Simulating the Demo Edge Plasma in a Compact High Heat Flux Experiment

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1. An Experiment to Fill the PMI Gaps to Demo

The Greenwald Report[1] has identified several gaps between ITER and Demo in the area of the plasma-material interface. The specific gaps identified are in 1) the sufficient understanding of plasma-wall interactions to predict the Demo PMI environment, 2) physical understanding and the technological ability to construct reliable plasma-facing components that can survive the nuclear, high heat flux environment of Demo, and 3) the ability to design in-vessel components that can survive and function in the Demo edge.

A concept for an experiment has been developed which would enable research on each of the identified gaps in a Demo-relevant PMI environment. This concept, called the National High-power Advanced Torus Experiment (NHTX)[2], aims to achieve very high levels of P/R in a compact, cost-effective device with excellent access for diagnostics and change-out of internal components. The mission of NHTX is to study the integration of high performance plasmas, with good confinement and stability in long-pulse discharges, with a high heat flux plasma boundary. The experiment is presently conceived with major and minor radii of 1.0 m and 0.55 m respectively, and it will operate with plasma current and magnetic field of up to 3.5 MA and 2.0 T, respectively. The total planned auxiliary heating power is 50 MW, with 30 MW from neutral beam injection. This white paper presents 2D modeling of the NHTX edge plasma to assess the ability of such a device to provide a Demo-like environment in which to carry out research to fill the PMI Gaps.

2. On the ability of NHTX to provide a Demo-like divertor

2D simulations have been performed of the scrape-off-layer (SOL) plasma parameters and target heat flux profiles for NHTX[3]. The four primary magnetic configurations considered, shown in Figure 1, differ in the number of nulls, flux expansion, and divertor target geometry. In the simulations, a pedestal density of $n_e=1.5\times10^{20}$ m^{-3} is assumed, and 30 MW of power is input into the SOL. This power level should be achievable in NHTX, given the 50 MW of available heating.

The predicted divertor temperatures for these four configurations are shown in Fig. 1. The simulations consistently show that transport is sheath-limited at this power level, and as a result the temperatures are very high (in the range of 200-300 eV). At these high plasma parameters, impurity radiation is very inefficient; calculations have shown that even a very large impurity content (up to 4% Argon concentration) does not radiate a significant fraction of the input power (less than 10%). The resulting heat fluxes on the target plates are also shown, and show that extremely large values in excess of 100 MW/m^2 are possible, in the case with combined single null and low flux expansion. Clearly, the high power of NHTX would allow material and PFC testing up to extremely high power loading. In the highest flux expansion configuration, the heat flux remains over 10 MW/m^2, illustrating the difficulty of the power handling problem at these very high power levels.

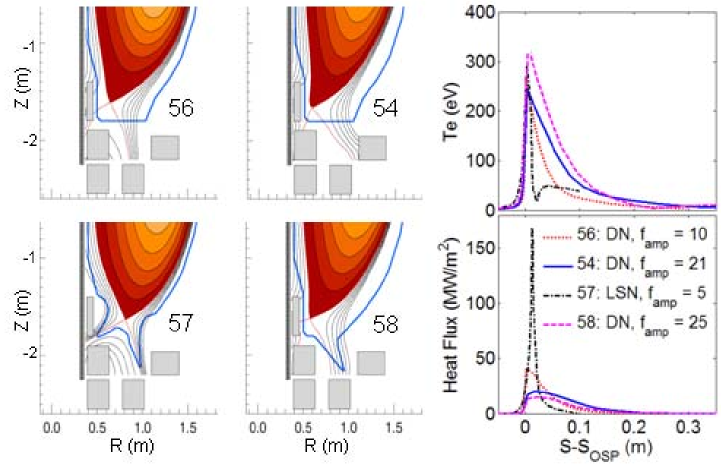

Figure 1) Divertor configurations and expected target temperatures, heat fluxes with P_{SOL}=30 MW

To simulate the Demo divertor, NHTX could be operated with a highly tilted target plate. In this configuration, the plate is oriented to promote partial detachment near the strike point; this is currently the planned divertor scenario for future large tokamaks including ITER. Simulations show that NHTX can achieve this operating point if sufficient gas is puffed in. In this case, an ITER-like divertor plasma is achieved, with high density (>10^{21} m^{-3}) near the separatrix strike point and low electron temperature (~1 eV); the heat flux is reduced to 10 MW/m^2, while the particle flux is more than 10^{24} D+/m^2/s. Assuming that Demo also operates at partial detachment (and that the maximum acceptable heat flux remains at ~10 MW/m^2), this scenario will allow a direct simulation of the Demo divertor, matching the plasma temperature and heat and particle fluxes. NHTX could thus test the current Demo heat flux mitigation strategy. Additionally, the compatibility of this divertor with the core plasma can be tested; it should be noted that the separatrix density in this detached case is very high (>5x10^{19} m^{-3}), which may not be consistent with high current drive efficiency or Advanced Tokamak scenarios.

3. Testing of Novel Heat Flux Mitigation Strategies

One of the mission elements of NHTX is to test alternative heat flux mitigations techniques, including new divertor geometries and alternative PFC materials or design. Two example novel scenarios have been simulated in the same manner as the conventional approaches described above.

The first of these is the Super-X Divertor[4], in which the divertor magnetic field structure is manipulated so that the separatrix is extended to much larger major radius. The higher radius of the target plate allows a simple geometric spreading of the heat flux, but the SXD additionally allows the field line connection length to be greatly increased. This can make detached operation more easily achieved, and also allows for higher radiated power levels. 2D modeling of an SXD has been performed in which the full 50 MW of heating power is put into the SOL, so that the divertor can tested at the highest possible power levels. These results can be compared with those for the highest flux expansion conventional divertor, also with 50 MW input power. It is found that the SXD calculations show a peak heat flux reduced by a factor of three, down to an acceptable 10 MW/m^2 from more than 30 MW/m^2. In addition, the plate of the SXD could be rotated in order to raise the strike-point density and lower the temperature, gaining the advantages of the vertical target described in the previous section. When this is done, the peak heat flux is reduced to only 5 MW/m^2. Clearly the SXD has the potential to reduce the power exhaust problem, and a successful test of this concept at Demo-relevant SOL parameters would help to close this gap to Demo.

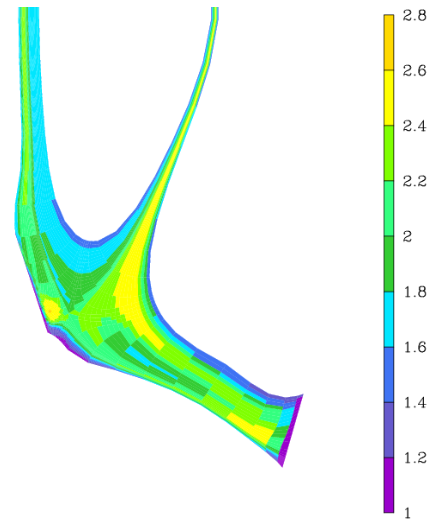

Figure 2) Z_{eff} with combined Li-evaporation and Super-X divertor

The second innovative heat flux mitigation strategy that has been simulated is the use of liquid lithium targets. In this modeling, a particular approach has been tested in which the surface of the lithium is allowed to heat up, until significant amounts of lithium evaporate from the target. In the simulations the vapor is trapped in the plasma due to rapid ionization, and radiates away a great deal of power[5]. This effectively shields the target from the incident heat flux so that further, excessive evaporation is avoided. As an example, a simulation was performed with the input power fixed at 50 MW, and 1 MW-equivalent of lithium vapor is released from the target (assuming a latent heat of 2.273x10^7 J/kg for the evaporated Li). The strong plasma fuelling from this vapor gives a high density, low temperature SOL plasma, which leads to strong radiation; in this case roughly 50% of the input power is lost via Li radiation. As a result, the heat flux on the target on the plate is reduced to less than 2.5 MW/m^2, showing that this scenario has promise as a heat flux mitigation strategy. The case described uses a Super-X divertor; similar scenarios can also be realized with conventional divertors, although more Li evaporation is required to reduce the heat fluxes. Although the lithium content of the plasma can be rather high, the low Z of Li combined with strong core hydrogenic fuelling might keep the dilution of the core plasma down to an acceptable level (core $Z_{eff} \sim 1.6$ in this case, see Fig. 2).

These are just a few examples of the sort of experiments that could be performed in NHTX. Realistically, the compatibility of these or any heat flux mitigation strategy with a high-performance core plasma needs to be demonstrated. In addition to the power handling aspect that has been the focus of this white paper, each divertor strategy must also be tested for its effectiveness in dealing with an array of PMI issues (for example, erosion and material migration, and helium pumping). An experiment such as NHTX could provide precisely this test, exploring the integrated PMI strategy in a Demo-like edge environment.

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[1] *Priorities, Gaps and Opportunities: Towards a Long-Range Strategic Plan for Magnetic Fusion Energy*, A Report to the Fusion Energy Sciences Advisory Committee, October 2007.
[2] R. Goldston et al., IAEA FEC 2008, FT/P3-12, Geneva, Switzerland, October 2008
[3] J.M Canik et al., PSI Conference, Toledo, Spain, May 2008.
[4] P. Valanju et al., submitted to Physics of Plasmas
[5] M.L. Apicella et al., EPS 2008, P-4.004, Hersonissos, Crete, Greece, June 2008