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**GENERAL ATOMICS**

# DISCLAIMER

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# ABSTRACT

Experiments on DIII-D confirm that the tritium breeding test blanket modules (TBMs) in ITER will lead to a decrease of the plasma rotation in H-modes [M.J. Schaffer, *et al.*, Nucl. Fusion **51** (2011) 103028]. Moreover, they suggest that long-wavelength correction fields applied with non-axisymmetric saddle coils will only be able to ameliorate a fraction of such a rotation reduction. The new finding obtained in rotating H-modes with parameters similar to the ITER baseline scenario contrasts previous experiments, which showed that saddle coils are very effective in restoring resilience to locked modes in L-mode plasmas. The experiments use a TBM mock-up coil that has been especially designed to simulate the error field induced by the ferromagnetic steel of a pair of TBMs in one ITER port. The $n = 1$ error field correction (EFC) is applied with a set of non-axisymmetric saddle coils (I-coil), whose currents are optimized in the presence of the TBM mock-up field using a newly developed non-disruptive technique that maximizes the angular momentum. However, a test of the effectiveness of the TBM EFC yields that the optimized EFC can only recover approximately a quarter of the ~20% rotation decrease attributed to the TBM error field. An alternative criterion to evaluate EFC has been its effectiveness in canceling the $n = 1$ plasma response to the field error. Plasma response measurements in the TBM experiment show that the I-coil can indeed cancel the magnetic measurements of the $n = 1$ plasma response to the TBM mock-up field. The required currents agree within the uncertainties of the estimates with the currents that maximize the angular momentum. The contrast between the limited effectiveness of $n = 1$ EFC in rotating H-modes and their ability to recover a low locking density in L-mode plasmas shows that the components of the non-axisymmetric field that brake the plasma at higher values of beta or higher rotation differ from the components that are responsible for the field penetration in low density L-modes.

# 1. INTRODUCTION

The present design of the tritium-breeding test blanket modules (TBMs) in ITER contains a significant amount of high-temperature and neutron tolerant, but also ferromagnetic, martensitic steel [1]. It is planned to install six TBMs grouped in pairs in three ITER midplane ports, which are toroidally separated by 40 deg. Each pair of TBMs will perturb the nearby plasma with a local ~1% reduction of the magnetic field. The spatial localization of the TBMs results in a broad toroidal mode spectrum of the perturbed field, with the strongest components having toroidal mode numbers $n$ in the range from seven to 30. Since such a spectrum differs significantly from common error fields in tokamaks, a coil set was built to mock-up the error field from the TBMs in one of the ITER ports, and installed in a DIII-D midplane port, Fig 1(a). The initial experiments [2,3] showed that in H-mode plasmas the TBM induced magnetic field error mainly decreases the plasma rotation and to a lesser degree energy and particle confinement. In addition the TBM field causes an enhanced loss of the fast beam ions in DIII-D leading to the formation of hot spots on the wall [4]. The TBM field was also found to degrade the tolerance to penetration of an $n = 1$ proxy error field, which can solely be explained by the detrimental effect of the TBM on plasma rotation [3]. In L-mode plasmas the TBM field increases the locking density. However, optimizing the correction of the $n = 1$ component of the TBM field alone is sufficient to restore the low locking density of plasmas without TBM field [2,3].
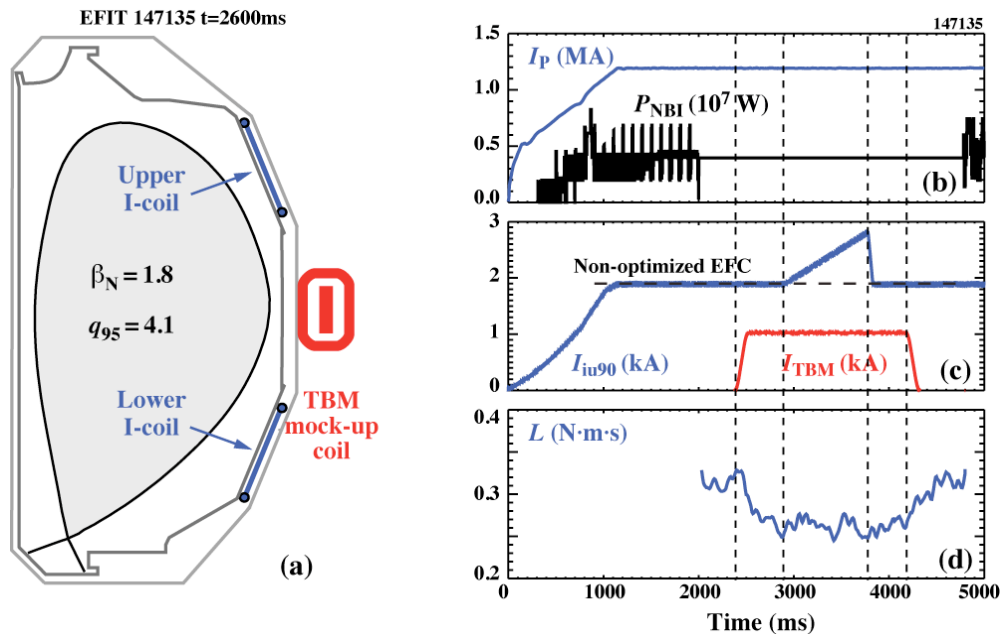


FIG. 1. (a) Poloidal cross section of an ITER similar shape plasma, TBM mock-up coil and internal non-axisymmetric saddle coils (I-coil) in DIII-D. During the (b) 2800 ms interval with constant NBI heating the (c) TBM field is applied and the I-coil field varied leading to (d) variations of the toroidal angular momentum of the plasma.

Motivated by the successful correction of the TBM induced error field in low-density L-mode plasmas a new set of experiments seeks to evaluate the effectiveness of TBM error field correction in ITER baseline relevant ELMy H-mode plasmas [5]. In Section 2 the effect of the TBM field on the target discharge is analyzed in detail. Section 3 describes a new non-disruptive technique to optimize the error field correction (EFC) in H-modes based on the maximizing the angular momentum and compares it with a technique based on the cancelation of the $n = 1$ magnetic plasma response to the TBM error. The empirical correction currents are compared with predictions in Section 4 and discussed in Section 5.

# 2.  EFFECT OF THE TBM INDUCED FIELD ERROR
# ON PLASMA ROTATION

In ELMy H-mode plasmas, the main effect of the field of the TBM mock-up coil in DIII-D is a decrease of the plasma rotation. The rotation braking is studied in discharges, which have an ITER-similar plasma shape, Fig. 1(a), and are heated with tangential neutral beam injection (NBI) in the direction of the plasma current. The discharges have a safety factor of $q_{95} = 4.1$ and contain a long interval with constant NBI power and tangential torque resulting in a normalized beta of $\beta_N = 1.85 \pm 0.10$ and a toroidal angular momentum of $L = 0.32 \pm 0.01$ N·m·s, Fig. 1(b). Fast plasma rotation and the choice of $q_{95}$ somewhat above the ITER baseline value decrease the effects of neoclassical tearing modes and lead to highly reproducible discharges. During the stationary interval, a current $I_{TBM}$ is applied in the TBM mock-up coil, Fig. 1(c). The amplitude of the current is chosen to generate a magnetic field ripple at the plasma surface of 3%, which exceeds the value expected in ITER by approximately a factor of three in order to make up for the three TBM ports in ITER. Recent calculations of ripple transport yield that the enhancement of the thermal ion loss rate scales linearly with the number of TBM ports and with the square of the ripple amplitude suggesting that a factor of $\sqrt{3}$ would be more appropriate [6]. The TBM field results in a decrease of $L$ on average by 20% ± 6%, Fig. 1(d). At the same time $\beta_N$ and the electron density only decrease on average by a few percent. The decrease of $L$ is therefore mainly caused by a decrease of the toroidal plasma rotation $\Omega$. The variation in $|\Delta L|/L$ among discharges is attributed to a variation of $\beta_N$ with higher values of $\beta_N$ leading to a larger $|\Delta L|/L$, similar to the momentum confinement degradation observed in the previous experiments [2,3]. With the application of the TBM field, the toroidal plasma rotation decreases across the entire profile, Fig. 2(a). Here, the rotation measurements are averaged over several discharges in stationary intervals before and after the turn-on of the TBM field.

The toroidal torque generated by the TBM field is characterized in more detail by analyzing the dynamic response to the perturbed field. A modulated TBM field with a modulation frequency of $f_{mod} = 5$ Hz is applied in the stationary phase of a discharge. The profiles of the amplitude $\delta\Omega$ and phase shift $\Phi$ of the resulting perturbation of the plasma rotation, Fig. 2(b,c), reveal a minimum of $\Phi$ and a maximum of $\delta\Omega$ at $\rho \approx 0.9$. The observed minimum phase shift of ~50 deg corresponds to a time scale $\tau = \tan(\Phi)/(2\pi f_{mod})$ of ~40 ms consistent with typical momentum transport time scales. Assuming that the perturbed field penetrates on a time scale, which is faster than the momentum transport time scale, the measurements are evidence of an edge localized torque.

Such an edge localized torque could be induced by enhanced fast ion losses at the edge. ASCOT simulations [7] together with experimental fast ion measurements [4], however, show that the fast ion loss due to the TBM field and hence its influence on the rotation is small. The rotation decrease in the DIII-D TBM experiment is therefore likely caused by a magnetic torque. Fast plasma rotation and edge localization rule out a resonant torque due to incomplete shielding

of low order resonant magnetic fields leaving enhanced neoclassical toroidal viscosity (NTV) [4,8] as a possible explanation.
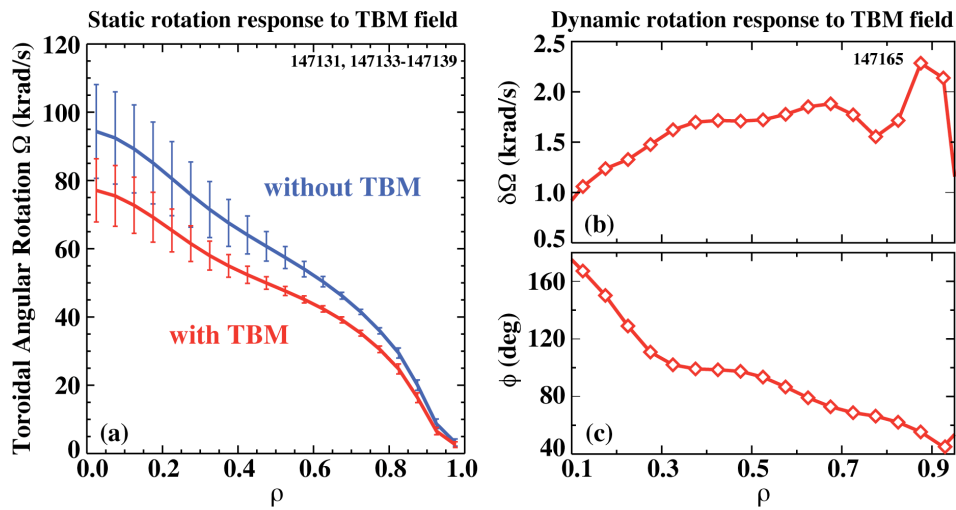


FIG. 2. (a) Change in the toroidal plasma rotation Ω due to the TBM field. The profiles are averaged over 200 ms intervals immediately before and 300 ms after the turn on of the TBM coil in eight similar discharges. Profiles of (b) amplitude δΩ and (c) phase shift Φ of the change in plasma rotation resulting from a 5 Hz modulation of the TBM coil current.

## 3.  EMPIRICAL OPTIMIZATION OF CORRECTION CURRENTS

The error field correction currents are optimized using a new non-disruptive technique that maximizes the toroidal angular momentum of the plasma. The correction field is applied with a set of internal saddle coils (I-coils), arranged in two arrays of six coils above and below the outboard mid plane, Fig. 1(a). Their geometry and capability to apply low $n$ magnetic fields are similar to the planned ELM correction coils in ITER. The currents in the individual coils are chosen to generate an $n = 1$ field, which can be expressed by an amplitude $I_n$ and a toroidal phase $\phi_n$. A complex notation $\mathbf{I}_n = I_n\, e^{i\phi_n}$ (with complex numbers denoted by bold symbols) is often used with the current at a toroidal location $\varphi$ being $I(\varphi) = \mathrm{Re}(\mathbf{I}_n\, e^{-in\varphi})$. In this work $n$ is always 1 and the index $n$ therefore dropped. The difference between the $n = 1$ phase of the current in the lower I-coil array (IL) with respect to the upper I-coil array (IU) is fixed to 240 deg, which guarantees a good coupling to the plasma [9]. The correction strategy is based on the assumption that for each $n$ there is a specific poloidal distribution of the external field the plasma is most sensitive to [10]. Correction coils are effective as long as their field is not orthogonal to this distribution. This "single mode" approximation has been supported by the EFC results in DIII-D H-modes, where the achievable rotation is independent of the poloidal spectrum of the correction field that cancels the perturbed field associated with the primary mode [11], while it falls short of explaining low density locked mode experiments [12,13]. In the single mode approximation each $n$ component of an error field source can be expressed by an equivalent non-axisymmetric coil current. In this work the upper I-coil is chosen as the reference and the total $n = 1$ field expressed as $\mathbf{I}_{\mathrm{tot}} = \mathbf{I}_{\mathrm{intrinsic}} + \mathbf{I}_{\mathrm{TBM}} + \mathbf{I}_{\mathrm{IU}}$, with $\mathbf{I}_{\mathrm{intrinsic}}$ describing the $n = 1$ component of the intrinsic field error and $\mathbf{I}_{\mathrm{TBM}}$ the $n = 1$ component of the TBM field error. In order to find the optimal I-coil current $\mathbf{I}_{\mathrm{IU,opt}}$ that maximizes the angular momentum, slow ramps $\Delta\mathbf{I}_{\mathrm{IU}}$ of the $n = 1$ amplitude with various toroidal phases are added to a reasonable guess of the correction of the intrinsic field error. Since it is impractical to sample the entire $I_{\mathrm{IU}}$-$\phi_{\mathrm{IU}}$ parameter space an appropriate model for the dependence of the angular momentum on $\mathbf{I}_{\mathrm{IU}}$ is applied. Assuming that the residual field error generates a torque that is proportional to $L\,|\mathbf{I}_{\mathrm{tot}}|^2$, angular momentum balance neglecting time derivatives yields,

$$L\left(\mathbf{I}_{\mathrm{IU}}\right) \;=\; \frac{L_{\mathrm{opt}}}{1 \,+\, \tau_{\mathrm{L}}\,c\left|\mathbf{I}_{\mathrm{IU}} \,-\, \mathbf{I}_{\mathrm{IU,opt}}\right|^2}\;, \tag{1}$$

where $\tau_{\mathrm{L}}$ is the angular momentum confinement time, $c$ the proportionality constant for the non-axisymmetric field torque and $L_{\mathrm{opt}}$ the achievable angular momentum with optimal correction currents. The assumed dependence of the magnetic braking torque on $L$ is characteristic for a non-resonant torque and consistent with NTV torque at high rotation [8] as well as with experimental results in DIII-D [14]. The optimization is carried out for the intrinsic field error only and for the combination of intrinsic and TBM field errors. The fits of the measured angular momentum to Eq. (1) yield good agreement, Fig. 3.
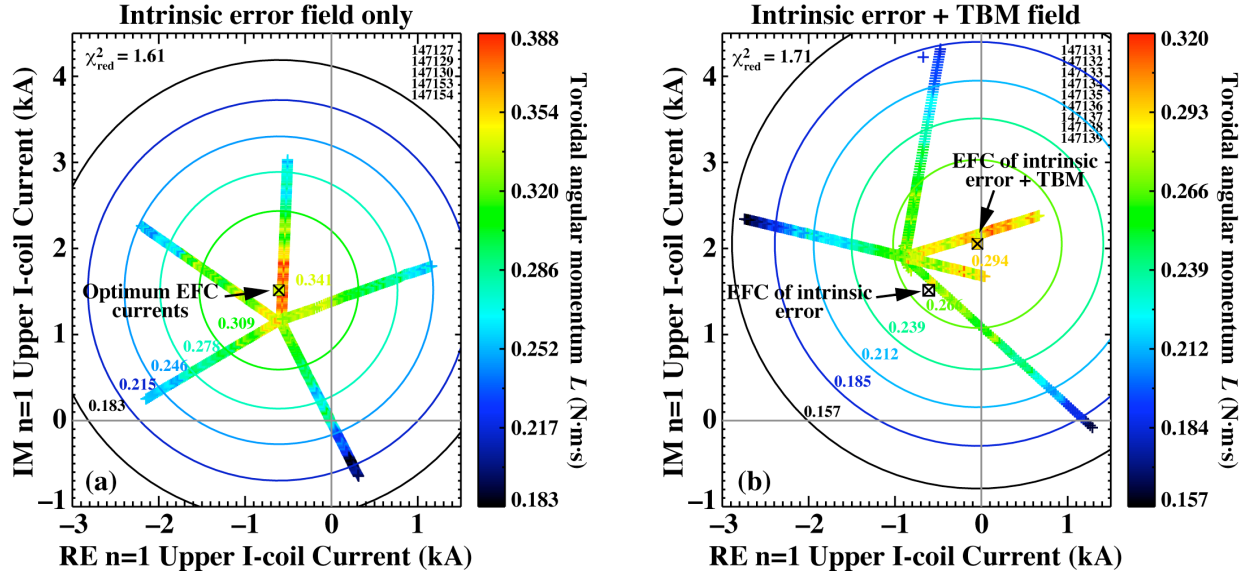
FIG. 3. *Dependence of the toroidal angular momentum L of the plasma without (a) and with the TBM field (b) on the amplitude and orientation of the n=1 component of the upper I-coil current. The concentric circles correspond to constant L contours of the fit using Eq. (1).*

The resulting optimal correction currents and maximum angular momentum are summarized in Table I. Without the TBM optimal EFC results in an angular momentum of 0.347 N·m·s. Adding the TBM field while keeping the I-coil currents unchanged reduces the angular momentum to 0.276 N·m·s, which is 20% lower than the value without the TBM and consistent with the results discussed in Section 2. Re-optimizing the EFC in the presence of the TBM field increases the angular momentum to 0.294 N·m·s, which is still 15% lower than $L_{opt}$ without the TBM. Thus, $n = 1$ EFC with the I-coil can recover approximately 25% of the TBM induced angular momentum decrease in the investigated fast rotating ITER baseline scenario in DIII-D. The difference between the optimal $n = 1$ correction current of the combined intrinsic and TBM field errors and the optimal $n = 1$ correction current of the intrinsic field error alone can be attributed to the TBM field error and has an amplitude of 780 A and a toroidal phase angle of 44 deg.

Table I: Result of the fits shown in Fig. 3

|  | Intrinsic EF only | Intrinsic + TBM EF |
|---|---|---|
| $I_{IU,opt}$ | 1.63 kA @ 110.6 deg | 2.05 kA @ 91.3 deg |
| $L_{opt}$ | 0.347 N·m·s | 0.294 N·m·s |
| $L(I_{IU,opt}[\text{intrinsic}])$ | 0.347 N·m·s | 0.276 N·m·s |

An alternative criterion to optimize EFC has been its effectiveness in canceling the $n = 1$ magnetic plasma response to the error field, mainly because it has been associated with a minimization of the residual magnetic braking torque [15]. The $n = 1$ plasma response $\mathbf{B}_p^{plas}$ is obtained by subtracting the known vacuum coupling to I- and TBM-coils from magnetic

measurements using a toroidal array of poloidal field probes. The measurements are zeroed before the turn on of the TBM coil and therefore only contain the response to the TBM field and the I-coil field resulting from the ramps $\Delta\mathbf{I}$. The capability of the I-coil to reduce the plasma response to the TBM field is demonstrated in Fig. 4, where an intermediate value within the I-coil ramp leads to an almost complete cancellation of $\mathbf{B}_p^{plas}$.
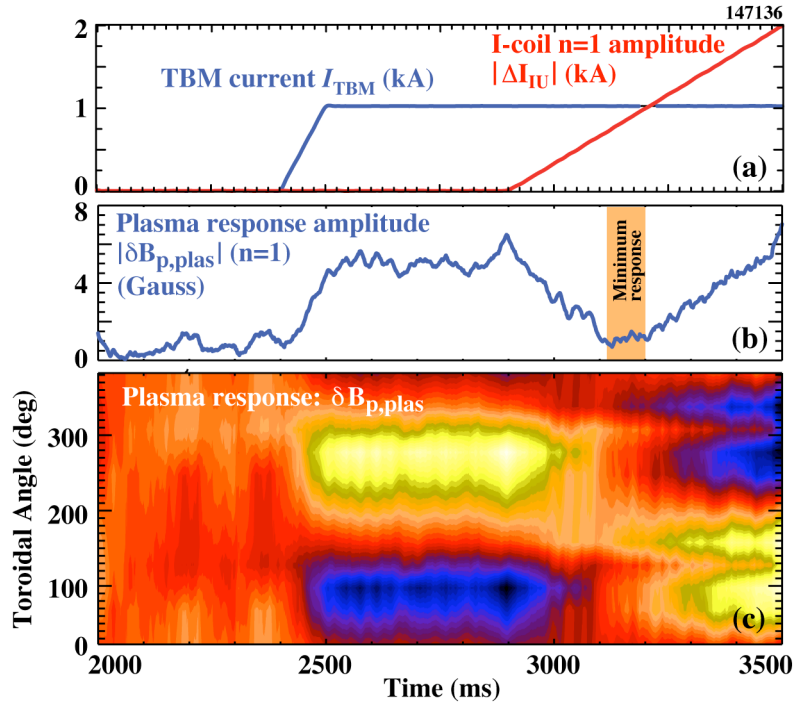


FIG. 4. Discharge with (a) an I-coil ramp $\Delta\mathbf{I}_{IU}$ at 77.5 deg during a TBM pulse. (b) Magnetic measurements show an n=1 plasma response $\mathbf{B}_p^{plas}$ to the TBM field, which can be almost cancelled by the I-coil. (c) Contours of the perturbed poloidal field as a function of time and toroidal angle show the phase inversion at the minimum response.

The optimal I-coil EFC currents are obtained by fitting $\mathbf{B}_p^{plas}$ measurements in several discharges to a model based on linear amplification of the TBM and I-coil fields,

$$\mathbf{B}_p^{plas} \;=\; \mathbf{B}_{p,\,TBM}^{plas} \;+\; \mathbf{b}_{p,\,IU}^{plas}\mathbf{I}_{IU} \quad . \tag{2}$$

The fitted plasma response to the TBM field $\mathbf{B}_{p,\,TBM}^{plas}$ and the complex coupling coefficient $\mathbf{b}_{p,\,IU}^{plas}$ then yield an estimate of the optimal I-coil TBM correction current $\mathbf{I}_{IU,\,opt} \;=\; -\mathbf{B}_{p,\,TBM}^{plas}\big/\mathbf{b}_{p,\,IU}^{plas}$, which cancels the $n=1$ plasma response to the TBM field. In the scenario investigated the optimum correction currents have an amplitude of approximately 960 A with a phase of 58 deg, which is close to the $n=1$ I-coil current applied at the time of the minimum response in discharge 147136, Fig. 4.

The I-coil currents that maximize the rotation in the presence of the TBM and the currents that cancel the plasma response are found to be in reasonable agreement, Fig. 5. Both measurements are fraught with significant uncertainties. The angular momentum has a shallow

maximum with the fit being largely determined by measurements with large $|\mathbf{I}_{IU} - \mathbf{I}_{IU,opt}|$ where deviations from the single mode approximation would lead to a systematic error. In addition the TBM contribution to $\mathbf{I}_{opt}$ is obtained from the difference of two measurements which both have a significant uncertainty. The determination of the I-coil currents that cancel the plasma response to the TBM field is easier since it is based on the identification of a zero crossing rather than an extremum. In the vicinity of the optimum correction currents the plasma response changes linearly with I-coil current, whereas the angular momentum is proportional to $|\mathbf{I}-\mathbf{I}_{opt}|^2$. However, the estimate of $\mathbf{I}_{opt}$ based on the cancellation of $\mathbf{B}_p^{plas}$ can also contain systematic errors, such as an incremental plasma response to the intrinsic error field caused by the TBM induced decrease in $\beta_N$.
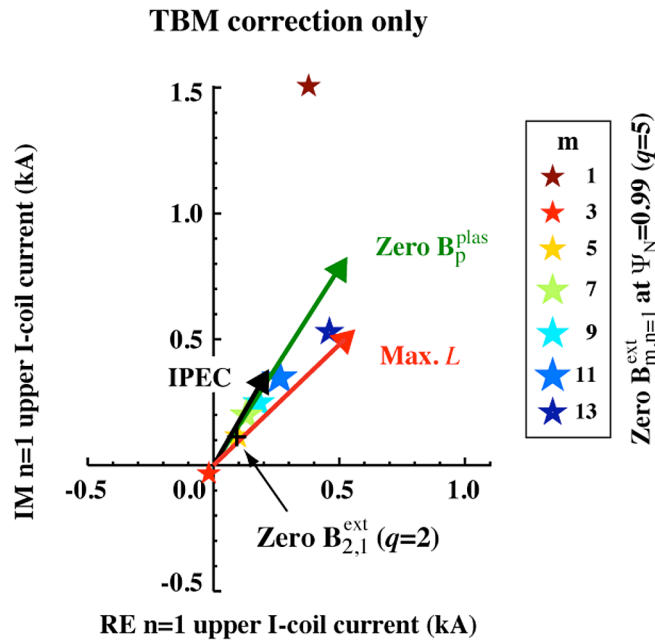


FIG. 5 Comparison of the optimal n=1 I-coil EFC of the TBM field obtained by maximizing the angular momentum L (red) and zeroing the magnetic plasma response $\mathbf{B}_p^{plas}$ (green) and an IPEC prediction (black) with I-coil currents that cancel various poloidal mode components with the same helicity as the equilibrium field.

While the minimization of the plasma response $\mathbf{B}_{p,TBM}^{plas}$ should also be amenable to a rapid optimization procedure based on magnetic feedback [15], TBM error field correction in ITER cannot take advantage of the turn on of the TBM field error. A correction strategy would therefore have to be based on the $\beta_N$ dependence of the plasma response, which decreases the signal amplitude and complicates its detection.

## 4.  COMPARISON WITH PREDICTED CORRECTION CURRENTS

The empirical correction currents described in Section 3 are compared with predictions based on vacuum and plasma response models. Decomposing the $n = 1$ component of the TBM and I-coil fields into poloidal mode components yields that the empirical correction currents approximately cancel the $m = 13$ component at the 99% flux surface, while they greatly overcorrect the resonant component at the same surface ($m = 5$) as well as at any other resonant surface, e.g. at $q = 2$, Fig. 5. This is consistent with previous experiments that showed that the plasma response is caused by the "kink-mode resonant" component of the external field, which has a higher $m$ number than the resonant field [9,14]. The ideal MHD plasma response is calculated using the IPEC code [16]. Minimizing sheet currents on resonant surfaces that prevent the formation of magnetic islands yields a correction current whose phase agrees well with the empirical correction currents, but whose amplitude is significantly lower, Fig. 5. While the shielding currents or the equivalent shielded resonant fields have often been associated with a torque, recent IPEC calculations show that a reduction of the shielded resonant field can increase non-resonant field components and thereby the NTV torque [13]. Further analysis is required in order to understand the relation between the empirical correction currents and the IPEC/NTV model.

# 5. SUMMARY AND DISCUSSION

Experiments on DIII-D confirm that the TBMs in ITER will lead to a decrease of the plasma rotation in H-mode scenarios. A dynamic analysis of the torque associated with the TBM field reveals a dominantly edge localized magnetic torque consistent with an NTV torque. Moreover, the DIII-D experiments suggest that in the ITER baseline scenario $n = 1$ correction fields applied with non-axisymmetric saddle coils will only be able to ameliorate a fraction of such a rotation reduction. This projection to ITER is based on experimental findings in DIII-D that in fast rotating H-modes with $\beta_N = 1.8$, $n = 1$ EFC with the I-coil can only recover 25% of the decrease in the toroidal angular momentum induced by the field of a TBM mock-up coil. The component of the $n = 1$ I-coil currents that recovers the most toroidal angular momentum is similar to the I-coil current that cancels the $n = 1$ plasma response to the TBM field supporting the hypothesis that 25% of the TBM induced rotation decrease is caused by an amplified $n = 1$ component of the TBM field. Consequently, the remaining 75% of the rotation decrease must be either caused by the remaining $n = 1$ field (corresponding to secondary mode components) or by $n > 1$ field components. If $n > 1$ components of TBM field contribute to the rotation braking, extending the EFC effort to higher $n$ components should recover further fractions of the rotation decrease.

The agreement between the $n = 1$ EFC currents that maximize the angular momentum and the currents that cancel the $n = 1$ plasmas response also demonstrates that the maximization of angular momentum is a viable new, non-disruptive technique to optimize the EFC.

The limited effectiveness of $n = 1$ EFC of the TBM field in rotating H-modes contrasts its ability to restore the resilience to locked modes in low density L-mode plasmas [2,3]. The mechanism that leads to the formation of the locked mode is thought to be a resonant torque and therefore different from the mainly non-resonant torque that decreases the rotation in fast rotating H-modes. The different effectiveness of $n = 1$ EFC therefore suggests a higher importance of either secondary $n = 1$ modes or $n > 1$ components of the TBM field for non-resonant magnetic braking than for resonant braking.

A particular concern for ITER is the observed significant increase of the detrimental effect of the TBM field error with $\beta_N$ once $\beta_N$ exceeds a value of approximately two [2,3]. A similar increase in the sensitivity to field errors was observed in previous H-mode error field studies and is explained by an increase in the $n = 1$ plasma amplification [14] suggesting that the TBM $\beta_N$ dependence is also caused by $n = 1$ amplification. Since EFC proved to be effective in cancelling the $n = 1$ plasma response, it can be speculated that $n = 1$ EFC will be able to suppress the deterioration with $\beta_N$.

While the recent experiments indicate that correcting the TBM induced magnetic field error with the long-wavelength saddle coils will only partially ameliorate its detrimental effect, the current understanding of the physics suggests that the effectiveness of long-wavelength EFC increases in regimes with lower rotation and higher $\beta_N$, which are both of concern for ITER. However, further experiments are needed to support these conjectures.

# REFERENCES

[1]  CHUYANOV, V.A., *et al.*, Fusion Eng. Design **83** (2008) 817.

[2]  SNIPES, J.A., *et al.*, Proc. 37th EPS Conference on Plasma Physics, Dublin, Vol. 34A (2010) P1.1093.

[3]  SCHAFFER, M.J., *et al.*, Nucl. Fusion **51** (2011) 103028.

[4]  KRAMER, G.J., *et al.*, Nucl. Fusion **51** (2011) 103029.

[5]  KESSEL, C.E., *et al.*, Nucl. Fusion **49** (2009) 085034.

[6]  OYAMA, N, *et al.*, "Effects of local toroidal field ripples due to test blanket modules for ITER on radial transport of thermal ions," accepted for publication in Nucl. Fusion.

[7]  EIKKINEN, J.A., and SIPILÄ, S.K., Phys. Plasmas **2** (1995) 3724.

[8]  PARK, J.-K., *et al.*, Phys. Rev. Lett. **102** (2009) 065002.

[9]  LANCTOT, M.J., *et al.*, Phys. Plasmas **17** (2010) 030701.

[10] BOOZER, A.H., Fusion Sci. Technol. **59** (2011) 561.

[11] OKABAYASHI, M., *et al.*, Nucl. Fusion **45** (2005) 1715.

[12] BUTTERY, R.J., *et al.*, Phys. Plasmas **19** (2012) 056111.

[13] BUTTERY, R.J, *et al.*, this conference, paper EX/P4-31.

[14] REIMERDES, H., *et al.*, Nucl. Fusion **49** (2009) 115001.

[15] GAROFALO, A.M., *et al.*, Nucl. Fusion **42** (2002) 1335.

[16] PARK, J.-K., *et al.*, Phys. Plasmas **14** (2007) 052110.

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