amount of time because the drop in \( I_p \) has the inverse effect of the drop in \( S \). This longer decay time, in conjunction with \( d_t I_p \) being comparable to \( d_t S \), then implies that \( I_p \) will have dropped
Figure A.1: Comparison of two example disruptions showing that a plasma that starts off smaller may actually have a larger cross-section when it crosses an unstable value of \(q\) \((q_u)\). The blue curves/points correspond the larger plasma and the orange curves/points correspond to the smaller one. (a) The evolution of \(I_p\), which is modeled as a linear decay. (b) The evolution of the plasma cross-sectional area, which is modeled as an exponential decay, with a decay rate faster than that for \(I_p\) by a factor of 3. The inset zooms in on the green region. (c) The evolution of \(q_{95}\), resulting from the given \(I_p\) and \(S\)-curves assuming \(q_{95} \propto S/I_p\). Some significant amount by the time \(q_{95}\) reaches the unstable value (at time \(t_u\) following the TQ). Exactly how much \(I_p\) changes over the time \(t_u\) will then influence what \(S\) at \(t_u\) \((S(t_u) = S_u)\) will be. We can compare what \(S\) needs to be to reach \(q_u\) for the cases where \(S\) at the onset of the disruption is either larger \((S_{0+})\) or smaller \((S_{0-})\). An example for this case is illustrated in Figure
A.1, where $S_-$ starts off only 3/4 the size of $S_+$, but both cases have the same $I_p$-evolution and $S$ exponential-decay rate (which is larger than the $I_p$ linear-decay rate by a factor of 3). Figure A.1(c) shows that the smaller case (the VDE) will reach the unstable $q$-value (in this case $q_u = 2$) earlier in time than the larger case (non-VDE). Now because both cases have the same $I_p$-evolution, Figure A.1(a) shows that the VDE case will have a larger plasma current when passing $q_u$ ($I_{p,-u}$) than the non-VDE case ($I_{p,+u}$), resulting in a larger plasma size at that time ($S_{-u} > S_{+u}$) by about a factor of 1.3x. The implications of this last statement can also be expressed quantitatively using the equality below

$$q_u \sim \frac{S_u}{I_{p,u}} = \frac{S_{+u}}{I_{p,+u}} = \frac{S_{-u}}{I_{p,-u}}$$

(A.2)

which uses the assertion that $q_u$ must be the same for both cases to constrain the relationship between $S_{+u}/I_{p,+u}$ and $S_{-u}/I_{p,-u}$. Since $I_{p,-u} > I_{p,+u}$, equation (A.2) enforces $S_{-u} > S_{+u}$, suggesting that the plasma at $q_u$ will be larger if it started off smaller at the onset of the disruption.