THE ADVANCED TOKAMAK PATH TO A COMPACT NET ELECTRIC FUSION PILOT PLANT


General Atomics,
San Diego, CA, USA.
Email: buttery@fusion.gat.com

J.M. PARK, J. CANIK
Oak Ridge National Laboratory,
Oak Ridge, TN, USA.

C.T. HOLCOMB
Lawrence Livermore National Laboratory,
Livermore, CA, USA.

Abstract

First of a kind physics based simulations project a compact net electric fusion pilot plant is possible at modest scale based on the advanced tokamak concept, and identify the key parameters for its optimization. These utilize a new integrated 1.5D core-edge approach for whole device modeling to predict plasma performance, by self-consistently applying transport, pedestal and current drive physics models to converge fully non-inductive stationary solutions without any significant free parameters. This contrasts with previous “systems code” approaches, where parameters can be simply chosen. This physics based approach has led to new insights and understanding of reactor optimization. In particular, results highlight a critical and very highly levering role of density, which increases fusion performance and self-driven ‘bootstrap currents’, thereby reducing current drive demands to enable high pressure solutions at the compact scale that make net electricity. Solutions at 6-T with ~4m major radius scale and 200MW net electricity are identified with margins and trade-offs possible in achievable parameters. Auxiliary current drive is projected from neutral beam and helicon ultra-high harmonic fast wave, though other advanced current drive approaches presently being developed also have potential. The resulting low recirculating power and double null configuration leads to a divertor heat flux challenge that is comparable to ITER, though reactor solutions may need to go further. Neutron wall loadings also appear tolerable. Strong H-mode access (factor >2 margin over the L-H transition scaling) and ITER-like heat fluxes are maintained with ~20-60% core radiation. The approach would benefit from high temperature superconductors, the higher fields of which increase performance margins, while their potential for demountability may facilitate a nuclear testing. However, solutions are possible with conventional superconductors. An advanced load sharing and reactive bucking approach in the main field and solenoid coils has been developed and would facilitate handling of mechanical stresses. The prospect of an affordable test device which could close the loop on net-electricity production and conduct essential nuclear materials and breeding studies is thus compelling, motivating research to prove the techniques projected here.

1. INTRODUCTION – THE ADVANCED TOKAMAK PATH

The Advanced Tokamak (‘AT’) concept [1, 2, 3, 4, 5, 6, 7] provides one of the most promising approaches to meet the challenge of sustained fusion energy. The essence of the approach is to modify the plasma configuration to obtain favorable confinement and stability properties, consistent with discharge sustainment. A key aspect is to replace inductive current with a combination of ‘bootstrap’ current [8], naturally arising from orbit effects at high pressure gradients, and auxiliary current drive such as from radiofrequency heating [9]. The concept exploits natural synergies that emerge between plasma stability, turbulence and the current and pressure profiles that are associated with fully or highly non-inductive operation. For instance [10], a broad current profile and strong plasma shaping lead to high β stability through a dissipative interaction with the device wall. Such profiles, with low or negative global magnetic shear, but high local shear, naturally stabilize turbulence, and lead to strong off-axis bootstrap currents that sustain the broad current profile. Because of such synergies, the AT is generally seen as the primary path to fusion energy amongst international partners engaged in the fusion endeavor, driving reactor concepts and associated scientific and engineering programs in each. Not least it is the approach for ITER’s steady state research toward fusion energy [11]. In Europe this serves as the basis on which various DEMO facilities are proposed to bridge the gap to commercial Fusion Power Plant (FPP) [12]. Options include a large, barely advanced EU-DEMO baseline with high recirculating power [13] and the somewhat advanced ‘stepladder’ approach [14, 15] with longer pulse length and more optimism in technology and physics assumptions. Japan proposes a more advanced SlimCS device [16], while South Korea targets its KSTAR program on the K-DEMO device [17, 18]. In the United States, the original ARIES design [9] has been updated with various ‘Advanced and Conservative Tokamak’ (ACT) versions [10], while proposals are also made for more compact lower power designs with the ARC [20] and ST Pilot Plant [21] facilities. Table 1 summarizes key parameters of these proposed devices.
TABLE 1: SUMMARY OF TOKAMAK FUSION POWER PLANT PARAMETERS.

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>R (m)</td>
<td>5.2</td>
<td>6.25</td>
<td>5.5</td>
<td>9.75</td>
<td>6.8</td>
<td>9</td>
<td>7.85</td>
</tr>
<tr>
<td>a (m)</td>
<td>1.25</td>
<td>1.56</td>
<td>2.1</td>
<td>2.44</td>
<td>2.1</td>
<td>3</td>
<td>2.5</td>
</tr>
<tr>
<td>B_R (T)</td>
<td>5.6</td>
<td>6</td>
<td>6</td>
<td>8.75</td>
<td>7.4</td>
<td>5.2</td>
<td>5.6</td>
</tr>
<tr>
<td>I_T (MA)</td>
<td>13</td>
<td>11</td>
<td>16.7</td>
<td>14</td>
<td>17</td>
<td>20</td>
<td>14</td>
</tr>
<tr>
<td>B_N</td>
<td>5.4</td>
<td>5.6</td>
<td>4.3</td>
<td>2.6</td>
<td>3.1</td>
<td>2.6</td>
<td>3.5</td>
</tr>
<tr>
<td>f_{Aux}</td>
<td>0.91</td>
<td>0.91</td>
<td>0.75</td>
<td>0.77</td>
<td>0.77</td>
<td>0.34</td>
<td>0.62</td>
</tr>
<tr>
<td>H_{Aux}</td>
<td>H_{Aux}=2</td>
<td>1.65</td>
<td>1.3</td>
<td>1.22</td>
<td>1.5</td>
<td>1.5</td>
<td>1.2</td>
</tr>
<tr>
<td>P_{Aux} (MW)</td>
<td>35</td>
<td>43</td>
<td>60-100</td>
<td>106</td>
<td>120</td>
<td>50</td>
<td>115</td>
</tr>
<tr>
<td>P_{EL} (MW)</td>
<td>1719</td>
<td>1800</td>
<td>2950</td>
<td>2600</td>
<td>2870</td>
<td>1800</td>
<td>1960</td>
</tr>
<tr>
<td>N_{0} (MW/m²)</td>
<td>3.2</td>
<td>2.45</td>
<td>3</td>
<td>1.5</td>
<td>2.3</td>
<td>0.9</td>
<td>1.2</td>
</tr>
</tbody>
</table>

These devices take advantage of the basic principle of utilizing high levels of bootstrap current (typically ~50-90%) and non-inductive current drive to sustain the plasma indefinitely (‘steady state’) or for long periods. However, their designs are based on systems code approaches, without a model for the pedestal, and without self-consistently solving for the transport, current drive and equilibrium, while plasma energy is simply chosen as desired. Further, many devices have diverged from the true AT approach embodied in the original ARIES-AT concept, proposing instead operation with lower B_N (=β_nAB/I_T) and bootstrap fraction, requiring more auxiliary current drive – thus driving up required fusion power and recirculating power (and heat and neutron loads) or other parameters, such as device scale, required current drive efficiency and/or H_{Aux}. By taking advantage of the natural benefits of the AT approach, more efficient and less demanding devices can be conceived. The potential to do so is highlighted by applying the simple analytic model of [14], where required reactor size is based on relations for fusion performance (P_{fus} ~ B_N^2 B_L R^3 / A^4 q^2), density limit, current drive, and net electricity in terms of field (B), pressure (B_N), safety factor (q), major radius (R), and thermodynamic and current drive efficiencies (η_{fus}, η_{CD}). One sees (Fig. 1) that required size falls rapidly with increasing B_N, B, and decreasing current drive need (‘half CD’ in figure) or net electric requirement. It is therefore important to scope out what is realistically possible in terms of physics of the core plasma, and indeed understand how this can be optimized to reduce demands on systems and aggressive technology assumptions.

The scale of many of the above devices is also driven by the desire to approach power plant performance levels, where low cost of electricity (COE) is projected to require electricity generation at the GW level (GWe) or higher [12]. However, a first electricity generating plant will inevitably not be low COE, due to its experimental nature and the investments required in its new technologies. Thus what is needed is a facility that can prove out the requirements and approach – a pilot plant – to establish the basis for successor low COE devices. Critical is to show that the energy loop can be closed with net electricity generation (somewhat arbitrarily to 200MWe here). In addition it is important to test and optimize nuclear materials and breeding with high neutron fluence. For efficiency, and to maximize progress to fusion energy, it is attractive to combine these missions into a single device, with phased operation to first proof the energy principle, prior to nuclear testing. It is also highly desirable to minimize capital cost and thus scale of the device, to remove barriers to funding. This requires careful physics based and engineering analysis.

In this paper we have conducted the first integrated physics simulations and further physics and engineering analysis, to resolve how to achieve sufficient performance to sustain 200MW net electricity in stationary ‘steady state’ conditions in a compact scale device. As described in §2, this uses tools developed and validated in the DIII-D program to project plasma performance and find self-consistent plasma-pedestal-transport-current drive solutions. The goal is not to pick out a single existence point, but rather to understand the optimization and benefits of various parameters in physics terms. Thus a range of solutions are obtained (e.g. Table 2), which not only confirm the benefits (and accessibility) of high β – high bootstrap fraction solutions, but also, interestingly, the importance of high density in achieving sufficient fusion performance and confinement, which can otherwise limit
such compact solutions. Higher field is also levering to pedestal and core transport, reducing required fusion power and plasma current, though it is noted that solutions exist with conventional (6T) and high temperature superconductors (HTS; 7T or higher). Thermodynamic and current drive efficiency are important in reducing recirculating power and thus required fusion performance, current or device scale. It is noted that many of these sensitivities are highlighted in a compact device, where the generation of net electricity relies on efficient confinement and power recycling. These trade-offs are explained in §3, with a discussion of exhaust handling in §4 and engineering approach in §5, before discussing conclusions and research needs motivated in §6.

2. ANALYSIS TOOLS

This study makes use of two principle analysis tools. First the FASTRAN code suite [22, 23] provides an efficient and robust iterative solution procedure to find a steady-state solution (d/dt~0) of core transport, self-consistent with external heating/current drive, MHD equilibrium, ideal MHD stability, and edge pedestal, built upon a modern integrated modeling framework, Integrated Plasma Simulator (IPS). This tool combines state of the art simulation of the theory-based core transport model, TGLF [24] for all transport channels (particle, energy, and momentum), and EPED for edge pedestal [25] to provide the boundary condition of the core transport. These are integrated with the well-established modeling of MHD equilibrium (EFIT), ideal kink stability (DCON, GATO) and external heating/current drive (NUBEAM, TORAY, GENRAY) to yield predictive capability for steady-state operating scenario development without significant reliance on any free input parameters. This contrasts with previous ‘systems code’ approaches that adjust performance to desired levels, instead predicting behavior based on the latest physics model to project key properties such as energy confinement and current drive. This suite has been extensively validated on DIII-D data [23, and references therein], used to project to ITER [26], and is used to design future DIII-D operating scenarios and upgrades [27]. This therefore represents the present state of the art, integrating the best available models for device projection. Nevertheless, further validation is important to constrain underlying models in more reactor-like configurations (e.g. high βn, broader current profiles, low rotation, T_e-T_i) – this is part of the future research advocated to enable the design of and a decision on an AT reactor. However, as demonstrated in this paper, use of this physics based predictive approach offers considerable insights over a traditional rotation, T

developments without

example compact AT optimizations found with FASTRAN simulations

<table>
<thead>
<tr>
<th>Case:</th>
<th>6T</th>
<th>6T</th>
<th>7T</th>
<th>7T</th>
<th>7T</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parameter:</td>
<td>( f_{GW}^{\text{ped}} = 1 )</td>
<td>( \eta_{TH} = 0.33 )</td>
<td>( \eta_{CD} = 0.25 )</td>
<td>( f_{GW}^{\text{ped}} = 1 )</td>
<td>( \eta_{TH} = 0.33 )</td>
</tr>
<tr>
<td>Current MA</td>
<td>9.5</td>
<td>11</td>
<td>8.2</td>
<td>9.5</td>
<td>9.6</td>
</tr>
<tr>
<td>( q_0 )</td>
<td>5.7</td>
<td>4</td>
<td>7.1</td>
<td>6.2</td>
<td>6.1</td>
</tr>
<tr>
<td>( \beta_n )</td>
<td>4.3</td>
<td>4</td>
<td>3.5</td>
<td>3.4</td>
<td>4</td>
</tr>
<tr>
<td>( H_m )</td>
<td>1.3</td>
<td>1.3</td>
<td>1.5</td>
<td>1.3</td>
<td>1.4</td>
</tr>
<tr>
<td>Q</td>
<td>10</td>
<td>17</td>
<td>13</td>
<td>9.5</td>
<td>17</td>
</tr>
<tr>
<td>( P_{\text{MW,CD}} )</td>
<td>74</td>
<td>73</td>
<td>51</td>
<td>82</td>
<td>115</td>
</tr>
<tr>
<td>( P_{\text{ne}} )</td>
<td>750</td>
<td>1280</td>
<td>640</td>
<td>775</td>
<td>1095</td>
</tr>
<tr>
<td>Neutron MW/m²</td>
<td>1.8</td>
<td>4</td>
<td>1.7</td>
<td>2.1</td>
<td>3</td>
</tr>
</tbody>
</table>

* All cases with 200MW, \( R=4m, R/a=3.1 \), with \( f_{GW}^{\text{ped}} = 1 \), and \( \eta_{TH} = \eta_{CD} = 0.4 \) except where otherwise stated.
3. CORE PERFORMANCE OPTIMIZATION

Initial simulations commenced with an intermediate scale 5m radius device, conservative current drive efficiencies (as used in EU-DEMO, with $\eta_{b}=0.33$, and $\eta_{CD}=0.25$). FASTRAN takes pedestal density as an input constraint in EPED, and uses TGLF to predict core transport and profiles self-consistently. Here we initially set a pedestal density normalized to Greenwald fraction, $f_{GW}$, to 0.85. However, as shown in Fig 2 blue curves this yielded poor performance due to confinement limits, as the device required considerable auxiliary power, struggling to reach high $\beta_N$ with net electricity production. Nevertheless, with 90MW of auxiliary heating a fully non-inductive point is realized ($f_{IW}$~1).

To explore the optimization, the parameter space was explored in toroidal field and $\beta_N$. In Fig. 3 are plotted solutions with each point adjusted in current and heating power (which therefore become predictions) to achieve the given $\beta_N$ and be fully non-inductive. Starting from the $P_{H/CD}$ panel it is observed that increased $\beta_N$ requires more auxiliary heating power. This in turn requires higher fusion power, which is achieved by raising plasma current (thus improving energy confinement). As a result of the higher current needed at high $\beta_N$, the bootstrap fraction actually falls as $\beta_N$ is raised! Further the high levels of fusion power lead to high neutron wall loading, likely to damage wall materials and also divertor heat loads (not shown). It is interesting to note that higher toroidal field also offered benefit in raising net electric production, due to increased pedestal and core performance. However, clearly better confinement is needed if we are to avoid raising current.

Returning to Fig. 2, additional scans were undertaken (green and red curves) which identified density as a key levering parameter. Increased density raises fusion performance and bootstrap fraction (which depends more strongly on density gradients than temperature gradients for a given pressure profile). The increased fusion power raises gross electricity production, while higher bootstrap fraction lowers current drive demands, leading to substantial gains in net electric power. As density is raised, fully non-inductive regimes become possible with lower auxiliary heating power and higher $\beta_S$. Even a modest rise in density, raising $f_{GW}$ 85% to 93 or 100% has dramatic effects on fusion performance (corresponding to line average Greenwald density fractions of 110-130%, due to the density peaking predicted). The benefits are more clearly observed in Fig. 4 where scans at fixed $\beta_N$ show a rapid decrease in required auxiliary power to maintain fully non-inductive conditions. These density levels are considered ambitious but tractable research challenges, given recent progress with techniques such as super-H mode, which show a path to raise pedestal with density [29].

Based on these promising results with higher density, a more compact device was attempted with 4m radius and 6T field (Fig. 5). However, the more compact size required higher current to reach 200MWe, driving $q_{95}$ down to ~4, raising disruption concerns as well as neutron wall loading. To progress further, less conservative thermal and current efficiencies are required than the relatively conservative ‘present-day physics’ assumptions of EU-DEMO. As shown bottom right, higher efficiencies ($\eta_{b}$ or $\eta_{CD}$) are greatly leveraging to net electric performance, suggesting improvements in wall loading and $q_{95}$ would be possible if one maintained 200MWe and lower current. Comparing
with other reactor concepts cited, $\eta_B=\eta_{CD}=0.4$ seems reasonable, though does represent a further research challenge to develop the required current drive technology and efficiency.

Building on these insights a final exploration of parameter space is presented in Fig 6. Here current and density are prescribed, while auxiliary heating and current drive power is allowed to float to ensure the solution stays fully non-inductive across the parameter space. Toroidal field is held constant at a now elevated value of 7T (which requires more advanced high temperature superconductor technology). These results highlight the synergy of density and high $\beta_N$, with higher $f_{GW}^{95}$ increasing fusion performance and net electric power, or allowing lower plasma current, auxiliary power and neutron loading (and implicitly divertor heat flux challenge) at constant net electric power. Higher $\beta_N$ also plays a strongly levering role on net electric power, as fusion performance rises proportionately. Bootstrap fraction increases substantially with density, but only rises weakly with $\beta_N$ at constant $f_{GW}^{95}$, as higher current (with lower $q_95$) is required to raise fusion performance to sustain that $\beta_N$. However, increased demands for auxiliary heating and current drive power at higher $\beta_N$ can be ameliorated by higher density operation.

As with all cases discussed in this section FASTRAN predicts a self-consistent solution consistent with bootstrap, heating and current drive models. The 7T $f_{GW}^{95}=1$ case is shown in Fig. 7 with 90% bootstrap fraction augmented by 750kV off-axis neutral beams and 1.2GHz helicon ultra high harmonic fast wave. 230GHz top launch ECH is also projected to be effective. While realistic current drive models are assumed, current drive efficiencies are set, as discussed above, posing a further research challenge to develop these technologies.

To summarize, these results demonstrate the levering roles of (i) higher $\beta_N$, (ii) higher density, (iii) higher current drive efficiency, and (iv) higher toroidal field for developing a compact ~200MWt device. Not discussed, but also important, were the benefits of strong shaping (elongation of 2 and triangularity of 0.6 were used here). Overall we see a family of solutions (e.g. Table 2) with improvements in one parameter providing margin in others, enabling a balanced risk that can avoid key extremes in parameters such as plasma current and low $q_95$, which might otherwise raise the specter of disruption risks. An important point is that while these solutions represent projections using the best presently available physics models, they nevertheless represent research challenges to validate the models for these advanced reactor profiles, and to demonstrate the required performance and parameters (notably the high pedestal Greenwald fraction and $\beta_N$), stability, control, density, current drive efficiencies and high field technologies. Such research can refine the models and show what is possible, to determine the optimal point based on the principles identified here.

4. MITIGATING THE PLASMA EXHAUST – DIVERTOR CHALLENGE METRICS

Preventing damage from the hot plasma exhaust represents a particular challenge for a compact device that must operate quasi-continuously. Not only should heat loads be controlled, but incident particle energies must also be reduced to virtually eliminate erosion, and ensure sufficient lifetime of plasma facing components. This can be
addressed by use of radiative impurity mantles within the separatrix and (different impurity) radiative divertors, leading to so-called ‘detached’ operation. The challenges for these techniques are to maintain sufficient power throughput to sustain the pedestal, while also avoiding divertor radiators reaching and collapsing the core. To characterize this challenge we consider divertor heat flux metrics and compare with ITER.

Considering power into the scrape-off layer, \( P_{\text{SOL}} = P_{\alpha \text{heam}} + P_{\text{H\&CD}} - P_{\text{brems/synch}} \), this leads to a poloidal heat flux that is dependent on scrape off layer width, which we take from the Eich scaling [30], \( q_\theta = 1.35B_0^{0.92}e^{-0.42P_{\text{SOL}}^{0.02}/R_0^{0.04}} \), to obtain \( q_\theta \sim P_{\text{SOL}}B_\theta/R_N \), where \( N \) is number of divertors (1 or 2). Mapping to parallel (to field line) heat flux cancels out this \( B_\theta \) dependence to yield \( q_{||} \sim P_{\text{SOL}}B_T/R_N \). If this power reaches divertor tiles the incident heat flux can be further mitigated by angling field lines with respect to those tiles, reaching a practical limit due to edges and shadowing of about 1 degree. Thus the tile heat load ‘divertor challenge metric’ is characterized by \( q_{||} \), and this can be mitigated by radiation and tile angling. However, if, as seems likely, a radiative detached divertor is required, this will modify the metric. Detachment requires sufficient radiation along the flux tube to cool the plasma to low temperature, and this depends not only on density and impurities but also connection length (with an exponent \( \sim 0.5-1 \) dependent on divertor physics regime). As connection length increases with \( B_T/B_\theta \), this reintroduces some \( B_\theta \) dependence in the divertor challenge metric. To bracket these possibilities we consider \( q_{||} \) and \( q_\theta \) as metric extremes, adjusting core radiation (through an assumed radiative impurity mantle) to match ITER values for each of these metrics in turn, based on the 7T \( f_{\text{med}}^{\text{IT}} = 1 \) case of Table 2. Scanning a range of hypothetical \( H_{\text{med}} \) values with GASC, it is found (Fig. 8) that ITER levels of divertor heat flux are readily achievable with modest amounts (20-60%) of core radiation for \( H_{\text{med}} \sim 1.3 \) or higher. Further, \( P_{\text{SOL}} \) remains well above empirical thresholds where H-mode quality is thought to be at risk (typically if \( P_{\text{SOL}}/P_{\text{LH}} < 2 \)). Indeed, it is found that core radiation fractions up to 80% appear possible while maintaining \( P_{\text{SOL}}/P_{\text{LH}} > 2 \). These encouraging trends arise from two principle elements: (i) the addition of a second divertor compared to ITER’s one, and (ii) the efficiency of the core solution, where fusion and recirculating power have been minimized. However, it is important to apply some qualifiers to this result. Firstly the factor 2 benefit in divertors may not be fully realized, as double null operation typically does not evenly distribute power between inner and outer divertors. Also, the use of high levels of core radiators might be expected to degrade pedestal and core performance somewhat, with effects from fuel ion dilution [31], and this has not yet been self-consistently fed back into the core solution of §3. Such considerations may be equivalent to lowering \( H_{\text{med}} \) in Fig 8. Finally, for a continuously operating pilot plant, it is likely that a level of radiative dissipation and detachment will need to be higher than ITER’s (the short pulse of ITER being able to accept some divertor erosion). These aspects strongly motivate further research to understand how to achieve a more dissipative (and dense) divertor, while containing the neutrals and impurities required within the divertor region, to avoid excessive deterioration of the pedestal and the core.

5. ENGINEERING CONSIDERATIONS

The GA systems code (GASC) explores the engineering design of the device using a set of algorithms as described in §3, based on [28]. These have been extended to evaluate the divertor heat load and L-H transition issues discussed above. Additionally, the code includes a newly developed High Temperature Superconducting (HTS) coil model [12], and has implemented more advanced stress handling techniques. The basic geometry of the coil set, based on relations for resistivity, heating and inductance, is set out in Fig. 9, where a 1m blanket provides sufficient shielding for the toroidal field (TF) coil and central solenoid (CS or OH). As set out in Table 2, FASTRAN finds existence points at 6T and 7T. 6T solutions could be addressed with conventional superconducting magnets. But it is attractive to move to High Temperature Superconductors (HTS) technology, as also proposed for ARC [20]. This not only improves device efficiency and margins, by enabling 7T (or higher) solutions. HTS also offer the potential for demountability (as the coils could withstand small temperature rises at joints). This technology is not yet proven, but if developed, would greatly facilitate the nuclear testing and breeding program, with the potential to change out walls and major components more rapidly. For instance, a vertical change out approach such as proposed in [32] may be attractive.
With the high fields and compactness of this device, mechanical stresses from electromagnetic loads will be a critical issue, with maximum fields in the magnets, \( B_{\text{max}} \sim 16-18 \text{T} \). For center post stresses we use a bucking cylinder solution to distribute the toroidal field (TF) loads over the CS and a central plug. This approach essentially engineers connections to support (‘bucks’) the TF and CS off of each other, with forces significantly cancelling when both coils are energized, and further support from a central plug. Assuming shear is transmitted between magnet interfaces, this configuration reduces the TF peak stress by a factor of \(~2\) over the free-standing TF/CS system. Energizing the CS reduces the overall stresses in the center post further, making the TF current the primary limiting load condition. This brings loads down to tolerable level, but it should be pointed out that this is a relatively advanced engineering solution that will require extensive R&D to implement. Additionally, a new quench protection model for the HTS solution has being implemented in GASC and preliminary results indicate machine size within the 4m range are achievable with additional copper on the TF and CS for quench protection below reasonable magnet thermal limits. Considering other design aspects, like, DIII-D, PF coils would be placed within the TF, to enable the device to achieve the strong shaping (elongation of 2 and triangularity of 0.6) that is instrumental in achieving an effective high performance core configuration. In addition, a copper coil, which requires less shielding could be placed closer to the plasma, to provide vertical control.

6. CONCLUSIONS AND IMPLICATIONS FOR FUTURE WORK

These studies reaffirm the advanced tokamak concept for fusion energy, using integrated predictive physics models for the first time to project reactor performance and self-consistent plasma solutions, that demonstrate that a net electric mission may be viable in a compact scale device. This is achieved by fully applying the principles of the advanced tokamak approach to increase \( \beta_n \) and bootstrap fraction in the presence of strong shaping, so that a predominantly self-sustaining plasma is established, developing a configuration in which favorable bootstrap, current drive, transport and stability properties naturally align. Interestingly, it is found that high density is an important element in achieving this solution with sufficient bootstrap and fusion performance for adequate self-heating and low auxiliary power requirements. Similarly, higher field is beneficial. These approaches reduce necessary current drive and heating, and thus required recirculating power, fusion performance, current, neutrons, and heat load, while offering improved stability in the form of higher safety factor. This enables the AT reactor concept to be scaled back to lower power and a more compact size. This offers the prospect of an affordable device that can prove the principle of fusion energy production, and sustain plasmas in steady state for a nuclear materials testing and breeding mission, on a path to a successor low cost of electricity facility.

Importantly, these solutions use integrated full physics simulations to \textit{predict} performance using the latest state of the art models, self-consistently, \textit{without free parameters}. While the solutions found do still point to considerable research challenges, discussed below, the approach identifies what is possible consistent with present physics understanding. This contrasts with other reactor concepts where key parameters such as plasma energy or stability are simply chosen to reach desired performance; it is important that this physics based approach be applied to such concepts so that their limits and challenges can be identified, and their specification optimized.

As highlighted in this paper, the physics and technical basis is not yet ready to proceed with a Compact AT pilot plant (or indeed, we would argue, any fusion power plant concept). Rather, these concepts and insights serve as motivation to guide the necessary research and technology development (the need for which is largely shared amongst all reactor concepts). Thus one foresees an “enabling research” program to develop this scientific and technological basis, which, with sufficient progress, will lead to a point where a solution simply becomes apparent, thus enabling a decision to proceed and parameters to be set. Design and construction would follow. Once operational, a phased exploitation is foreseen, with initial short pulse demonstrations of the required performance and efficiencies of various systems. This could include net electricity generation, thus meeting an early proof of principle for the approach. The facility would then likely be upgraded for an extended pulse nuclear testing and breeding mission, involving testing and possible changes of materials and components:

```
Enabling Research  Decision  Design  Build  Net electric  Nuclear testing
```

This concept motivates further work including: (i) the construction and initial phases of exploitation of ITER, to learn the practical lessons of reactor design and operation, critical to being able to pursue this future device, (ii) development of a high performance, high density and \( \beta \) steady state core physics basis, with sufficient control of transients and safe termination, (iii) resolution of an advanced divertor solution capable of reducing erosion and heat flux, (iv) validation of suitable materials, particularly for plasma facing components that are compatible with core performance and the harsh nuclear environment, (v) demonstration of effective current drive technologies that can withstand reactor conditions and achieve high efficiency, (vi) development of high temperature demountable superconductors, (v) reactor design and engineering studies, including concepts for tritium breeding.
Thus a compact AT pilot plant, returning to the true principles of the advanced tokamak approach, provides important motivation for fusion research and understanding of the optimization of the approach, while offering compelling prospects that could speed the path to fusion energy. Such a path should be the focus of world research in coming years, alongside the critical science and technology pioneered in and flowing from ITER.

ACKNOWLEDGEMENTS

This material is based upon work supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences, using the DIII-D National Fusion Facility, a DOE Office of Science user facility, under Awards DE-FC02-04ER54698, DE-AC05-00OR22725 and DE-AC52-07NA27344. DIII-D data shown in this paper can be obtained in digital format by following the links at https://fusion.gat.com/global/D3D_DMP.

Disclaimer: This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of its employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

REFERENCES