Physics and engineering research needs for 3D coil systems for a tokamak DEMO

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Conventional and low-aspect-ratio tokamaks are sensitive to 3D fields as small as $\delta B/B \sim 10^{-4}$, and the application of 3D fields has effectively become a requirement for stable high-β tokamak operation. Present tokamak experiments are successfully utilizing 3D fields to correct low-n field errors arising from coil imperfections, stabilize resistive wall modes (RWMs) with active feedback control, and suppress edge localized modes (ELMs). Coils utilized for the above applications may also be beneficial for providing axi-symmetric radial field for vertical instability control, and could hypothetically be used for increasing energetic electron transport during disruptions – possibly avoiding generation of large runaway electron plasma currents. However, the coil systems for producing these 3D fields may not be viable in the DEMO environment. In particular, the high neutron fluence in DEMO would be damaging to any insulating material used in control coils and also generates strong nuclear heating of the coils. Thus, it will be required to place 3D control coils behind sufficient nuclear and thermal shielding – i.e. behind the blanket modules and likely the high-temperature (HT) component shields. The resulting potentially significant distance between the 3D field coils and the plasma could have serious consequences for the physics performance and engineering design of DEMO.

Error field correction (EFC) coils

Low toroidal mode number (n=1-2) resonant magnetic field perturbations arising from imperfections in the nominally 2D poloidal field (PF) and toroidal field (TF) coil systems can result in the formation of magnetic islands (locked modes), resonant field amplification (RFA) of error fields, and strong toroidal flow damping – all of which can degrade plasma confinement and/or lead to a plasma termination. For major radii inside the radius R_0 of a toroidal array of 3D radial field coils, the radial field decays as $(R/R_0)^n$. Since low-n – in particular n=1 – fields are most important to correct, maintaining a small plasma-coil separation is generally less critical for error field correction, and the ITER EFCC system is a good example of how outboard ex-vessel mid-plane coils are useful for controlling low-n error fields.

However, there are two recent findings that motivate additional EF physics research and engineering design relevant to DEMO. First, inclusion of the plasma response to low-n 3D fields **[1]** has recently been shown to be critical to achieving predictive capability (versus empirical) for optimal error field correction and coil design. Plasma response effects have only recently been considered for ITER EFC analysis **[2]**, and have not been considered for any tokamak DEMO. Second, NSTX results have recently indicated that $n > 1$ (n=3) intrinsic error fields can negatively impact the sustainment of high β scenarios operating above the no-wall stability limit, and that correction of n=3 error fields significantly improves performance. The present ITER EFC system is limited to correcting n=1, and DEMO designs have not considered the possible need for this capability. The very high beta operating points (near the ideal wall limit) proposed for tokamak DEMOs could amplify $n > 1$ fields – potentially increasing the need $n > 1$ EFC capability. Such higher-n correction will be more difficult to achieve in DEMO due to the larger plasma-coil separation than is common in present experiments. The potentially high coil currents required for n > 1 EF correction with far-away coils could represent a significant engineering challenge for DEMO.

Resistive Wall Mode (RWM) control coils

At low plasma rotation and with beta above the no-wall stability limit – as expected in ITER advanced operating scenarios and in a tokmak DEMO – the n=1 RWM is commonly the most unstable mode that requires continuous and sufficiently fast 3D active feedback control. While the off-midplane in-vessel control coils proposed for ITER may be sufficient for n=1 RWM control in ITER, the situation in a tokmak DEMO will be much more challenging. First, the significant electrically conducting structure in the shielding between the plasma and RWM control coils will likely decrease the coil-plasma mode coupling efficiency making active feedback control more challenging. Further, operation near the ideal wall limit – as proposed in DEMO scenarios could increase the intrinsic RWM growth rate significantly – up to perhaps ~0.1-1% of the Alfvénic growth rate. Previous RWM stabilization analysis for tokamak-based DEMOs such as ARIES-RS and ARIES-ST largely ignored the RWM control issue, and for ARIES-AT neglected the important impact of the blanket modules, HT shields, and copper vertical stabilization shells on RWM control **[3]**. VALEN calculations for n=1 RWM control for ITER using ex-vessel control coils and including the blanket modules indicate that simple PD control techniques are ineffective at achieving C_β above 0.25 **[4]** which is unacceptably low for the estimated expected $C_\beta \approx 0.7$-0.9 values required for DEMO. At the very least, it is likely that advanced state-space controllers **[5]** (presently experimentally untested) would be required in DEMO to operate near the ideal-wall limit and to minimize control power. Because of the very broad pressure and current profiles and very high C_β values expected in a tokamak DEMO, n > 1 RWMs are also expected to be unstable and require active feedback control. Thus, the coils, power supplies, and mode detection systems will require multi-n capability, as perhaps required for EFC as described above. Finally, additional design work is also needed to determine if the same coil system used for EFC (typically high current, low voltage) can be utilized for RWM control (typically high voltage/slew-rate, low current) in DEMO.

ELM control with Resonant Magnetic Perturbation (RMP) coils

The high-confinement mode (H-mode) of operation is needed to achieve high fusion gain in ITER and DEMO. The steep gradients of the edge transport barrier associated with H-mode can trigger ELMs which rapidly (sub-ms time-scale) release a significant fraction (up to ~10%) of the plasma stored energy. The very high transient edge heat fluxes associated with ELMs can damage the divertor and first-wall components of ITER and DEMO necessitating the reduction/elimination of ELMs. For example, to avoid ablation/melting of the ITER tungsten divertor, ELM energy losses must be held below approximately ~1MJ or 0.3% of ITER stored energy. A tokamak DEMO **[3]** will have 2-3 times higher total plasma stored energy with similar plasma size, implying ELM size control to the 0.1% level is required. Thus, ELMs must effectively be eliminated.

The application of non-axisymmetric RMP fields in the edge region is a promising technique to suppress ELMs **[6]**; however, significant additional research is required to provide a firm scientific and technical basis for extrapolation to DEMO and support the ITER program. In particular, a deeper understanding of how the 3D fields modify edge transport and stability is needed, and at pedestal collisionality and normalized machine size (ρ^*) relevant/extrapolable to DEMO. At least three physics mechanisms have been identified as possible means of affecting the edge transport: 1) formation of stochastic magnetic field lines in the H-mode pedestal that lead to enhanced radial transport and/or from the electric fields arising from quasi-neutrality, 2) variations in the magnetic field strength which directly enhance the neoclassical thermal and particle transport, and 3) non-axisymmetric variations in the magnetic field strength which damp toroidal rotation and zonal flows and thereby enhance the transport. With regard to stability, 3D equilibrium effects could impact the ELM stability directly, and modifications to the rotation and rotation shear by 3D fields could also impact ELM stability.

Beyond suppressing ELMs, there are additional requirements for RMP fields to be beneficial to DEMO performance. 1) the reduction in thermal confinement by RMP must be minimized to minimize the loss of fusion gain, 2) it may be required to have separate control of the thermal and particle transport to minimize impurity accumulation during ELM-free operation while retaining high thermal confinement, 3) the 3D field must be sufficiently well localized to the plasma edge to avoid the excitation of core locked-modes and/or core flow damping.

The edge localization of 3D magnetic fields in DEMO using conventional coils may be the most challenging requirement, since n=3-6 perturbations are likely needed for RMP transport control, and the rapid fall-off in field magnitude with plasma-coil distance is problematic. If high-n field perturbations are created by coils far from the plasma, high coil currents are required, and the amplitudes of lower-n side-bands must be carefully controlled to minimize unwanted field components that penetrate deeper into the plasma core. Further, mid-plane coil systems have not suppressed ELMs in any tokamak device, so having at least upper and lower off-midplane coils is likely a requirement. Whether far-away off-mid-plane coils can mitigate ELMs is presently an unanswered question – both theoretically and experimentally. Designing close-fitting ELM control coils has proven to be extremely challenging in ITER, and viability in DEMO is questionable.

Other applications of 3D coil systems

The in-vessel 3D coil system presently proposed for ITER also has capability to provide n=0 radial field for vertical control. If RMP coils can be placed sufficiently close to the plasma in DEMO, the same coil systems may also be able to provide vertical control in DEMO. This possibility has not been explored for DEMO. Further, runaway formation during the quench phase of a high current disruption is a very serious issue for ITER, and likely must be completely avoided in DEMO. Enhanced radial transport of fast-electrons by residual 3D ergodic fields in the plasma core produced by 3D RMP fields already present to suppress ELMs may be capable of reducing the danger of runaways in DEMO.

Summary of design issues with 3D coils

In summary, there are several cross-cutting issues and performance trade-offs associated with 3D current carrying coils that must be considered in a tokamak DEMO design.

1. 3D coil maintainability and survivability will be very challenging for coils placed near the plasma.
2. For 3D coils placed close to the plasma, the disruption forces on the coils, and what the coils are attached to, are likely increased.
3. 3D coils near/among the breeding blankets will reduce the tritium breeding ratio.
4. Far-away coils will require higher coil currents, and also may not provide adequate physics performance.
5. The higher coil currents of far-away 3D coils increase the electromagnetic forces on the 3D coils and require more space.
6. 3D fields will also impact the SOL and possibly the divertor, and could result in toroidally asymmetric heat and particle loads on the PFCs. This could be beneficial by spreading the heat flux, or deleterious, by increasing the peak heat flux. The impact of these effects has not been considered for ITER or DEMO.
7. A single 3D coil set may not be able to provide for simultaneous EF, RWM, RMP, and vertical control, thus multiple coil sets may be required.

Elements of a possible research thrust for 3D fields in tokamaks

1. Enhanced theory, modeling, and diagnostic support, and additional 3D coils systems on existing, upgraded, and future facilities to understand the plasma equilibrium, transport, and stability response to 3D fields with plasma ν^* and ρ^* relevant/extrapolable to DEMO and including advanced control methods.
2. A systematic comparison – both experimentally and theoretically – of near and far-away coils systems (both on and off-midplane) with a full assessment of the engineering and physics trade-offs for a tokamak-based DEMO.
3. For close-fitting coils, exploration and development of coil/insulator technologies that can survive the nuclear environment of DEMO – including coil designs without insulators (i.e. integrated into the vacuum vessel/other components)
4. Understanding of the interaction between EF/RWM/RMP 3D fields and any permeable ferritic materials in inserts or blanket modules near the plasma edge.
5. Assessment of a wider range of uses and implications of 3D coils – including runaway suppression and 3D heat loads on PFCs from EF and RMP fields.

References

[1] J.-K. Park, et al. Phys. Plasmas **14** (2007) 052110
[2] J.-K. Park, et al. Nucl. Fusion **48** (2008) 045006
[3] S.C. Jardin et al. Fusion Engineering and Design **80** (2006) 25–62
[4] J. Bialek, 6th ITPA MHD task group meeting, Tarragona, Spain (2005)
[5] O. Katsuro-Hopkins et al. Nucl. Fusion 47 (2007) 1157–1165
[6] T. E. Evans et al. Phys. Rev. Lett. **47** (2004) 235003-1