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Physics Design of the National High-power Advanced Torus eXperiment

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Introduction

Moving beyond ITER toward a demonstration power reactor (Demo) will require the integration of stable high fusion gain in steady-state, advanced methods for dissipating very high divertor heat-fluxes, and adherence to strict limits on in-vessel tritium retention. While ITER will clearly address the issue of high fusion gain, and new and planned long-pulse experiments (EAST, JT60-SA, KSTAR, SST-1) will collectively address stable steady-state high-performance operation, none of these devices will adequately address the integrated heat-flux, tritium retention, and plasma performance requirements needed for extrapolation to Demo. Expressing power exhaust requirements in terms of P_{heat}/R, future ARIES reactors are projected to operate with 60-200MW/m, a Component Test Facility (CTF) or Fusion Development Facility (FDF) for nuclear component testing (NCT) with 40-50MW/m, and ITER 20-25MW/m. However, new and planned long-pulse experiments are currently projected to operate at values of P_{heat}/R no more than 16MW/m. Furthermore, none of the existing or planned experiments are capable of operating with very high temperature first-wall (T_{wall} = 600-1000C) which may be critical for understanding and ultimately minimizing tritium retention with a reactor-relevant metallic first-wall. The considerable gap between present and near-term experiments and the performance needed for NCT and Demo motivates the development of the concept for a new experiment - the National High-power advanced-Torus eXperiment (NHTX) - whose mission is to study the integration of a fusion-relevant plasma-material interface with stable steady-state high-performance plasma operation. Such a device would not have a high-fluence NCT mission, but would advance the science and technology necessary to accelerate the NCT mission at reduced risk in a separate nuclear facility. For the NHTX mission, flexibility to test multiple divertor configurations and first-wall components is critical, and flexibility in plasma exhaust configuration and boundary shape is important for understanding the plasma-wall interaction. Sufficient profile control must be available to generate high-performance fully non-inductive plasmas with high $P_{heat}/R \leq$ 50MW/m and long pulses=200-1000s. Incorporation of hot walls, trace-tritium, liquid metals, and ELM and disruption control are additional design goals.

Physics Design

To achieve a high P_{heat}/R mission while minimizing the cost of auxiliary heating systems and magnet operation, small major radius is clearly favorable. Access for heating systems and diagnostics also place a practical lower bound on the plasma minor radius $\geq$ 0.5m. Systems code studies have been performed to determine the optimal design of such a device assuming normally-conducting actively-water-cooled magnets, ITER-98PBY2 H-mode confinement scaling, aspect-ratio-dependent elongation and no-wall stability limit scalings, and fully non-inductive (NI) current drive from bootstrap (BS) current and neutral beam injection current drive (NBICD) with up to 32MW of 110keV deuterium NBI. Inboard space for a half-swing solenoid capable of ramping I_P to 3.5MA is also included. As shown in Figure 1, the systems code studies find that an optimal aspect ratio A = 1.8-2.0 simultaneously maximizes the achievable P_{heat} and I_P (and W_{tot} - not shown) at fixed R_0. For aspect ratio A=1.8, the resultant NHTX design point is P_{heat} = 50MW at R_0=1m for P_{heat}/R = 50MW/m, I_P=3-4MA, B_T = 2T, κ = 2.7-3, $H_{98Y,2}$ = 1.3, β_N=4.5, β_T=14%, Greenwald density fraction f_{GW}=0.4-0.5, f_{BS} = 65%, and f_{NI} = 100%. Higher β_N and f_{BS} are possible with resistive wall mode stabilization and enhanced confinement.

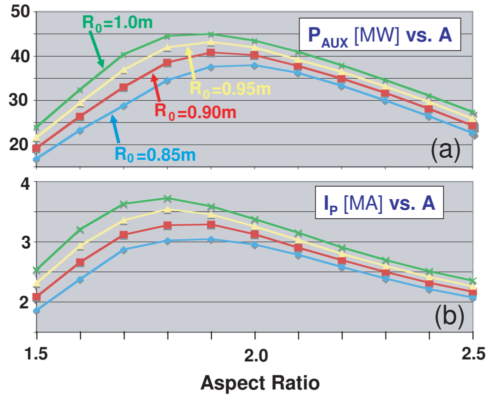

Figure 1: (a) Auxiliary power vs. aspect ratio A for a range of R_0, and (b) total I_P from BS and NBICD vs. A.

Component and diagnostic accessibility is particularly important for the NHTX mission, and Figure 2a shows the large separation between toroidal field (TF) coils (gray) and vessel (green) and the large ΔZ=2m vertical gap between outboard poloidal field (PF) coils (orange). Figure 2b shows the divertor PF coil design can accommodate an ITER-like lower-single-null plasma and divertor geometry (left), a JET-like divertor (right), and other concepts including a liquid-lithium divertor module. Figure 2c shows the PF coil set provides considerable shape flexibility, for example a squareness range of -0.15 to 0.25. A possible location for the NHTX high-temperature wall/liner is outlined in blue in Figure 2a.

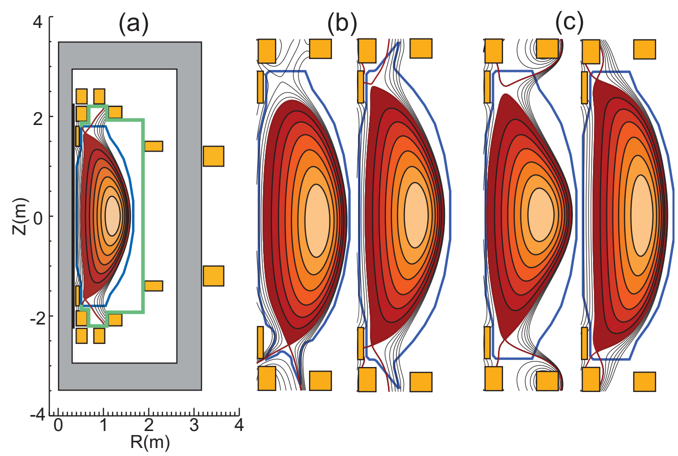

Figure 2: (a) NHTX cross-section, (b) example divertor configurations, (c) example plasma shaping flexibility.

Four 8MW NBI boxes from TFTR upgraded to long-pulse capability using the TPX design could provide a majority of the auxiliary heating power of NHTX (60-65%) and the central current drive necessary to supplement the bootstrap current. The remaining 18MW would come from RF sources to be specified. Ten TF coils of the geometry shown in Figure 3 can limit the TF ripple to 0.5% at the plasma boundary, and as shown in Figure 3, provide sufficient access for radially and vertically steerable tangential NBI.

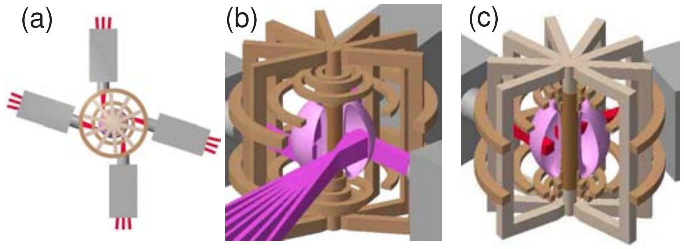

Figure 3: (a) Top view of NHTX with tangential NBI, (b) side view showing allowable R_{tan} range = $R_0 \pm 0.2$m, (c) allowable $Z_{tan} = \pm 0.4$m (note shifted NBI in red).

TRANSP calculations of NBICD have been used for benchmarking the 0D formulas used in the systems code. The kinetic profile shapes used are shown in Figures 4a and b and are taken from $f_{NI} = 65\%$ NSTX discharges with $Z_{eff}(0) \approx 2$, $T_i(0)/T_e(0)$ is fixed at 1.5, and the density chosen to achieve β_T = 14%. For these profiles, the pedestal electron collisionality ν_e^* = 0.01-0.05 is comparable to the ITER pedestal value. Figure 4c shows the current density profiles with R_{tan} = 1.15m for the middle NBI source. The associated q profile is weakly-reversed with $q_{min} > 2$, and low-n TAE modes are calculated to be stable with the NOVA-K code, while n=4, 5, and 7 modes are unstable with $\gamma/\omega \leq 2\%$. As R_{tan} is varied from 0.7m to 1.3m, the bulk current drive efficiency increases by as much as a factor of three (with fixed target q profile) highlighting the potential advantages of off-axis NBICD. Self-consistent BS+NBI safety factor profiles for different R_{TAN} are shown in Figure 4d. As seen in the figure, R_{TAN} variation can provide control over the core magnetic shear which can influence both the core thermal transport and MHD stability. Vertical NBI shifting is calculated to provide similar q profile control.

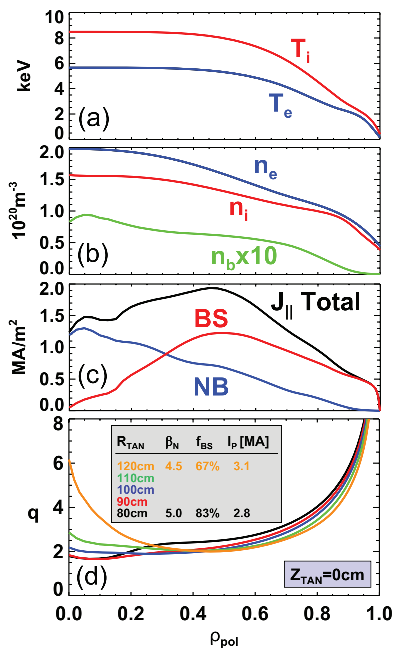

Figure 4: (a) T(ρ), (b) n(ρ), (c) $J_{||}(\rho)$, and (d) $q(\rho)$ vs. R_{tan}.

To gain a better sense of the challenge of managing high-heat flux at the reactor level, Figure 5a shows the peak heat flux estimated using a 2-point Borass model for an ITER-like LSN divertor in NHTX. The assumed Bohm $\chi_\perp$ (consistent with NSTX data [1]) results in a $\lambda_{q_{||}-midplane}$=0.8-1.3cm which is roughly twice the value expected in ITER. In the sheath-limited regime, this standard flux expansion geometry (poloidal flux expansion = 3) has peak heat-fluxes $\leq$ 70MW/m^2. Even at low T$_{divertor}$ with partial detachment due to significant radiation in the SOL, the peak heat-flux of 15-20MW/m^2 exceeds ITER design limit of 10MW/m^2. A key goal of the NHTX mission is to test if high core and divertor radiation fractions (as possible means of divertor heat-flux reduction) are compatible with H-mode access, low Z_{eff}, and efficient pumping and particle control [2]. Another approach to heat flux reduction is large magnetic flux expansion [3], and Figure 5b shows an NHTX double-null divertor configuration with poloidal flux expansion = 35 at the strike point. As seen in the figure, in the sheath-limited regime, peak heat-fluxes are reduced a factor of 5 relative to the ITER-like divertor. In this divertor configuration, liquid lithium could be tested as both a high heat-flux target and as a large-area pump for hydrogenic species.

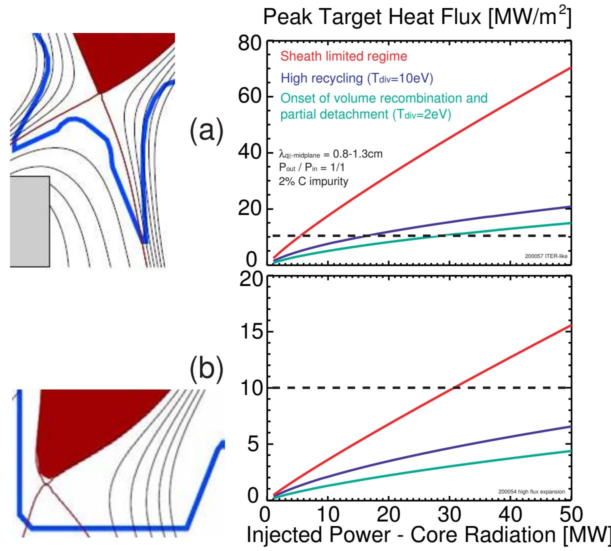

Figure 5: Peak heat-flux estimates in NHTX with (a) ITER-like and (b) high flux-expansion divertors.

Summary

The physics design described above provides an existence proof of a device capable of pursuing a mission of studying the integration of a fusion-relevant plasma-material interface with stable steady-state high-performance plasma operation. The design demonstrates flexibility to test multiple divertors, first-wall components, plasma exhaust configurations, boundary shapes, and plasma current profiles. Future design activities will focus on the design implications of high-temperature walls, the choice of wall material, trace-tritium for retention studies, and liquid metals for high heat flux and particle control. The avoidance of transient heat-loads to the divertor and first wall is essential in Demo. Thus, coil designs for ELM suppression and resistive wall mode control will also be pursued in addition to disruption avoidance and mitigation techniques. With these integrated design features, the NHTX device would advance the science and technology necessary to accelerate a nuclear component testing mission at reduced risk.

References

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