# Highlights from Sherwood 2018

International Sherwood Fusion Theory Conference
April 23 – 25, 2018

Hosted by the physics department at Auburn University
Held at the The Hotel at Auburn University and Dixon Conference Center

In total, 111 scientists attended the conference, which, as is customary for the Sherwood Fusion Theory Conference, had 3 plenary talks, 12 invited talks and three poster sessions, over a span of two and a half days.


In his plenary talk on Monday morning, titled "*What will we learn from ITER?*", **Richard Hawryluk** discussed the physics goals of the ITER experiment and highlighted open questions which the fusion community expects the ITER experiment to help answer.

In the second plenary talk, on Tuesday morning, titled "Dust in fusion plasmas: Insights from the Magnetized Dusty Plasma Experiment", **Edward Thomas** presented the capabilities of the magnetized dusty plasma experiment (MDPX) at Auburn University and suggested ways to use MDPX for the study of long-duration plasma-wall interactions, which represent a key area of interest for magnetic confinement fusion experiments.

In his plenary talk on Wednesday morning, titled "*Tokamak Disruption Simulation: Progress toward Comprehensive Modeling*", **Carl Sovinec** gave a summary of the disruption modeling capabilities the fusion theory community has developed in the last forty years, and presented unresolved issues which lie ahead on the path to predictive comprehensive modeling.

The topics of the invited talks covered the whole range of topics which have historically formed the core of Sherwood Fusion Theory conferences: MHD stability of tokamaks and disruption simulation, modeling of runaway electron population dynamics, transport physics in the core and edge of fusion devices, and numerical methods to compute non-axisymmetric equilibria and design efficient stellarator coils.
Author-provided summaries of the invited talks and Carl Sovinec's plenary talk are included at the end of this document.

There was a very strong showing by graduate students, postdocs, and young scientists at the meeting. 25 students attended the conference and presented their work, either in invited talks or in poster sessions.

Six "**Student Poster Awards**" were given to the following students for their exceptional presentations:

**Mike Martin (University of Maryland)** "The Parallel Boundary Condition for Turbulence Simulations in Low Magnetic Shear Devices"

**Joseph Jepson (University of Wisconsin-Madison)** "NIMROD Modeling of Poloidal Flow Damping Using a Delta-F Kinetic Closure"

**Kyle Bunkers (University of Wisconsin-Madison)** "Investigation of Boundary Conditions for Vertical Displacement Events with NIMROD"

**Hongxuan Zhu (Princeton University)** "Wave kinetics of drift-wave turbulence and zonal flows beyond the ray approximation"

**Torrin Bechtel (University of Wisconsin-Madison)** "Finite Parallel Transport on Stochastic Fieldlines Changes Global Stellarator Beta"

**Xiang Fan (UC San Diego)** "Cascades, "Blobby" Turbulence, and Target Pattern Formation in Elastic Systems: A New Take on Classic Themes in Plasma Turbulence"

Student poster award winners. From left to right:
Matt Landreman (University of Maryland, Chair of Sherwood Executive Committee), Kyle Bunkers, Joseph Jepson, Torrin Bechtel, Hongxuan Zhu, Mike Martin, and Jungpyo Lee (MIT PSFC, now Hanyang University – Chair of the Sherwood Program committee)

# Critical Role of Sonic Rotation on Ion and Impurity Transport

Emily A. Belli and J. Candy, General Atomics, P.O. Box 85608, San Diego, CA 92186-5, USA

## Abstract

The influence of sonic rotation on gyrokinetic stability and transport is studied, with important implications for heavy impurity dynamics. Sonic toroidal plasma flow, on the order of the ion sound speed, arises in tokamaks due to external torque driven by neutral beam injection and can have a profound effect on the intensity of drift-wave turbulence. It is common in gyrokinetics to consider the weak rotation limit [1], retaining only the ExB flow, Coriolis drift and toroidal rotation shear. However, correct treatment of the sonic rotation regime [2] requires the additional consideration of centrifugal effects. Because of their complexity, these new sonic terms (quadratic in the Mach number), are ignored in most codes and widely-used reduced models of transport. In this work, the impact of rotation on ion and impurity transport is explored with the gyrokinetic code CGYRO and the drift-kinetic code NEO, both of which implement full sonic rotation. It is found that including only weak rotation terms, while neglecting centrifugal terms, leads to a large error. While the ITG drive is dominantly affected by the Coriolis drift, centrifugal drifts and electrostatic trapping corrections induced by the rotation lead to significant modifications to the heavy impurity particle transport. For impurities in a rotating plasma, both gyrokinetic and neoclassical transport must be considered. The turbulent transport is enhanced by the complex interaction between the Mach number and toroidal rotation shear in the drifts, while the neoclassical transport becomes competitive with the turbulent transport through enhanced effective toroidal curvature drifts. This has significant implications, for example, on detrimental core tungsten accumulation in a reactor.

This work was funded by the U.S. DoE under DE-FC02-06ER54873.

[1] R. Waltz, et al., Phys. Plasmas 14, 122507 (2007).
[2] H. Sugama, Phys. Plasmas 5, 2560 (1998).

# Toroidal drift modes driven by the magnetic drift resonance

Jan Weiland, Chalmers University of Technology and Euratom-VR Association, SE 41296 Gothenburg, Sweden

## Abstract

We here find that the kinetic and fluid linear drift resonances have several similarities. The reason for our interest in this is that our fluid model has recently been shown to be exact for drift waves and other modes in that frequency range. Thus transport is driven by the fluid linear growth rate and our drift wave system behaves like a cold beam-plasma system although it has a finite temperature. A main similarity is that neither fluid nor kinetic responses should be expanded in the curvature in the bulk interior of tokamaks. That we can use the fluid response close to the magnetic drift resonance is a consequence of the fact that the closure is exact. A systematic orbit integration technique is introduced for deriving the fluid model and for evaluating the effects of nonlinearities. This leads to the conclusion that we may use quasilinear theory in fluid descriptions of drift waves while a kinetic description requires a strongly nonlinear formulation.

# Detached plasma regimes in innovative long-legged divertor configurations

M.V. Umansky[a†], B. LaBombard[b], D. Brunner[b], T. Golfinopoulos[b], A.Q. Kuang[b], M.E. Rensink[a], J.L. Terry[b], M. Wigram[c], D.G. Whyte[b]

[a]*Lawrence Livermore National Laboratory, Livermore, CA 94550, USA*
[b]*MIT Plasma Science and Fusion Center, Cambridge, MA 02139, USA*
[c]*York Plasma Institute, Dept. of Physics, Univ. of York, York, YO10 5DD, UK*

Simulations carried out with the tokamak edge transport code UEDGE [1] for parameters specified in the ADX tokamak design [2] show that tightly-baffled long-legged divertors may provide an order of magnitude increase in peak power handling capability compared to conventional divertors, particularly if a secondary X-point is included in the leg volume [3]. Furthermore, a fully detached plasma state can be passively maintained over a wide range of input power from the core. As the power from the core is varied, the detachment front merely shifts up or down in the leg but remains stable. These features are highly desirable for handling power exhaust in a DT fusion reactor where options for detachment control are limited. The key physics for attaining the passively stable, fully detached regime in these simulations involves the interplay of strong convective plasma transport to the divertor leg outer sidewall, confinement of neutral gas in the divertor volume, geometric effects including a secondary X-point, and atomic radiation. Detachment front location is set by the balance between the power entering the divertor leg and the losses to the walls of the divertor channel. Therefore, for a fixed level of power exhaust, the location of the detachment front is insensitive to divertor leg length (Fig.1) – as long as the leg length exceeds the front location. Correspondingly, the maximum power that can be accommodated by the divertor while still staying detached increases with the leg length. In response to variation of model assumptions (magnitude of anomalous radial transport, impurity radiation, neutral transport model, geometry of plasma-facing components), the overall divertor plasma behavior remains qualitatively similar: a stable fully detached regime is maintained, lending confidence in the modeling results. Remarkably, a very similar fully detached long-legged divertor regime can be reproduced for parameters of the ARC reactor [4], as demonstrated by recent ARC divertor modeling [5].

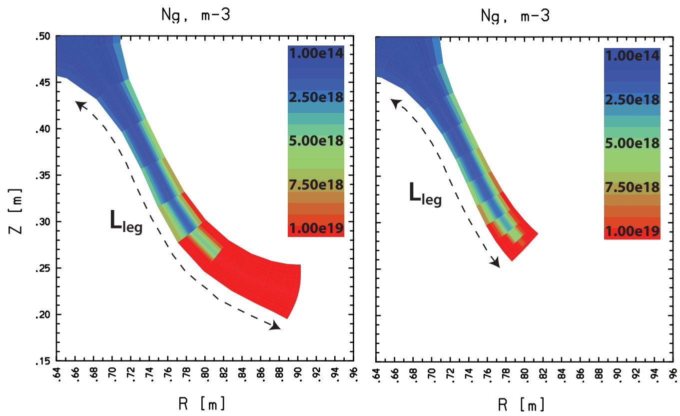


*Fig. 1 – Distribution of neutral gas density is shown for the outer divertor leg, indicating the steady-state location of the detachment front. Regardless of the leg length, the detachment front location is similar for the same exhaust power.*

[1] T.D. Rognlien et al., J. Nuc. Mat. 196, 347–123 (1992); [2] B. LaBombard et al., Nucl. Fusion 55, 053020 (2015); [3] M.V. Umansky et al., Phys. Plasmas 24, 056112 (2017); [4] Kuang et al., Fus. Eng. Design, 137, 221-242 (2018); [5] M. Wigram et al., Contrib. Plasma Phys. 58, 791–797 (2018), also IAEA FEC 2018 conference presentation P7-20, to be submitted to Nuclear Fusion.

## A gyrokinetic model for the tokamak periphery

R. Jorge[1,2], B. Frei[2], A. Baillod [2], P. Ricci[2] and N. F. Loureiro[3]

[1]*Instituto de Plasmas e Fusão Nuclear, Instituto Superior Técnico, Universidade de Lisboa, 1049-001 Lisboa, Portugal*
[2]*École Polytechnique Fédérale de Lausanne (EPFL), Swiss Plasma Center (SPC), CH-1015 Lausanne, Switzerland*
[3]*Plasma Science and Fusion Center, Massachusetts Institute of Technology, Cambridge MA 02139, USA*

Despite significant development over the last decades, a model able to describe the tokamak periphery region extending from the edge to the far scrape-off layer is still missing. This is because this region is characterised by the presence of electromagnetic fluctuations at all scales, the presence of strong flows, comparable amplitudes of background and fluctuating components, and a large range of collisionality regimes. The lack of a proper model has undermined our ability to properly simulate the plasma dynamics in this region, which is necessary to predict the heat flux to the vessel wall of future fusion devices, L-H transition, and ELM dynamics. These are some of the most important issues on the way to a successful fusion reactor. A drift-kinetic model able to describe the plasma dynamics in the tokamak periphery has been recently developed, and is now extended to account for gyrokinetic fluctuations, to retain electromagnetic fluctuations at all scales and allow for comparable amplitudes of background and fluctuating components. In addition, the model implements a full Coulomb collision operator, and is therefore valid at arbitrary collisionality regimes. For an efficient numerical implementation of the model, the model equation is projected onto a Hermite-Laguerre velocity space polynomial basis, obtaining a moment hierarchy [1]. The treatment of arbitrary colisionalities is performed by expressing the full Coulomb collision operator in gyrocentre phase-space coordinates, and by providing a closed formula for its gyroaverage in terms of the gyrokinetic moments, therefore filling a long standing gap in the gyrokinetic literature [2]. The use of systematic closures to truncate the moment hierarchy equation, such as the semicollisional closure, allows for the straightforward adjustment of the kinetic physics content of the model. In the electrostatic high collisionality regime, our model reduces to an improved set of drift-reduced Braginskii equations, which have been widely used in scrape-off layer simulations. The first numerical studies based on our model were carried out, shedding light on the interplay between collisional and collisionless mechanisms that characterise the tokamak periphery. In particular, a numerical study on the linear properties of drift-waves was performed, which allowed to compare the Coulomb collision operator to collision operators used in state-of-the-art turbulence simulation codes [3]. Established collisional and collisionless limits were retrieved and an analysis on both the growth rate and eigenmode spectrum shows the need for retaining the Coulomb collision operator, specially at the intermediate levels of collisionality relevant for present and future magnetic confinement fusion devices.

[1] R. Jorge, P. Ricci, N. F. Loureiro, Journal of Plasma Physics **83**, 6 (2017)
[2] R. Jorge *et al.*, Full-Coulomb Gyrokinetic Collision Operator, *to be submitted to Journal of Plasma Physics*
[3] R. Jorge, P. Ricci, N. F. Loureiro, Physical Review Letters **121**, 16 (2018)

# Computational modelling of quasi-single helicity states in an RFP

A.S. Ware and C. Miele

*Department of Physics & Astronomy, University of Montana, Missoula MT 59812, USA*

Numerical modeling of the impact of boundary shaping on access into and out of quasi-single helicity states in reversed-field-pinch (RFP) plasmas is explored. Experiments have shown that RFP plasmas can self-organize to a quasi-single helicity (QSH) equilibrium with a helical axis [1,2]. These states have improved confinement and lower magnetic turbulence levels compared to a standard RFP plasma which has multiple helicities in the magnetic spectrum. These experiments all have circular, or nearly-circular cross-sections. The VMEC code can obtain computational ideal MHD equilibria with a helical axis and a symmetric boundary [3]. In this work, we analyze the VMEC input parameters that control access to QSH states and test the impact of 2D-shaping of the boundary on RFP equilibria. One of the parameters used to assess the QSH state is the radial excursion of the magnetic axis (i.e., the amplitude of the deviation from an axisymmetric magnetic axis). A key factor in determining if a QSH state is obtained is the safety factor profile being near resonance (e.g., the $q$-profile is close to the $m/n = 1/5$ resonance but does not reach it). Particular attention is paid to the impact that shaping has on access to quasi-single helicity states. The effect of increasing elongation and triangularity are tested systematically. Increased elongation results in lower plasma current for the same safety factor profile and a larger radial excursion of the helical axis in a QHS state. Optimization of the boundary coefficients targeting an increased radial excursion of the helical axis is undertaken. The results are shown in Figure 1 which indicates lower plasma current and a larger excursion of the helical axis.

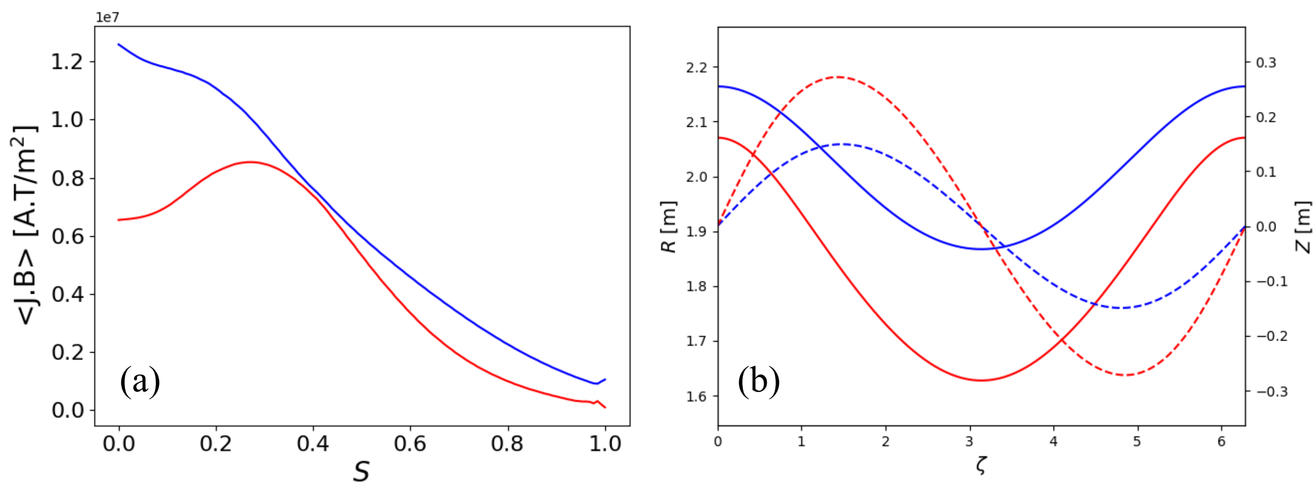


**Figure 2.** (a) The field-aligned current and (b) the magnetic axis position are shown for an initial QSH state (blue) and a case optimized for the excursion of the helical axis (blue). Both the radial (solid) and vertical (dashed) positions of the magnetic axis are shown.

*Acknowledgement*: Work supported by the U.S. Department of Energy under Grant DE-FG02-03ER54699 at the University of Montana.

**References**

[1] D.F. Escande, et al., *Phys. Rev. Lett.* **85**, 1662 (2000)

[2] L. Marrelli, L., et al. *Phys. Plasmas* **9**, 2868 (2002)

[3] J.D. Hanson, et al., *Nucl. Fusion* **53**, 083016 (2013)

# Hessian matrix approach for determining error field sensitivity to coil deviations.

Caoxiang Zhu, Stuart R. Hudson, Samuel A. Lazerson, and David A. Gates

*Princeton Plasma Physics Laboratory, Princeton University, P.O. Box 451, NJ 08543, USA*

The presence of error fields has been shown to degrade plasma confinement and drive in stabilities [1]. Error fields can arise from many sources, but are predominantly attributed to deviations in the coil geometry. Controlling the error field is critical for stellarators where most the confining magnetic field is produced by carefully optimized coils. The accuracy requirements were the largest cost growth of the NCSX stellarator, which was cancelled because of cost overrun [2].

Conventional approaches to studying the error field caused by coil deviations are to apply possible displacements and calculate the resulting error fields [3]. Heavy computations are required and only certain limited deviations could be explored Here, we introduce a Hessian matrix approach for determining error field sensitivity to coil deviations. The FOCUS code [4], provides fast and accurate calculations of an error field evaluation function ($f_B$) and its second derivatives (i.e. Hessian). Near a local minimum, $f_B$ is linearly proportional to the eigenvalues of the Hessian matrix, when decomposing the coil perturbation in the basis of eigenvectors. The sensitivities of error fields to coil displacements are then determined by the eigenvalues.

A proof-of-principle example is given on the CNT configuration [5]. The results, as shown in Fig. 1, show that misalignments at the inner parts of the interlinked coils will cause significant error fields while others are relatively less important. We anticipate that this new method could provide information to avoid dominant coil misalignments, simplify coil designs and ultimately reduce the cost of stellarator coils.

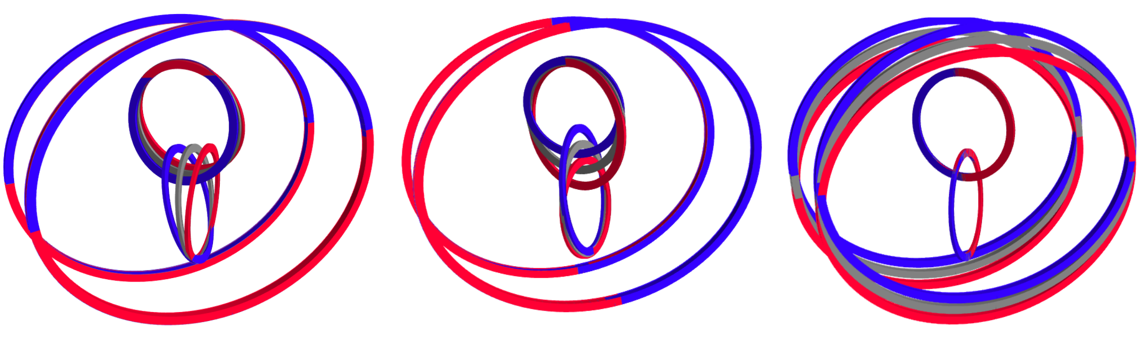

Figure 1. Perturbed coils under the different principal eigenvectors, left: the 1st; middle: the 2nd; right: the 30th. The equilibrium, negatively and positively perturbed coils are in grey, blue and red respectively.

[1] Buttery et al., Nuclear Fusion, 39(11Y):1827–1835, 1999.
[2] Strykowsky et al., 23rd IEEE/NPSS SOFE, IEEE 2009.
[3] Hammond et al., Plasma Physics & Controlled Fusion, 58(7):074002, 2016.
[4] Zhu et al., Nuclear Fusion, 58(1):016008, 2018.
[5] Pedersen et al., Fusion Science & Technology, 46(1):200–208, 2004.

# New insights on SOL plasma turbulence

Paolo Ricci, C.F. Beadle, F.D. Halpern, J. Loizu, A. Mosetto, P. Paruta, F. Riva, C. Wersal,
Swiss Plasma Center, EPFL, Station 13, Lausanne, 1015, Switzerland

## Abstract

Understanding the behaviour of the fusion fuel in the SOL region is a crucial step on the way to fusion energy. For example, the SOL dynamics determine the heat load to the tokamak vessel walls – a showstopper for the whole fusion program if material requirements cannot be met. With the goal of improving our understanding of the SOL dynamics, the GBS code was developed during the past years [1]. GBS simulates the SOL plasma turbulence by solving the drift-reduced Braginskii equations self-consistently with the kinetic neutral atom dynamics [2]. The simulations evolve the SOL dynamics as it results from the plasma outflowing from the core, turbulent transport, plasma losses and recycling at the walls. Simulations with realistic tokamak parameters can be performed. Thanks to GBS simulation results, the instabilities driving transport were identified [3] as well as the turbulence saturation mechanisms [4] and the role of velocity shear in suppressing turbulence [5]. Plasma current circulation and toroidal rotation mechanisms at play in the SOL were clarified [7]. These advances have led to a first principles scaling of the SOL width [6] and to the understanding of the physics mechanisms responsible for the narrow feature observed in the proximity of the last closed flux surface [5] (an observation that led to the redesign of the ITER first wall). We will present an overview of our simulation and theoretical results, as well as their comparison with experimental measurements from several tokamaks worldwide.

[1] Ricci et al, Plasma Phys. Contr. Fusion 54, 124047; Halpern et al, J. Comp. Phys. 315, 388.

[2] Wersal and Ricci, Nucl. Fusion 57, 116018.

[3] Mosetto et al, Phys. Plasmas 20, 092308.

[4] Ricci and Rogers, Phys. Plasmas 20, 010702.

[5] Halpern and Ricci, Nucl. Fusion 57, 034001.

[6] Loizu et al, J. Plasma Phys. 83, 575830601; Loizu et al, Phys. Plasmas 21, 062309.

[7] Halpern et al, Nucl. Fusion 53, 122001; Halpern et al, Nucl. Fusion 54, 043003.

# Towards a first-principles-based Whole Device Model for fusion plasmas

Gabriele Merlo, The University of Texas at Austin, 201 E 24th St, Austin 78712 Texas, USA

with J. Dominski, A. Bhattacharjee, C. S. Chang, E. D'Azevedo, S. Ethier, R. Hager, A. Hakim, J. Hittinger, F. Jenko, S. Klasky, S. Ku, M. Parashar, S. Parker, L. Ricketson, A. Siegel, B. Sturdevant, and E. Suchyta

## Abstract

Understanding, predicting, and optimizing the performance of ITER and of future fusion power plants is a necessary step towards the success of nuclear fusion. Within the Exascale Computing Project (ECP), the Whole Device Model Application is a project that aims at providing a first-principles-based computational tool that integrates all the crucial elements required to simulate a burning plasma.

The first stage of this project consists in coupling the two existing state-of-the-art gyrokinetic codes GENE and XGC, which are used to simulate turbulence in respectively the core and the edge of a tokamak. (During a second stage, other codes will be coupled to this basic framework.) This task has been accomplished, and the two codes are now coupled, enabling one to carry out simulations of turbulent transport from the magnetic axis to the wall.

The coupling algorithm, based on exchanging moments and electromagnetic fields, allows to couple the core-edge codes in a tight and self-consistent way at the level of the distribution function, regardless of the very different numerical schemes and discretization techniques employed by the two codes (GENE is a continuum code, whereas XGC is a particle-in-cell code).

The coupling scheme initially implemented for XGC-XGC coupled simulations has also been applied to GENE-GENE runs. These simplified code-coupling allowed us to learn on the behavior of the coupled system, including the effect of boundary conditions, avoiding the complications due to mappings between the different meshes. The interpolation scheme used for transferring data back and forth between GENE's structured grid and the unstructured XGC one will also be introduced and discussed. First results obtained with the coupled GENE-XGC code will be presented.

This research was supported by the Exascale Computing Project (17-SC-20-SC), a collaborative effort of the U.S. Department of Energy Office of Science and the National Nuclear Security Administration.

# Nonlinear Mode Penetration Caused by Transient Magnetic Perturbations

M.T. Beidler, J.D. Callen, C.C. Hegna, and C.R. Sovinec
*University of Wisconsin-Madison*

Experimental observations in tokamaks suggest the destabilization of NTMs and the transition to ELM suppression are caused by mode penetration of 3D magnetic fields prompted by transient MHD events. Three dimensional magnetic fields in tokamaks can induce forced magnetic reconnection (FMR) and produce magnetic islands on resonant surfaces. Conventional analytic solutions to FMR focus on describing the time asymptotic state [1]. The focus of this work is to understand the nonlinear dynamics of mode penetration, evolution of a metastable equilibrium from a high-slip, flow-screened state into a low-slip, field-penetrated state. In particular, this work addresses how transient MHD events, such as sawteeth and ELMs, trigger mode penetration.

In this work, we present nonlinear computations employing the extended-MHD code NIMROD, building on previous work [2] by incorporating a temporally varying external magnetic field as a simple model for an MHD event that produces resonant magnetic perturbations. Proof-of-principle computations vary parameterizations of the transient external field and initial metastable state to probe the threshold for mode penetration, a locked mode response and magnetic island evolution. See Fig. 1 for results of changing the magnitude of the external field.

We test these computational results against a quasi-linear analytical theory that captures the temporal evolution properties of the electromagnetic and viscous forces during and after a transient. We find qualitative agreement between our computational and analytical results. However, computational tools are necessary to accurately capture the threshold conditions for mode penetration induced by an MHD transient due to the transition from linear to nonlinear evolution.

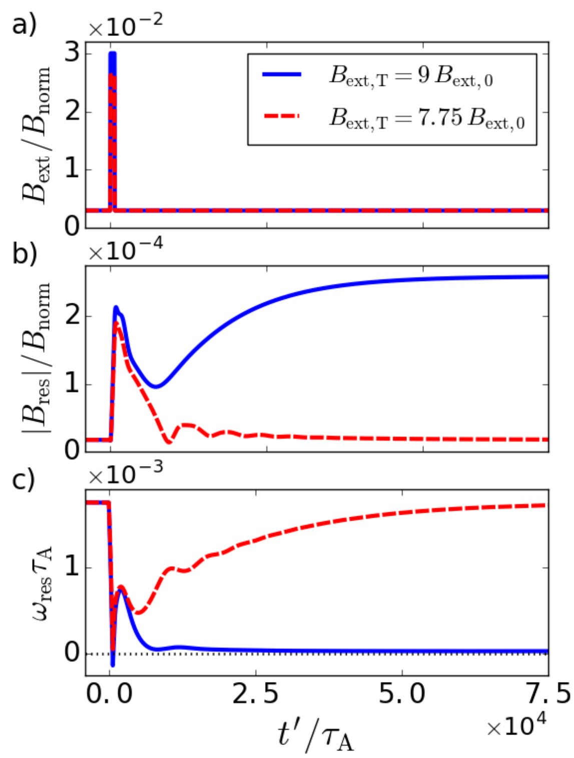


Figure 1. Comparison between a) the externally applied magnetic field $B_{ext}$, b) the field response $|B_{res}|$, and c) the flow frequency at the rational surface $\omega_{res}$ for two computations with different magnitudes of the magnetic transient $B_{ext}$. The effect of the larger transient is to precipitate a transition to a low-slip, field-penetrated state, while the smaller transient returns to the high-slip, flow-screened equilibrium.

[1] R.Fitzpatrick, Nucl. Fusion 33, 1049 (1993)
[2] M.T.Beidler, J.D.Callen, C.C.Hegna, and C.R.Sovinec, Phys. Plasmas 24, 052508 (2017)

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# Fast instabilities by nonlinear interchange instability

Linda E. Sugiyama, M.I.T., Bldg. 26-557, 77 Massachusetts Avenue, Cambridge, MA 02319 U.S.A.

## Abstract

Important instabilities in magnetically confined plasmas develop an accelerating, faster-than-exponential growth rate that leads to a rapid ``crash" of the background plasma, expelling substantial amounts of plasma and energy and reducing the destabilizing gradients. Examples include the 1/1 internal kink (sawtooth) over a central $q<1$ region and large Edge Localized Modes (ELMs) with a steep near-edge pressure gradient. They develop a highly localized radial bulge (or several) of the unstable surface. Strong compression against the surrounding plasma increasingly aligns the magnetic field lines, which already have a magnetic shear close to that of the mode resonant surface, and strengthens the transverse pressure gradient. Above a threshold amplitude, a mostly ideal, nonlinear interchange can rapidly push plasma across this field without changing the orientation of the magnetic field vectors or field lines. A protruding ``magnetic pipe" produced by reconnection is unnecessary, although reconnection may ultimately occur. Time scales and plasma redistribution are similar to experimental observations. The bulge develops through the nonlinear interaction of the mode harmonics, which create successively higher toroidal and poloidal harmonics that further localize the instability, a process that feeds back on itself. The mechanism exists in MHD and can be demonstrated analytically and numerically. The fast stage of the sawtooth crash is independent of resistivity [1], despite strongly resistive growth rates at small amplitude. It does not follow the Kadomtsev model, because density is not tied to the field lines. For sawteeth[1] and ELMs[2], no magnetic flux tubes connect the expelled plasma to its original location, beyond the immediate vicinity of the resonant surface, unlike some theories of explosive growth.

Work partially supported by U.S. DOE DE-SC0007883.

[1] L. Sugiyama, Phys. Plasmas 21, 022410 (2014).
[2] L. Sugiyama, Phys. Plasmas 17, 062505 (2010).

# MHD Stability of Negative Triangularity Tokamaks

L. J. Zheng, M. T. Kotschenreuther, F. L. Waelbroeck,
M. E. Austin, W. L. Rowan, P. Valanju, and X. Liu
Institute for Fusion Studies, University of Texas at Austin, Austin, TX 78712, USA

The divertor heat load is a major concern for fusion reactors. This led Ref. [1] to propose a negative triangularity tokamak as the product of a design philosophy that shifts the focus of optimization efforts away from the plasma core, prioritizing instead the reactor boundary. The negative triangularity tokamaks can have a larger separatrix wetted area, more flexible divertor configuration design, wider trapped particle-free scrape-off layer, lower background magnetic field for internal poloidal field coils, and larger pumping conductance from the divertor plenum. However, the stability beta limit is a concern. Recent TCV and DIII-D experiments have increased the interest in negative triangularity by showing that such discharges exhibit H-mode-level confinement features with L-mode-like edge behavior without ELMs [2,3].

First, using the numerically reconstructed experimental equilibrium, our AEGIS/DCON computation confirmed the stability of the beta normal of 2.6 achieved in DIII-D experiments against low-n MHD kink modes. This is slightly lower than the positive triangularity cases with the same types of density and temperature profiles.

Next, we analyzed the equilibrium parameters in greater depth. We found that the negative triangularity configuration leads to a lower safety factor value especially near the edge. This feature tends to favor the advanced tokamak scenario with high bootstrap current fraction and peak pressure profile. Indeed, we were surprised to find that the negative triangularity configuration can actually achieve even higher beta normal than the positive triangularity case in certain cases. As shown in Fig. 1, our calculations show that in some higher bootstrap fraction, high poloidal beta, negative triangularity cases the beta normal limit can reach 8 li(I/aB) for low n (1-5) modes, twice the value of the so-called Troyon limit encountered by the positive triangularity tokamaks. The beta value seems to be limited by the high n ballooning modes, which may be improved by the profile optimization and non-local effects.

In conclusion, our results indicate that the negative triangularity tokamaks are not only good for divertor design, but also for stability. They can potentially achieve:

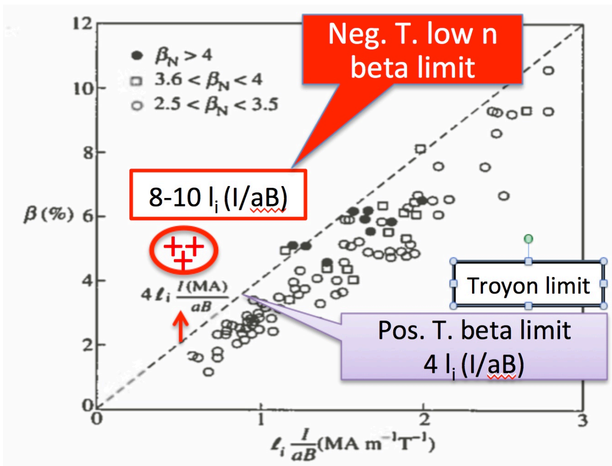
1. High beta, steady state confinement with very high bootstrap current fraction.
2. ELM free performance.
3. Reduced disruptivity, since it is stable against low-n kink modes, and its beta limit is determined by high n modes.
4. High resistive wall mode beta limit.
5. High reactor fusion productivity with peak pressure profile, etc.

Nevertheless, the high n mode instabilities may occur, which can lead to the so-called "soft" beta limit.

An exciting sign has been observed in the D3D negative triangularity experiments with reduced current as indicated by our theory. In the positive triangularity tokamak case, a reduction of plasma current leads to a reduced energy confinement as shown by the dashed line in Fig. 1. The further negative trigaularity experiments in D3D show that the confinement remains unchanged during the reduction of palsma current.

References
[1] M. Kikuchi, T. Takizuka, M. Furukawa, JPS Conf. Proc. 1, 015014 (2014).
[2] Y. Camenen et al, Nucl. Fusion 47, 510 (2007).
[3] M. E. Austin et al., Bull. Am. Phys. Soc. vol. 62 (2017).

Figure 1:

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**Production rate of runaway electrons in dynamic scenarios: a probabilistic backward Monte Carlo method**

D. del-Castillo-Negrete and G. Zhang
Oak Ridge National Laboratory

The understanding and control of runaway electrons (RE) is a top priority of the nuclear fusion program because, if not avoided or mitigated, RE can severely damage the plasma facing components. A problem of particular interest is the computation of the RE production rate for given conditions, e.g. the plasma temperature and the electric field. Recently we proposed a novel approach to solve this problem using the backward Monte Carlo (BMC) method [1]. BMC is based on the direct numerical solution of the Feynman-Kac formula that establishes a link between the solution of the adjoint Fokker-Planck problem (which gives the probability of runaway $P_{RE}$) and the stochastic differential equations describing the trajectories of RE in the presence of collisions, electric field acceleration, and radiation damping. Computationally, the BMC is a deterministic algorithm that reduces the problem to the evaluation of Gaussian integrals that can be efficiently computed with high accuracy using Gauss-Hermite quadrature rules. Going beyond [1], we have extended the BMC to dynamic scenarios where the electric field and the plasma temperature exhibit time dependence. The BMC method can be implemented in any number of dimensions, N, and compared with other approaches (e.g., direct Monte-Carlo simulations or solution of the adjoint Fokker-Planck partial differential equation) exhibits a superior scaling with N. The method is also unconditionally stable, and like the direct Monte-Carlo approach, it is fully parallelizable. Once $P_{RE}$ is computed, the production rate is evaluated by integrating $P_{RE}$ over the distribution function. For illustration purposes, here we present a 2-D case involving the dependence of $P_{RE}$ on the total momentum, p, and the pitch angle θ. As Fig.1 shows, the electric field dynamics has a significant impact on the probability of runaway at short times (t=0.6) and in the long-time (asymptotic) equilibrium state. An electron with initial p and θ in a red region will most likely become a runaway electron, i.e. its momentum will reach a prescribed large value. On the other hand, blue regions correspond to initial conditions for which Coulomb drag and synchrotron radiation damping overcome the electric field acceleration. We have also used the BMC method to compute $P_{RE}$ including time dependence in both the electric field and the plasma temperature. The results also show a nontrivial evolution of $P_{RE}$, not captured in simplified descriptions. Current work includes the sparse-grid based numerical implementation in higher dimensions to incorporate the full RE orbit dynamics.

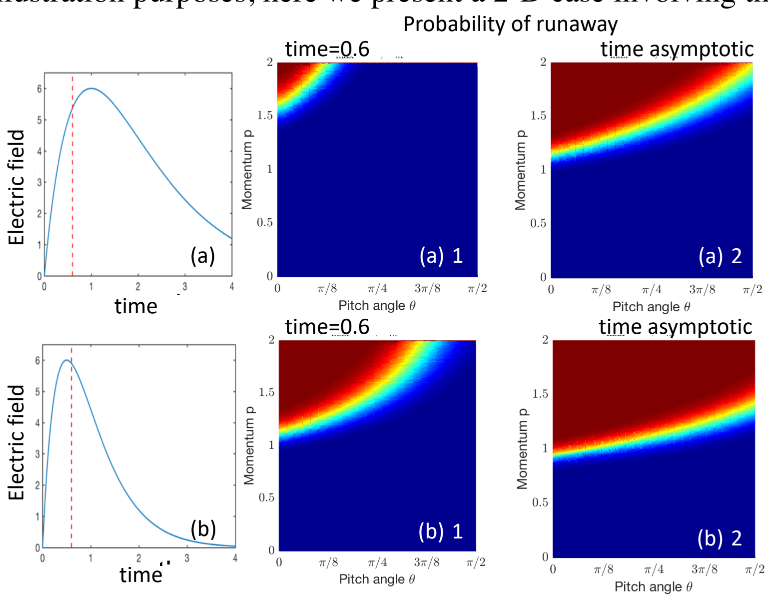


*Figure 1. Time evolution of probability of runaway, $P_{RE}$, in time-dependent electric fields. Panels (a) and (b) show two examples of electric field evolution. Panels (a)-1 and (a)-2 show the short time and long-time $P_{RE}$ for case (a). Panels (b)-1 and (b)-2 show the corresponding results for (b). Dark red (dark blue) corresponds to $P_{RE}$=1 ($P_{RE}$=0).*

[1] G. Zhang and D. del-Castillo-Negrete, "A backward Monte-Carlo method for time-dependent show runaway electron simulations" Phys. of Plasmas **24**, 092511 (2017).

**Tokamak Disruption Simulation: Progress toward Comprehensive Modeling**

Carl R. Sovinec, University of Wisconsin-Madison, 1500 Engineering Drive, Madison, Wisconsin 53706-1, USA

Full-scale operation of the ITER experiment will produce plasma thermal energy and releasable magnetic energy on the order of hundreds of mega-Joules. Unplanned disruptions to these discharges will be capable of causing significant material damage to plasma-facing components and electrically conducting structures. Efforts to understand disruptive dynamics, and to engineer mitigation systems, include the development of numerical models that can make predictions without destructive testing. Because many different physical effects influence plasma dynamics during disruptions, this modeling will require integrated simulation.[1] This presentation reviews computational results that describe the macroscopic dynamics that occur in disruptions. It also describes the modeling components that will need to be assembled for predictive disruption simulation.

While tokamak disruptions take different forms and progress through different "trigger"-"adverse event"-"disruption precipice" sequences,[2] macroscopic instability is common to all disruptions. Considering magnetic relaxation[3] while minimizing relative magnetic helicity provides insight into why. As shown for results with different tokamak shapes, magnetic relaxation tends to reduce magnetic shear and edge-$q$, which increase susceptibility to magnetic-island overlap and external-kink. These tendencies make the tokamak more prone to disruptive macroscopic dynamics than reversed-field pinch profiles, where relaxation tends to increase magnetic shear.

Numerical computations have been applied to study the nonlinear effects of macroscopic instability over the past four decades. Results reviewed in the presentation include 1) bubble-swallowing and asymmetric plasma-surface current conduction from external kink, 2) nonlinear resistive-wall mode evolution, 3) mixing from nonlinear ballooning, 3) toroidally symmetric and asymmetric vertical displacement, 4) magnetic stochasticity resulting from magnetic-island overlap, 5) mode-penetration and locking with field errors, 6) runaway electron confinement, and 7) mixing of injected impurities due to MHD instability. A recent 3D computation performed with the NIMROD code serves as an example. It demonstrates the characteristic current bump from the thermal quench (TQ) followed by a relatively long current quench (CQ). Horizontal forcing that would result from the modeled dynamics is inferred from the asymmetric magnetic flux that penetrates the resistive wall.

More accurate predictive models will require significant extensions of present-day macroscopic modeling. The most important extensions are kinetic closures, runaway-electron modeling, impurity flows and radiation, plasma-surface interaction, and external electromagnetics. Integrated disruption simulation will likely need reduced modeling of at least some of these effects. It will also require advanced algorithms for multiphysics and multiscale applications. Choices will need to consider which models need to be tightly coupled and which may be loosely coupled. Taking advantage of new computational hardware will also be essential, but the increasing architectural complexity introduces its own set of challenges.

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[1] P. Bonoli and L. Curfman McInnes, *Report of the Workshop on Integrated Simulations for Magnetic Fusion Energy Sciences*, Rockville, Maryland, US Dept. of Energy Office of Science, 2015.
[2] C. Greenfield and R. Nazikian, *Report of the Workshop on Transients in Tokamak Plasmas*, Rockville, Maryland, US Dept. of Energy Office of Science, 2015.
[3] J. B. Taylor, Phys. Rev. Lett. **33**, 1139 (1974).