An Experiment to Tame the Plasma Material Interface

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Abstract. The plasma material interface in Demo will be more challenging than that in ITER, due to requirements for approximately four times higher heat flux from the plasma and approximately five times higher average duty factor. The scientific and technological solutions employed in ITER may not extrapolate to Demo. The key questions to be resolved for Demo and the resulting key requirements for an experiment to "tame the plasma material interface" are analyzed. A possible design point for such an experiment is outlined.

1. Introduction

The operational requirements for a Demonstration Power Plant (Demo) considerably exceed those for ITER. A 1 GWe Demo about the size of ITER must produce ~ 2500 MW of fusion power at Q ~ 25 and demonstrate techniques to achieve average duty factor of ~ 75%. Thus the challenge of "Taming the Plasma Material Interface[1]" of Demo is daunting. Compared with ITER's operation at 500 MW, Q = 10 and maximum planned average duty factor of ~15%, Demo's plasma material interface must accept approximately four times higher power and operate for extended periods at approximately five times higher duty factor. Furthermore, because of the wide difference in performance requirements for ITER and Demo, ITER will use scientific and technological solutions that may not extrapolate to Demo, such as a moderate fraction of radiated power and water-cooling of both the first wall and divertor. In particular the low first-wall temperature of ITER (~200C) compared with Demo (~700C) presents a different environment with respect to tritium codeposition and bulk retention, while ITER's low duty factor and relatively short pulse provide opportunities for tritium removal that are not available for Demo.

2. Key Questions for an Experiment to "Tame the Plasma Material Interface"

Can high-performance, fully steady state plasma operation avoid high-energy ELMs and damaging disruptions? ITER itself cannot tolerate repetitive ELMs of significant amplitude, nor many high-energy disruptions. For Demo it will be critical to have in hand techniques capable of reducing the frequency of these events to extremely low levels. High performance plasma operation is required for these studies, but ELMs and disruptions should not be so energetic as to make it difficult to explore ELMy and disruptive regimes in the process of developing the techniques for ELM and disruption control. Non-axisymmetric coil systems should be explored for long-pulse MHD control.

Can extremely high radiated power fraction be consistent with high confinement and low Z_{eff}? The recent European Power Plant Conceptual Study[2] relies in some designs on very high radiated power to reduce heat flux to the divertor plate. EU-B, for example, has core plus edge radiated power of 58%, total radiated power of 80 – 90%, and $Z_{eff} = 2.7$. The plasma dilution and core radiation result in a requirement for an H-factor of 1.2, at a Greenwald density parameter of 1.2 and plasma current of 28 MA. Operation with such high radiated power fraction is not consistent with high gain in ITER. It is anticipated that high core radiated power, reducing the heat flux across the separatrix, will raise the overall input power requirements for sustaining H-mode performance. Thus, like fusion reactor designs and unlike ITER, an experiment to “tame the plasma material interface” must have heating power far exceeding its L to H transition power, in order to be able to explore this approach.

Can magnetic flux expansion and/or stellarator-like edge ergodization reduce heat loads? A number of innovative divertor magnetic configurations have recently been developed[3,4,5], which promise to increase the field line length to the divertor target and spread out the heat flux at the target itself. The Super-X divertor concept also offers the possibility of shielding the outer divertor from neutron fluence, by placing its target plate at large major radius. A new device should be designed to be flexible enough to test these ideas. It is also highly desirable that a device of this type be able to test edge magnetic field structures of the type expected in stellarators, both to support the stellarator line of development, and to take advantage of the possible favorable characteristics of such field structures[6].

Can tungsten or other solid materials provide acceptable erosion rates, core radiation and tritium retention? There are concerns about erosion rates of W in the presence of seed impurities, and W can suffer a “foaming” of its surface structure in the presence of He bombardment[7], shown in FIG. 1, possibly resulting in increased erosion and dust formation. The erosion rate on W surfaces of 0.1nm/sec reported for ASDEX-U[8] would be unacceptable in Demo, at much higher heat and particle fluxes. ASDEX-U also experiences limitations in its operational scenarios with W[8], due to core radiation. Results from C-Mod[9] suggest that tritium retention may be problematic even with a refractory metal divertor and first wall. Tritium diffusion in refractory metals is extremely sensitive to temperature, and Demo will operate with a hot first wall. Thus the new device must be able to operate with solid divertor and first wall surfaces at temperatures up to ~ 1000K in regions away from the divertor strike points, which will be at higher temperature, and be capable of flexible material change-out, including the use of mixed materials.

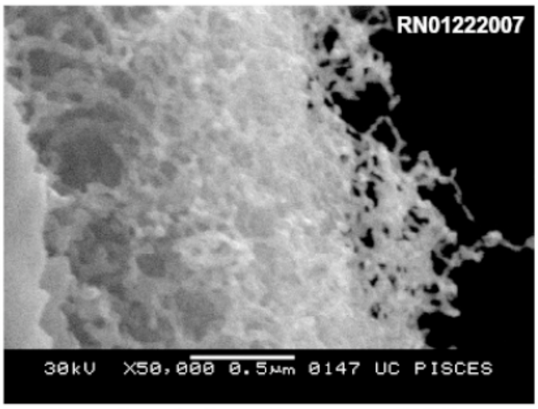

FIG. 1. Foaming of tungsten surface due to bombardment with helium.

Can dust production be limited, and can dust be removed? As indicated in ITER studies, and even more seriously for Demo, dust in the vacuum vessel can present a major safety and tritium retention issue. A new device must be configured to carefully monitor dust formation and to test techniques to remove dust actively.

Can liquid surfaces effectively handle high heat flux and provide adequate tritium exhaust, while limiting dust production? Recent results on CDX-U[10], T-11M[11] and FTU[12] are encouraging with respect to plasma operation with liquid lithium surfaces. For example there is evidence that self-shielding is effective in clamping the surface temperature of a capillary porous system on FTU. Test stand results[13] suggest that very high power densities, up to at least 50 MW/m^2, can be handled on capillary porous systems, albeit with high evaporation rates, and that high short-pulse heat loads up to at least 60 MJ/m^2 can also be tolerated. A new device should be capable of testing both capillary porous and fast-flowing liquid metal divertor targets, including assessment of liquid-surface stability and surface temperature limits, as well as development of techniques to assure that metal does not accumulate in the vacuum vessel. Results from TFTR[14], CDX-U[10], T-11M[11], FTU[12] and NSTX[15] suggest that reduced hydrogenic recycling associated with a lithium surface can improve confinement, and suppress ELMs in the case of NSTX. These results should be able to be followed up in a new device.

Can plasma-material interface solutions developed at low neutron fluence be made compatible with the high neutron fluence of Demo? Bulk material properties are significantly altered by neutron irradiation, and damage sites retain tritium. Complex joining technologies need to be qualified under significant neutron fluence. Any new facility must work in conjunction with a Fusion Materials Irradiation Facility to resolve these issues.

3. Quantitative Research Capability Requirements to Address Key Questions

Input power / surface area P/S ~ 1 MW/m^2; Input power / major radius P/R ~ 50 MW/m: Our putative demonstration power plant would be sustained by 100 MW of external heating and current drive power, plus 500 MW of internally produced alpha power. For a plasma surface area equal to ITER's, this corresponds to P/S = 1.05 MW/m^2 of average heating power; for the U.S. ARIES-AT study P/S = 0.85 MW/m^2. Thus the scale of the challenge to the first wall sets the requirement for P/S at ~ 1 MW/m^2 in our device.

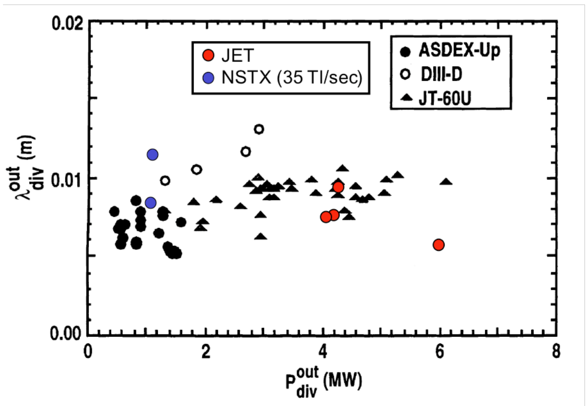

FIG 2. Outer midplane power scrape-off width vs. power delivered to outer divertor.

It is difficult to estimate the peak heat flux at the divertor strike point, which is likely the more critical parameter. There is evidence that the edge and SOL electron temperature gradient scale length varies with the major radius of tokamaks[16]. However the available evidence based on mapping heat flux from the divertor surface to the outer midplane scrape off layer does not support a scaling of the power scrape-off width with device size. Since gas puffing and impurity seeding can result in partial detachment, it seems best to compare devices operating in deuterium ELMy H-mode with no or minimum gas puff and without externally injected impurities. An initial compilation[17], and more recent results from JET[18] and NSTX[19] under these conditions (FIG. 2) show little variation with machine size. Assuming the intrinsic outer-mid-plane power scrape-off width is independent of machine size, pending further results in this area, the relevant parameter for inter-device comparison is P/R.

Our putative power plant has input power divided by major radius of 97 MW/m. For the U.S. ARIES-AT study it is 74 MW/m. However synchrotron and bremsstrahlung losses should not be included in assessment of the challenge to the divertor (although they should be included in the challenge to the first wall discussed above). These are likely to be in the range of 20-40% in Demo, but negligible in our experiment. It seems that a reasonable requirement for P/R is thus in the range of 50 MW/m.

Heating power / H-mode threshold power $P/P_{LH} > 5$, close to $n = n_G$: While projections of the L to H transition threshold power are very uncertain, power plant designs generally have large margins, making the simultaneous use of high core radiated power and H-mode confinement more credible. Thus a high ratio of heating power to L to H transition power is a central requirement for a machine with a mission to test power-handling concepts. A margin of a factor of five at high density seems to be a minimum specification.

Stored energy / major radius W/R ~ 5 MJ/m: Repetitive ELM energy loss in ITER must be reduced to < 1 MJ[20] (0.3% of total stored energy) to assure acceptable divertor lifetime. The width of the ELM heat deposition is assumed to be equal to the power flux width between ELMs, so a relevant scaling parameter is W/R. To understand the physics of ELM suppression it would be desirable to be able to study long-pulse operation with ELMs up to ~ 3% of the total stored energy. W/R should then be limited to ~ 5 MJ/m, which is ~ 10x less than in ITER. This analysis ignores the deposition time of the ELM energy, which is estimated to scale as $L_{//}/C_s$. However both $L_{//}$ and C_s are variable and likely to be smaller in the device under consideration than in ITER. In the very simplest analysis this factor comes in as the square root. If the deposition time is substantially shorter than the 500 μsec assumed for ITER, the range of studies of ELMy plasmas would need to be reduced, or larger damage accepted. A similar set of arguments pertains to disruption suppression studies, and 5 MJ/m, corresponding to 15 MJ total in JET (R = 3m), should allow studies of disruptive regimes and their control without unacceptable damage due to rapid heat deposition.

Flexible poloidal field system capable of wide variation in flux expansion and the ability to direct divertor field lines to large R: New ideas for divertor magnetic configurations must be able to be tested in such a device. An initial set of highly flexible PF coils will need to be installed, but it is also necessary to be able to replace or augment these coils as understanding evolves. This will likely require readily demountable TF coils.

Non-axisymmetric coils for stellarator-like edge and MHD stability: The requirements for such coils require further definition, in collaboration with the stellarator community. Non-axisymmetric coils would also be used for ELM suppression, control of resistive wall modes and possibly for improved MHD stability [21,22].

High temperature operation: The diffusivity of hydrogenic species in materials is very sensitive to ambient temperature[23] (FIG. 3). Since for reasons of thermal efficiency a power plant must operate with a hot first wall, up to ~1000K, it is important to examine the effects on hydrogenic retention at relevant material temperatures. It would be acceptable to cycle the temperature of the plasma facing components to ~ 600K between shots to limit diffusion during off periods.

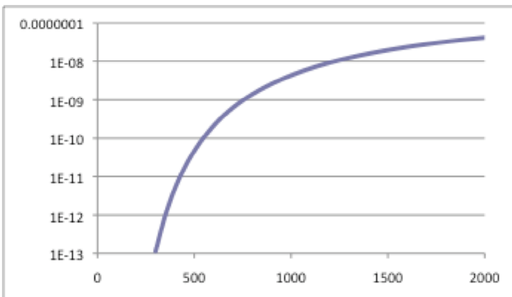

FIG 3. Diffusivity (m^2/sec) of H in W vs. temperature (K).

Replaceable first wall and divertor, solid and liquid: New ideas for divertor material configurations, including the use of mixed materials, must be able to be tested in such a device. An initial flexible system should be installed, but it is important to be able to reconfigure the first wall and divertor, *e.g.,* to new materials, solid or liquid, between experimental campaigns. This will require extremely good access to internal components.

Pulse length ~ 200 – 1000 sec; Total On-time ~ 10^6 sec / year: Results from long-pulse Tore-Supra discharges suggest that particle balance comes into steady state after ~ 100 seconds[24]. Tore Supra operates with carbon walls, so its "steady-state" includes steady hydrogenic accumulation in the torus by codeposition. It is likely that the high particle fluxes anticipated in the machine we are considering, particularly with metallic walls, will bring particle balance into equilibrium relatively quickly. It is also likely that thermal equilibrium of the first wall will be reached relatively quickly, since aggressive temperature control will be required in a device of this sort. Internal plasma processes for a relatively small device should come well into equilibrium over a time scale of 200 seconds. It also seems reasonable to project that 200 seconds of very high power operation will greatly dominate the plasma-wall interactions associated with start-up and shut-down. Longer and therefore fewer pulses, however, may be desirable to limit thermal fatigue of components.

The total on-time per year is set by the desire to fully diagnose at least one plasma-wall configuration, and preferably more, in a year's operation. Since tritium accumulation must come to equilibrium in of order 10,000 seconds in Demo (dividing allowed torus inventory by flow rate), 10^6 sec should be adequate time for a number of such studies in different conditions. Indeed much shorter periods of operation have been very valuable in current experiments. The characteristic diffusion time for hydrogenic species in W at 1000K, over the ~ 4mm width contemplated in power plants, is also ~ 10,000 seconds. A requirement of 10^6 sec / year is equivalent to about 10 hours of operation at 25% duty factor, 120 days per year. This would correspond, for example, to a single 15-minute pulse or five 3-minute pulses per hour. If the plasma facing components are cycled to 600K between pulses, and to lower temperature overnight, the diffusion in the bulk metal should be negligible during off times.

Erosion and redeposition studies will be challenging. Assuming an erosion limit of 3mm in Demo, an erosion rate of 0.1nm/sec would give a lifetime of less than one full-power year. Techniques will need to be developed to measure erosion and redeposition to very high accuracy. One can imagine a large number of probes based on the DiMES technology[25] modified for much higher power, long-pulse operation. It would be desirable also to develop highly accurate ranging techniques, with wide accessibility, capable of measuring net erosion in the range of 10's of microns after as few as 12 days, or 10^5 seconds, of operation. Spectroscopic techniques have also been used to measure local erosion rates[8].

Excellent access for surface and plasma diagnostics, as well as PFC services: The discussion above, and general considerations about the mission to "tame the plasma material interface", make clear that extensive diagnostic access will be critical for success. In addition to all of the modern plasma diagnostics that will be needed to develop and sustain high-performance steady-state plasmas, it will also be necessary to provide an accurate understanding of the plasma-material interface at essentially all locations around the torus. It will be critical to develop the design of such a facility in concert with careful assessment of diagnostic requirements. Access will be required as well for heating and cooling systems, for pumping, for liquid metal, and to develop techniques for dust removal.

Extensive deuterium and trace tritium operational capability: Some of the work described here can be accomplished with H plasma operation. However edge plasma conditions are experimentally observed to be different in D compared with H (and much less different in DT compared with D). D operation is also required to study hydrogenic retention, since H is ubiquitous in many materials. A problem may arise even with D, in that after years of D operation it may be difficult to determine if accumulated D in the vessel is a result of the recent operation, or has accumulated over time. The experience on TFTR and JET has shown that tritium is a very valuable tracer for hydrogenic accumulation. Since tritium is so easily detected, even trace tritium experiments should be helpful, and at the low levels involved, the radioactivity from the DT reactions should not be higher than from extensive D operation.

Synergy with a Fusion Materials Irradiation Facility: There would be strong synergy between a device to “tame the plasma material interface” and a facility with the goals and capabilities of the International Fusion Materials Irradiation Facility, IFMIF. Changes in bulk materials properties measured after irradiation in IFMIF can be taken into account in the design of plasma facing components. The complex joining technologies required can be tested in the neutron penumbra of IFMIF. Material samples damaged in the neutron fluence of IMFIF can be exposed to Demo-relevant divertor plasmas to study their tritium retention and any changes in their plasma material interaction properties.

4. Required Parallel Activities

A number of activities in experiment, theory and modeling, and technology development will be needed both to prepare for and ultimately to support the operation of a new experiment to “tame the plasma-material interface.” In addition to parallel and synergistic research on ITER and a Fusion Materials Irradiation Facility, these include:

Experiment:

- Experiments on short-pulse fusion confinement facilities developing a reliable scaling of the SOL heat flux, and providing data for detailed tests of divertor theory and modeling.
- Experiments on long-pulse fusion confinement facilities to gain experience with fully non-inductive operational scenarios, including ELM and disruption control.
- Tests of divertor concepts and materials, such as the Snowflake[4] and Super-X divertor[5], liquid lithium, and high-temperature tungsten, at moderate P/R.
- Tests of non-axisymmetric coils to mimic stellarator edge, suppress ELMs, control resistive wall modes and possibly improve MHD stability.
- Diagnostic development to allow accurate local measurement of plasma-material interactions, including erosion, redeposition, dust generation, and hydrogenic retention.

Theory and Modeling:

- Modeling and code development for plasma surface interaction, including detailed verification and validation of models of erosion and redeposition, dust generation, hydrogenic retention, with uniform and mixed materials and with liquid metals.
- Divertor and scrape-off-layer theory and code development, including detailed verification and validation of parallel and perpendicular transport of impurities and heat.
- Divertor simulation and innovation, for example developing the magnetic, neutral baffle and target configuration for a Snowflake or Super-X divertor with a liquid metal target.
- Design of non-axisymmetric coils to mimic stellarator edge conditions, suppress ELMs, control resistive wall modes and possibly improve MHD stability.

- Theoretical materials development, for example developing improved W alloys with reduced surface deformation due to He bombardment and reduced tritium retention.
- Modeling of flowing liquid metals both for capillary porous systems and fast jets.

Technology Development:
- Materials testing in a powerful plasma-surface interaction research facility, for example confirming plasma-surface interaction properties of improved refractory materials.
- PFC technology development and testing, for both solid and liquid systems, for example of appropriate joining technologies and methods to reduce oxygen for high temperature He cooling, and of steady-state and pulsed power handling by liquid surfaces, as well as practical approaches to lithium recycling.
- Development of long-pulse heating and current drive technologies.
- Development of technologies to remove dust in the Demo environment.

4. Possible Experiment to "Tame the Plasma Material Interface"

A candidate configuration for a facility to "tame the plasma material interface", meeting the requirements outlined here, has been developed[26] (FIG 4.). It has tentatively been given the name "National High Power Advanced Torus Experiment", NHTX. The basic machine parameters are R = 1m, P_{in} = 50 MW, a = 0.55m, κ = 2.7 – 3.0, B_T = 2T, I_p = 3 – 4 MA and arbitrary pulse length. It uses steady-state water-cooled demountable copper coils for maximum flexibility and accessibility. The steady-state power requirement for this device is 300 MW: 166 MW for heating and current drive systems, 88 MW for TF power, 37 MW for PF power and 10 MW for balance of plant. Its key characteristics are:

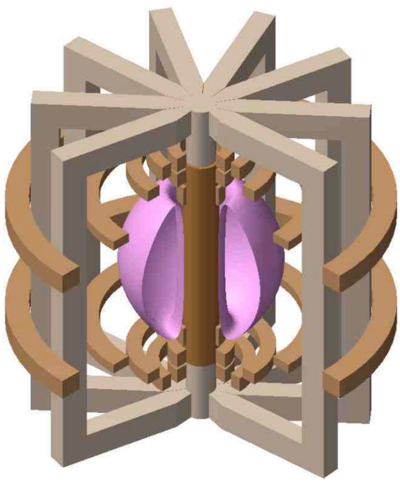

FIG. 4. An experiment to "Tame the Plasma Material Interface".

- P/S = 1.1 MW/m^2; P/R = 50 MW/m
- 30 MW of 110 keV NBI sustains 3 MA plasma current with 70% f_{bs} and q > 2 across the profile.
- 20 MW is available for RF heating and current drive.
- W/R ~ 4 – 5 MJ.
- Solenoidal flux for plasma start up and ramp up.
- P/P_{LH} = 8 @ n/n_{GW} = 1[27]
- Flexible poloidal field coil system capable of heat flux expansion variable up to 40.
- Rectangular TF coils with external PF coils providing the dominant vertical field. Alternate internal PF coil configuration supports Super-X divertor.
- Wide radial access for surface and plasma diagnostics
- Demountable upper TF coil legs give vertical access for change-out of internal components by crane lift.
- Removable inner high-temperature vacuum vessel.
- Outer vacuum vessel incorporating neutron shielding for extensive deuterium and trace tritium capability.

Calculations with SOLPS[28] for this device show that even with a flux-expanded magnetic configuration, with 30 MW of power crossing the separatrix and no impurity seeding, a peak heat flux of 18 MW/m^2 would be incident on the divertor plate[29]. However if 20% of the divertor target heat flux is absorbed into evaporative cooling of lithium, 50% of the power that would otherwise have struck the divertor plate will be lost by radiation, Z_{eff} at the plasma will be 2.0, and the peak plasma heat flux will be reduced to 6 MW/m^2. Calculations of the flux expansion possible with the Super-X divertor (FIG. 5) indicate that even with 50 MW of power crossing the separatrix, the peak heat flux is reduced to about 10 MW/m^2 and with only 4% of the incident power absorbed in lithium evaporation, the radiated power is 50%, Z_{eff} at the plasma is 1.6, and the peak incident plasma heat flux is reduced to 2.5 MW/m^2.

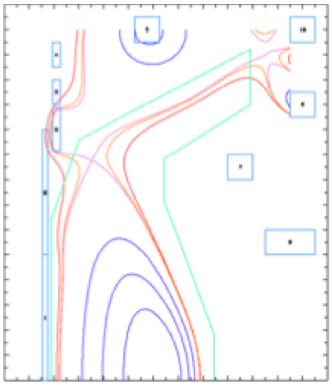

FIG 5. Super-X divertor.

5. Conclusion

The device discussed here, in conjunction with ITER, IFMIF and the work discussed above in experiment, theory and modeling, and technology development, would provide a cost-effective program for qualifying the scientific techniques, materials and technologies to "tame the plasma material interface" in support of Demo or a Nuclear Component Test Facility. The low aspect ratio of this device would provide physics support for a breadth of choice in aspect ratio for either or both of these. The plasma-material interface results from this experiment would be directly applicable to a tokamak, spherical torus or stellarator Demo.

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