The High-Field Path to Practical Fusion Energy

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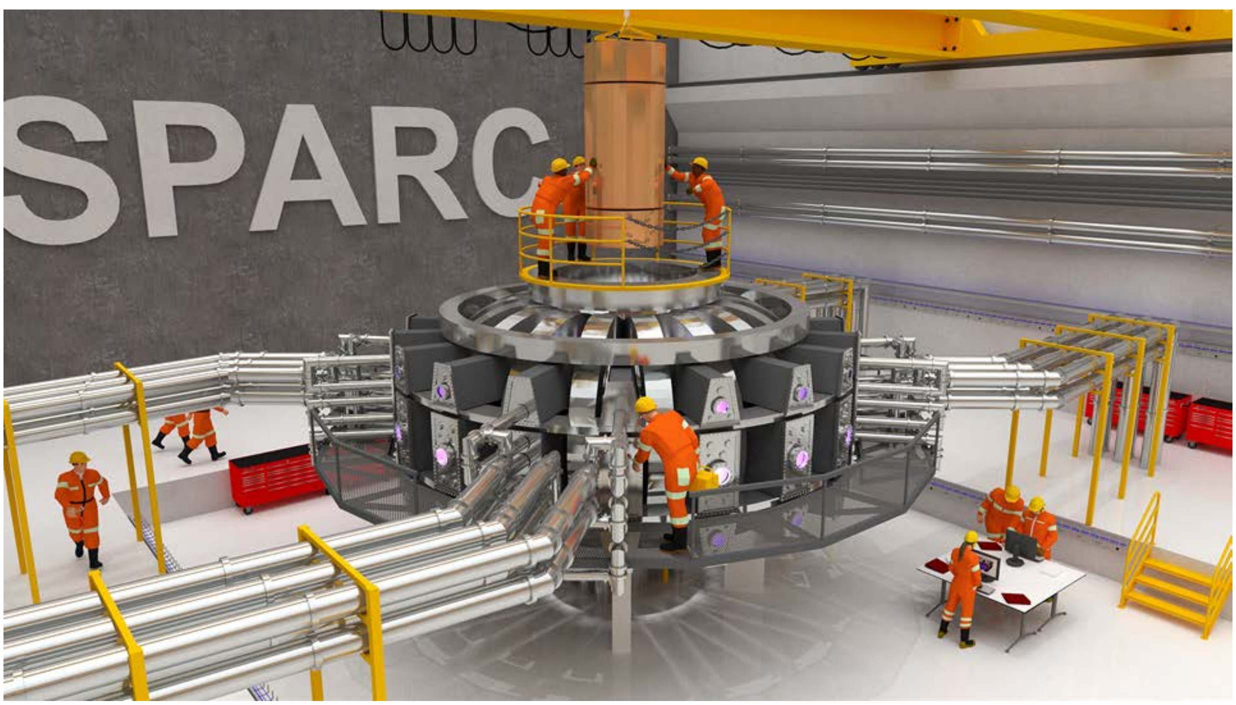
This white paper outlines a vision and describes a coherent roadmap to achieve practical fusion energy in less time and at less expense than the current pathway. The key technologies that enable this path are high-field, high-temperature superconducting magnets, a molten salt liquid blanket, high-field-side lower hybrid current drive and a long-legged, advanced divertor. We argue that successful development and integration of these innovative and synergistic technologies would dramatically change the outlook for fusion, providing a real opportunity to positively impact global climate change and to reverse the loss of U.S. leadership in fusion research suffered in recent decades.



An illustration of the SPARC tokamak – a compact, net-energy tokamak that would be a key step on the high field path to fusion. Credit: Ken Filar
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1. Executive summary

We propose a faster, lower cost development path for fusion energy by exploiting High-Temperature Superconductors (HTS) and several other innovative technologies. The need for reliable, scalable carbon-free energy sources is clear, but the question before us is whether fusion can contribute in time to impact climate change. The current international roadmaps going from ITER to an even larger DEMO, perhaps through an additional FNSF (Fusion Nuclear Science Facility) or CTF (Component Test Facility) device, do not promise electricity on the grid in less than 50 years. We note that: "At some point delay is equivalent to failure, as stakeholders in government and industry conclude that no solution will be forthcoming [1]".

Fortunately, we think that the basis for a breakthrough is here. HTS has reached a level of industrial maturity sufficient for high-field fusion magnets. At the same time, the physics basis of the tokamak is well-developed and well-documented [2,3] - the world's fusion community has expressed confidence in that basis through its willingness to invest the equivalent of $40-50B in ITER. Harnessing tokamak physics with the new high-field superconducting technology would allow drastic reduction in the size of a tokamak fusion system, accelerate the pace and decrease the cost for each step, and lead to a much more attractive and economically viable power plant.

The plan outlined in this document would exploit this new magnet technology and blaze a new path, allowing the US to take the lead in development of fusion magnets, blanket technology, construction and maintenance schemes, plasma-material interactions and plasma sustainment. By ensuring US leadership, this path would

- Attract collaborations from the world program
- Reinvigorate a declining research enterprise in all of our institutions
- Create excitement and enhance our ability to attract the best students, scientists & engineers
- Reignite interest from policy makers and the public
- Provide a path for fruitful public-private partnerships.

Scientific and Engineering opportunities:

The advantages of high-field for magnetic confinement are clear [4]. The "size" of a plasma is properly measured in gyro-radii; doubling the field leads to a proportional decrease in the linear size for fixed performance and so roughly an order of magnitude drop in the reactor volume and weight. Neutron power loading, which is a critical metric for fusion power economics, goes like B^4 for fixed normalized plasma pressure, β. At the time ITER was designed (1995-2005), the best superconductor available was Nb_3Sn which effectively limited the magnetic field at the coil to ~12T with field at the plasma center at less than 6T. Alternate designs for compact burning plasma experiments using high-field copper magnets were also considered by the U.S. program

[5,6] but were deprecated as "dead-ends" for fusion development. What has changed is the availability of REBCO (rare-earth barium copper oxide), a type of superconductor with critical temperatures, critical magnetic fields and critical current densities each an order of magnitude higher than Nb_3Sn. The new superconducting materials are deposited as a thin layer on a steel substrate producing strong, flexible tapes, ideal for winding magnets (see section 2).

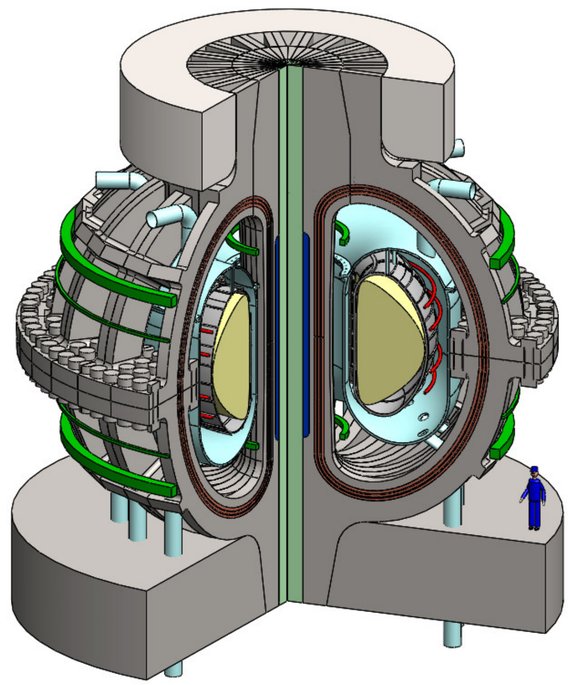


Figure 1. ARC is a concept for a fusion pilot plant that could produce 500 MW of fusion power and over 200 MW of electricity in a device roughly the size of the JET tokamak [6]

Another distinct advantage arising from the use of high-temperature superconductors is that the higher specific heat of materials compared to operation at 4°K, combined with additional operational margin, should allow construction of superconducting coils with demountable joints (as described in section 4). This concept could revolutionize construction and maintenance of a toroidal confinement reactor and is synergistic with an additional innovation – the provision of all blanket functions through immersion of the fusion core in a bath of molten salt (see section 5). This blanket concept improves shielding, by removing all cracks and gaps; dramatically reduces the volume of solid material exposed to high neutron flux (and thus the quantity of radioactive waste by about 50x) and further simplifies maintenance. The molten salt would be drained and the toroidal magnet disassembled to allow all core components to be accessible by a vertical lift. In doing so, this concept enables a development path for fusion materials in which replaceable cores are installed successively as part of the R&D program. All of these features have been captured in the ARC fusion pilot plant design concept [7].

The plasma scenarios required for steady-state ARC operation have already been demonstrated, though for short pulse only [8]. To achieve steady-state, ARC could take advantage of the improved Lower-Hybrid Current Drive (LHCD) efficiency predicted for higher field, especially if combined with an innovative high-field-side launch [9, and section 6]. Modeling suggests that 25% of the current could be driven with only 5% of the total power, allowing good control at high Q and reasonable requirements for bootstrap current [7]. Placing the LHCD launchers on the high-field side also eases plasma-materials interactions which challenge low-field-side structures [10]. Heat and particle loads at the divertor would be a generic challenge in ARC as they are for all practical fusion schemes, since economics dictate a high average neutron wall loading and thus an overall high fusion power density. The community consensus is that finding a solution to this problem will require a dedicated divertor test tokamak, perhaps along the lines of the proposed ADX experiment [11, and section 7]. Simulations have suggested that its long-legged divertor and x-point target greatly expands the operational range for a robust detached divertor solution [12]. Recent studies have further suggested that detachment is easier to obtain at high magnetic fields, requiring a lower impurity fraction [13]. The ARC concept is compatible with an advanced divertor of this type [14].

Disruptions might seem to be a particular problem in a high-field device, due to the magnitude of **JxB**. However, we note that the I_P x B_T forces for ARC is about the same as for much larger machines with conventional fields and performance, while lever arms in ARC would be shorter, reducing stress. Smaller size allows more rapid response for disruption mitigation techniques, but quench times are likely to be faster as well. Runaway electron growth, which depends exponentially on the total plasma current would be much smaller. Overall, one might conclude that compact, high field reactors are somewhat more likely to survive disruptions, but their lifetimes are still threatened. Thus, the big advantage for the high-field approach is the ability to operate at high fusion power while staying far away from well-known operational limits, significantly reducing the probability of disruption. With fusion gain scaling like B^3 and fusion power density like B^4, ample headroom is available in the design and operating space. Finally, while ELMs represent a transient heat load that is particularly hard to handle, we note that the C-Mod experiment, which ran H-modes at higher field than any other device in the world, never experienced large ELMs (or other significant MHD). It is likely that operating at lower β reduced the ELM drive to the point where other, less damaging transport mechanisms could dominate.

While it is an appealing concept, we are not yet ready today, to design or build an ARC-class device. In section 8 we lay out a development path that includes the elements described in this white paper which would provide the necessary scientific and engineering R&D required before proceeding to the pilot plant. A key element in that roadmap is a compact, high-field, net energy experiment that we are calling SPARC. SPARC would be the size of ASDEX-Upgrade or DIII-D, but with HTS coils, would achieve a toroidal field in the plasma center of 12T and plasma current of 7.5 MA. Using standard (ITER) physics rules, the device is predicted to produce 50-100 MW of fusion power, achieving a Q ~ 3-5. Such a device would demonstrate net energy gain and burning plasma control and would validate the promise of high-field devices built with the new superconducting technology. It fits into an overall strategy driven by the principle of retiring risks early, at the lowest possible cost. More details on SPARC can be found in section 3.

2. High Temperature Superconductors

An ideal technology for fusion magnets

Figure 2.1 REBCO superconductors are available in thin, flexible tapes. The superconducting material is deposited in a thin layer on a strong steel substrate.

Magnet systems are the ultimate enabling technology for magnetic confinement fusion devices. All design concepts for power producing commercial fusion reactors rely on superconducting magnets for efficient and reliable production of the magnetic fields. HTS represents a new, game changing opportunity that could significantly advance the economic and technical status of magnetic confinement physics experiments and fusion reactors. It could revolutionize the design of magnetic fusion devices leading to very high performance in compact devices with simpler maintenance methods and enhanced reliability. This new technology could lead to significant acceleration of fusion energy development [4].

The advantages of HTS are that they can operate at very high magnetic field, high cryogenic temperature, high current densities, and under larger mechanical stresses and strains compared to existing low-temperature superconductors (LTS). The expanded volume of operating space in these critical parameters opens options for magnet design in a manner never previously available in fusion magnet technology. We note that the advantages of HTS magnets would apply to almost any type of magnetic confinement or plasma physics device including stellarators.

Advantages of HTS for a Fusion Device:

The B^3 dependence of fusion gain and the B^4 dependence of fusion power density allows reactor level performance in much smaller devices and may be crucial for fusion's eventual commercial realization. The maximum field at the coil (which is limited by achievable current density in the superconductor) has been a critical input for the design of magnetic fusion devices [2,3,15] as found in systems codes and studies [16,17] and tokamak magnet design studies [18]. A tokamak with HTS (for example, Rare-Earth Barium Copper Oxide - REBCO) would allow an increase in B_T at the center of the plasma, over LTS technology, from ≈ 5.5T to more than 12T. The field at the coil would increase from 12 T to ≈ 25 T. Compact, high-field devices were, in fact, the proposed route to burning plasmas in the U.S. magnetic fusion program for 20 years prior to entering ITER. Community consensus was reached that a small high-field burning plasma experiment, with copper magnets, could be successful [19,20,21]. (Subsequent research has shown this to be correct.) HTS enables even smaller devices at higher field and provides a technological path to a commercially attractive steady-state fusion power system. Other advantages include:

1. *Demountable magnets*: The higher critical temperature and higher heat capacities of materials at higher temperatures enables fusion magnets that incorporate demountable resistive joints that lead to vastly improved access for construction and maintenance, important for experiments and reactors. (See section 4)
2. *Operational Robustness*: High-field compact devices operate far from all intrinsic disruptive kink, pressure, density, and shaping limits, and use normalized plasma regimes (β_N, H, q) already integrally demonstrated in present devices.
3. *Steady-State Physics*: Analysis shows that high-gain, robust steady-state operation, with significant external control of the current, will arise from the combination of small size, high field, high safety factor, and associated improvements in current drive at high magnetic field. (See section 6)

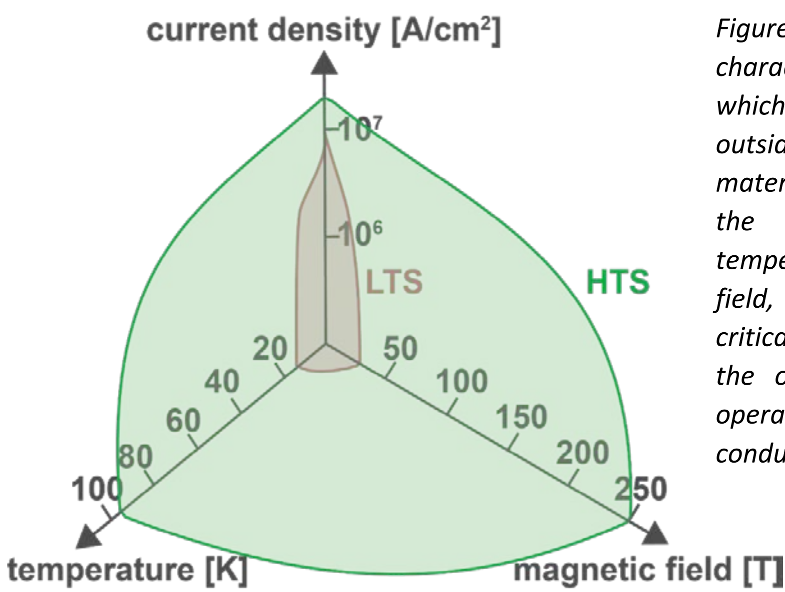


Figure 2.2 Practical superconductors can be characterized by a critical surface below which the material is a superconductor, and outside of which it is a normal conducting material. The primary variables that define the critical surface are the critical temperature, the upper critical magnetic field, and the critical current density. The critical surface of the HTS conductor shows the orders of magnitude advantage in operating space gained over LTS conductors.

Favorable characteristics of HTS over other magnet technologies

1. ***High field.*** REBCO superconductor carries sufficient current density for magnet applications at fields up to 100T [22]. It has recently been incorporated into solenoid magnets at fields over 40T [23]. This level of performance surpasses the requirement of ~20T on coil for compact high-field tokamaks.
2. ***High temperature.*** REBCO can operate at over 90K but performs much better when sub-cooled; high-field fusion and accelerator magnets often target 20-30K. The advantage of high temperature operation goes well beyond the thermodynamic efficiency of the

cryogenic system. Operation at temperatures well above those limited by liquid helium and the relative insensitivity of the critical current to temperature results in magnets with much higher operating stability, a critical consideration for the long-life operation required in a dynamic fusion environment. Further, these properties have enabled some REBCO magnets to forgo incorporating electrical insulation [24] and allows the incorporation of resistive joints [25]. The high critical temperature could also allow operating in a nuclear heating environment significantly higher than possible in LTS magnets.

3. ***High engineering current density.*** REBCO has been incorporated into magnets at over 40T at engineering current densities exceeding 1000 A/mm^2 [23]. This is an order of magnitude higher current densities compared to LTS equivalent magnets. This leads to much smaller magnets for the same magnetic field, taken to distinct advantage in compact all REBCO NMR user magnets at fields over 35T now under construction [26]. In fusion applications, this leads to more room for structure in the magnet.

4. ***High strength and high modulus.*** REBCO's primary constituent material (~50-90% by volume) is high strength steels. The superconductor remains reversibly superconducting at stresses over 600 MPa and strains up to 0.45% [27], factors of two improvement over LTS, thus enabling smaller magnets and more compact designs.

5. ***No reaction process as part of winding.*** Unlike LTS Nb_3Sn which needs to be combined with other materials, wound into final shape, heat treated at high temperatures for long periods, and then carefully handled, HTS is ready for operation as manufactured, since its superconductor layer is directly deposited in the manufacturing process, and can be wound into final position in a single operation. This results in a simpler assembly process for the magnet, and opens up a wider choice of structural materials.

6. ***Radiation resistance***. Numerous studies have been performed verifying that REBCO has resistance to neutron damage similar to that of LTS [28,29].

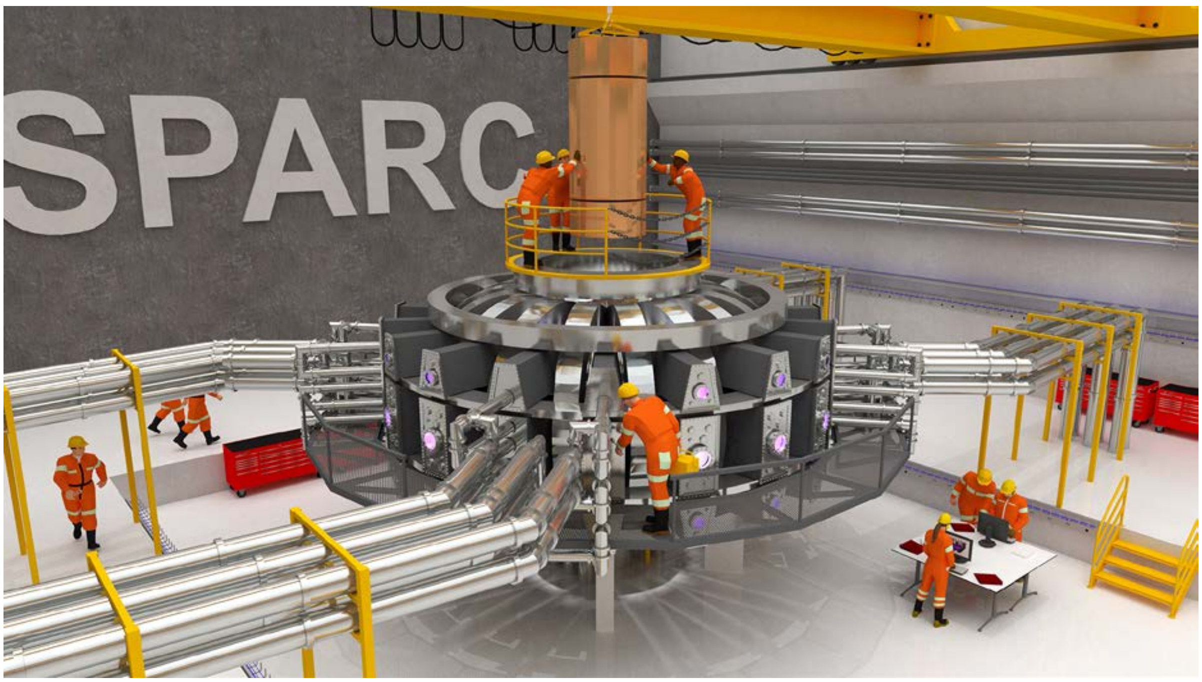
R&D for a Path Forward

An integrated program to develop fusion-class HTS magnets will be needed. The key near-term targets for the magnet R&D will be driven by the requirements of the SPARC device. Development of fusion-class magnets at the appropriate fields would require a significant effort as was done across multiple labs and universities for Nb_3Sn for ITER. Since the critical field of the REBCO superconductor is so high, the ultimate magnet, and thus fusion device performance, is primarily limited by the mechanical strength of structure around it. Existing high strength stainless steel and superalloy materials are adequate for projected fusion requirements. If exotic new, nano-strengthened materials or composites can be developed with increased tensile strength, elastic modulus, and fracture toughness, further performance improvement in the form of reduced magnet build and higher field operation may be possible.

In the present configuration of the superconductor as a flat tape, AC losses and current distribution are not ideal for fast transient, AC or pulsed operation. This can be improved with further R&D investment. However, performance is already sufficient for the TF coils of a tokamak. Although extremely stable in operation, quench detection may be a significant issue, due to very slow propagation of a normal zone. The present standard use of inductively balanced voltage taps could be a limitation on safe performance and further R&D into innovative methods for quench detection is warranted. For example, normal zone sensing by the use of optical fibers is presently being studied at laboratory scale [30]. Self-protecting NI REBCO pancake magnets have recently been successfully built and demonstrated [*e.g.* 31, 32]. Although further R&D is necessary, such magnets could provide an unprecedented degree of engineering robustness and survivability for superconducting fusion magnets.

3. SPARC – A Compact, Net-Energy Fusion Experiment

Once the basic engineering of HTS fusion magnets is established, the next step would be to use that technology to build a DT burning, compact, high-field net fusion energy experiment. Preliminary analysis has led to a conceptual design, SPARC V0, with 1.65m major radius and 0.5m minor radius operating at an on-axis toroidal field of 12 T and plasma current of 7.5 MA. A set of conceptual design parameters is shown in Table I. Such a device is predicted to produce 50-100 MW of fusion power. Its mission would be to demonstrate break-even fusion production in high-field, high-density scenarios and to demonstrate the integrated engineering of fusion-relevant HTS magnets at scale. While taking advantage of the new magnet technology, SPARC leverages the decades of international experience with tokamak physics.



An illustration of the SPARC device as it might appear during construction, showing the coaxial feed lines for 30 MW of ICRF heating. Credit: Ken Filar

SPARC further contributes to the high-field (ARC-like) fusion road map described in this white paper by retiring risks in 1) high-field scenario development, 2) I-mode extrapolation, 3) fast particle stability at high-density and high-field, 4) disruption avoidance and mitigation at high field, 5) DT/nuclear operation and control, 6) tritium handling, 7) high-performance ELM-suppressed regimes and 8) ICRF heating. Research in advanced confinement modes, while not required for the basic mission, should be accessible and could help retire additional performance risks for ARC.

SPARC would be in the same class as previously proposed compact burning plasma experiments such as FIRE [6], BPX [5] or IGNITOR [33]. Similar in size to mid-scale devices like ASDEX-Upgrade and DIII-D and only a modest extrapolation from the 8 T operation on C-Mod, SPARC is well within the envelope of tokamak systems engineering experience. The baseline heating system would provision up to 30MW ICRF with antennas similar to the field-aligned design proven on C-Mod [34]. Various ICRF scenarios look promising – including the recently demonstrated three ion hybrid, using DT(3He) in the nuclear regime [35]. The RF technology is basically in hand. The diagnostics strategy will borrow heavily on experience from actual machines including C-Mod, TFTR and JET and planned tokamaks like ITER and FIRE.

Table I: Nominal parameters for the Strawman SPARC V0

R_o	1.65 m
a	0.5 m
ε	0.33
κ	1.8
δ	0.4
B_T	12 T
I_P	7.5 MA
q_{95}	3.05
P_{FUSION}	50-100 MW
P_{EXT}	30 MW
Flattop	10 sec
Q_{FUSION}	2-5
$T_i(0)$	15-22 keV
$\langle T_i \rangle$	6-9 keV
$n_e(0)$	4 - 6.5 x $10^{20}/m^3$
$\langle n_e \rangle$	3 - 5 x $10^{20}/m^3$

Despite using superconducting coils, SPARC is designed for a relatively short pulse – roughly 10 seconds of flat-top at full performance. Operating in a short pulsed mode provides a number of important advantages, primarily by minimizing the impact of the nuclear mission. It allows for high fusion power production without significant shielding, which would otherwise drive up the size and cost. The shorter pulse and smaller size keeps the site tritium inventory lower and the neutron fluence low enough to avoid any significant damage to structural components. Overall it falls roughly within the TFTR and JET nuclear precedents. The experiment will be designed for ~1,000 full power DT shots and more than 10,000 in DD. Pulsed operation obviates the need for active cooling of the first wall, which also simplifies engineering. It is important to note, however, that given its small size, the SPARC pulse length is several current relaxation times and thus steady-state with respect to plasma physics.

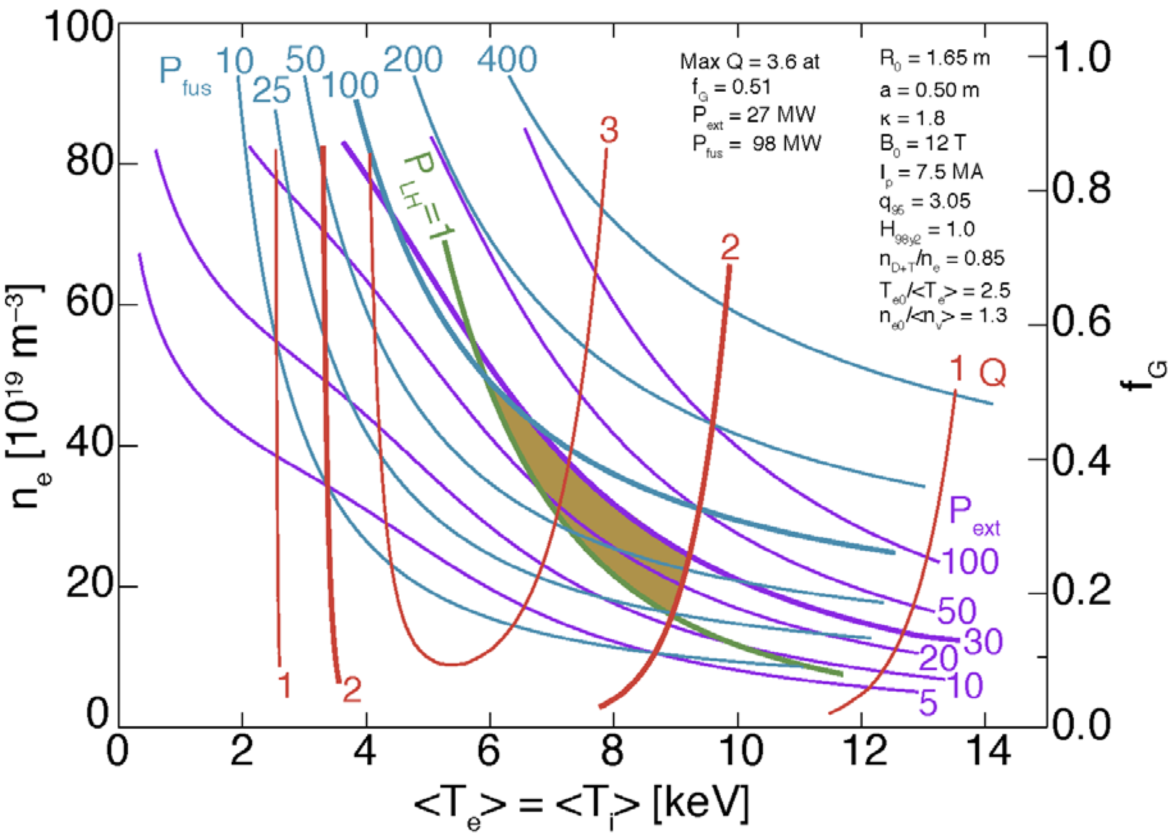
This raises the converse question; if the machine is to be run in a pulsed mode, why build it with superconducting magnets? After all one notes, that when the U.S. was looking in the past at lower cost burning plasma options, compact, high-field designs with cryogenically cooled copper magnets were chosen and detailed engineering was carried out (essentially validating aspects of the high-field approach that we are describing here).

There are two compelling reasons for choosing superconductors for SPARC. First, SPARC will serve as a test bed for magnet systems at full scale and with full integration into a fusion system, allowing exploration of burning plasma physics at the higher field, higher density regimes we envision for future power plants. Secondly and perhaps surprising, the engineering looks to be easier and the costs lower with HTS superconductors. Copper is a difficult material with which to build high field magnets. It is inherently mechanically weak, requiring rather innovative approaches to keep the structures within allowable stresses (e.g. note the sliding joint design of C-Mod). Magnet heating in a copper magnet also gets harder and harder to overcome in high field designs, severely limiting pulse length. It is quite difficult to get the pulse length as long as a current relaxation time in a burning device with passively cooled copper magnets. Thermal and mechanical cycling are also severe, imposing serious limits on the total number of full performance discharges. These issues are present whenever a copper magnet pulses, whether in DT or not – or even in tests without plasma. The specification for FIRE was 3,000 full-field pulses and only 525 for Ignitor [36,37]. The challenging cooling requirement for copper magnets also reduces the shot rate, even for non-DT operation. FIRE would have been able to run full field shots every 3 hours, dramatically slowing the rate of learning during the commissioning and DD phases. The copper magnet for FIRE would have needed huge and expensive pulsed power systems, requiring a 1,000 MVA power system from an alternator-flywheel system that stored up to 20 GJ of energy. It would have consumed a prodigious quantity of LN_2. boiling a quarter of million liters for each pulse [36]. In contrast, SPARC would operate 3-4 shots per hour and require only modest power and cryogenic systems.

The projected SPARC performance is based on conservative plasma physics – essentially the ITER baseline scenario [2, 3]. It is more conservative with respect to physics operational limits, running at lower β_N and far below the density limit. In terms of dimensionless plasma parameters, SPARC is closer to the database median than ITER, notably for normalized size (ρ^*) and density, (n/n_G). In fact, a number of shots exist in the ITER confinement database that simultaneously match all of the dimensionless parameters (β, ν^*, ρ^*, q_{95}, n_G, ε, κ, δ). The precedent of these early JET discharges with $B_T > 3T$ and $I_P > 3MA$, increases confidence in the confinement projections. Under the baseline assumptions for confinement and access to H-mode, SPARC would achieve $Q \approx 3.6$ generating up to 100 MW of fusion power. A POPCON (Plasma Operation Contour) plot for SPARC is shown in the figure below. The figure shows the available operational space and boundaries under standard ITER physics assumptions. Using different confinement assumptions, we find that SPARC has robust access to $Q > 2$ even at $H_{98} \approx 0.9$, that is, one standard deviation below the scaling mean. It would reach $Q = 1$ in L-mode ($H_{89} = 1$) and $Q \approx 2.6$ in a modestly improved L-mode ($H_{89} = 1.4$). There is a long tail on its high end performance, with $Q = 5$ in reach with H_{98} of 1.1 (one standard deviation above the scaling mean) and perhaps higher in I-mode. Advanced operation with hybrid modes or ITBs could be explored, but are not required for the mission. In these regimes, the operational limit would be on allowable fusion power, though higher performance might be possible for shorter pulses.

Power loading on the first wall is a widely recognized challenge for any high-performance fusion device. For SPARC, the strategy involves ensuring a moderate level of radiation, $P_{RAD}/P_{LOSS} \sim 0.5$, continuous sweeping of the divertor strike point and non-actively cooled internal components. Under these conditions proposed divertor materials easily survive a full power pulse. We further note that C-Mod, AUG, JET have demonstrated $P_{RAD}/P_{LOSS} > 0.9$ while maintaining good core confinement.

By all measures, achieving the SPARC goal represents less of a physics extrapolation than ITER – at about 1% of the cost – following a demonstration of the new HTS magnet performance. Going forward with this step is notably conservative when compared to the decisions to build TFTR and JET based on 1970's physics and technology databases. Thus we conclude that once the magnet technology is demonstrated, achieving the SPARC fusion mission is less risky than other major fusion initiatives.



POPCON plot for SPARC baseline operation, showing robust access to Q > 2 under standard ITER physics assumptions, including standard H-mode confinement. The operational space is bounded by the L-H threshold and available heating power. Maximum performance under these assumptions is Q = 3.6 with 100 MW of fusion power.

4. Demountable Superconducting Magnet Coils

A further innovation that allows a radical change in maintenance, blanket technology and the whole fusion pathway

With regards to modularity, jointed, *demountable superconducting magnets* are proposed for devices beyond SPARC. Many of the engineering challenges associated with the toroidal geometry of a magnetic confinement device would be eliminated since the dismounted Toroidal Field (TF) magnet is a cylindrical rather than toroidal structure – i.e. it does not link the vacuum vessel or poloidal field coils. Jointed TF coils would allow vertical maintenance inside the coils, the construction of poloidal field (PF) coils internal to the TF coil, and the possibility of a single-piece, modular vacuum vessel. Early stage R&D suggest that this approach is technically feasible [25,38,39,40]

Scientific and/or engineering opportunity:
Demountable superconducting magnets would be highly advantageous in any magnetic fusion concept. In tokamaks and stellarators, for example, demountable toroidal field coils would allow vertical maintenance on subsystems contained within the TF magnet "cage", greatly simplifying and accelerating the installation, maintenance, and replacement of internal components. This contrasts strongly with traditional sector maintenance schemes that require components internal to the TF magnet to be cut, welded, inspected and assembled in a nuclear environment in order to be removed and replaced.

Furthermore, when coupled with a liquid immersion blanket and a compact, high-field design [4,7], the vertical maintenance scheme enabled by demountable toroidal field coils will allow simplified removal and replacement of the entire power core (*e.g.* first wall, divertors, RF antennas, vacuum vessel, internal poloidal field coils) as a single, modular component. An example of such an approach is the recent ARC reactor design study [7]. This proposed maintenance scheme would allow the entire power core, including the primary vacuum vessel, in a reactor to be considered a non-lifetime, or consumable, component. The power core could be completely fabricated and qualified at an offsite facility, without the need for extensive remote handling, shipped to the reactor, and replaced vertically as maintenance schedules require. This approach, which reduces the required operational lifetime of the power core, particularly the vacuum vessel, has three principal advantages:

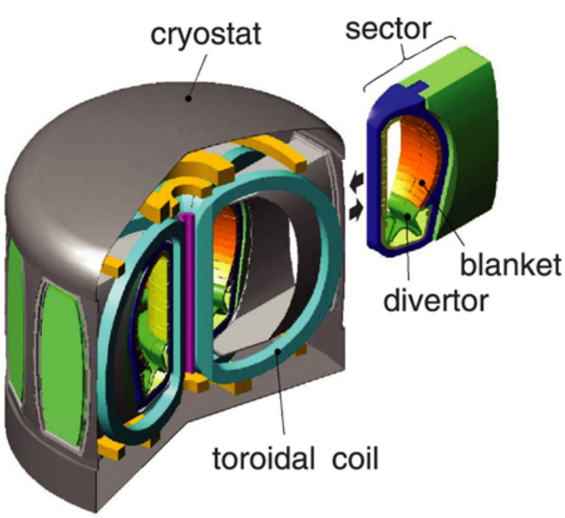
1. Significantly lowers the survivability requirements of plasma-facing and structural materials that will experience significant plasma-material interactions and radiation damage, reducing the total amount of activation in the vacuum vessel.
2. Allows the power core to be manufactured and quality controlled off-site - while the reactor continues to generate electricity - with no need for remote handling as opposed to reconstructing the power core remotely inside the TF magnet during maintenance periods. While remote handling will be required to mount/de-mount the coils, this design

requirement is significantly less demanding than the in-situ welding, installation, alignment etc. required in sector maintenance.

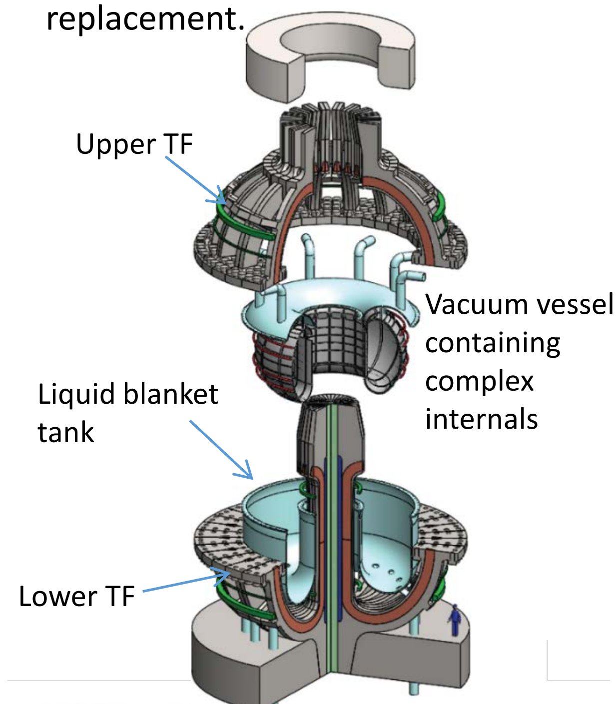
3. Reduces the risk and consequence of an off-normal event (such as an unexpected large disruption) permanently damaging the vacuum vessel and disabling the reactor.

Figure 3.

Sector maintenance:
The standard, where the fusion blanket is cut into sections in place, removed through the TF magnet in many pieces then new pieces are inserted one by one and re-welded.



Vertical maintenance:
Enabled by joints, the TF cage is removed and only the vacuum vessel is removed in one piece, swapped with a ready-made replacement.



Because demountable TF magnets enable significant power core modularity, early fusion reactors would be able to simultaneously perform as fusion nuclear science facilities (FNSFs). By incorporating fusion material science advances learned during device operation into each successive power core, the first fusion reactors would provide cost-effective, integrated, and perfect fidelity test beds, bootstrapping the performance of fusion materials while simultaneously providing fusion electricity. This breaks out of the “chicken and egg” problem of having to design an FNSF with structural materials which are known beforehand to be sufficiently robust to the hostile fusion nuclear environment to survive for the required lifetime.

With regards to tokamak design, demountable TF coils would allow PF coils to be installed within the TF magnet cage, as was done on Alcator C-Mod, where their closer proximity to the plasma would significantly lower coil currents and forces. Furthermore, internal PF coils would enable novel divertor topologies such as long leg divertors, a promising solution to the severe challenge of heat exhaust and plasma-material interactions in tokamak divertors [11]. Demountable TF coils would also allow replacement of single TF coils, reducing the risk that a TF fault would permanently disable an entire fusion reactor. Indeed it is for these reasons that many tokamak experiments with copper coils have been designed as demountable. It is worth noting that for stellarator configurations, with their more complicated 3D magnets, demountable coils might offer an even greater advantage for construction and maintenance.

Looking ahead to commercial power plants, a vertical maintenance scheme as described above would have substantially less complex remote handling challenges and faster maintenance times than a device with sector maintenance, allowing the availability of the fusion reactor to be higher. Plant availability is a key driver for power plant economics, especially fission [41], and would be the case with fusion reactors as well. Increased availability would greatly improve the economic viability of fusion reactors, leading to a higher chance of fusion being adopted as a source of clean power.

5. Molten-Salt Liquid Blankets

The transformational impact on fusion power plant design, maintenance, safety, and economics

The liquid immersion blanket is an innovative design concept in which the fusion power core (*e.g.* plasma, first wall, divertor, heating and current drive antennae and vacuum vessel) is completely immersed in a continuous, structure-free liquid volume of molten salt. The liquid immersion molten-salt blanket provides net tritium production, high-efficiency thermal conversion for electricity production, and ideal, nearly penetration-free radiation shielding for the device and personnel [7]. The concept shows very favorable improvements over traditional blanket technology, resulting in substantial opportunities to maximize plant availability through accelerated maintenance schemes, minimizing cost by reducing engineering complexity, and improving the reality and perception of fusion as a safe and clean energy source by dramatically reducing the volume of radiological waste. The availability of demountable superconducting toroidal field (TF) coils made with high-temperature superconductors fundamentally improves the design of tokamak fusion power plants and is synergetic with a liquid blanket, making it a cornerstone technology within the accelerated high magnetic field path to practical fusion energy.

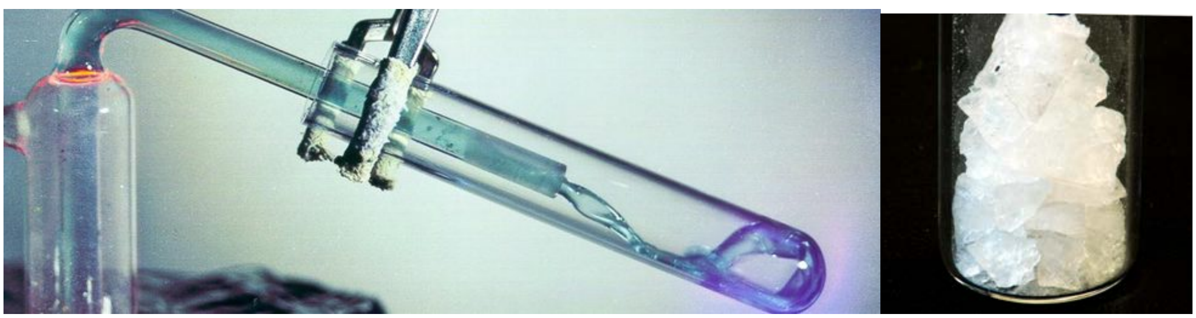


Figure 4. FLiBe in its liquid and solid forms

Scientific and/or engineering opportunity:

Traditional blanket designs have been implemented within "sector maintenance" schemes, in which non-lifetime components placed within the TF magnet "cage" must be extractable through ports, squeezing in between oversized TF coils. This approach generally results in blanket subsystems that:

- Have high engineering complexity (high cost and high risk of assembly failure)
- Have extremely challenging remote handling requirements (prolonged maintenance schemes, low plant availability, and likely poor economics)
- Have significant internal structural material (leading to difficulties in tritium breeding, high-volumes of highly activated material, and high material disposal costs)

- Must be contained within the vacuum vessel (design challenges and serious accident potential in worst-case loss-of-vacuum incidents), which provides both the safety, vacuum barrier as well as structural support.
- Must be compatible with operating in a vacuum environment.

Conversely, an immersion liquid blanket resides outside of the vacuum vessel and within a larger blanket tank. The liquid blanket can be easily drained when it is time to replace the power core due to neutron damage. Thus the vacuum vessel becomes a replaceable component that does not need to contain an internal blanket to protect it from radiation damage. Instead, the liquid outside of it essentially eliminates many of the traditional blanket engineering challenges by enabling the power core to simply be immersed in a single, continuous volume, or pool, of molten salt. Molten salts like FLiBe have many advantages for fusion blankets, including very low activation, low chemical reactivity, low electrical conductivity, excellent neutronics properties, and high operating temperature [42,43].

Impacts of an immersion, molten-salt blanket on fusion power plant design:

- The absence of internal non-breeding structural material and very high solid angle (close to 4π) of the immersion blanket enables larger choice for net-tritium breeder materials than traditional blankets.
- Unlike liquid metal blankets, molten salts have low electrical conductivity, substantially mitigating MHD-induced problems, reducing pumping power requirements and enabling faster plasma control from external coils [44].
- Molten salt provides mechanical damping and enhanced structural stability to the vertically suspended vacuum vessel during major plasma disruptions.
- The liquid blanket conforms to long-leg divertor topologies without sacrificing tritium breeding.
- Liquid molten salt provides a near-perfect radiation shield with zero free-streaming pathways.
- Highly exothermic nuclear reactions within the molten salt provide additional heat for electricity.

Impacts on maintenance and economics:

- The immersion blanket drastically simplifies what traditionally are extremely complex subsystems, improving economics since the blanket cost is much closer to raw material cost.
- The liquid blanket eliminates the large amount of structural material contained in traditional blankets that becomes highly radioactive, minimizing decommissioning costs and complexity.

- Low-Z molten salts rapidly moderate the fusion neutrons and minimize neutron backscatter into the first wall and vacuum vessel, significantly reducing radiation damage in these components.
- High operating temperatures (>630 degrees C) enable high efficiency thermal conversion.
- With demountable TF coils the liquid blanket aids a vertical maintenance scheme where the power core is treated as a single, modular entity that is quickly replaced wholesale after several years by draining the molten salt, replacing the power core, and refilling the molten salt blanket tank.

Impacts on safety:

- The liquid blanket operates near atmospheric pressure. It is outside the vacuum vessel and does not require high-vacuum engineering, minimizing the probability and severity of loss-of-vacuum accidents.
- The radiation dose from activated molten salts such as FLiBe are dominated by radioisotopes with short half-lives (< 24 hours), significantly reducing worst-case accident scenarios [45].
- FLiBe freezes at temperatures below 460 degrees C, mitigating liquid radiological waste issues.
- The blanket tank acts as the lifetime safety barrier and does not provide the primary vacuum. This tank receives low radiation doses due to the effective shielding by the liquid blanket.

6. High-Field Launch RF for Heating and Current Drive

A new path towards RF sustainment of high-field fusion reactor plasmas

High-field-side Lower hybrid current drive offers the possibility of higher efficiency and better current profile control while protecting the launching structures from damaging plasma interactions. RF (radiofrequency) actuators with high system efficiency (wall-plug to plasma) and the ability for continuous operation have long been recognized as essential tools for realizing a steady state tokamak. [1]. The power required to maintain a tokamak's plasma current directly impacts the net power plant output. Thus, the economic and engineering viability is critically impacted by the amount and efficiency of the required externally driven current. The requirement for auxiliary current drive is dictated by the need to augment the self-generated, bootstrap current fraction (scaling $\propto \sqrt{\varepsilon}\beta_P$, where ε is the inverse aspect ratio, β_P is the ratio of plasma to poloidal magnetic field pressure). Consideration of MHD stability suggests efficient off-axis current drive at normalized minor radius, $r/a = \rho \approx 0.6$-0.8 is necessary for a reactor.

Current drive and heating technologies must achieve high system efficiency ($\approx$ 70%), have high system availability and reliability, and operate continuously in a thermonuclear environment. Among a number of heating and current drive actuators, lower hybrid range of frequency (LHRF) is among the most promising for off axis current drive due to its high efficiency. While the LHRF core and coupling physics are well understood, integrated physics/engineering solutions for a reactor environment are currently lacking demonstration. A number of physics and technology challenges to utilization remain including efficient coupling, the location and efficiency of the driven current, and minimization of RF/launcher-generated impurity contamination. In a reactor environment, PMI issues associated with coupling structures are similar to the first wall issues and have been identified as a potential show-stopper [46]. The impurity contamination associated with RF operation must be low to avoid excess impurity radiation losses; for metals; typically $<10^{-4}$ per plasma ion must be achieved. The launching structures must also minimally impact tritium breeding and experience erosion rates less than 1 mm/year.

Scientific and/or engineering opportunity:

At present, the efficiency for off-axis LHCD is theoretically the highest among all CD techniques. Ion cyclotron, neutral beam, and electron cyclotron (ECCD) actuators could potentially provide off-axis current drive but with significantly reduced current drive efficiencies and non-optimal driven current location. (Neutral beams also significantly expands the nuclear envelope of a fusion reactor.) For example in DIII-D, the proposed helicon RF current drive is calculated to provide ~65 kA/MW peaked at $\rho<0.5$, off axis neutral beam injection provides 18 kA/MW peaked at $\rho\sim0.5$, vertical launch ECCD is calculated to provide 28 kA/MW peaked at $\rho\sim0.6$. In contrast, HFS LHCD is calculated to provide 150-200 kA/MW, roughly an order of magnitude improvement at the needed location ($\rho \approx 0.7$).

High-field-side (HFS) LHCD is a game changer for RF current drive physics. Low field side (LFS) LH couplers have matured significantly and a passive active coupler has demonstrated remarkable coupling resilience, <1% power reflection, over a broad range of plasma conditions at reactor relevant power densities [47]. However, LFS LHCD couplers have traditionally been located on the mid-plane, where estimated erosion rates are >5 mm/year. Further, impurity contamination from fast ions, RF induced fast electrons and convective cells are tenuously controlled and expected to be problematic in any reactor.

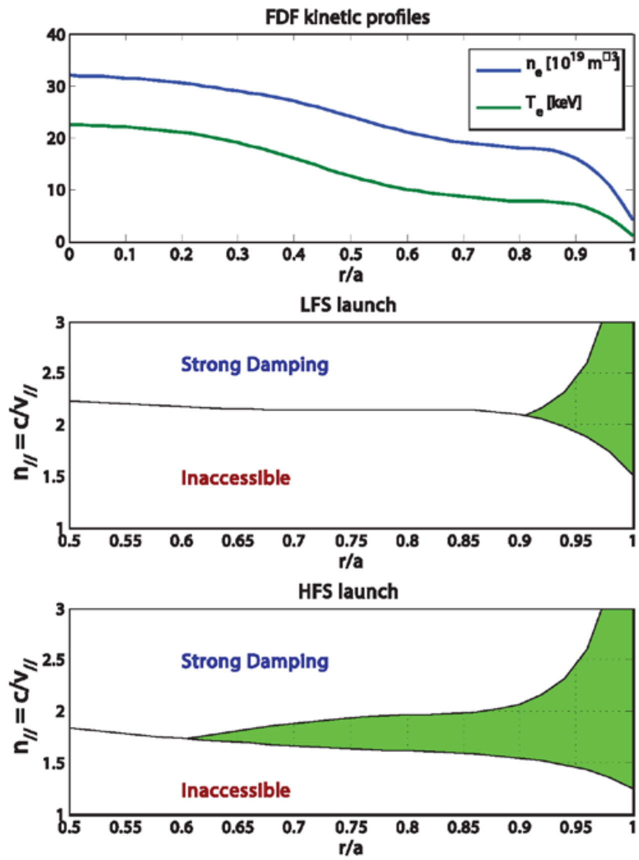


Figure 5 Accessibility window for LFS and HFS LHCD shows the HFS penetrates to $\rho \approx 0.6$ at low $n_\parallel$ for FDF temperature and density profiles.

Placing the launcher on the high-field side vastly improves wave penetration, resulting in driven current at mid-minor radius [9,48], precisely the location where current needs to be to create flat or reverse magnetic shear profiles [49]. Wave access is bracketed by wave accessibility [50] and the condition for electron Landau damping [51]. For a given wave frequency, the minimum index of refraction, $n_\parallel$, that will penetrate into the plasma is proportional to $n_e^{1/2}/B$. Ideally, the waves penetrate into the plasma until electron Landau damped which occurs when $n_\parallel \leq (30/T_e)^{1/2}$ where T_e is the electron temperature in keV. On the HFS, the B_T-field is significantly higher allowing for the launch of lower $n_\parallel$ waves that penetrate farther into the plasma core before damping. An example is shown in figure 5 where plasma profiles from FDF design are shown and the wave accessibility window is depicted in green [52]. The HFS LHCD penetrates up to $\rho \approx 0.6$ at low $n_\parallel$. The lower $n_\parallel$ is a significant benefit since the waves are absorbed at higher T_e yielding a higher current drive efficiency, $\propto 1/n_\parallel^2$, due to wave momentum being transferred to less collisional electrons [53]. To optimize current drive efficiency, the poloidal position is selected to balance the effects of toroidicity and poloidal field to maintain the smallest $n_\parallel$ possible. In this way, the current drive efficiency can be increased up to $\approx 50\%$ compared to low-field side launch [10], which is a critical advantage in lowering the recirculating power fraction in steady-state operation.

HFS RF launch is also a game changer for controlling plasma-material interactions, providing a means to dramatically improve launcher robustness in a reactor environment. For near-double null plasmas, a low heat-flux, quiescent boundary layer naturally forms on the high field side:

- no heat/particle pulses reach there from ELMs [55], with essentially zero fluctuation-induced fluxes [56] and no filamentary, cross-field transport
- local density at high-field side launch structures can be precisely controlled via upper/lower X-point flux balance due to steep SOL density profile [57], and/or distance from the last-closed flux surface (LCFS) to the launcher
- reduced heat and particle fluxes (including from neutrons) on the HFS wall will improve long-term survivability of the antenna structures
- the flux of energetic ion orbit loss to the HFS is virtually nonexistent
- there is no impact from runaway electrons at this location
- impurity ions produced from the launcher are expected to be very well screened based on results from impurity transport experiments [58]
- local plasma recycling fluxes are low, which minimizes neutral pressures in the vicinity of antenna/waveguide structures, leading to improved RF voltage handling [59]
- RF driven convective cells that can lead to RF enhanced heat loads and impurity sources at the antenna are half as strong as to those on the Low-Field-Side [60].

R&D for this concept could begin early with experiments proposed on the DIII-D and WEST tokamaks. These experiments would validate RF models for wave coupling, propagation and damping. Eventually, tests in reactor-relevant plasma regimes on a device like ADX would be required before implementing them in a reactor/pilot plant.

7. Long-Leg Divertors and ADX, the Advanced Divertor Experiment

A strategy to solve divertor heat flux and plasma-material interaction challenges for fusion energy development – enabled by testing in a high-field, high power density Divertor Test Tokamak (DTT)

Extreme levels of power exhaust and PMI have long been recognized as critical challenges for fusion development. "*Taming the plasma-material interface"* has always ranked among the highest priorities in the FES program [1,61,62]. Recent experimental results indicate these challenges are even more severe than originally anticipated [63]; divertor power handling in ITER will be marginal at best; there has been a growing concern that a fusion reactor may not be feasible, based on power exhaust and long-pulse PMI considerations alone. As noted in the EFDA roadmap [64], "*If ITER shows that the baseline strategy cannot be extrapolated to DEMO, the lack of an alternative solution would delay the realization of fusion by 10-20 years*."

Modeling indicates that tightly-baffled, long-leg divertors with embedded secondary x-points have potential to provide an order-of-magnitude increase in power exhaust handling over conventional divertors, while fully suppressing plasma-material interaction (PMI) damage [12,65]. A passively-stable, fully-detached state may be maintained in the divertor leg over a large power window. This result, if confirmed by experiment, would solve the divertor heat flux and PMI challenges for tokamaks. Demountable magnets, made possible by high-temperature superconductors (HTS), allow reactors to employ blanket geometries and coil configurations that can accommodate long-leg divertors – with no impact on core volume, while maintaining acceptable coil currents, and keeping the coils fully shielded from neutrons [14].

However, the projected performance under reactor conditions is highly uncertain. The dimensionless parameter space (set by plasma, atomic and PMI physics) is well beyond the validation range of current models – particularly considering the role of turbulence. Impacts on pedestal/core plasma performance and operational windows (e.g. confinement regimes) are impossible to predict. A dedicated experimental platform is needed to test these and other promising divertor concepts and to validate physics models applicable for a reactor. Standard dimensionless plasma similarity techniques cannot be applied. Divertor parameters must be identical – plasma pressure, parallel heat flux density, neutral density, magnetic field, mechanical geometry, magnetic geometry/topology, field line lengths, heat flux width, electron temperature and gradients, and radiation emissivities.

A sensible strategic plan for fusion development must therefore include a purpose-built Divertor Test Tokamak (DTT) to test long-leg and other advanced divertor concepts, as recognized by community experts in the FES 2015 PMI workshop report [66]. Boundary heat flux widths are found to scale inversely with poloidal field and independently of machine size – spanning a factor of 5 in size [63] (C-Mod vs. JET). This means that a compact, high field DTT can produce *reactor-identical divertor conditions*. In this way, promising concepts can be tested in the actual geometry (in cross-section) that would be used in a full-scale reactor [11].

Moreover, a compact, high-field DTT could also function as an RF Sustainment Test Tokamak (STT). High-field side RF launch combined with operation in near double-null configurations is a transformative idea for achieving efficient, low PMI RF actuators for heating and current drive. This idea takes advantage of the 'quiescent scrape-off layer' that naturally forms at this location to reduce PMI and provide unprecedented external control of local conditions at the antenna-plasma interface. Most importantly, as noted above in section 6, RF wave physics is highly favorable with high-field side launch. Lower hybrid waves may penetrate to mid minor radius and at the same time attain a 40% or more improvement in current drive efficiency [9]. A host of other potential benefits accrue as well, including the elimination of energetic particle loads, ELM heat pulses and runaway electron damage on launch structures. These are among the approaches needed to make fusion energy a reality.

A compact, high-field DTT would be prototypical in many ways for an ARC-like reactor concept, combining:

- Tightly baffled, long-leg divertors with embedded secondary x-points
- Divertor plasma conditions (pressure, power density, geometry) nearly identical to ARC
- Pedestal pressures approaching that of ARC
- Exclusive use of RF systems for heating and current drive, including inside-launch RF
- Operation at high absolute plasma pressure but moderate plasma beta
- Equilibrated and strongly coupled electrons and ions
- Operation in regimes with low or no external torque
- No fueling from external heating and current drive actuators

Such an integrated demonstration of reactor-relevant core-pedestal-divertor operation with steady state sustainment (i.e., pulse length greater than current relaxation time) is precisely what is necessary to inform a high field development pathway towards an ARC-like pilot plant.

Scientific and engineering opportunities

With HTS opening up a new high field pathway for magnetic fusion energy development [4,7], critical research shifts towards finding viable physics and engineering solutions for support systems: (1) advanced divertors for heat exhaust/erosion control that can handle order-of-magnitude increases in power density over present experiments, (2) efficient (i.e., wall-plug to plasma) RF systems for steady-state current drive and heating that survive the onslaught of PMI in a reactor.

Demountable toroidal field (TF) magnets, facilitated by HTS [4] have the potential to change this outlook considerably. For example, ARC employs an immersion FLiBe blanket for tritium breeding and neutron shielding. The demountable TF allows poloidal field (PF) coils to be placed inside the TF and the vacuum vessel/first-wall/divertor assembly to be removed as a single unit for service and replacement, mitigating component lifetime requirements. This enables ARC to function as a fusion nuclear science R&D facility. Taking advantage of these features, a tightly-baffled, long-leg divertor with an "X-point target" [11] has recently been incorporated into the

ARC concept [14,67] – with no impact on core plasma volume or TF magnet size, acceptable PF coil currents and lifetimes (> 5 full power years), and adequate tritium breeding ratio (1.08). Such tightly-baffled, long-leg divertor geometries may enhance peak heat flux handling by a factor of 10 over conventional approaches. Just as important, access to a passively-stable, fully detached divertor condition may be possible over a wide power range. This latter feature is essential because, unlike in present experiments, detection and feedback control of divertor conditions in response to transients may not be possible – the neutron environment disallows standard diagnostics and system response times are much too long.

Unfortunately, our present experience and models are inadequate for reliable extrapolation to reactor conditions; these have not been developed in appropriate regimes. Divertor physics involves the interplay among plasma, neutrals and impurities with the potential for very strong levels of turbulence and transport – critically affecting, for example, the interaction with and power loss to the sidewalls of a long-leg divertor chamber. Additional variables include mechanical geometry, magnetic topology, 'upstream' conditions and interactions with the pedestal (e.g., transport barriers). Because atomic processes are involved, plasma physics dimensionless similarity techniques [68,69] are not appropriate for divertors [70,71,72]. Instead, a 'divertor plasma simulator' must reproduce conditions identical to those that will be encountered in a reactor. Yet no facility exists in the world, nor is any being planned, that can test long-leg divertors with embedded secondary x-points – or other advanced concepts – in the relevant regimes.

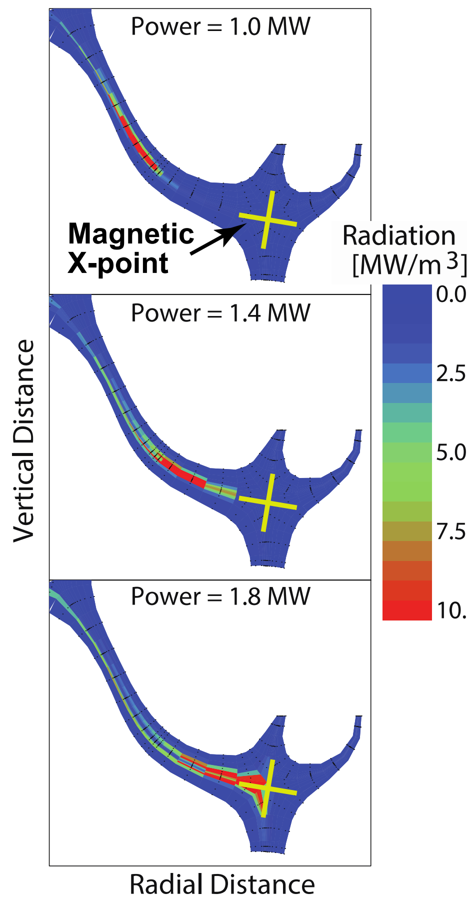


Figure 6.1 The X-point target divetor is a promising, long-legged divertor concept that has the potential to solve the plasma-material interface challenge – attaining a passively stable detached divertor condition that accommodates a wide range in power exhaust [65][55].

These considerations have led the US fusion community to conclude that a purpose-built, Divertor Test Tokamak (DTT) is necessary to solve the challenges in this area [66]. Such a facility would have the divertor volume and flexibility to explore a wide variety of promising concepts, including liquid metal target options. The ADX proposal [11] is recognized as a strong candidate. In addition, a purpose-built ADX could incorporate other innovations highlighted in the PMI workshop report [66], such as high-field side RF systems combined with near double-null operation –

a potentially transformative technique to solve RF current drive and actuator PMI challenges.

ADX

ADX (the Advanced Divertor and RF tokamak eXperiment) [11] is conceived as a compact, high-field, high power density Divertor Test Tokamak (DTT) and RF sustainment test tokamak (STT) specifically designed to fill these gaps in the world fusion research program; it has a large, flexible divertor volume with the ability to deploy RF launchers on the *high-field side*; its access to high performance, reactor prototypical core plasma conditions with short current relaxation times, make it an ideal platform for exploration of reactor-relevant RF wave physics and current drive actuator development.

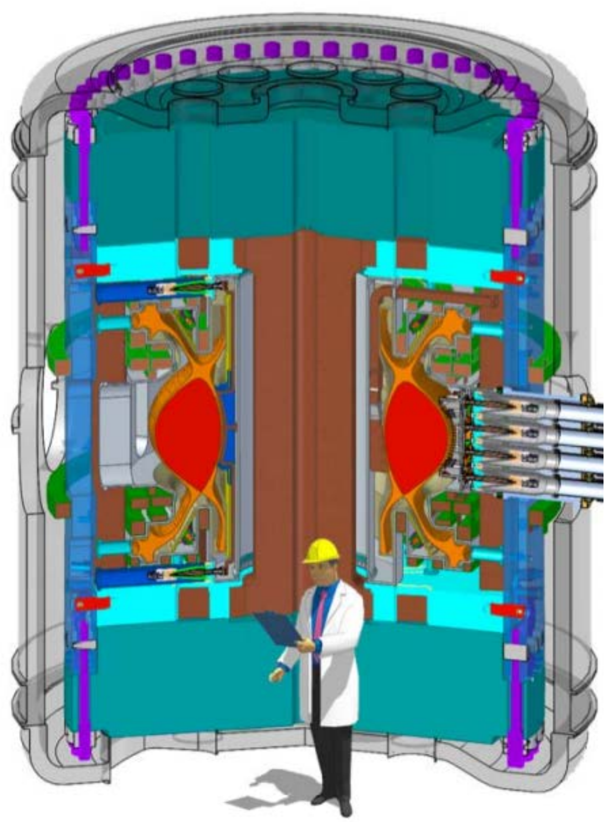


Figure 6.2 The ADX device would provide a reactor relevant test bed for advanced divertors and RF sustainment technologies.

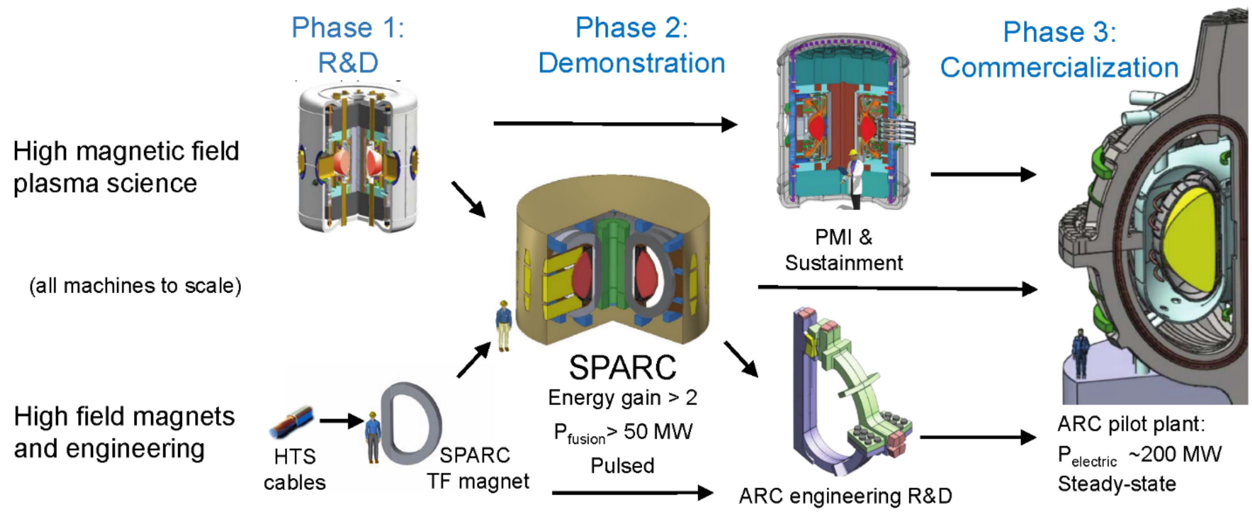
World research has established that boundary heat flux widths are independent of machine size and scale inversely with poloidal magnetic field [63], and that pedestal pressures scale with poloidal magnetic field squared [73]. These scalings enable an ADX – operating at the poloidal field, plasma pressure and exhaust power density of a pilot plant – to perform *divertor identity experiments*, and thus test the most promising concepts in the actual size and geometry (in cross-section) that would be implemented in a reactor. If a divertor solution is proven in an ADX, its performance could be projected with low risk to a pilot plant.

The US is presently in a position to take the lead in this area with the construction of a dedicated Divertor Test Tokamak (DTT), as recognized in the FES 2015 PMI workshop report [66]: ***"In our judgment, the development of this science and technology is the most critical issue for advancement to DEMO, and the country that leads here will be in a leading scientific and technological position for the future."*** An ADX would build on US expertise in advanced divertors, physics, plasma-material interactions, liquid metals, RF actuators and RF wave physics. It would provide critical information for next-step reactor designs and ensure US leadership in these areas. *"Through the process of experiment-driven science and discovery, a DTT would rapidly advance fundamental understanding, stimulate game-changing innovations, and facilitate U.S. world leadership in these most important science areas." [66]*

ADX (or a similar DTT device) is a necessary element of a strategic plan for U.S. fusion energy development. Such a purpose-built device can function *as both a DTT and an RF sustainment test tokamak* (STT). The ability to produce reactor-level conditions in a full-scale divertor mockup is not only practical but *essential* for developing reliable physics models. Similarly, a tokamak with the ability to implement high-field side RF launch systems at reactor parameters is essential – it would help determine if RF sustainment is in fact technically viable for a tokamak.

8. The High-Field Fusion Development Path

The following chart summarizes the development plan described in this monograph. The goal is to dramatically speed up the rate of development, drastically reduce the cost and arrive at a more attractive final product in terms of unit size, modularity and ease of construction and maintenance – all leading to commercially competitive electricity production from fusion.



Starting with the existing physics and technology basis, the elements of this roadmap target risk retirement across the full set of technical issues requiring resolution for fusion power. The first phase of HTS magnet development, leading to the construction and operation of the SPARC device will demonstrate the integrated operation of high-field net energy gain and address burning plasma physics in the high-field, high-density regime.

Parallel research on a set of focused physics and plasma technology topics will also be required. Of particular importance is the need to develop and demonstrate a solution to the intense heat and particle losses that are present at the plasma edge. The solution must simultaneously 1) deal with steady state and transient heat loads; 2) reduce erosion to nearly zero levels, 3) demonstrate tritium retention at sufficiently low levels and 4) demonstrate the integration with acceptable core performance. A key part of this research will be the exploration of advanced divertor geometries, which have shown promise in computer simulations [11,12]. The edge/divertor solution must be integrated with adequate core plasma performance and sustainment with reactor relevant driver technologies. Finally solutions for disruption avoidance and mitigation must be developed and tested.

In parallel, a program of engineering technology R&D needs to be undertaken. This R&D would develop 1) demountable superconducting magnets, 2) molten salt blanket including validation of schemes for thermal hydraulics, breeding and tritium extraction, 3) technology to support the proposed maintenance scheme and 4) materials for first fusion core.

The ARC pilot plant would integrate these technologies and demonstrate, at scale, full system integration, long lifetime components, the maintenance scheme and energy conversion to electricity, while the process of licensing the ARC device would be a key step toward commercialization.

9. Summary

In this white paper we have described an innovative and coherent vision for the future of fusion energy. We believe that, in contrast to the pathway we're currently on, this roadmap could allow the U.S. to develop fusion energy in time to make a difference for global climate change. Moreover, this path leads to a fusion power plant that would be more attractive economically with more robust engineering than the conventional designs under consideration. The next steps in this roadmap are clear – we need to:

1) Develop large volume, high-field HTS superconducting magnets for fusion applications
2) Demonstrate net energy gain in a compact, high-field burning plasma experiment (SPARC)
3) Develop and demonstrate practical designs for demountable superconducting magnets
4) Develop the technology for the molten salt blanket
5) Demonstrate the physics and engineering for high-field side LHCD
6) Develop and demonstrate robust solutions for particle and heat loads at the interface between the hot plasma and ordinary matter with a new high-field, high-heat flux divertor test tokamak along the lines of ADX
7) Integrate these elements in a compact, high-field fusion pilot plant along the lines of the ARC concept.

A sensible and prudent strategic plan is one that determines, as early as possible and as well informed as possible, the likelihood for success in these critical areas. And while this roadmap implies some significant program re-direction, that is to be expected with the adoption of such transformative R&D.

Embarking on such a path would ensure U.S. leadership in many critical aspects of plasma physics and fusion development. Seizing this leadership, the US would attract excellent scientists and students and drive international collaborations. This new vision for fusion development is ripe for exploration and innovation – a very exciting field for universities, national labs, and industry – with world-changing impact and importance.

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