An Experiment to Tame the Plasma Material Interface

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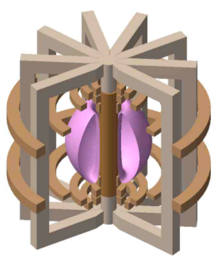
The plasma-material interface is the untamed frontier of fusion science. Approaches to heat flux handling and tritium retention that may work for ITER do not generally extrapolate to Demo. Defining questions at this frontier include: Can extremely high radiated-power fraction be consistent with high confinement and low Z_{eff}? Can magnetic flux expansion and/or stellarator-like edge ergodization reduce heat loads sufficiently? Can tungsten or other solid materials provide acceptable erosion rates, core radiation and tritium retention? Can dust production be limited, and can dust be removed? Can liquid surfaces more effectively handle high heat flux, off-normal loads and tritium exhaust, while limiting dust production? Does the reduction of hydrogenic recycling from liquid lithium surfaces improve plasma performance? Answers must be developed and validated in the context of high-performance, fully steady state plasma operation, avoiding high-energy ELMs and eliminating almost all disruptions. Ultimately compatibility with high neutron fluence must be demonstrated in a Component Test Facility or in Demo itself. A test vehicle to explore this untamed frontier is a high-performance, high-power-density plasma with long pulses, excellent diagnostic access, replaceable first wall and divertor, flexible poloidal field coils, powerful heating and current drive systems, extensive deuterium and trace tritium operation, and the ability to test a range of plasma-facing materials, both solid and liquid, at reactor-relevant temperature.

Draft requirements for such a facility include:

- Input power / major radius ~ 50 MW/m
- Heating power / H-mode threshold power > 5, close to $n = n_G$
- Flexible poloidal field system capable of wide variation in flux expansion
- Non-axisymmetric coils to produce stellarator-like edge field structure
- Replaceable first wall and divertor, solid and liquid
- High temperature ~ 600C first wall operational capability
- Pulse length ~ 200 – 1000 sec
- Excellent access for surface diagnostics
- A range of heating and current drive systems
- Extensive deuterium and trace tritium operational capability

A candidate configuration for such a facility has been identified using water-cooled demountable copper coils for flexibility and accessibility. At R ~ 1m, a ~ 0.55m, B_t ~ 2T, I_p ~ 3.5 MA, this device, called the National High-Power Advanced Torus Experiment (NHTX), would provide a cost-effective platform, in conjunction with an enhanced program in plasma facing surface science and component technology, for developing the materials and techniques to tame the plasma material interface. Its results would be applicable to tokamaks, spherical torus's and stellarators.



This work supported in part by U.S. DOE Contract # DE-AC02-76CH03073