Research Needs Workshop (ReNeW)

White Paper: Some issues related to core and divertor control for DEMO

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Issues

A DEMO-class tokamak must produce significant fusion power, while at the same time justifying economic viability. To optimize the return on investment, it is important for a DEMO reactor to operate in a regime favorable to "high performance," particularly with regard to optimizing the plasma energy confinement time (τ_E) and the plasma beta (β_T). Higher values of τ_E and β_T are more readily obtained as the plasma shape becomes increasingly "triangular," and this is achieved more naturally in a balanced double-null (DN) rather than in a single-null (SN) . Since ITER will rely on the SN configuration and generate considerably less plasma heating and fewer neutrons than a DEMO, core and divertor control solutions that may be sufficient for ITER may not carry over to DEMO, e.g., ARIES-AT [1].

While the SN shape has only one divertor and the DN two distinct divertors, it does not necessarily follow that a design based on the DN shape is significantly more costly or complicated to build than a design based on the SN. For example, because pumping on the two inner divertor legs of a DN plasma has been shown to be largely superfluous in maintaining density control, there appears to be little need to install pumping hardware on the difficult to engineer high-field side of the core plasma [2]. Similarly, the DN requires much less heat protective armor at the inner divertor targets, because ≥80% of the power from the core flows into the scrape-off layer on the low-field side, which is magnetically disconnected from the inner divertor targets [3]. Finally, the plasma scrape-off width is much narrower and more quiescent on the high field side for DNs than SNs. These properties of DN justify tighter placement of the core plasma with the centerpost and a more effective use of space on the high field side than with the SN. When the physics and engineering considerations are taken together, the DN offers a greater promise of high performance (and economic viability) for DEMO, as evidenced by the implementation of DN for the ARIES series.

As with ITER, successful DEMO operation requires independent means of achieving and maintaining favorable performance characteristics in both plasma core and divertor. Three characteristics are particularly important. First, there is a need for active control over the fusion power output, as quantified, for example, by the neutron production rate R_N. Second, access into the H-mode energy confinement regime, as well as maintaining adequate confinement throughout the burn cycle, requires control over the power flow through the pedestal into the scrape-off layer P_{SEP}; for example, variation in radiated power in the core P_{RC} changes P_{SEP}, and that, in turn, can change the energy confinement of the main plasma. Because DEMO is an ignition device, control over P_{SEP} with additional auxiliary power is unlikely to be nearly as effective as it would be for ITER. Third, the divertor structure must be protected against excessive local heating from the power outflow from the main plasma, and this would necessitate a means of maintaining an acceptable level of divertor heating at the divertor targets, as quantified by the peak heat flux at the divertor targets Q_{TAR}. Whether DN or SN, a DEMO will likely have to consider the real possibility of thermal heat loads at the divertor target(s) large enough to result in unacceptable erosion and/or melting. Predicting the degree of this damage in a DEMO-class tokamak is made more challenging by (a) the uncertainty in the physics involved in predicting the width of the divertor heat flux profile at the divertor targets under DEMO conditions, (b) the uncertainty of how the physical properties of the divertor materials change over time due to the expected high levels of neutron flux bombardment that can decrease the effectiveness of heat removal from the divertor, and (c) the uncertainty in the number of highly energetic ions entering the scrape-off layer from the core plasma, striking the divertor targets, and leading to unacceptable rates of material sputtering. The "detached" radiating divertor solution for heat flux reduction in ITER discharges may not carryover to DEMO: (1) a DN is very hard to detach simultaneously at both upper and lower divertor targets, because the particle drifts, which can either help or hinder the detachment process are very different in each divertor, and (2) the presence of relatively cold "divertor MARFEs" in proximity to the main plasma could have a deleterious effect on energy confinement. Promising new divertor concepts, such as Snowflake [4] (Lawrence Livermore National Laboratory) and Super-X [5] (U. of Texas) , may be able to circumvent some of these problems, but they require fine control over both the magnetic shaping of the divertor and magnetic balance between divertors. For example, control over divertor magnetic balance is significant, because the heat flux width in the scrape-off layer may be very narrow at the midplane (e.g., ≤ 5 mm) and even small deviations from magnetic balance between divertors can result in a large proportion of the power flow in the scrape-off layer plasma being dumped into one or the other divertor. Because the coils that are needed to maintain the correct configuration may be susceptible to neutron damage over time in a DEMO environment, the coils that control magnetic balance and divertor shaping may not be able to be placed close enough to the X-point and divertor plasmas to be effective.

Contribution of ITER and future AT tokamaks to bridge the gap to DEMO

Although ITER is a sub-ignition tokamak, ITER will still provide DEMO with a significant amount of useful data in evaluating control algorithms for R_N, P_{SEP}, and Q_{TAR}. As mentioned above, control over P_{SEP} by direct means, i.e., by auxiliary power, is no longer possible in DEMO. Hence, the ITER program should devote effort to address reliable methods that can control P_{SEP} by indirect methods, at the same time that operators have independent control over R_N and Q_{TAR}.

More daunting is how best to handle the very high levels of loading power expected at the divertor targets of a DEMO device. A useful figure of merit for comparing the localized divertor heating between tokamaks is the ratio of total heating power (P) to major radius (R). For ITER, P/R ≈ 21 MW/m, and for an ARIES-AT DEMO, P/R ≈ 74 MW/m. While the values of the divertor heat flux depends on the details of how the divertor is configured and on the operating mode, their P/R values suggest that the localized divertor heating can be expected to be much greater for an ARIES-AT DEMO than for ITER using similar divertor designs. In addition, since ITER would be expected to generate roughly one-sixth of the neutron flux density on the divertor structure as an ARIES-AT DEMO, it is unclear whether information about the effect of neutron damage on materials used in the divertor could be applied to DEMO design in a timely manner. Finally, the SN shape in ITER provides little input into the myriad shaping and control issues involved in operating a DEMO in DN.

Research thrust possibility --- divertor program

While ITER will provide important information concerning control over several aspects of core physics that could be transferred to a DEMO tokamak device, it is apparent that the divertor heating issue may severely constrain the operation of a DEMO. Because of this, it is advisable to explore alternative divertor concepts, such as Snowflake and Super-X. To this end, the tokamak fusion community should consider a divertor program for DEMO that has near-term and long-term objectives.

<u>Shorter term (~10 – 15 years) objectives: Basic feasibility study</u>

1. Experimental shaping studies in present day tokamaks should be initiated to provide the relevant data needed for a preliminary analysis of these alternative divertor. Candidate tokamaks for this task, such as DIII-D. NSTX, K-STAR, EAST, etc., would need to have very good plasma shaping capability and may require modification of the vacuum vessel.
2. Dedicated effort would be needed to model the acquired data by using existing transport analysis tools, such as UEDGE and SOLPS5. Identify the critical concerns, e.g., the sensitivity of the divertor configuration to the location of the shaping coils.
3. If (2) is successful, extrapolate the candidate divertor concept to a DEMO, e.g., ARIES-AT. Determine the optimal shaping coil placement and assess survivability of these coils to the DEMO radioactive environment. After studying the tradeoffs, determine the optimal placement of the coils with regard to survivability of the coils. Determine whether or not the magnetic balance (*dRsep*) near DN can be controlled in finer detail than the power scrape-off width.

<u>Longer term (> 15 years) objective: Field testing the divertor concept</u>

No matter how well the above studies are executed, the plasma edge, scrape-off layer, and divertor physics of a DEMO tokamak is very likely to have significant uncertainty, e.g., in the power scrape-off width. In addition, the longer term effectiveness of the shaping coils in a highly radioactive environment adds further uncertainty to the prospects of a DEMO divertor that requires a high degree of shape control. In this regard, it may be prudent to bridge the gap between ITER and DEMO with an intermediate step that can be designed to address these concerns. One possibility is the Fusion Development Facility (FDF) tokamak [6]. The heat loading parameter of FDF is more representative of a DEMO ARIES-AT device. For example, $P/R \approx 43$ for FDF in its "2 MW/m^2" base case, which is roughly twice that of ITER or other present day tokamaks. In addition, FDF would be expected to generate neutron fluxes on the divertor structure more in-line with those anticipated for DEMO, thereby providing a realistic way of assessing how the divertor materials may change over time in a DEMO environment and how these changes might be manifested by changes in divertor performance.

Like DEMO, FDF is a DN tokamak. The success of the DN depends crucially on how well the shaping/control coils can maintain divertor balance. In both FDF and DEMO, protecting these coils from neutron damage is crucial. Even with shielding, these coils could not be as close to the X-points as in present day DN tokamaks, meaning that maintaining control over the magnetic balance would be a serious task for DEMO. FDF would encounter a similar set of problems and the solutions found for FDF would have direct application to DEMO. This would also include data on how neutron damage over time affects shaping and control.

Applying advanced divertor designs, such as Super-X and Snowflake to control divertor power exhaust would be relatively straightforward in DN devices, such as FDF or DEMO, because high field side pumping of the inner divertor legs is unnecessary and the power flow to the inner divertor targets is much smaller than that to the outer divertor targets. Each divertor concept uses a much larger wetted area at the outer divertor targets to spread out the heat loads. A test of either the Super-X and Snowflake or any other concept on FDF would be a real indicator of how effective these concepts would be in a DEMO device. A successful demonstration of power exhaust handling on FDF would remove one of the biggest uncertainties in designing a successful DEMO.

References

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