09 April 2018

To: Members of the National Academies Burning Plasma Panel

Over the past year, a fusion community-based "strategic directions" activity has been under way. The aim of this activity has been to foster debate within the U.S. fusion community about the future of fusion research in the United States. Over 200 members of the community have participated in this activity so far, submitting white papers and debating technical initiatives, missions, research pathways, and strategic principles in working groups and in two week-long workshops. The participants in this activity are to some extent reflective of the community's makeup in terms of institutional affiliation (national laboratories, universities, companies) and of physics and technology disciplines. However, they comprise a largely self-selected group, not one that was specifically designed to be representative of the overall community's institutional or topical balance.

We have made progress by engaging constructively in and debating some of the challenging technical and political issues that face our field. Yet, it has become clear that a sustained effort well beyond the time horizon of your panel will be necessary for us to reach community consensus on key aspects of a strategic plan for the U.S. program. Nonetheless, several research thrusts have received considerable attention in our discussions and workshop summaries. We term these "strategic elements" because they are widely seen as exciting and having the potential to be important components of a U.S. strategy for fusion energy development. We have identified nine such strategic elements to document in white papers for submission to your panel, and are submitting the first six of these herewith. The complete list of strategic element white papers that we will submit is as follows:

1. Burning plasma *still in preparation*
2. Developing HTS magnets for fusion applications
3. Configuration research
4. Stellarators
5. Theory/computation
6. Plasma-material interactions and divertor
7. Fusion nuclear materials *still in preparation*
8. Tritium fuel cycle
9. Sustained high performance tokamaks *still in preparation*

It should be stressed that these nine do not constitute a complete list all of initiatives that might merit inclusion in a strategic roadmap to fusion. But they are clearly important and of wide interest in the fusion community. Furthermore, while we have followed a process for vetting these white papers to the extent possible (see below), there has not been dedicated time for debating the details contained in them and therefore they should not be interpreted as expressing consensus community views.

Each of these papers was drafted by a small team of authors, then peer reviewed by two knowledgeable colleagues, then updated to accommodate review comments. The papers have been posted for a period of time on a public web site:

https://sites.google.com/site/usmfrstrategicdirections/strategic-element-white-papers where readers can download the papers and submit chits, also public. Authors were given some guidelines regarding content and length, but basically the aim was to convey to you the reasons why the element is exciting and the research agenda for it within a U.S. fusion energy strategy. The intent of these papers is to inform rather than to advocate and to be representable, as far as possible, as products of the community's strategic directions activity. We hope to submit the three remaining strategic element white papers and several working group reports as soon as possible.

Submitted by:

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Developing HTS Magnets for Fusion Applications

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1 Description of HTS Magnet Technology

All design concepts for power producing commercial fusion reactors rely on superconducting magnets for efficient and reliable production of the magnetic fields. High Temperature Superconductors (HTS) represent 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 could lead to significant acceleration of fusion energy development [1].

The advantages of HTS are that they can operate at very high magnetic field, high cryogenic temperature, high current densities, and larger mechanical stresses and strains compared to existing low-temperature superconductors (LTS). Each of these parameters is extremely important and constraining to a fusion reactor design. The expanded volume of operating space in these critical parameters opens a large space for enhanced magnet design. The most revolutionary aspect of HTS, particularly Rare-Earth Barium Copper Oxide (REBCO) superconductors, is their ability to maintain high current-carrying capability at very high magnetic fields. Historically, the maximum field on coil (limited by achievable current density in the superconductor) has been a primary driver for designing magnetic fusion devices. Consider a tokamak: HTS allows an increase in B_T over LTS technology from ~5.5 T to 10-12 T. (The field at the coil increases from ~12 T to 20 T).

2 Benefits to Fusion Program

High-field, high-temperature superconductors would enable a new generation of compact fusion experiments and power plants, dramatically speeding the development path and improving the overall attractiveness of fusion energy. Since magnet systems are the ultimate enabling technology, HTS could significantly enhance the performance and feasibility of almost any type of magnetic confinement or plasma physics device including, Spherical Tokamaks, Field Reverse Concepts, gas dynamic trap, magnetic mirrors levitating dipoles, etc. HTS can be used with any magnetic field configuration including 3-D shaped devices such stellarators and helical devices.

1. *Smaller burning plasma experiments:* High magnetic field at small size formed the basis of the US magnetic fusion program for 20 years prior to entering ITER. The science was successfully demonstrated on the Alcator devices and the US planned for flagship devices such as CIT, BPX, and FIRE. Community consensus was reached that a small high-field burning plasma could be successful with copper magnets [2,3,4]. HTS enables even smaller devices at higher field without the issues associated with copper magnets, accelerating fusion development.
2. *Performance vs. Cost*: The B^3 - B^4 dependence of well-known fusion parameters (power density, Lawson criterion) allows both high energy gain and power density in much smaller devices and may be crucial for fusion's eventual commercial realization.

3. *Operational Robustness*: High-field compact devices operate far from all intrinsic disruptive kink, pressure, density, and shaping limits, and use normalized plasma regimes (βN, H, q) already integrally demonstrated in present devices.
4. *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.
5. *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.

3 Current status of R&D and Readiness

All 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 three primary variables that define the critical surface are the critical temperature T_c (K), the upper critical magnetic field, B_{c2} (T), and the critical current density, J_c (A/mm^2). The critical surface of the HTS conductor gives an order of magnitude advantage in operating space over LTS conductors.

The thin-film type of superconductor is purchased from suppliers who produce it in thin strips instead of wires in automated thin-film processes which build up the constituent layers. Characteristics already achieved and well documented include:

1. **High field.** REBCO superconductor carries sufficient current density for magnet applications at fields up to 100 T [5]. It has recently been incorporated into solenoid magnets at fields of 35 T [6] and very recently over 40 T [7]. This surpasses the requirement of ~20 T on coil for very compact high-field tokamaks.
2. **High temperature operation.** REBCO, with critical temperature at 90 K can operate near 77K but performs much better when subcooled and thus high-field fusion and accelerator magnets often target 20-30 K or lower. The significance of the high temperature operation goes well beyond the thermodynamic advantages in 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 [8] eliminate cyrogens for low heat load devices [9] and allows the incorporation of resistive joints [10]. The high critical temperature and stability margin could also allow operating in a nuclear heating environment significantly higher than allowed in LTS magnets.
3. **High engineering current density.** REBCO has been incorporated into magnets at over 40 T at engineering current densities exceeding 1000 A/mm^2 [7]. 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 user magnets at fields of 32 T just being commissioned [11]. In fusion applications this leads to more room for structure in the magnet and nuclear radiation shielding.

4. **High strength and high modulus.** REBCO's primary constituent material (~50-90% by volume) is high strength nickel alloys or steels. The superconductor remains reversibly superconducting at tensile stresses over 600 MPa, which is comparable to the supporting steel structure, and strains up to 0.45% [12], factors of two improvement over LTS, thus enabling smaller magnets and more compact designs.
5. **No reaction process as part of winding.** Unlike LTS materials like Nb_3Sn where additional processing optimization and controls including high temperature, long duration heat treatments are required, REBCO conductor is ready for operation directly from the manufacturer and can be wound into final position in a single operation. This feature has the potential to simplify the manufacturing process and widen candidate magnet materials for electrical insulation and structural purposes.

4 National and International Programmatic Context

The U.S. has the opportunity to develop a world leading HTS superconducting magnet development program that will attract the best researchers from the U.S. scientific community, along with a strong industrial component. The U.S. took the initiative with development of the Cable-In-Conduit-Conductor (CICC) concept when the rest of the world was straggling along with outdated pool-cooled magnet technologies. The CICC concept was revolutionary and now dominates the international fusion magnet technology. The same will be true if the U.S. fusion program in collaboration with other US government scientific programs investments in and leads in development of HTS conductor and magnet technology. The U.S. community is also developing high-field solenoid and accelerator magnets using REBCO conductors and cables for other, non-fusion applications, which can be leveraged for the fusion magnet program. In fact, an excellent opportunity exists now to coordinate HTS technology development across multiple DOE-SC programs.

The time frame for HTS technology to be made ready for use in a next step device depends primarily on the funding rate. If it is desired to be used in an FNSF device then the tech nology development should be accelerated. It is most likely to be in time for any type of DEMO device, but the engineering and operation feasibility, as well as the economic value should be demonstrated on a much smaller device if one is needed in support of a DEMO reactor.

5 Possible 15-year U.S. Research Agenda

Operation of HTS materials has already been demonstrated for small-bore superconducting magnets at fields, current densities, stresses and JxB forces larger than required for fusion magnets [7]. Commercially available HTS conductor based on REBCO must be packaged into cable, suitable for large volume, high-field fusion magnet system. It then has to be incorporated into large bore magnets along with the engineering systems required to safely operate the magnet with significant stored energy. The challenges in this area are primarily electro-mechanical in nature involving integrated mechanical engineering of high strength structures and manufacturing and assembly processes. Many of these engineering decisions share strong similarity to the experience gained from LTS development [13][1]. It must be noted that existing tokamaks (e.g. C-Mod) and burning plasma designs (BPX, FIRE) have successfully

[1] "Taking advantage of a large experience gained in the course of a ten-year activity of supervision of CICC manufacture in industrial environment. [it] can be envisioned for further CICC development employing HTS material… opening completely new routes in the design of large-size, larger-current superconducting systems" [13]

dealt with similar mechanical stresses and doing so requires engineering discipline but not advances in materials or physics [14].

Recent studies indicate that HTS magnets could be made demountable [10] which would have large impact on fusion reactor operation due to improved ability to maintain the machine, increasing reliability and availability. Demountable coils require relatively short lengths of REBCO, effectively increasing conductor production yield, and lowering conductor cost. A strong synergy exists between the high-B, smaller size, and demountable coils, allowing for simplified and improved fusion engineering choices: e.g. immersion liquid blankets, and a modular vacuum vessel, which then becomes the only replacement item in the reactor.

6 Research Directions Beyond 15 Years

Once HTS conductor and magnet technology is developed through a phased and well-funded R&D program, the technology should be transferred to industry through one or more large-scale magnet prototypes, followed by series fabrication for a burning plasma physics experiment. At that point the industrial scale will be demonstrated, as has been the case for ITER magnet construction.

7 Critics' Objections and Advocates' Responses

REBCO materials are sufficiently advanced for next-step fusion applications. The technology has progressed out of the laboratory and into industrial production. Present performance of commercially produced REBCO tape is already sufficient for use in practical fusion experimental devices now. These conductors have been operated in conditions they would encounter in a fusion magnet in solenoids.

1. **Very high operating stresses at high magnetic field.** 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. REBCO does not require heat treatment and allows more flexible choice of structural materials. 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 is possible.
2. **Thin, flat tape geometry is not convenient for multi-strand, high current conductors**. In the present configuration of the superconductor as a flat tape, AC losses and current distribution, are non-ideal for fast transient or ac or pulsed operation. These can be improved with further R&D investment, e.g., the demonstrated striation process. However, current performance is sufficient for the TF coils of a tokamak where field is most important and operation can accommodate the increased loss.
3. **Quench detection is difficult in HTS magnets.** Although extremely stable in operation, quench detection is 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. Therefore, further R&D of 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 [15].
4. **Insufficient production piece length**. For a fusion magnet the typical cable lengths are 200-700 m. (ITER TF conductors are 700 m). REBCO with uniform critical current along the

length is regularly available in lengths of 100-300 m with continuous lengths approaching 1000 m. Lengths longer than this can easily be achieved with small resistance joints. This has no relevant performance penalty in a multi-tape cable, which easily share current, and the loss is insignificant compared with nuclear heat loads.

5. **Sufficient production volume.** Although REBCO conductors are in commercial production by at least 11 companies around the world, production rates are relatively low and product costs are high. Production rates need to be significantly increased and defect rates in the conductor reduced to increase yield and lower costs. This can be done with increased investment in capital equipment for production and improved conductor process control and quality insurance. A capital expenditure of order $10M is sufficient for most companies to make a factor of 2 or 4 increase in production and a factor of 2 decrease in cost per kA-m [16]. A fusion reactor requires ~5,000,000 kA-m of tape. Current single manufacturer annual production is approximately 1/50th of this but is scaling fast with doubling rates of a few years [16]. Needs from other magnet applications (e.g., HEP and medical) can help increase the production volume and reduce the conductor cost.
6. **Radiation resistance.** Numerous studies have been performed verifying REBCO has similar resistance to neutron damage as the leading LTS candidate [17, 18]. Until now, the radiation damage to organic insulators has been the life limiting component for the superconducting confinement magnets. If radiation damage to HTS materials is proven to be no worse than to Nb_3Sn, then this could actually be considered a positive attribute.
7. **High cost.** Current prices for REBCO are ~$100/kA-m which is a factor of 5-10 higher than price parity for Nb3Sn. Price by this figure of merit has decreased significantly year over year due to increased current carrying capability and process improvements. At production levels anticipated in market adoption or for a fusion device, REBCO manufacturer's and market researchers predict costs to reduce to price parity [16, 19]. Further, the superconductor itself represents a small fraction of the cost of the device, so spending here to shrink the device size is prudent.

8 References

1. Whyte, D.G., Minervini, J., LaBombard, B., Marmar, E., Bromberg, L., and Greenwald, M., "Smaller & Sooner: Exploiting High Magnetic Fields from New Superconductors for a More Attractive Fusion Energy Development Path," Journal of Fusion Energy (2016). https://link.springer.com/article/10.1007%2Fs10894-015-0050-1.
2. Summary of MFE study: Major conclusions from Snowmass 2002. https://fire.pppl.gov/snowmass02_exec_summ080402.pdf
3. FESAC Report on Burning Plasma Strategy, 2002 Executive summary, https://science.energy.gov/~/media/fes/fesac/pdf/2002/Austin_final_full_2002.pdf
4. National Research Council. 2004. *Burning Plasma: Bringing a Star to Earth*. Washington, DC: The National Academies Press. https://doi.org/10.17226/10816.
5. Y. Iwasa and SeungYong Hahn, "First-cut design of an all-superconducting 100-T direct current magnet", *Applied Physics Letters*, **103** 253507 (2013) https://doi.org/10.1063/1.4852596
6. U.P. Trociewitz, *et al.*, "35.4 T field generated using a layer--wound superconducting coil made of (RE) Ba2Cu3O7− x (RE= rare earth) coated conductor," *Appl. Phys. Lett*. **99**, 202506 (2011) https://doi.org/10.1063/1.3662963

7. SeungYong. Hahn, et al, "Overview of Recent Progress in No-Insulation REBCO Magnet," presented at MT25, Amsterdam, Sept 2017, to be published in IEEE Trans. On Appl. Supercond.
8. S Yoon, et al., "26 T 35 mm all $GdBa_2Cu_3O_{7-s}$ multi-width no-insulation superconducting magnet", Supercond. Sci. Technol. 29 04LT04 (2016) https://doi.org/10.1088/0953-2048/29/4/04LT04
9. S. Awaji, et al., "First performance test of a 25T cryogen-free superconducting magnet", Supercond. Sci. Technol. 30 065001 (2017) https://doi.org/10.1088/1361-6668/aa6676
10. F. Mangiarotti, "Design of demountable toroidal field coils with REBCO superconductors for a fusion reactor" MIT PhD Thesis (2016) https://dspace.mit.edu/handle/1721.1/103659
11. K. Kim, et al., "Design and performance estimation of a 35T 40m no-insulation all-REBCO user magnet", Supercond. Sci. Technol. 30 065008 (2017) https://doi.org/10.1088/1361-6668/aa6677
12. C. Barth, G. Mondonico, and C. Senatore, "Electro-mechanical properties of REBCO coated conductors from various industrial manufacturers at 77K, self-field and 4.2K, 19T", Supercond. Sci. Technol. 28 045011 (2015) https://doi.org/10.1088/0953-2048/28/4/045011
13. L. Muzzi, *et al.,* "Cable-in-conduit conductors: lessons from the recent past for future developments with low and high temperature superconductors", *Supercond. Sci. Technol.,* **28** 053001 (2015) https://doi.org/10.1088/0953-2048/28/5/053001
14. P. Titus, "Structural Design of High Field Tokamaks", PSFC report JA-03-9 (2003) http://library.psfc.mit.edu/catalog/reports/2000/03ja/03ja009/03ja009_full.pdf
15. Scurti, F., Ishmael, S., Flanagan, G., and Schwartz, J., "Quench detection for high temperature superconductor magnets: a novel technique based on Rayleigh-backscattering interrogated optical fibers," Superconductor Science and Technology, 29(3), (2016). http://iopscience.iop.org/article/10.1088/0953-2048/29/3/03LT01/meta
16. SuNAM presentation at 2014 Kyoto Workshop on HTS Magnet Technology for High Energy Physics – The 2nd Workshop on Accelerator Magnet in HTS (WAMHTS-2), see slides 22,23,39,41. https://indico.cern.ch/event/319762/
17. R. Prokopec, et al, "Suitability of coated conductors for fusion magnets in view of their radiation response", Supercond. Sci. Technol. 28 014005 (2014) https://doi.org/10.1088/0953-2048/28/1/014005
18. J. Emhofer, M. Eisterer and H. W. Weber, "Stress dependence of the critical currents in neutron irradiated (RE)BCO coated conductors", Supercond. Sci. Technol. 26 035009 (2013) https://doi.org/10.1088/0953-2048/26/3/035009
19. "Superconductors: Global Markets to 2022." (Note: Report must be purchased) https://www.researchandmarkets.com/research/6pdjj9/superconductors

Magnetic Configuration Research: A Foundation Element for the Development of Magnetic Fusion Energy

Configuration research is a primary driver of innovation and discovery and must remain a foundational element of any U.S. fusion energy program strategy. Research on the magnetic confinement of fusion plasmas tends to differentiate a handful of named configurations by their separate operating points and relative advantages in solving key challenges toward fusion power. In part, this results from technical differences that make it impractical to study multiple, optimized configurations in a single laboratory setting. There is a strong desire to answer, "What is the best configuration?". However, coordinated configuration research is essential to achieve true predictive fusion science, otherwise our knowledge is limited to the narrow ranges that define the tokamak configuration. Furthermore, the large technological gap between present-day experiments and a commercial fusion reactor implies there are challenges yet to be exposed. The tradeoffs represented in different magnetic configurations offer enormous potential to address these challenges as they arise, increasing the odds to achieve practical fusion power that is competitive in the future energy market. A reduction in the scope of configuration research only reduces the possibility to achieve fusion energy.

The prospects to achieve predictive fusion science are bright, given the combination of mature basic understanding and ever-increasing computational capabilities that help us understand the nonlinear nature of fusion plasma behavior. A suite of well-diagnosed configuration experiments is essential to validate plasma models. Predictive fusion science should embrace multiple configurations as close cousins, not just view them as competitors for fusion, since they represent particular combinations of the fundamental variables that govern magnetic confinement. Importantly, the base plasma models are universal so that the inevitable physics and technological tradeoffs can be understood, increasing the opportunity for innovation and allowing the possibility for optimized configurations yet to be discovered.

Configuration research is also one of the most important risk mitigation strategies for the development of fusion energy. The relative advantages and challenges associated with different configurations inherently broaden the possibilities for achieving a practical fusion power source. The U.S. strategic plan must have a spectrum of risk in its elements, including elements that foster the opportunity for disruptive innovation.

Overview of Magnetic Configuration Research: There are only two magnetic field geometries useful for confining a sustained fusion plasma: the torus and the magnetic mirror. Both geometries rely on fundamental momentum and energy conservation principles, and each takes advantage of the enormous anisotropy for collisional transport in a strong magnetic field. Non-tokamak toroidal configurations are sometimes referred to as "alternates", which reflects the relative degree in development and investment (not alternate fusion power). Stellarator configurations are second in development maturity and are discussed in a separate paper. Here we emphasize other toroidal configurations, mirror configurations, and "magneto-inertial" concepts that fall between sustained fusion plasmas and purely inertial fusion concepts.

The U.S. has been a leader in advancing multiple configurations, but the present funding from DOE-FES supports only the tokamak and stellarator configurations. Sharpened focus on tokamaks and stellarators is occurring worldwide as well. In recent years, the former U.S. Innovative Confinement Concepts (ICC) program on non-tokamak research generated a number of new experiments and innovative scientific results, but it was guided primarily by "is there something better than a tokamak?" and emphasized the distinction in concepts. A broader scientific goal to establish predictive science that spans multiple configurations was not a driving force. Most of these ICC experiments were located at universities, which created visible on-campus leadership opportunities in the highly competitive academic environment. Theory, modeling, and diagnostic capabilities were a modest part of the program, but they fell short of the requirements for a coordinated program with predictive science goals. Looking ahead, multi-configuration research creates a fantastic opportunity for the whole community. Well-diagnosed experiments with predictive science goals are feasible at intermediate scale, which helps create opportunity for multiple institutions. The challenge is greatest for larger facilities that must be constructed to validate fusion science close to burning plasma conditions. It will help greatly if we can say, reliably, what the investment cost needs to be and why.

Planning for toroidal configurations is most developed. The FESAC Priorities, Gaps, and Opportunities study[1] provided a thorough analysis of the technical gaps to fusion power with an emphasis on the tokamak configuration. This was followed by the FESAC Toroidal Alternates Panel[2], which considered the issues and opportunities for the stellarator, reversed field pinch, spheromak, and field-reversed configuration in the ITER era. The sum of these efforts was expanded on in the MFE ReNeW workshop report[3]. Note that mirror configurations and the Levitated Dipole were not included in these exercises. The science for pulsed, magnetized, high-energy-density configurations are discussed in a separate ReNeW workshop report[4].

In terms of gap closure, non-tokamak configurations offer the possibility to eliminate key gaps that occur for the tokamak configuration. All gaps must be addressed, and therefore gap elimination is high leverage in the development of fusion power. Since the elimination of any one gap often comes at the expense of widening other gaps or creating new gaps, there is no way to understand the fundamental tradeoffs inherent to the variables in configuration space unless the fusion program maintains research on multiple magnetic configurations.

Status, Benefits, and Near-Term Opportunities: Prior to the 1990's, the U.S. pursued fusion energy research on a variety of configurations at multiple scales. Experiments were located at national labs, universities, and in industry. Much of this research was terminated in FY 1993 to narrow the program on the tokamak configuration. Following the U.S.'s exit of ITER, there was a rebirth in "alternative concept" research in the late 1990's, which reinvigorated the non-tokamak program with new and under-explored configurations. However, configuration research again declined over the period 2010-2017, and today non-tokamak, non-stellarator support for fusion energy development by the U.S. DoE has all but ceased. The ARPA-E ALPHA program presently supports intermediate-density, magnetized, pulsed fusion concepts through fixed-term funding for technology transfer and collaboration with the private sector.

The non-tokamak, non-stellarator configurations[5] that define this "configuration research" strategic element are identified in Table 1. While this might seem a long list, these configurations derive from a few incontrovertible principles: symmetry, the need for a poloidal field in a torus, and a requirement to pre-heat a burning fusion plasma[6]. Key characteristics that distinguish these configurations from the tokamak are tabulated. Each of these characteristics represents opportunity to simplify and improve the vision for a magnetic fusion reactor. Equally important, the configurations, along with the tokamak and stellarator, provide a basis set needed to validate fusion science. The configurations include (a) those with open magnetic field topologies, e.g. the gas dynamic trap (axisymmetric mirror) and centrifugally confined mirror, (b) those with closed magnetic fields topologies having a moderate toroidal field, i.e., reversed field pinch and spheromak, and with little-to-no toroidal field, e.g. field reversed configuration, levitated dipole, and flow Z-pinch, and (c) pulsed concepts that rely on magnetic insulation and imploding liners, i.e., magneto-inertial fusion (a.k.a. magnetized target fusion).

Configuration	**Near-Unity Beta**	**More Compact**	**Reduced Field at Magnet**	**Auxiliary Heating Not Required**	**Simply- Connected Geometry**	**Advanced Fusion Fuels**	**Steady State (S) Pulsed (P)**
Gas Dynamic Trap (GDT)	◆				◆		S
Centrifugal Mirror		◆			◆		S
Reversed Field Pinch (RFP)		◆	◆	◆			S
Spheromak		◆	◆	◆	◆		S
Field-Reversed Configuration (FRC)	◆	◆	◆		◆	◆	S
Levitated Dipole	◆				◆	◆	S
Flow Z-pinch	◆	◆	◆	◆	◆	◆	P
Magneto-Inertial Fusion (MIF)	◆	◆	◆	◆	◆	◆	P

Table 1. Advantages of non-tokamak, non-stellarator magnetic fusion configurations. The last column identifies inherently pulsed (P) or a steady-state sustainment scenario is identified (S).

Listed below in a common format are key benefits, fusion science highlights, connections to gaps, world program context, status, and next steps for research on each configuration. Given that the support for non-tokamak, non-stellarator research has been drastically reduced, there is an immediate need to assess and rejuvenate magnetic configuration research. The scope of and coordination between "configuration research" and other strategic elements including tokamak configurations, stellarator configurations, theory, materials, fusion technology, etc. must be formulated in a complete strategic plan for the U.S. fusion energy program.

GAS DYNAMIC TRAP (GDT)[7]: An axisymmetric mirror defined by a long mirror-to-mirror distance (compared to ion mean free path) and high mirror ratio. MHD stability is provided by plasma escaping through the mirror throat into a region of good curvature[8].

Key benefits: Simple engineering; steady state operation; no plasma current

Fusion science highlights: A short-pulse (5 ms) experiment at modest magnetic field and heating beam energy (0.3-15T; 25keV) has demonstrated MHD stability at β~60% with classical fast ion behavior and an electron temperature up to 0.9 keV[9], which meet the requirements of a designed GDT-based fusion neutron source that operates at higher field, beam ENERGY and in steady state.

Connections to gaps: A next-step GDT will press the state of the art in steady state operation, will present the ideal test bed for new high temperature superconducting magnets (simple, small bore, and axisymmetric), and its potential as a fusion neutron source addresses major gaps for materials and component development.

Gap elimination: The axisymmetric mirror has viable reactor scenarios (both tandem[10] and GDT[11] variants) that feature a stable, plasma-current free equilibrium that cannot disrupt. The primary gap elimination is through creation of a fusion neutron source.

World program: Only one GDT experiment has been built to date, in Russia. More broadly, tandem mirrors are used to study fusion science at Gamma10 (Japan) and the new KMAX (China, under construction). There are also several material-plasma interaction experiments in the mirror configuration, including Proto-MPEX/ MPEX (USA), JULE-SIM (Germany) and PLAMIS (South Korea).

Status and next steps (<15 year): The GDT at BINP has a planned upgrade to use a multi-mirror end cell to improve axial confinement and the corresponding reactor scenario. The next step in this path is to create a high flux fusion neutron source[12]. Construction of a proof-of-principle steady state, fully superconducting, high field GDT with pulsed heating systems including 80keV deuterium beam injection can begin immediately (cost estimate $50M). It must confirm low secondary electron emission from the end cells (for electron thermal confinement) and low neutral pressure in the central chamber (for fast ion confinement). Implementation of steady state heating upgrades (totaling 50 MW) and DT operation can commence in about 10 years.

CENTRIFUGAL MIRROR[13]: An axisymmetric magnetic mirror configuration is rotated azimuthally at supersonic speeds. The radial centrifugal force confines plasma along the field, closing out loss cones. Velocity shear suppresses flute interchange instability. Pastukhov loss theory predicts Lawson conditions at Mach 6. 3D MHD simulations show confined toroidal plasma that is MHD stable due to velocity shear (V′). This is an underexplored concept.

Key benefits: Simple geometry, steady state, and no abrupt terminations. The axial length is comparable to toroidal geometry circumference. V′ shear is large enough to also suppress microturbulence, resulting in classical cross-field transport (no neoclassical transport enhancements). Non-conventional, physics-based concept makes a study of this novel system highly attractive academically.

Fusion science highlights: MCX experiment[14,15] (2000-2010) was rotated at supersonic speeds and showed quiescent confinement at Mach 1-3, with a 12-fold drop in density axially. Key parameters: $n \sim 3\times10^{20}/m^3$, $T \sim 40$eV, $\tau \sim 0.4$ms, mirror ratio < 8, peak field 1.2 T. Cost < 0.5M/yr.

Connections to gaps: Broaden predictive understanding of transport and scaling in V′ shear dominated plasmas[16], and develop the plasma-material interface in a poloidal-field geometry, e.g., liquid wall concepts.

Gap elimination: Disruption-free sustained plasma; greatly decreased axial length compared with a static mirror system

World program: Some similarity with the Novosibirsk GDT experiment, which is an elongated mirror rotated subsonic by tailored electrostatic biasing, thus providing V′ shear to stabilize the flute mode.

Status and next steps (<15 year): Currently, there is no centrifugally confined plasma research. The centrifugal concept is in infancy. Next steps after MCX are exploration beyond the neutral-dominated regime (possibly using Li pumping), driving rotation by NBI, test if V′ shear suppresses drift modes and if this implies classical cross-B transport. Scaling studies point to high B operation with high mirror ratio, B^m. Over a 10-year horizon, costs would be <$2M/yr. The centrifugal mirror could possibly be interesting to private venture, but a concept this nascent would require several value-added, high-risk phases. There is synergy with proposals for a GDT-based neutron source. Key challenges: V′ shear must suppress flutes at high Reynolds #s, the atomic speed barrier must be overcome, and insulators must sustain 1-10 MV/m, possibly with flared B fields.

REVERSED FIELD PINCH (RFP)[17]: A toroidal, axisymmetric configuration with a highly sheared magnetic field generated primarily by plasma current rather than external coils

Key benefits: Ohmic ignition[18] and inductive steady-state[19,20] are possible if the tokamak-like confinement achieved in present experiments[21] endures at fusion conditions. Heating and sustainment are provided by robust, reliable, axisymmetric transformers that do not require perforations in vessel materials surrounding the plasma. The magnetic field strength at coils is minimized, and high beta is demonstrated[22].

Fusion science highlights: Seminal development of active MHD control[23,24,25], validation of nonlinear extended MHD models[26] of fusion plasmas[27], magnetic self-organization[28,29], and demonstration of the classical confinement of energetic ions in a toroidal plasma[30].

Connections to gaps: Broaden predictive understanding of transport and scaling associated with microturbulence and multi-scale interactions[31], develop robust mode control[32,33], demonstrate inductive steady-state sustainment (oscillating field current drive[34]), develop the plasma-material interface in a poloidal-field-dominated geometry, e.g., liquid wall concepts.

Gap elimination: Obviate auxiliary heating by rf or neutral beam injection, greatly simplifying a reactor first-wall and enhancing overall maintainability and reliability; minimize the magnetic field at magnets

World program: Five experiments: MST (I_p =0.6 MA, R/a=1.5/0.5, *USA*), RFX-mod (I_p =2MA, R/a=2.0/0.4, *Italy*), KTX (I_p <0.5 MA, R/a=1.4/0.4, *China*), Extrap-T2R (I_p <0.3MA, R/a=1.24/0.18, *Sweden*), RELAX (I_p =0.125MA, R/a=0.51/0.25, *Japan*)

Status and next steps (<15 year): Federal funding for RFP fusion research on MST is being terminated, undermining U.S. leadership in RFP research. An upgrade to the shell, boundary, and control coils on RFX-mod has recently been approved. The KTX program is new, with emphasis on completing power supplies and diagnostics. Resolving key gaps for the RFP requires a larger, high-current device with I_p ≥4MA, as described in the FESAC Toroidal Alternates Panel report. This facility would address understanding transport mechanisms, confinement scaling, and steady-state inductive current drive. It would begin the development of integrated boundary control. The estimated cost is several $100M and could be staged to reduce risk.

SPHEROMAK[35,36]: A toroidal, axisymmetric plasma configuration contained within a simply-connected vacuum chamber with no externally applied toroidal magnetic flux[37]

Key benefits: Sufficiently large plasma currents allow for Ohmic ignition provided that sufficient energy confinement quality is achieved at fusion conditions. Reduction of technological complexity due to the elimination of the toroidal field coil set and central solenoid may allow for reductions in fusion reactor costs. Modest peak magnetic field on coil allows for flexibility in superconducting material for the poloidal field coil set which is required for steady-state operation.

Fusion science highlights: Platform for study of plasma self-organization, magnetic relaxation and magnetohydrodynamic (MHD) dynamos, verification and validation (V&V) of nonlinear, non-ideal MHD models for fusion plasmas[38], study of helicity injection current drive[39].

Connections to gaps: Study of advanced, energy-efficient current drive to address gap in magnetic configuration sustainment, greater degrees of plasma current profile control. V&V of nonlinear, non-ideal MHD models on small-scale spheromak experiments to enable predictive modeling of fusion systems. Simpler geometry allows for easier optimization of blanket assemblies and first-wall power loadings for eventual fusion reactor systems.

Gap Elimination: Usage of high plasma current magnetic configuration with energy efficient current drive may allow for Ohmic heating to ignition, eliminating the need for auxiliary heating systems. Reducing overall fusion system complexity to enable easier maintainability and potentially lower capital and maintenance costs to enable economic competitiveness.

World Program: HIT-SI3 (R =0.33 m, R/a =1.4, I_p ~30-90 kA, T <100 eV, B ~30 mT, *U. Washington*), SSX (R =0.25 m, R/a =1.2, I_p ~30 kA, T_i =40 eV, B ~100 mT, *Swarthmore*), FAMU-STPX (I_p ~600 kA, T ~300 eV, *FAMU*), TS-4 (R =0.5 m, R/a =1.5, I_p ~30-100 kA, B ~100 mT, *U. Tokyo*), and the Caltech Spheromak Experiment.

Status and next steps: Federal funding for spheromak experiments is small and insecure across all agencies (DOE OFES, ARPA-E, DOE/NSF Partnership). Investment in a new, upgraded sustained-spheromak facility should be made to enable both mainline and spheromak-specific gaps to be resolved. Transient spheromak experiments (e.g. SSPX at LLNL) have produced transient spheromaks with peak electron temperatures between 500-600 eV[40]. A new sustained-spheromak experiment would help address scaling of advanced, power efficient current drive methods to larger, higher temperature plasmas with sufficient energy confinement quality. Additionally, this facility would provide a greater separation of timescales of plasma dynamics at higher Lundquist number ($S = Lv_A/\eta$). A national sustained spheromak program with $5-15M/yr would greatly improve spheromak R&D progress and gap resolution efforts.

FIELD-REVERSED CONFIGURATION (FRC): A toroidal, axisymmetric, extremely high beta configuration in a simply-connected geometry with poloidal magnetic field generated by plasma current[41,42]

Key benefits: Compact toroidal system with (i) simple axisymmetric geometry that facilitates a translation along a central axis, (ii) extremely high β and associated economic attractiveness, (iii) unrestricted natural divertor system facilitating heat removal and exhaust engineering that could enable direct-energy conversion, and (iv) potential for advanced, aneutronic fuel cycle

Fusion science highlights: Demonstration of various reliable FRC formations such as field-reversed theta pinch (FRTP), rotating magnetic field (RMF) driven, FRC collisional merging, and counter-helicity spheromak merging. Demonstration of macroscopically stable, hot plasma sustainment up to 5+ ms via high-power neutral-beam injection (NBI) whose fast ions are classically confined in an FRC, which also exhibits a favorable energy confinement scaling that is proportional to positive power of electron temperature (unlike Bohm scaling)[43].

Connections to gaps: Study of efficient plasma heating (by NBI, RF, compression, etc.), current drive, and stability / plasma control. Broaden understanding of transport and scaling inside and outside of FRC separatrix. Demonstrate steady-state plasma sustainment or pulsed magnetic/inductive plasma compression for breakeven (magnetized target fusion).

Gap elimination: Eliminate extreme material challenges via aneutronic fuel cycle; eliminate linked-magnet constraints to improve system maintainability and reliability

***World program**:* Ten experiments: C-2U/C-2W (FRTP/FRC merging/NBI, USA), PFRC (RMF, USA), MSX (FRTP, USA), NUCTE/FAT (FRTP/FRC merging, Japan), IPA/Grande (FRTP/FRC merging/MTF, USA), TS-3/TS-4 (Spheromak merging, Japan), MRX/FLARE

(Spheromak merging, USA), SSX (Spheromak merging, USA), KMAX (FRTP/FRC merging, China), Yingguang-I (FRTP/MTF, China)

Status and next steps (<15 year): Two different FRC-based fusion approaches are currently underway in the U.S. and Asia by private/government funding: beam-driven FRC for steady-state operation and pulsed-compressional FRC for MTF. For the beam-driven FRC, near-term objective is to demonstrate steady-state high temperature FRCs by high power NBI and other auxiliary heating; while, for MTF approach, effective high-pulsed compressional magnetic field (up to ~50 T) will be designed and applied to achieve high temperature/density fusion condition. Both of which require device upgrade / scale-up with some R&D; however, experimental span of FRC research can be relatively short / aggressive because of system simplicity.

LEVITATED DIPOLE: Toroidal configuration with a purely poloidal magnetic field generated by a single coil suspended within the plasma by magnetic levitation[44]. The concept was motivated by the understanding gained from satellite observations of magnetospheric plasmas and advances in high-field superconducting magnets.

Key benefits: Provides steady-state, disruption-free, and near-unity beta plasma confinement. It is most relevant for use with aneutronic fusion fuel cycles to accommodate a floating coil within the plasma. The dipole's inherently larger particle transport relative to heat transport bolsters tritium-suppressed D-D fusion, in particular.

Fusion science highlights: Demonstrated robust steady-state operation with good plasma confinement. Observation of inward turbulent pinch[45,46,47]; concept driver for advanced-fuel fusion reactor development[48]; concept driver for fusion space propulsion[49]

Connections to gaps: Broadens understanding of self-organized plasma turbulence, motivates the development of high-field, high-performance magnets, stimulates fusion plasma conditions with advanced fuel cycle

Gap elimination: Simple plasma sustainment that eliminates current disruptions; aneutronic fuel cycle eliminates many fusion material challenges; inherent plasma expansion simplifies the plasma-material interactions, including the interface for auxiliary heating sources

World program: The LDX[50] (MIT) was the largest dipole experiment with a 0.66 m diameter, 1.2 MA superconducting (Nb_3Sn) coil. The RT-1 device (U. Tokyo) has a 0.50 m diameter 0.25 MA high-Tc Bi-2223 superconducting coil. Steady-state discharges are maintained with 10-50 kW of ECRH. Recently, low-power ICRH experiments have begun at RT-1.

Status and next steps (<15 year): Laboratory experimental tests of the dipole concept with high-power heating must be conducted to verify confinement properties at fusion-relevant conditions. Several experiments have been proposed but not yet funded. These projects have total project costs ranging between $6M USD and $25M USD. A fusion-performance experiment requires a device that can be built using existing superconducting magnet technology in a scaled experiment, e.g., a 4 m diameter, 15 MA coil coupled with 10 MW of auxiliary heating could achieve Q(DT) ≈1. The required containment vessel is large but uses simple, low-cost technology.

FLOW Z-PINCH[51,52]: A linear configuration relying solely on sheared axial flows to provide plasma stability

Key benefits: No external magnetic field coils and purely azimuthal magnetic fields leads to $\langle\beta\rangle = 100\%$ with perpendicular transport towards any material structure. Resulting high energy densities naturally lead to a compact and low-cost device.

Fusion science highlights: Demonstrated high performance sheared-flow-stabilized Z-pinch plasmas[53,54,55] with quiescent lifetimes greater than 1000 V_A and with plasma parameters that are $n_e \approx 2\times10^{23}$ m^{-3}, $\tau \approx 50$ μs, and $T_e \approx 1$ keV[56]. Produced sustained 5-10 μs pulses of DD neutrons, suggesting thermonuclear origin.

Connections to gaps: Investigate sheared-flow stabilization in a simple configuration with potential applications to other configurations. Develop high beta concepts with no magnetic field coils. Study plasma-material interactions, including liquid metal walls.

Gap elimination: High beta operation avails advanced fusion fuels. No external field coil and linear configuration greatly simplify fusion core design.

World program: UW-Seattle/LLNL experiments: ZaP, ZaP-HD, FuZE. Previous experiments of continuous flow pinch and quasi-steady-state plasma accelerator existed at LANL[57] and Kurchatov Institute[58].

Status and next steps (<15 year): Federal funding (ARPA-E) for fusion research on the flow Z-pinch is scheduled to terminate August 2018. Next steps include demonstrating shear flow stabilization of the Z-pinch with increasing plasma current and driving to fusion-grade plasmas, designing plasma-facing electrodes, and researching plasma interactions with liquid metal walls.

MAGNETO-INERTIAL FUSION (MIF), a.k.a. MAGNETIZED TARGET FUSION (MTF)[52]: This is a class[59,60] of pulsed, imploding fusion concepts, i.e., liner compression of a magnetized plasma[61,62,63,64] utilizing magnetic field to reduce thermal transport and enhance alpha-particle deposition in the stagnated fusion plasma.

Key benefits: Intermediate-density MIF optimizes the combination of required stored energy and heating power to achieve Lawson conditions[65], thus potentially offering a lower-cost, faster development path to economical fusion power. Key benefits are (1) use of low-cost pulsed power, (2) heating via compression, and (3) compatibility with a thick liquid blanket.

Fusion science highlights: Simple, low-cost means to access magnetized high-energy-density (HED) regimes[4], enabling advances in fundamental plasma and HED physics.

Connections to gaps: Because MIF has many challenges orthogonal to those of MFE, MIF represents an important piece of a diverse portfolio to mitigate risk in fusion-energy development. MIF also shares common challenges with MFE, e.g., power extraction (G-10, G-11, G-12), predictive modeling (G-1, G-6), measurement (G-3), and RAMI (G-14, G-15).

Gap elimination: MIF, by virtue of its pulsed nature, elimination of auxiliary heating, and likely use of a thick, flowing liquid blanket, strongly mitigates many Greenwald et al.[1] gaps (G-2, G-4, G-5, G-7, G-8, G-13). There are of course new gaps, e.g., robust, repetitive pulsed power.

World program: Z machine (e.g., MagLIF), Russian MAGO, Chinese solid-liner compression of FRC and interest in MagLIF and other MIF concepts, ARPA-E ALPHA program (early-stage development of several MIF variants), and magnetized ICF (LLE/Rochester and NIF).

Status and next steps (<15 year): Continued NNSA funding will allow timely, further studies of crucial physics at fusion-relevant densities and temperatures on the Z machine or other NNSA facilities, benefitting MIF development but not direct support of its fusion energy potential. A combination of ARPA-E follow-on funding and/or reinstatement of support for MIF within FES could allow the most promising CE-level MIF concepts, presently supported by ARPA-E, to possibly progress to POP- and then PE-level performance, which should be a primary objective over the next 10 years. The goal should be to put us on a path to enable DEMO-level performance in 15-20 years. A budget of ~$10M increasing to $20M/year in the next 3-5 years would allow meaningful and timely progress toward POP performance for several of the ongoing CE efforts.

Programmatic Implications: History shows that the strong drive to identify "the best" configuration makes it difficult to coordinate research on different configurations. The loss, rebirth, and subsequent loss of non-tokamak research correlates with the challenge in realizing facilities on the scale of ITER. While it is important to expose the benefits of different configurations, since this may in fact be essential to realize fusion energy, the maturity of fusion science allows the possibility to understand and predict fusion plasma behavior across configuration boundaries. This is a programmatic vision that demands greater coordination and less institutional identity associated with any one configuration. To succeed, theory and computation must be made as universal as possible within the bounds defined by the principles governing fusion plasma confinement and heating. There is an opportunity to organize experimental facilities with greater national ownership. If an appropriate strategy is adopted, the program can support universities, national laboratories, and coordination with the growing private sector's investment in fusion energy development. Given the wide range in relative maturity, experimental facilities at small and intermediate scale are appropriate for many of the next steps described above. New facilities at multiple scales will generate scientific interest and allow rapid progress that complements the inherent longer timescale associated with projects like ITER. The U.S. fusion program needs to regain trust, and successfully completing a number of projects on different scales will help rebuild this trust.

Critics' Concerns and Advocates' Responses:
Concern: The tokamak configuration clearly performs the best. Why do we need to investigate configurations that do not perform as well?
Response: A fusion reactor does not yet exist. It is difficult to prove that any configuration will or will not work. Given fusion's importance, we need risk mitigation strategies, including validated science that reliably determines what is possible or not. Configuration research is fundamental to this science and to overall risk mitigation of fusion energy development.

Concern: We cannot afford research on configurations other than the mainline.
Response: We need arguments that can grow support for fusion energy. Configuration research is a fundamental approach to fusion energy that everyone can embrace for the essential science it provides and for its potential to enable robust, simple, and smaller reactor concepts. The required resources are not large for every element in a balanced portfolio.

Concern: Alternate configurations might help optimize second generation fusion power, but we should concentrate on the tokamak now so that fusion's importance is demonstrated as quickly as possible.
Response: By any metric, the world's fusion programs are already very concentrated on the tokamak and have been for decades. Unless we research alternatives, a second-generation reactor cannot be based on a different configuration. There are legitimate concerns that the present tokamak path will not lead to competitive fusion energy. Developing the scientific and technical understanding that produces economically viable fusion reactors should be a priority, so that a first-generation, non-competitive reactor does not eliminate fusion as a future energy source.

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[1] "*Priorities, Gaps and Opportunities: Towards a Long-Range Strategic Plan for Magnetic Fusion Energy*", 2007, https://science.energy.gov/~/media/fes/fesac/pdf/2007/Fesac_planning_report.pdf (2007)

[2] "*Advancing the Science of Alternate Toroidal Magnetic Fusion Concepts*", 2008, https://science.energy.gov/~/media/fes/fesac/pdf/2008/Toroidal_alternates_panel_report.pdf

[3] "*Magnetic Fusion Energy Sciences ReNeW Workshop Report*," 2009, https://science.energy.gov/~/media/fes/pdf/workshop-reports/Res_needs_mag_fusion_report_june_2009.pdf

[4] "*Basic Research Needs for High Energy Density Laboratory Physics*", 2010, Chapter 2, https://science.energy.gov/~/media/fes/pdf/workshop-reports/Hedlp_brn_workshop_report_oct_2010.pdf

[5] Brown et al. Madison Workshop White Paper, 2017

[6] Sarff. Madison Workshop White Paper, 2017

[7] Anderson et al. Madison Workshop White Paper, 2017

[8] V.V. Mirnov and D. Ryutov, *Soviet Technical Physics Letters* **5,** 279 (1979)

[9] P.A. Bagryansky et al., *Phys. Rev. Lett*. **114** 205001 (2015)

[10] T.K. Fowler, R.W. Moir, and T.C. Simonen, *Nucl. Fusion* **57** 056014 (2017)

[11] A.A. Ivanov and V.V. Prikhodko, *Physics Uspekhi* **60** 5 (2017)

[12] D.D. Ryutov, *Plasma Phys. Control. Fusion* **32** 11 (1990); P.A. Bagryansky et al. *Fus. Eng. Des*. **70** 13 (2004); T.C. Simonen et al., *Nucl. Fusion* **53** 6 (2013); A.V. Anikeev et al., *Materials* **8** 12 (2015)

[13] Hassam et al. Madison Workshop White Paper, 2017

[14] R.F. Ellis et al., *Phys. Plasmas* **12**, 055704 (2005)

[15] A.B. Hassam, *Phys. Fluids B* **3**, 485 (1992); W.C. Young et al., *Phys. Plasmas* **18**, 112505 (2011)

[16] Y.M. Huang and A.B. Hassam, *Phys. Rev. Lett*. **87**, 5002 (2001)

[17] McCollam et al. Madison Workshop White Paper, 2017

[18] J. P. Christiansen and K.V. Roberts, *Nucl. Fusion* **22**, 77 (1982)

[19] M.K. Bevir, C.G. Gimblett, G. Miller, *Phys. Fluids* **28**, 1826 (1985)

[20] F. Ebrahimi et al., *Phys. Plasmas* **10**, 999 (2003)

[21] J.S. Sarff et al., *Plasma Phys. Control. Fusion* **45**, A457-470 (2003)

[22] M.D. Wyman et al., *Phys. Plasmas* **15**, 010701 (2008)

[23] B. Alper, *Phys. Fluids B* **2**, 1338 (1990)

[24] P. R. Brunsell et al., *Phys. Rev. Lett.* **93**, 225001 (2004)

[25] R. Paccagnella et al., *Phys. Rev. Lett.* **97**, 075001 (2006)

[26] C. R. Sovinec et al., *J. Comput. Phys.* **229**, 5803 (2010)

[27] J.R. King et al., *Phys. Plasmas* **19**, 055905 (2012); J.P. Sauppe and C.R. Sovinec, *Phys. Plasmas* **24**, 056107 (2017)

[28] Y.L. Ho and G.G. Craddock, *Phys. Fluids B* **3**, 721 (1991)

[29] R. Lorenzini et al., *Nature Phys*. **5**, 570 (2009)

[30] G. Fiksel et al., *Phys. Rev. Lett.* **95**, 125001 (2005); J.K. Anderson et al., *Phys. Plasmas* **20**, 056102 (2013)

[31] Z.R. Williams et al., *Phys. Plasmas* **24**, 122309 (2017); J.R. Duff et al., *Phys. Plasmas* **25**, 010701 (2018)

[32] P. Zanca, *Nucl. Fusion* **47**, 1425 (2007)

[33] K.E.J. Olofsson et al., *Plasma Phys. Control. Fusion* **52**, 104005 (2010)

[34] K. J. McCollam et al., *Phys. Plasmas* **17**, 082506 (2010)

[35] Jarboe et al. Madison Workshop White Paper, 2017

[36] Sutherland et al. Madison Workshop White Paper, 2017
[37] T.R. Jarboe et al., *Nucl. Fusion,* **52**, 083017 (2012)
[38] K.D. Morgan et al., *Phys. Plasmas* **24**, 122510 (2017)
[39] A.C. Hossack et al., *Nucl. Fusion* **57**, 076026 (2017)
[40] B. Hudson et al., *Phys. Plasmas* **15**, 056112 (2008)
[41] M. W. Binderbauer, et al., *Phys. Rev. Lett*. **105**, 045003 (2010)
[42] J.A. Romero et al., *Nature Comm.* **9**, 691 (2018)
[43] H. Gota et al., *Nucl. Fusion* **57**, 116021 (2017)
[44] D.T. Garnier et al., *Nucl. Fusion* **49** 055023 (2009)
[45] Boxer et al., *Nature Phys.*, **207** (2010)
[46] Garnier et al., *Phys. Plasmas* **24**, 012506 (2017)
[47] Saitoh et al., *J Fusion Energy* **29**, 553 (2010)
[48] Kesner et al., *Nucl. Fusion* **44**, 193-203 (2004)
[49] Teller et al., *Fusion Tech*. **22**. 82 (1992)
[50] Garnier et al., *Fusion Eng. Des*. **81**, 2371 (2006)
[51] U. Shumlak et al., *Nucl. Fusion* **49**, 075039 (2009)
[52] Hsu et al. Madison Workshop White Paper, 2017
[53] U. Shumlak et al., *Phys. Rev. Lett.* **87**, 205005 (2001)
[54] U. Shumlak et al., *Phys. Plasmas* **10**, 1683 (2003)
[55] R.P. Golingo, U. Shumlak, and B.A. Nelson, *Phys. Plasmas* **12**, 062505 (2005)
[56] U. Shumlak et al., *Phys. Plasmas* **24**, 055702 (2017)
[57] J. Marshall, *Phys. Fluids* **3**, 134 (1960).
[58] A.I. Morozov, *Sov. J Plasma Phys.* **16**, 69 (1990)
[59] R.C. Kirkpatrick, I.R. Lindemuth, M.S. Ward, *Fusion Tech*. **27**, 201 (1995)
[60] I.R. Lindemuth, *Phys. Plasmas* **24**, 055602 (2017)
[61] J. H. Degnan et al., *Nucl. Fus*. **53**, 093003 (2013)
[62] S.C. Hsu et al., *IEEE Trans. Plasma Sci.* **40,** 1287–1298 (2012)
[63] M. R. Gomez et al., *Phys. Rev. Lett.* **113**, 155003 (2014)
[64] M. Laberge et al., 25th Symposium on Fusion Engineering (SOFE), San Francisco, CA, 2013; doi: 10.1109/SOFE.2013.6635495
[65] I.R. Lindemuth and R.E. Siemon, *Amer. J. Phys*. **77**, 407 (2009)

Quasi-Symmetric Stellarators as a Strategic Element in the US Fusion Energy Research Plan

Quasi-Symmetric Stellarator Research

The stellarator offers ready solutions to critical challenges for toroidal confinement fusion: it provides a steady-state, major-disruption free reactor concept with minimal recirculating power requirements for plasma sustainment. The stellarator concept has undergone a rebirth in recent years as a result of major advances in theoretical understanding, the advent of enhanced computational capabilities, and new experimental research that have substantially furthered our predictive understanding of many aspects of three dimensional magnetic confinement systems. The configurational flexibility afforded by allowing 3D shaping opens up new possible confinement regimes and optimization opportunities. This 3D magnetic design freedom allows us to test our understanding of symmetry effects on plasma confinement and to produce physics-optimized fusion configurations not possible under the constraints of axisymmetry. Historically, classical stellarators have lagged behind tokamaks in performance due to their relatively poor neoclassical confinement at low collisionality. Groundbreaking optimized designs from the 1980's, such as the W7-AS [1] in Garching, Germany and then the quasi-helically symmetric HSX [2] device in Madison, Wisconsin demonstrated that neoclassical optimization improves the thermal confinement of stellarators up to a level similar to tokamaks. The success of the initial 2016 and 2017 campaigns on the W7-X [3,4] stellarator at IPP in Greifswald, Germany, the world's first large scale neoclassically-optimized stellarator, is the most recent advance on the path to a 3D solution to the problem of maintaining fusion in steady-state. More is needed however. While the LHD stellarator in Japan and W7-X are demonstrating various advantages of the stellarator approach, *neither will explore the possible advantages of quasi-symmetry in stellarators, which is what this Strategic Element proposes.* The virtues of this Strategic Element are detailed below, and its implementation will lead to a faster, more attractive path to fusion energy realized via the stellarator concept.

Benefits of Quasi-Symmetric Stellarator Research

The fact that the stellarator generates most of its rotational transform from external coils yields significant fusion benefits. These benefits include a magnetic configuration that is inherently steady state, without the need for significant current drive or current profile control. This leads to a reactor with low recirculating power allowing an easier attainment of net electricity output [5]. Stellarators provide the ability to serve as a test bed for physics issues that arise from long pulse operation given their steady-state nature, such as plasma material interaction and impurity control. The external control of the plasma configuration, given the rigid magnetic cage provided by a field from external coils, implies that loss of equilibrium due to plasma instability and major disruption is avoided. Given the lack of major disruptions, generation of their associated runaway electrons is not of concern as in tokamaks. In addition, the stellarator has a radiative density limit set by the available heating power [6], thus allowing high density operation not constrained by Greenwald limit type phenomena [7]. This high density operation has associated benefits in terms of decreased thermalization times for energetic particles and improved energetic particle stability as well as being very desirable for divertor operation. In terms of divertor operation, long connection lengths in the 3D edge plasma can yield wider scrape-off layer (SOL) widths and heat deposition profiles. The broad range of edge magnetic configuration properties provides flexibility for edge/SOL optimization in future devices. Finally, the external control of the magnetic configuration inherent to the stellarator concept allows for more confidence in attaining the final plasma configuration based on the computational design.

Current Status of Quasi-Symmetric Stellarator Research

There are topical areas in quasi-symmetric (QS) stellarator physics for which research gaps exist [8]. The US stellarator community is well positioned to address many of these gaps. Since the design of the HSX, W7-X and NCSX configurations, there has been considerable theoretical and computational activity in these areas. These advances can be employed to embark on a new era of QS stellarator physics research with an expanded theory/computational effort, focused design activity and new experimental facilities. The main technical challenges for the existing stellarator program include:

- There has not yet been an experimental demonstration of adequate energetic ion confinement in any stellarator suitable for a reactor. Promising ideas for stellarator optimization have not been adequately explored (see e.g. [9,10,11,12]).
- A new opportunity in stellarator optimization is use of 3D shaping to affect turbulent transport (see e.g. [13,14]).
- There are unexplained low impurity regimes observed in experiments (see e.g. [15,16]).
- Divertor design is not a closed issue in the stellarator or tokamak, but potential solutions are emerging (see e.g. [17,18]). Methods for automating divertor design should be pursued.
- QS optimization allows for the presence of large flows that could benefit various confinement properties. There is a need to assess the virtues of these flows in high performance stellarators.
- It is a challenge to find reactor relevant coil designs that enable improvements in plasma confinement. However promising new coil design tools are developing (see e.g. [19,20]).

Programmatic Context

The world stellarator program is currently dominated by the large superconducting-coil facilities LHD (Japan) and W7-X (Germany). The U.S. remains active in international experimental stellarator research through a robust partnership with W7-X and targeted collaborations with LHD, both involving multiple U.S. institutions. While these programs have and will demonstrate some advantages of the stellarator approach, neither of their design approaches scale to attractive reactors. In particular, energetic ion confinement may not be adequately addressed and they only explore two of three major divertor concepts that have been identified [8]. While HSX has demonstrated the benefits of QS for electron transport, there are significant issues (ion transport, turbulence optimization, divertor design, flow physics) that need to be resolved to realize a stellarator vision for DEMO. So far, only China is pursing a QS stellarator experimental program with a budgetary commitment at the concept exploration scale. In order to fully evaluate and exploit the potential of QS stellarators for fusion, U.S. leadership and a robust, broad-based US program are required.

Proposed 15-year U.S. Research Agenda

The STELLCON report [8] outlines an approximately 20-year research plan that is summarized by the timeline in Figure 1. There are 3 basic elements of the plan:

An optimization and design initiative: A national stellarator design project should be established as soon as possible to guide the design of the two proposed new experimental facilities. A similar joint effort launched in the late 1990's produced large advances in stellarator analysis and design tools [21], deepened the understanding of QS stellarators, and produced two machine designs, for NCSX and for QPS. In the intervening years there have been advances in design tools, providing new capabilities to improve coil designs and reduce turbulent transport, resulting in better designs. At the same time, the design goals have become more challenging– new configuration designs must integrate the core, divertor, and coils in the optimization; and reactor-relevant metrics such as alpha losses and maintainability must have greater weight in the design process. In order to pool capabilities and develop designs for new experiments in the most efficient manner, the task of advancing stellarator designs is best carried out by a national team, including both university, industry, and national laboratory participants. The following elements would define the optimization strategy for this initiative:

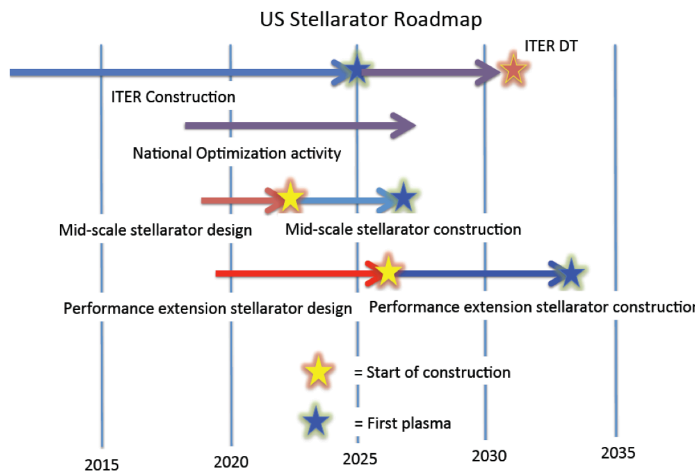

Figure 1 *Possible timeline for the major elements of the proposed reinvigorated US stellarator*

1) Optimization of neoclassical confinement through quasi-symmetry
2) Elimination of MHD instability and maintenance of good flux surfaces at finite β
3) Reduction of fast particle loss by optimization
4) Turbulent transport minimization
5) Automated divertor design consistent with optimized core
6) Coil simplification with engineering constraints to improve constructability and maintenance

A Mid-scale Facility: The design activity would lead directly to the design and construction of a new mid-scale device as soon as possible to demonstrate and test the physics basis of QS optimization by:

- Examining the physics of quasi-symmetric confinement in fusion-relevant collisionless-ion regimes with $T_i \sim T_e$
- Focusing on other issues that are not addressed in W7-X or LHD (fast particles, etc.)
- Implementing innovative design choices based on the optimization activity

Research Directions Beyond the 15-year Horizon

A Large-scale Facility: A large-scale device based on proven design principles is needed to demonstrate the required performance in fusion relevant regimes. A definitive international assessment of the potential of quasi-symmetry requires an integrated experiment, one that can answer equilibrium, stability, divertor, and energetic-particle related issues simultaneously and self-consistently. The exact requirements can only be determined by carrying out a multi-disciplinary conceptual design activity, but examples of this class of facility abound. One can anticipate that a plasma radius in the ≥ 0.5-1.0 m range, magnetic field strength in the 4 to 6 T range, and multi-10s of MW of plasma heating will be needed. Pulse length requirements are not so easily anticipated; much can be learned about divertors and plasma evolution in ~10 s pulses, but a convincing demonstration of reliable steady-state performance will likely require minutes to hours. The design may or may not include capability for DT operation, but nonetheless must be shown to be on a path to steady state nuclear facilities that are practical with respect to engineering issues such as fabrication and maintainability.

Successful implementation of this plan would place the US in a leadership position to develop an attractive stellarator-based fusion power plant in the post-ITER era.

Critics' Objections and Advocate Responses

• *Stellarators are too complicated and expensive. The 3D nature of stellarator coils make them more difficult to engineer and build.* We note that fabrication accuracies are high for all fusion systems and many devices have suffered cost overruns. The dominant source of the cost-overruns and schedule delays have been associated with high precision construction requirements, not 3D complexity [22,23]. There are several examples (W7-X, LHD) of successfully constructed large superconducting stellarator systems. Significant recent work has been done in simplifying coil designs in 3D systems [24]. Also, recent results have shown the ability to trim out error fields [25]. Stellarators coils are complex, but they provide the enormous offsetting advantages of simple operation and low recirculating power once the magnetic surfaces are created. This results in fewer and less complex auxiliary systems, greater availability, and improved operating economics. In addition, the ability to avoid major disruptions and the corresponding large transient forces, the elimination of the need for mitigation techniques simplifies the overall design. The ability to design the q-profile to avoid low order resonances is also an offsetting advantage.

• *Poor neoclassical and fast particle confinement in fusion grade plasma*. Experimental solutions with quasi-symmetry have demonstrated good confinement in smaller scale experiments and W7-X will demonstrate another optimization method (quasi-omnigeneity) to improve neoclassical confinement in a performance class device. A definitive experimental test of the efficacy of QS optimization in high performance plasmas (be it quasi-helical or quasi-axisymmetric) is needed by the world-wide fusion program. A program describing fast particle confinement optimization is described in this document.

• *Stellarators have high aspect ratio resulting in larger reactor unit sizes.* Designs exist with lower aspect ratio than present-day devices [26]. The possibility of improved confinement, due to turbulent transport optimization, could permit smaller minor radii in future devices. Moderate aspect ratio can also lower first-wall replacement demands.

• *3D divertor solutions are not yet demonstrated*. This area is a key focus of ongoing experiments and theoretical investigations. External control of the plasma edge may allow increased divertor heat flux width. Stable detachment has been demonstrated in experiments without adverse core effects [27]. High density operation also permits better divertor solutions. W7-X will demonstrate the viability of the island divertor, but the non-resonant divertor requires additional study. A reactor-relevant divertor solution is a critical area for all of magnetic fusion, and we note that such a solution has also not yet been demonstrated for tokamaks.

• *Stellarators are behind – we can't wait for them to catch up.* Because of the lower recirculating power, it may be possible to combine the FNSF and burning plasma mission with the electricity demonstration step [5]. This advantage could reduce the number of steps to a reactor.

References

[1] M. Hirsch, *et al., Plasma Phys. Control. Fusion* **50**, (2008) 053001
[2] J. M. Canik, *et al., Phys. Rev. Lett*. **98**, (2007) 085002
[3] H.-S. Bosch, *et al., Nucl. Fusion* **57**, (2017) 116015
[4] R.C. Wolf, *et al., Nucl. Fusion* **57**, (2017) 102020
[5] J.E. Menard, *et al., Nucl. Fusion* **51**, (2011) 103014
[6] S. Sudo, *et al., Nucl. Fusion* **30**, (1990) 11
[7] M. Greenwald, *Plasma Phys. Control. Fusion* **44**, (2002) R27-R53
[8] D. A. Gates, *et al., Journal of Fusion Energy* **37**, (2018) 51
[9] H. E. Mynick, *et al., Phys. Plasmas* **13**, (2006) 064505
[10] M. McMillan, S. A. Lazerson, *Plasma Phys. and Controlled Fusion* **56**, (2014) 095019
[11] D. Spong, S. Hirshman, J. Whitson, *Phys. Plasmas* **5**, (1998) 1752
[12] M. Drevlak, *et al., Nuclear Fusion* **54**, (2014) 073002
[13] H. Mynick *et al., Phys. Rev. Lett.* **105**, (2010) 095004
[14] C. C. Hegna, *et al., Phys. Plasmas,* **25**, (2018) 022511
[15] K. McCormack, *et al., Phys. Rev Lett.,* **89**, (2002) 015001
[16] K. Ida, *et al., Phys. Plasmas* **16**, (2009) 056111
[17] A. H. Boozer, *J. Plasma Physics* **81**, (2015) 515810606
[18] A. Bader, *et al., Phys. Plasmas* **24**, (2017) 032506
[19] M. Landreman and A. H. Boozer, *Phys. Plasmas* **23**, (2016) 032506
[20] Zhu, C., *et al., Nucl. Fusion* **58**, (2017) 016008.
[21] Spong D.A., *et al., Nucl. Fusion* **41** (2001) 711
[22] H.-S. Bosch *et al. Nucl. Fusion* **53 (**2013) 126001
[23] R. L. Strykowsky, *et al.,* in the *Proceedings of the Symposium on Fusion Energy, San Diego, CA* (2009): http://ieeexplore.ieee.org/stamp/stamp.jsp?tp=&arnumber=5226449
[24] D. A. Gates, *et al., Nucl. Fusion* **57**, (2017) 126064
[25] S. A. Lazerson, *et al., Nucl. Fusion* **57** (2017) 046026
[26] M. C. Zarnstorff, *et al., Plasma Phys. Control. Fusion* **43** (2001) A237
[27] Y. Feng, *et al., Nucl. Fusion* **45** (2005) 89

IMPORTANCE OF THEORY, COMPUTATION AND PREDICTIVE MODELING IN THE US MAGNETIC FUSION ENERGY STRATEGIC PLAN

Fatima Ebrahimi (PPPL/PU), Gary Staebler (GA), Paul Bonoli (MIT), Francois Waelbroeck (UT), Chris Hegna (UW-Madison), Lynda LoDestro (LLNL): Based on the community input at the Madison and Austin workshops

1. Description of the element:

Due to the complexity of fusion plasmas, gaps in scientific understanding of the underlying plasma physics remain one of the most fundamental challenges for achieving a viable fusion reactor. Models of self-heated burning plasmas confined by magnetic fields require nonlinear interaction of multiple physics processes over spatial and temporal scales spanning many orders of magnitude. To overcome these multi-physics and multi-scale challenges, understanding through theory and computation, combined with advances expected in extreme-scale computing in the coming decade, is a necessity to accelerate development toward a fusion-energy source. At the two recent community workshops "theory and computation" was recognized as a key strategic element that constitutes a natural foundation for any US strategic approach toward the development of magnetic fusion energy. In the US MFE program, the five main missions of this element are to: **1- continually broaden and deepen our understanding of the physics of fusion burning plasmas, 2- develop physics-based validated predictive capability, 3- discover new modes of operation, 4- explore and optimize device design, and 5- develop real-time plasma control systems**.

The transformative capabilities through innovative analytical techniques, reduced models and advances in high performance computing could lead to

- closing some of the remaining gaps for reliable prediction for burning plasmas including ITER
- optimization of tokamak, stellarator and other alternative MFE concepts, for furtherance toward fusion energy.

Historically, the fundamental scientific impact of fusion theory has extended far beyond the MFE community. There are many examples of conceptual advances, pioneering analytical techniques and high-end computational plasma physics models developed in the US MFE community that have also contributed to significant advances in other subfields of plasma physics as well as in the international MFE programs.

2. Benefits:

Some major areas of a MFE strategic plan that benefit from theory and computation are:

- **Deployment of predictive theory as a tool for discovery and support of existing experiments**: The US has invested in a world leading diagnostic measurement capability that has greatly advanced the validation of fusion plasma theory leading to the development of predictive reduced models. This fusion energy science mission has strengthened the confidence in the success of ITER and is now being utilized by China to design their next step CFETR machine. This science mission is essential to a successful US strategic plan to develop fusion energy within the next few decades. Existing MFE tokamaks DIII-D and NSTX-U are highly diagnosed and provide detailed multi-scale validation data for physics models. Physics codes with synthetic diagnostics enable more detailed analysis of experiments. Efficient validation and analysis workflow have made large data set uncertainty quantification possible. The resulting confidence in theoretical predictions will enable new discoveries ranging from innovative control methods to the unforeseen modes and regimes of operation that they will enable.

- **Next large-scale US fusion experiments**: The US fusion energy science mission has developed validated predictive models that will enable a better-informed selection of the next generation of US fusion experiments. The prediction by high performance computing and modeling can be used to identify a specific configuration designed for its operational performance limits, for example confinement time or pulse length, and its ultimate viability for a fusion reactor. Theory and predictive computing, whether used for the performance extension and control scenarios on existing large-scale devices or for design innovations for the next step experiments, will reduce the cost and shorten the timeline of the US path forward for fusion energy and provide guidance for a strategic plan.
- **ITER**: With ITER now under construction, the next decade provides opportunities for theory, computation and predictive modeling to have impact on critical aspects during burning plasma operation, including tolerance of heat and force loads on the first wall, and control of transient events. Prediction of ITER operation from discharge startup to rampdown is a strategic goal for the US program that will enable improving the fusion performance of ITER.
- **Cross-field interaction and educational enrichment**: Theory and computation have had a significant role in promoting synergy between the fusion program and other branches of plasma physics research. Pioneering theories such as spontaneous tearing reconnection or kinetic sub-scale gyro-averaged models, which are applicable in many branches of plasmas physics, have been initiated (established) in the fusion community. Advances in computation and theory will further promote cross-field interaction in plasma physics as a unifying field of fusion, low temperature, high-energy-density, space, and astrophysical sciences. This interaction enriches both the educational and scientific aspects, and attracts younger students and scientists to pursue their careers in developing the groundbreaking solutions necessary to achieve fusion power reactor.
- **Leadership in fusion and plasma science:** Theory and computation has greatly contributed to the US leadership in fusion and plasma science. Some recent *scientific advances in fusion* enabled through combined theoretical and experimental effort, and strong and essential partnership between the USA and Europe are:
    - *theoretical prediction and experimental demonstration of neoclassical tearing mode stabilization by localized electron cyclotron current drive [1]*
    - *understanding and quantitative verification of global mode stability in experimental high performance tokamak plasmas, based on drift-kinetic MHD theory [2]*

A defining characteristic of the US fusion program has been its strong emphasis on constantly advancing the *frontiers of plasma physics*. Two general areas of US leadership enabled through strong engagement of fusion theorists are:

- *leading-edge plasma-physics research through NSF-DOE partnership*
- *high-end computing and the establishment of fusion integrated simulation through SciDAC/ASCR partnership*

The advent of computation at the exascale in the US presents opportunities to advance all areas of plasma and fusion material science. To maintain a leading role for the US in the pursuit of a viable controlled thermonuclear reactor, it is critical to maintain, support and encourage the interplay between reactor design and high-quality, *leading-edge plasma-physics research and computer science*. Major efforts must be placed on developing codes and capabilities for simulating plasma behavior. These efforts place great demands on computational methods, advanced algorithms and hardware, and are undoubtedly useful well beyond fusion physics.

3. Current status:

High performance computing (HPC) is critically important for our present US fusion theory program. Over the past decades, the US fusion program has led the world in developing new physics models in the areas of gyrofluids, gyrokinetics, wave-particle interactions and extended MHD, and numerical methods to exploit advanced computing. In particular, numerous SciDAC/ASCR partnerships over the previous decade facilitated advances in high-performance computing using petaflop architectures. To maintain this scientific leadership position additional pioneering plasma science is critical in the areas of 1) analytical theory for the understanding of the physics of plasmas, for the development of the basis of computational models, as well as to interpret, reinforce and verify computation, 2) high performance computing utilizing a multi-fidelity hierarchy of physics models ranging from high degree of freedom to reduced models and neural networks, and 3) validated predictive integrated modeling. Currently, complementary approaches are being pursued in order to achieve the mission of our program (understand, predict, explore, and control):

- **Standalone models**: Individual models ranging from fluid models to full kinetic models can be used to simulate the entire device. Fluid MHD models in particular, such as extended MHD, are beneficial to describe global nonlinear macroscopic plasma behavior, and address the challenges of controlling ELMs and disruptions. Standalone models can also have some loose coupling with external systems. To gain confidence for prediction for reactor scales, high fidelity standalone models should be validated on smaller-scale devices (or simpler experiments). Historically *different MFE magnetic configurations*, including reversed field pinches, spheromaks, and field reversed configurations have been successfully used as validation targets for validation of nonlinear extended MHD (and hybrid kinetic-MHD models). Even non-MFE devices (e.g. LAPD) can play a valuable role in validation at relevant physics parameters and allow further extrapolation.
- **Integrated modeling for fast prediction**: This approach is integrated modeling through direct multiphysics-multiscale coupling of individual high-fidelity models. An integrated model should contain the core confined burning plasma, plasma edge (including scrape-off layer) and the external systems (i.e. plasma facing material, vessel wall, RF antennas, beams, coil controllers). There are different types of coupling of high-fidelity codes such as RF-MHD, kinetic core-edge, MHD-kinetic, and edge plasma – multimaterial (coupling EM gyrokinetic to comprehensive models of neutral particle and radiation transport), some of which have been supported by the SciDAC program. These couplings are challenging and require an extensive applied math and computer science effort, which are on the 15-year time line.[4] Reduced-fidelity models calibrated by highest fidelity physics simulations, and experimentally validated, could favorably be used for fast prediction of plasma performance.

By utilizing these approaches, some of *the main objectives of our current program* are 1) to understand and predict the operational limits of the existing experiments, 2) code verification/validation with uncertainty quantification, and 3) physics-based predictive modeling leading to performance optimization and controlling the transients in ITER.

4. Programmatic context:

Partnership with other agencies: Partnerships with other US agencies, such as ASCR, NSF, NASA enhance the scientific and the educational breadth of the MFE program. Our fusion theory program is very much strengthened by synergy with other subfields of plasma physics.
International partnership: The US theory program due to its current scientific leadership in areas such as stellarator optimization and integrated simulation, could further benefit by actively engaging in the rising international fusion programs and the newly built or upgraded fusion experiments around the world: W7X, JET, WEST, JT60-SA, EAST and KSTAR, to name a few. Engaging with the ITER modeling group through the ITPA and the EU, Japanese and Chinese theory communities enhances the productivity of the US MFE theory and computation program.

5. Possible 15-year U.S. research agenda

To get the full benefits of the fusion theory program outlined in section 2, we envision that our effort in high performance computing combined with integrated modeling should continue to be pursued for the next 15-years. To achieve theoretical physics-based predictions for a fusion reactor with quantified confidence, at every level the individual models should be validated on various MFE smaller-scale highly diagnosed devices performing specific validation experiments. There are major challenges and opportunities, which have been identified by the community, as critical gaps in theory (and associated gaps in simulation capability)[3]. Here, impactful major opportunities for theory and simulations are:

- **Understand, predict, and control plasma transients:** At present, the number one challenge for burning plasmas including ITER is disruption. Advanced simulations need to model all forms of disruption from instability to final wall deposition. Models should be developed to understand and address some of the challenges, including runaway electron generation and evolution, rotation physics and mode locking, disruption-related plasma-wall interaction and open-field currents. Understanding the control and mitigation techniques such as, Edge Localized Mode control with external coils, pellet fueling and disruption mitigation, and active control of MHD instabilities are essential. Real-time disruption forecasting from theory-based stability boundary maps and plasma control systems based on neural network and machine learning techniques could provide robust disruption avoidance. In the next 15-years, computational modeling of transients would have a direct impact on the ITER research program.
- **Modeling for long pulse operation:** High performance computation of non-inductive current-drive techniques should be integrated from the edge to the core, and shown that current and heat could be built up in the plasma core and form a steady state. Challenges are predictions for solenoid-free current-drive techniques (through various helicity injection techniques and subsequent RF- neutral beam ramp-up), prediction and mitigation of RF interactions with the plasma-material-interface at the plasma boundary and integrated modeling to predict effective alpha particle heating and possible energetic particle instabilities.

- **Design optimization toward disruption-free configurations:** Optimizations through modeling will guide us to configurations of ultra-optimized stellarators that are inherently steady-state and avoid disruptions. Development of computational tools should be pursued to further exploit the potential of stellarators and to determine the effect of the magnetic configuration on turbulent transport, magnetic surface fragility, macroscopic instabilities, energetic ion confinement, impurity control and edge/divertor physics. Theory and modeling could also investigate 1) the advantages of high temperature superconductors on confinement, and 2) engineering design improvements for advanced divertor, blanket, RF launchers, and outside fluid loops.
- **Improved modeling for plasma-material interaction:** Boundary models should advance to integrate multiple physical processes that cover a wide range of overlapping spatial and temporal scales. This includes integration from the hot, confined pedestal zone with sharp gradients, to the cooler unconfined scrape off layer and divertor plasma where heat fluxes reaching the walls must be within material limits, and finally the first few microns of the wall itself.

The ultimate goal is to achieve optimization/prediction/control for burning plasmas through whole device modeling (WDM). [3,4] It should be noted that the state of readiness varies for the different elements in the areas outlined above, and some would require further resources to mature for integration into WDM. While a comprehensive assessment of the readiness of different components for WDM is being performed, continued development of analytical theory combined with validated standalone simulations is still necessary. A whole device model could be an assembly of physics models with a range of fidelity, which ultimately allows simulating from the plasma core to the wall during plasma discharge and from start up to ramp down. In order to achieve the 15-year physics objectives, advances in mathematical and computational technologies are essential. With the move to exascale computing, further interaction among computer scientists, applied mathematicians and plasma physicists is essential and could ultimately help to overcome the challenges of integrated modeling. [3,4] In particular, predictive modeling could be critical for enabling innovative concepts. A summary of the theory and computation challenges and the required R&D discussed during the Madison and Austin community workshops [5-9] is organized in table 1.

6. Research directions beyond the 15-year horizon

We envision predictive models of the whole device, which include all components that describe plasma, from macroscopic equilibrium to micro-turbulence and plasma-surface interactions, and ultimately including all components that describe the evolution of a plasma discharge from start-up to termination. Whole device models are required for assessments of reactor performance in order to minimize risk and qualify operating scenarios for next-step burning plasma experiments, as well as time-dependent or single-time-slice interpretive analysis of experimental discharges. In ITER, the exploration of burning plasma and pulse control scenarios will be guided by modeling, as it is not feasible to determine operational limits by running trial discharges. The goals of the theory program should include models that provide 1) options for interpretive as well as predictive modes, and synthetic diagnostics, 2) an environment for connecting to experimental databases, possibly on remote platforms, and 3) infrastructure to support the above, as well as for machine and scenario design and operation.

7. Critics' objections and advocates' responses

Objection: Reduced models would be sufficient for burning plasma predictions.
Response: The objectives for the next 15-year outlined above will require complementary use of all the approaches, including theory/high performance computing as well as reduced models.

Objective	Challenges	R&D needed
Understand, predict, and control plasma transients	• Disruptions and runaway electron generation and evolution, rotation physics and mode locking, disruption-related plasma-wall interaction and open-field currents • Pellet fueling and disruption mitigation • Power threshold for the H-mode transition • Edge Localized Mode control with external coils • Active control of MHD instabilities	• Further development of analytical theory and validated modeling for all challenges listed in the left column Innovations in the areas of integrated coupled models: ❖ High-fidelity coupling of core-pedestal-SOL system through kinetic (EM gyrokinetic – full kinetic) or MHD-kinetic core-edge coupling for transients such as ELM growth and ejection, and stabilizing physics effects of energetic particles and runaway electrons ❖ Real-time disruption forecasting from theory-based stability boundary maps ❖ Plasma control systems based on neural network and machine learning techniques to provide robust disruption avoidance
Modeling for long pulse operation	• Steady-state coupling of core, edge, and plasma material interactions • Fast ion instabilities and transport • Interaction of fast particles with thermal plasma waves	• Validated predictive extended MHD simulations for non-inductive "solenoid-free" current-drive (through various helicity injection techniques and subsequent RF- neutral beam ramp-up) • Modeling to investigate high-field LHCD launch and its impact on microturbulence in the SOL Innovations in the areas of integrated coupled models: ❖ A predictive capability for self-consistent interaction of RF power with the scrape-off layer and wall, including realistic antenna and first wall geometry ❖ Integrated modeling to predict effective alpha particle heating and possible energetic particle instabilities
Design optimization toward disruption-free configurations	• Stellarator fast-ion and thermal confinement optimization • Impact of high-Tc superconducting magnets on confinement configurations • Explore new magnetic configurations	• Further development of analytical theory and simulations for: - improved stellarator optimization - evaluating the implications of HTS on stability and the heat flux width - compact tokamak/ST design to lower aspect ratio for greater magnetic field utilization, improved stability, and reduced TF magnet mass - optimization of all existing MFE concepts to assess their potential for improved stability and confinement and to explore new magnetic concepts Innovations in the areas of integrated coupled models: ❖ Develop computational tools to couple EM GK codes to 3-D (MHD) equilibrium conditions for the purpose of minimizing turbulence in stellarators ❖ Development of nonlinear MHD and further development of transport codes (such as TASK3D) for stellarators ❖ Integrated physics and engineering optimization design for advanced divertor, blanket, RF launchers, and outside fluid loops for reactor design and safety

Improved modeling for plasma-material interaction	• Reliably predict scrape-off layer transport and beyond • Plasma material interaction • Material resilience to neutron damage	• Develop codes to examine advanced divertor concepts, including alternate magnetic-geometry divertors and liquid walls Innovations in the areas of integrated coupled models: ❖ Multi-scale SOL models including molecular dynamics and kinetic Monte Carlo codes, 2D and 3D plasma transport codes, and 4-5D EM gyrokinetic codes ❖ Plasma codes to couple with efficient wall models for erosion/redeposition of surfaces, impurity release, and tritium trapping within the wall

Table 1: Summary of the theory and computation challenges in the four areas in Section 5.

Referees: David Newman (U. Alaska) and John Canik (ORNL)

Thanks to: M. J. Pueschel (UW), B. Grierson (PPPL) and W. Horton (UT) for comments and endorsements.

8. References

[1] John Dawson Award for Excellence in Plasma Physics Research, C. Hegna, H. Zohm, J. D. Callen, O. Sauter, R. J. LaHaye (2014)

[2] Landau-Spitzer Award, H. Reimerdes, J. W. Berkery, S. A. Sabbagh, Y. Liu (2016)

[3] P. Bonoli, L. C. McInnes, C. Sovinec, D. Brennan, T. Rognlien, P. Snyder, J. Candy, C. Kessel, J. Hittinger, L. Chacon, D. Estep, T. Munson, W. Bethel, M. Greenwald, D. Bernholdt, and B.Lucas. Report of the workshop on integrated simulations for magnetic fusion energy sciences, June 2015. http://science.energy.gov/~/media/fes/pdf/workshop-reports/2016/\ISFusionWorkshopReport_11-12-2015.pdf.

[4] A white paper on whole device modeling will be submitted.

[5] C. C Hegna "Stellarator Research: Challenges and Opportunities" Madison workshop July 25 2017. Hegna_theory.pdf

[6] A. Bhattacharjee, et al. "Strategic Role of Exascale Computing in US Magnetic Fusion Research" Madison workshop July 25 2017. Bhattacharjee_exascale.pdf

[7] C. R. Sovinec et al. "Theoretically Integrated Predictive Modeling" Madison workshop July 25 2017. Sovinec.pdf

[8]G. Staebler, et al. "A Plan to establish the physics basis for validated predictive whole device modeling" Madison workshop July 25 2017. Staebler.pdf

[9] F. Ebrahimi et al. "Transformative Theory and Predictive Modeling-- a pathway toward fusion energy" Austin workshop, Dec. 13 2017. Ebrahimi_innovations_theory_SA3.pdf

Elements of a US R&D Plan to Solve Plasma-Material Interaction Challenges for Magnetic Fusion Energy

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Divertor and main chamber wall components in a DT fusion reactor must be capable of handling extreme levels of plasma heat exhaust and plasma-material interactions (PMI). A central question for magnetic fusion energy development is: does there exist a combination of plasma physics scenarios and material technologies that can make this happen? US fusion researchers are pioneering world-leading approaches to solve PMI challenges, as highlighted in the 2015 FES Workshop on PMI [1]. Elements of this research were presented at the Madison and Austin Strategic Planning Workshops. A consensus view was: *"PMI/divertor problems [are] very important, compelling options need to be evaluated"* [2]. This paper discusses 6 PMI program sub-elements that received considerable discussion at the workshops [3-5]: (1) Advanced Divertors, (2) Advanced Solid PFC Materials and Manufacturing, (3) Liquid Metal PFCs, (4) Linear Plasma Test Stand for long-pulse PMI, (5) High Field Side RF systems, and (6) Divertor Test Tokamak. These sub-elements are not competing proposals. Integrated closely together, they would form the underpinnings of a compelling, world-leading, PMI R&D program.

1. <u>Advanced Divertors</u> [6]: Present experiments indicate an unavoidable tradeoff between good core confinement and protecting conventional divertor targets. This prohibits the use of conventional divertor solutions for DEMO since power handling must be increased by an order-of-magnitude while nearly complete suppression of target plate erosion must also be attained.

Benefits: New physics ideas embodied in advanced divertor concepts could meet this challenge, including: passive or active control mechanisms to keep the "divertor detachment front" from degrading the core; operating highly dissipative attached divertor regimes; using a liquid or vapor as the divertor target. The US has been the primary innovator in this area, proposing a number of options: snowflake, X, super-X, active liquid Li replenishment, Li vapor box, long-leg X-pt target, small angle slot (SAS); as well as from QST, the v-shaped deep slot [7-12].

Current Status: Exploratory, proof-of-concept experiments have been performed at low and moderate power on ~half these concepts in US tokamaks and tokamaks overseas, some accessing DEMO-relevant PMI conditions at their target plates ($T_{et} < \sim 5$ eV, $n_{et} \sim 10^{21}$m^{-3}) [13].

International context: TCV, MAST-U, AUG plan to continue proof-of-concept experiments.

Possible 15-year U.S. research agenda: From 2015 FES Workshop Report on PMI [1]: (1) exploit and upgrade existing divertor experiments for enhanced runtime, diagnostics and personnel; explore power handling limits of existing divertor configurations; upgrade divertor configurations and materials (solid and liquid) and explore power handling limits; (2) complement with targeted collaborations on overseas experiments; maximize U.S. benefits from ITER; (3) establish national working group to examine design options for DTT; implement DTT.

Research directions beyond the 15-year horizon: US DTT; further overseas collaborations.

Critics' objections: No existing experiment can produce reactor-level plasma conditions throughout its divertor and so cannot provide experimental access to the integrated, dissipative physics regimes that will likely exist in a reactor.

Advocates' responses: Well diagnosed existing divertor experiments can improve our understanding and validate codes for more reliable predictions for reactor conditions.

2. <u>Advanced Solid PFC Materials and Manufacturing</u> [14]: New materials are potential game changers for fusion. Large impacts on concepts and performance are possible for plasma-facing components (PFCs), structural and blanket materials. In many cases, conventional materials technology cannot meet the requirements. Incident plasma heat fluxes of 100's of MWm^{-2} and particle fluxes of ~10^{24} $m^{-2}s^{-1}$ are anticipated. The fusion reactor wall and PFCs must withstand incident particle energies varying from a few eV ions to MeV neutrons [15, 16] and some reactor designs call for operation at temperatures up to ~1000 C to obtain high thermal efficiencies. Expected rates of net erosion and deposition of solid PFC material in reactors are projected to be 10^2-10^5 kg/yr for all elements and compounds. Heavy deposits (slag) can interfere with operation (e.g. UFO-induced disruptions) making PFC slag management critical.

Benefits: Robust Advanced Manufacturing (AM) including additive manufacturing processes build parts layer by layer using lasers or other techniques that fuse powders or fibers. AM is expected to transform the world's industrial output and enable new materials and products [17]. Desired microstructure, PMI properties, self-healing and radiation resistant properties, can be *designed* into complex geometries and hierarchical structures addressing surface/bulk functions in a single graded system [18-20]. AM is potentially transformative for PFCs by enabling low-Z material integration with complex high-Z substrates that provide PMI protection for high-Z components. Flow-through solid PFCs could provide in-situ replenishable clad-like designs using weakly bonded ceramic-based material at the plasma material interface.

Current status: PMI research is being performed on tungsten-based materials: advanced W-based composites, ductile W metal-matrix, W particulate, W laminate, and continuous W fiber. Work on non-W composites: SiC/UTHC, SiC/MAX, mostly in bulk with some PMI efforts. Recent efforts in innovative PMI materials include: nanostructured and mesoporous refractory-based materials; carbon nanostructures and 2D materials; ultra-high temperature ceramics (UHTC), B_4C, SiC, ZrB2, ZrC, and high-entropy alloys (HEAs) [21].

Programmatic context: AM has a strong technology pull in the aerospace structure and automotive sectors but very little synergy with DOE FES programs. DOE FES has some effort in PMI technology development and leverages international collaboration (DIFFER, FZ Julich).

Possible 15-year U.S. research agenda: A panel is needed to examine the wide range of options and to set short-term and long-term priorities for AM PFC R&D. Early-stage research in high-risk materials could include self-healing and adaptive PFCs, amorphous metals, advanced ceramic composites, such as MAX-phase (layered, hexagonal carbides and nitrides) composites. Determining linkages between AM and PMI properties through process/structure/function relationships could expedite development, along with a robust testing program on several platforms, including: linear plasma test stands, current tokamaks, DTT.

Research directions beyond the 15-year horizon: This area will require a growing level of R&D effort to fully exploit the coupling of advanced manufacturing techniques with enhanced component function – ultimately delivering PMI tolerant PFCs, integrated into bulk radiation-resistant heat sink materials and incorporated into complex blanket geometries.

3. <u>Liquid Metal (LM) PFCs</u> [22]: Liquid metal plasma-facing components (PFCs) have the potential to solve PMI challenges for fusion – self-healing, renewable surfaces that accommodate high heat loads (including transients) while potentially enhancing plasma energy confinement.

Benefits: Lithium-plasma interactions are found favorable: reduction in SOL recycling, plasma impurities and ELMs (by modifying pedestal); increase in edge plasma temperature. Lithium has a self-shielding response to plasma heat fluxes (divertor vapor target concept). Tin is a higher temperature alternative liquid metal. Both may be combined in an alloy.

Current status: Research is focused on controlling LM MHD effects [23, 24] and self-shielding [25] response to lithium PFCs (recycling, edge temperatures, pedestal, confinement) [26, 27]. Techniques include *slow-flow* and *fast-flow. Slow-flow* – liquid metal wets a cooled substrate and is slowly replenished [28]. Near term issues include: vapor shielding, substrate and flow control, lithium vs. tin or tin-lithium alloys, and integrated closed-loop testing on confinement devices. *Fast-flow* – liquid metal flow provides heat and particle removal [29]. Issues include MHD effects in magnetic fields, and material ejection from plasma-induced transients. MHD flows in narrow channels have been investigated in test stands [30], but not for toroidal flows. Efficient tritium separation is required for lithium PFCs, but few techniques have been studied [31, 32]. Effects of lithium on confinement and equilibrium have been noted, but technical solutions to slow and fast flow have undergone little testing. Test stands are needed to develop control approaches, and test/optimize ideas for deployment in a confinement facility [33, 34]. EUROfusion and China [35] are developing LM PFCs, although it is clear that the US is the world leader in this area.

Possible 15-year U.S. research agenda: Technological development of flowing LM PFCs, and vapor shielded systems. Fuel recovery/control demonstrated at large scales; assessment of tritium removal systems, material corrosion and embrittlement issues. Recycling of eroded materials (e.g., Sn, Li) demonstrated, performance limits of lithium vapor shielding [36] defined. Integrated performance and response/recovery from transients assessed experimentally.

Research directions beyond the 15-year horizon: Evaluate added complexity of LM PFCs against gains in erosion resilience, power handling, and confinement. Impacts on thermal-to-electricity conversion efficiency for reactors, safety, and economics studied. Inform reactor designs based on parallel development of advanced divertors and in-situ renewal of solid PFCs. Liquid metal concepts tested under the stress (PMI, heat fluxes) of a high-power, linear plasma test stand. Most promising ideas tested on a DTT, at reactor levels in an integrated tokamak environment.

Critics' objections: LM PFCs add complexity and may restrict operating temperatures of first wall components.

Advocates' responses: Liquid lithium PFCs may increase confinement significantly over solid high-Z walls; SOL modifications may be favorable for power handling. Technology development and scoping studies are certainly needed for all LM PFC implementations.

4. <u>Linear Plasma Test Stand (LPTS) for long-pulse PMI</u> [37]: Candidate PFCs must be tested for ability to withstand PMI under steady-state and transient heat loads, including thermo-mechanical properties (thermal conductivity, creep strength, He and H embrittlement), plasma-induced effects (erosion, redeposition, surface modification, dust formation) and hydrogen retention. Neutron damage effects (dpa, He production, transmutation) must also be considered.

Benefits: A high power density LPTS can expose small samples and mock-up modules to plasma conditions anticipated at reactor divertor targets. Operating in steady state, they can extend the PMI knowledge gained from short pulse exposure, e.g., tokamaks, to very long pulse and high plasma fluence, as needed for reactors. The performance of a wide range of materials now available – advanced PFCs, AM materials, liquid metals – can be rapidly tested, including samples previously exposed to neutron irradiation. Testing of prototype mockup modules at performance parameters, including liquid metal technologies, is necessary before deployment on tokamaks. A dedicated, high power LPTS facility with excellent diagnostic access would work synergistically with a solid/liquid PFC R&D program and DTT, to expedite PFC development.

Current status: Existing LPTSs have proven successful in providing basic data on PMI, e.g., PISCES (US) at low power density and Magnum (EU) at low target T_{et} and T_{it}. FZ-Juelich is proceeding with JULE-PSI [38], based on their PSI-2 with plans to include radioactive hot cells. China is also formulating plans for a high power LPTS.

Possible 15-year U.S. research agenda: The Material Plasma Exposure eXperiment (MPEX) [39] is proposed to perform this function. ORNL has built a prototype device, proto-MPEX, with the aim of assembling three key components for MPEX: (1) high power helicon source, (2) the means to heat electrons in an overdense plasma (EBW and/or whistler waves), (3) ICRH ion heating. Tests of (1-3) have shown necessary performance albeit not simultaneously.

Critics' objections: It's not certain MPEX can achieve its performance objectives. Other facilities in the world will be similarly capable to MPEX.

Advocates' responses: Capabilities are distributed over several devices in the world. MPEX aims for integrating all those capabilities in one device. Materials testing is often the rate-limiting step. The possibilities for advanced materials are exploding; intellectual property will likely extend far beyond fusion applications. These considerations, in addition to practicalities of shipping and handling neutron-activated materials, call for a dedicated US LPTS facility as part of an integrated PMI R&D plan.

5. <u>High field side RF systems</u> [40]: High Field Side RF launch (HFS RF) is identified as a potentially transformative approach to solve PMI challenges for RF launch structures, and also to enable efficient non-inductive current drive, which is essential for a steady state tokamak reactor.

Benefits: PMI on RF launchers – regarded as a potential show-stopper for application in a reactor – may be mitigated by placing RF structures on the HFS [41]. A quiescent scrape-off layer naturally forms there, producing steep SOL density gradients in near double-null configurations. Plasma density, and RF coupling, at the launcher may be *actively controlled* via external control knobs of magnetic flux balance and wall gap. Fluxes of energetic particles from various origins (e.g. runaway electrons, trapped ions, ELMs) are largely absent at the HFS. In addition, the RF wave physics for HFS launch is projected to be highly favorable. For lower hybrid current drive (LHCD), high magnetic field allows waves with low $n_{//}$ to penetrate deep into the plasma before damping, driving current where it is needed (0.6 < r/a < 0.9). CD efficiency, which scales as $1/n_{//}$, may be increased ~40% or more compared to LFS launch. RF waveguides are relatively small and may be embedded in the neutron shield blanket of a reactor. Locating HFS launchers off mid-plane may reduce neutron fluxes relative to the LFS. HFS launch is also favorable for mode conversion current drive in the ion cyclotron range of frequencies (ICRF) [42], with similar PMI advantages.

Current status: To date, no HFS LHCD experiments and virtually no HFS ICRF experiments have ever been performed. The technical means exist today to perform proof of concept experiments on existing tokamaks.

International context: The US is the innovator and leader in the world program.

Possible 15-year U.S. research agenda: Proof-of-concept HFS LHCD experiments are proposed for DIII-D in a 2 to 5 year time frame. CD efficiencies are projected to be 2 to 10 times higher than NBI, vertical ECCD or Helicon wave. WEST, operating at long pulse and with high-Z walls, could test HFS RF efficacy at higher fields (3.7 T) as well as coupler technologies with active cooling. A purpose-built DTT could serve as a platform to test HFS RF at reactor-level magnetic fields, plasma densities, PMI fluxes and surface power loadings. Additional R&D is required to improve RF source and antenna efficiencies for reactor application. RF and material testing/R&D programs are also required to investigate/develop manufacturing techniques for couplers, waveguides, and antennas using reactor relevant materials.

Critics' objections: HFS location and antenna feeds are difficult to access and service.

Advocates' responses: Radial build of couplers is modest, HFS couplers can be installed even on smaller present-day tokamaks, performance must be demonstrated in tests.

6. Divertor Test Tokamak (DTT) [43]: Exciting new advanced divertor ideas have potential to increase power exhaust handling to reactor levels ($q_{//}$ > 10 GW m^{-2}) while suppressing material erosion and damage. These include magnetic geometries with optimized target plate geometries, embedded x-points, extended legs, tight gas baffling, and various combinations of the above [7-11]. Liquid metal divertor schemes have also been proposed [36]. Present experiments cannot achieve upstream parameters of plasma pressure or heat flux approaching those of fusion power systems. In addition, present devices lack the flexibility to provide high power-density tests of advanced divertor options, and cannot readily vary solid and liquid plasma-facing materials.

Benefits: From 2015 FES Workshop Report on PMI[1]: "*We recommend establishing within the FES strategic plan a national working group to examine design options for a DTT facility. This facility should be capable of producing reactor-level plasma parameters in its divertor – while at the same time having the divertor volume and flexibility to explore a variety of advanced divertor concepts: magnetic geometries, topologies, mechanical shapes, gas dynamic options, and different target materials including liquid metals. 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.*" The consensus position of the Madison NAS Workshop is in resonance with the 2015 FES PMI Workshop. Both Workshops also noted that a new high-power-density DTT facility has been analyzed recently, featuring long divertor legs and a flexible poloidal field configuration, along with flexibility in gas dynamics and the use of solid and liquid plasma-facing materials [44].

Current status: No community-wide activity to date on examining design options for a DTT.

Possible 15-year U.S. research agenda: Establish within the FES strategic plan a national working group to examine design options for a DTT facility; implement a US DTT.

Research directions beyond the 15-year horizon: Taken together as part of a coordinated strategic plan, a DTT would work synergistically with solid/liquid PFC R&D programs and a high power linear plasma test stand – identifying and developing new concepts, testing them first in a high power linear plasma test stand and then deploying the most promising ideas for testing in an integrated tokamak reactor environment at the required performance levels on a DTT.

Critics' objections: Need for a DTT vs divertor studies in existing devices. The cost, which is estimated to be ~ $70M for an ADX-DTT and ~500M€ for the DTT recently proposed by Italy. Next step studies should also include neutrons. The challenge of power exhaust might not be as severe as the PB/R scaling suggests, possibly mitigated by cross field transport in the divertor region under high density detached divertor conditions.

Advocates' responses: Divertor physics involves interplay among plasma turbulence, neutral dynamics and atomic physics, which is impossible to model reliably. A U.S. DTT would address divertor designs specific to the more compact reactor concepts favored here. The strategic advantage of a DTT is that many concepts can be tested quickly, at relevant scale and at the plasma physics parameters required; rapid test cycles are precluded by neutron activation.

References

[1] Maingi, *et al.,* Report on Science Challenges and Research Opportunities in PMI, from Fusion Energy Sciences Workshop on Plasma Materials Interactions, May 4-7, 2015.

[2] David Maurer's summary slides from MFRSD Workshop – Madison, July 24-28, 2017

[3] Amanda Hubbard's WG4 presentation at MFRSD Workshop – Austin, December 11-15, 2017

[4] Madison discussion groups summary slides and comments on 6 PMI sub-elements

[5] Austin discussion groups summary slides and comments on 6 PMI sub-elements

[6] **Advanced Divertors**

Presentations made at MFRSD Workshop -- Madison, July 24-28, 2017

Menard Increase emphasis on physics and technology innovations for compact tokamak fusion
Hill U.S. Tokamak Facilities: Powerful Tools for Developing a Faster Path to Fusion Energy

White papers submitted to MFRSD Workshop -- Madison, July 24-28, 2017

Guo Developing and Validating Heat Flux Solutions for Steady-State Fusion
Meade The Need for Innovation in Fusion and Strategy for PFC Development
Menard Increase emphasis on physics and technology innovations for compact tokamak fusion
Navratil Perspectives for a strategic plan for a reinvigorated US fusion energy program

White papers submitted to Fusion Energy Sciences Advisory Committee on Transformative Enabling Capabilities

Guo Development of Advanced Divertor Concepts for Steady-State Fusion

Presentations at FESAC TEC Meeting June 20-22, 2017

Guo Development of Advanced Divertor Concepts for Steady-State Fusion
LaBombard Long-leg divertors with secondary x-points: a potential solution for divertor heat flux and PMI challenges - aided by the development of demountable HTS magnets

[7] Kotschenreuther, M., Valanju, P.M., Mahajan, S.M., and Wiley, J.C., *"On heat loading, novel divertors, and fusion reactors,"* Phys. Plasmas **14** (2007) 072502. https://doi.org/10.1063/1.2739422

[8] Ryutov, D.D., *"Geometrical properties of a "snowflake" divertor,"* Phys. Plasmas **14** (2007) 064502. http://aip.scitation.org/doi/abs/10.1063/1.2738399

[9] Valanju, P.M., Kotschenreuther, M., Mahajan, S.M., and Canik, J., *"Super-X divertors and high power density fusion devices,"* Phys. Plasmas **16** (2009) 056110. http://dx.doi.org/10.1063/1.3110984

[10] Umansky, M.V., LaBombard, B., Brunner, D., Rensink, M.E., Rognlien, T.D., Terry, J.L., and Whyte, D.G., *"Attainment of a stable, fully detached plasma state in innovative divertor configurations,"* Phys. Plasmas **24** (2017) 056112. http://aip.scitation.org/doi/abs/10.1063/1.4979193

[11] Guo, H.Y., Sang, C.F., Stangeby, P.C., Lao, L.L., Taylor, T.S., and Thomas, D.M., *"Small angle slot divertor concept for long pulse advanced tokamaks,"* Nucl. Fusion **57** (2017) 044001. http://stacks.iop.org/0029-5515/57/i=4/a=044001

[12] Nobuyuki, A., Katsuhiro, S., Kazuo, H., Kenji, T., Shinsuke, T., and Tomonori, T., *"A simulation study of large power handling in the divertor for a Demo reactor,"* Nucl. Fusion **53** (2013) 123013. http://stacks.iop.org/0029-5515/53/i=12/a=123013

[13] Stangeby, P.C. and Leonard, A.W., *"Obtaining reactor-relevant divertor conditions in tokamaks,"* Nucl. Fusion **51** (2011) 063001. http://stacks.iop.org/0029-5515/51/i=6/a=063001

[14] **Advanced Solid PFC Materials and Manufacturing**

Presentations made at MFRSD Workshop -- Madison, July 24-28, 2017

Stangeby Developing refractory low-Z ceramic-clad Plasma Facing Components (PFCs) for robust burning plasma device operation
Henager Enhanced Fusion Energy Materials Program to Explore Potential Game-Changing Materials

White papers submitted to MFRSD Workshop -- Madison, July 24-28, 2017

Dehoff Advancing Fusion with Advanced Manufacturing
Henager Enhanced Fusion Energy Materials Program to Explore Potential Game-Changing Materials

Lumsdaine	Topology optimization for enabling innovative, high performance designs for additively manufactured first wall and divertor components
Nygren	High Impact on Fusion - Multiple Transformative Enabling Capabilities
Stangeby	Developing refractory low-Z ceramic-clad plasma facing components for robust burning plasma device operation

White papers submitted to Fusion Energy Sciences Advisory Committee on Transformative Enabling Capabilities

El-Atwani	Advanced Material Design for Fusion Applications
Henager	Plasma-Facing Materials by Design and Rapid Prototyping via Additive Manufacturing
Katoh	Advanced Manufacturing for Fusion PFC and Blanket Materials
Katoh	Emerging High Temperature Materials for Potential Application to Fusion
Nygren	Advancing Fusion Power -- Smart Tiles and Fast Data
Nygren	Development of Fusion Sub-components with Additive Manufacturing
Nygren	High Impact on Fusion - Multiple Transformative Enabling Capabilities
Tillack	Accelerated Deployment of Silicon Carbide Composites for an Attractive Fusion Energy Source
Wang	Laser powder-bed-fusion additive manufacturing as a transformative technology for plasma-facing materials and components
Youchison	Advanced Cooling Technologies through Additive Manufacturing

Presentations at FESAC TEC Meeting May 30 - June 1, 2017

Tillack	Accelerated Deployment of Silicon Carbide Composites for an Attractive Fusion Energy Source

Presentations at FESAC TEC Meeting June 20-22, 2017

Dehoff	Additive Manufacturing: Manufacturing Demonstration Facility
Henager	Plasma-Facing Materials-by-Design and Rapid Prototyping via Additive Manufacturing
Katoh	Advanced Manufacturing for Fusion PFC and Blanket Materials
Katoh	Emerging High Temperature Materials for Potential Application to Fusion
Martinez	Advanced Material Design for Fusion Applications
Nygren	Development of Fusion Subcomponents with Additive Manufacturing
Wang	Additive manufacturing for fusion energy sciences
Youchison	Additive Manufacturing for Advanced Cooling Technologies

Presentations at FESAC TEC Meeting July 19-21, 2017

Nygren	High Impact on Fusion - Multiple Transformative Enabling Capabilities
Rochau	Advanced Energy Conversion Cycles
Sarrao	Materials by Design; Materials for Harsh Environments

[15] Federici, G., Biel, W., Gilbert, M.R., Kemp, R., Taylor, N., and Wenninger, R., *"European DEMO design strategy and consequences for materials,"* Nucl. Fusion **57** (2017) 092002. http://stacks.iop.org/0029-5515/57/i=9/a=092002

[16] Zinkle, S.J., *"Fusion materials science: Overview of challenges and recent progress,"* Phys. Plasmas **12** (2005) 058101. http://aip.scitation.org/doi/abs/10.1063/1.1880013

[17] GE Additive Press Release, *"GE well positioned to accelerate the additive revolution,"* November 15, Boston MA. (2016) https://www.ge.com/additive/press-releases/ge-well-positioned-accelerate-additive-revolution

[18] Nygren, R.E., Youchison, D.L., Wirth, B.D., and Snead, L.L., *"A new vision of plasma facing components,"* Fusion Engineering and Design **109-111** (2016) 192. http://www.sciencedirect.com/science/article/pii/S0920379616302277

[19] Dr. Ir. Sybrand van der Zwaag, "*Self healing materials : an alternative approach to 20 centuries of materials science"*,Springer series in materials science (Springer, Dordrecht, The Netherlands, 2007).

[20] Tan, L., Snead, L.L., and Katoh, Y., *"Development of new generation reduced activation ferritic-martensitic steels for advanced fusion reactors,"* J. Nucl. Mater. **478** (2016) 42. http://www.sciencedirect.com/science/article/pii/S002231151630232X

[21] Zou, Y., Ma, H., and Spolenak, R., *"Ultrastrong ductile and stable high-entropy alloys at small scales,"* Nature Communications **6** (2015) 7748 http://dx.doi.org/10.1038/ncomms8748

[22] **Liquid Metal PFCs**

Presentations made at MFRSD Workshop -- Madison, July 24-28, 2017

Andruczyk The Case for Liquid Lithium as a Surface Material in Fusion Devices
Smolentsev Integrated approach to the development of liquid metal systems for FNSF and next fusion machines

White papers submitted to MFRSD Workshop -- Madison, July 24-28, 2017

Andruczyk The Case for Liquid Lithium as a Surface Material in Fusion Devices
Jaworski Breaking through to Reactor Solutions with a Focused, Liquid Metal Program
Majeski Magnetic confinement without anomalous transport

White papers submitted to Fusion Energy Sciences Advisory Committee on Transformative Enabling Capabilities

Jaworski Slowly flowing and high temperature liquid metals as plasma-facing materials
Kolemen Fast Liquid Metal Program for Fusion Reactor Divertor
Majeski Mitigation of scrape-off layer power flow with lithium plasma-facing surfaces
Majeski Recycling reduction for control of anomalous transport
Ruzic Liquid - Lithium as a Plasma Facing Material for Fusion Reactors
Williams Self-Healing Liquid Metal Protection System for Plasma-Facing Components

Presentations at FESAC TEC Meeting May 30 - June 1, 2017

Majeski Recycling reduction for control of anomalous transport

Presentations at FESAC TEC Meeting June 20-22, 2017

Ghoniem Self-Healing Liquid Metal Protection System for Plasma-Facing Components
Jaworski Slowly flowing and high temperature liquid metals as plasma-facing materials
Kolemen Fast Liquid Metal Program for Fusion Reactor Divertor
Majeski Mitigation of scrape-off layer power flow with lithium plasma-facing surfaces
Ruzic The Case for a Lithium-Surface Divertor

[23] Morley, N.B., Smolentsev, S., and Gao, D., *"Modeling infinite/axisymmetric liquid metal magnetohydrodynamic free surface flows,"* Fusion Engineering and Design **63-64** (2002) 343. http://www.sciencedirect.com/science/article/pii/S0920379602001862

[24] Brooks, J.N., Allain, J.P., Bastasz, R., Doerner, R., Evans, T., Hassanein, A., Kaita, R., Luckhardt, S., Maingi, R., Majeski, R., Morley, N.B., Narula, M., Rognlien, T., Ruzic, D., Stubbers, R., Ulrickson, M., Wong, C.P.C., Whyte, D., and Ying, A., *"Overview of the ALPS Program,"* Fusion Science and Technology **47** (2005) 669. https://doi.org/10.13182/FST05-A763

[25] van Eden, G.G., Kvon, V., van de Sanden, M.C.M., and Morgan, T.W., *"Oscillatory vapour shielding of liquid metal walls in nuclear fusion devices,"* Nature Communications **8** (2017) 192. https://doi.org/10.1038/s41467-017-00288-y

[26] Maingi, R., Boyle, D.P., Canik, J.M., Kaye, S.M., Skinner, C.H., Allain, J.P., Bell, M.G., Bell, R.E., Gerhardt, S.P., Gray, T.K., Jaworski, M.A., Kaita, R., Kugel, H.W., LeBlanc, B.P., Manickam, J., Mansfield, D.K., Menard, J.E., Osborne, T.H., Raman, R., Roquemore, A.L., Sabbagh, S.A., Snyder, P.B., and Soukhanovskii, V.A., *"The effect of progressively increasing lithium coatings on plasma discharge characteristics, transport, edge profiles and ELM stability in the National Spherical Torus Experiment,"* Nucl. Fusion **52** (2012) 083001. http://stacks.iop.org/0029-5515/52/i=8/a=083001

[27] Boyle, D.P., Majeski, R., Schmitt, J.C., Hansen, C., Kaita, R., Kubota, S., Lucia, M., and Rognlien, T.D., *"Observation of Flat Electron Temperature Profiles in the Lithium Tokamak Experiment,"* Phys. Rev. Lett. **119** (2017) 015001. https://link.aps.org/doi/10.1103/PhysRevLett.119.015001

[28] Vertkov, A., Luyblinski, I., Evtikhin, V., Mazzitelli, G., Apicella, M.L., Lazarev, V., Alekseyev, A., and Khomyakov, S., *"Technological aspects of liquid lithium limiter experiment on FTU tokamak,"* Fusion Engineering and Design **82** (2007) 1627. http://www.sciencedirect.com/science/article/pii/S0920379607002372

[29] Zakharov, L.E., *"Magnetic Propulsion of Intense Lithium Streams in a Tokamak Magnetic Field,"* Phys. Rev. Lett. **90** (2003) 045001. https://link.aps.org/doi/10.1103/PhysRevLett.90.045001

[30] Kusumi, K., Kunugi, T., Yokomine, T., Kawara, Z., Kolemen, E., Ji, H., and Gilson, E.P., *"Study on Thermal Mixing of MHD Liquid Metal Free-Surface Film Flow,"* Fusion Science and Technology **72** (2017) 796. https://doi.org/10.1080/15361055.2017.1347457

[31] Maroni, V.A., Wolson, R.D., and Staahl, G.E., *"Some Preliminary Considerations of A Molten-Salt Extraction Process to Remove Tritium from Liquid Lithium Fusion Reactor Blankets,"* Nuclear Technology **25** (1975) 83. http://www.tandfonline.com/doi/abs/10.13182/NT75-A24351

[32] Teprovich Jr., J.A., *"Electrochemical extraction of hydrogen isotopes from molten lithium,"* submitted to Fus. Eng. Design (2018)

[33] Andruczy, D., Ruzic, D.N., Curreli, D., and Allain, J.P., *"HIDRA: Hybrid Illinois Device for Research and Applications,"* Fusion Science and Technology **68** (2015) 497. https://doi.org/10.13182/FST14-989

[34] Majeski, R., Kolemen, E., Hvasta, M., Kozub, T., and Winkelman, J., *"Development of Liquid Metal Divertors for Fusion,"* Am. Nucl. Soc. Fus. Energy Div. Newsleter (2017) http://fed.ans.org/wp-content/uploads/2017/12/Dec-2017-Newsletter.pdf

[35] Hu, J.S., Zuo, G.Z., Ren, J., Yang, Q.X., Chen, Z.X., Xu, H., Zakharov, L.E., Maingi, R., Gentile, C., Meng, X.C., Sun, Z., Xu, W., Chen, Y., Fan, D., Yan, N., Duan, Y.M., Yang, Z.D., Zhao, H.L., Song, Y.T., Zhang, X.D., Wan, B.N., Li, J.G., and Team, E., *"First results of the use of a continuously flowing lithium limiter in high performance discharges in the EAST device,"* Nucl. Fusion **56** (2016) 046011. http://stacks.iop.org/0029-5515/56/i=4/a=046011

[36] Goldston, R.J., Myers, R., and Schwartz, J., *"The lithium vapor box divertor,"* Physica Scripta **2016** (2016) 014017. http://stacks.iop.org/1402-4896/2016/i=T167/a=014017

[37] **Linear Plasma Test Stand**

Presentations made at MFRSD Workshop -- Madison, July 24-28, 2017

Rapp — Development of the PMI science and PFC technology for fusion reactors with steady-state linear plasma devices

White papers submitted to MFRSD Workshop -- Madison, July 24-28, 2017

Rapp — Development of the PMI science and PFC technology for fusion reactors with steady-state linear plasma devices

[38] JULE-PSI, a LPTS planned by the Institute of Energy and Climate Research Plasma Physics IEK-4, Forschungszentrum Jülich. http://www.fz-juelich.de/iek/iek-4/EN/Research/03_HML_ENG/03d_JULE-PSI/_node.html

[39] Rapp, J., Biewer, T.M., Bigelow, T.S., Caughman, J.B.O., Duckworth, R.C., Ellis, R.J., Giuliano, D.R., Goulding, R.H., Hillis, D.L., Howard, R.H., Lessard, T.L., Lore, J.D., Lumsdaine, A., Martin, E.J., McGinnis, W.D., Meitner, S.J., Owen, L.W., Ray, H.B., Shaw, G.C., and Varma, V.K., *"The Development of the Material Plasma Exposure Experiment,"* IEEE Transactions on Plasma Science **44** (2016) 3456. https://doi.org/10.1109/TPS.2016.2628326

[40] **High Field Side RF Systems**

Presentations made at MFRSD Workshop -- Madison, July 24-28, 2017

Bonoli — PMI Challenges and Path towards RF Sustainment of Steady State Fusion Reactor Plasmas

White papers submitted to MFRSD Workshop -- Madison, July 24-28, 2017

Caughman — Reliable Long-Pulse Plasma Heating and Current Drive using ICRF

Bonoli — PMI Challenges and Path towards RF Sustainment of Steady State Fusion Reactor Plasmas

White papers submitted to Fusion Energy Sciences Advisory Committee on Transformative Enabling Capabilities

Wukitch — Path towards RF Sustainment of Steady State Fusion Reactor Plasmas

Presentations at FESAC TEC Meeting July 19-21, 2017

Wukitch — Path towards RF Sustainment of Steady State Fusion Reactor Plasmas

[41] Wallace, G.M., Bonoli, P.T., Wukitch, S.J., Wright, J.C., Kessel, C.E., Davis, A., and Rognlien, T., *"Heating and current drive actuators study for FNSF in the ion cyclotron and lower hybrid range of frequency,"* Fusion Engineering and Design**, in press** (2017) http://www.sciencedirect.com/science/article/pii/S0920379617307251

[42] Bonoli, P.T., Baek, S.G., LaBombard, B., Greenwald, M., Leccacorvi, R., Lin, Y., Marmar, E.S., Palmer, T.R., Parker, R.R., Porkolab, M., Shiraiwa, S., Sorbom, B., Vieira, R., Wallace, G.M., White, A.E., Whyte, D.G., Wilson, J.R., Wright, J.C., and Wukitch, S.J., *"Novel Reactor Relevant RF Actuator Schemes for the Lower Hybrid and the Ion Cyclotron Range of Frequencies,"* presented at the 2016 IAEA Fusion Energy Conference, Kyoto [TH/5-1], 2016. https://conferences.iaea.org/indico/event/98/session/17/contribution/754.pdf

[43] **Divertor Test Tokamak**

Presentations made at MFRSD Workshop -- Madison, July 24-28, 2017

Brunner — A strategy to solve divertor heat flux and plasma-material interaction challenges for fusion

White papers submitted to MFRSD Workshop -- Madison, July 24-28, 2017

LaBombard — A strategy to solve divertor heat flux and plasma-material interaction challenges for fusion - aided by the development of demountable superconducting magnets

LaBombard — ADX: A compact, high-field, high power density Divertor Test Tokamak (DTT) and RF Sustainment Test Tokamak (STT) for fusion energy development

White papers submitted to Fusion Energy Sciences Advisory Committee on Transformative Enabling Capabilities

Brunner — Developing a reactor power exhaust solution by testing advanced divertors in a compact, divertor test tokamak

LaBombard — Long-leg divertors with secondary x-points: a potential solution for divertor heat flux and PMI challenges -- aided by the development of demountable HTS magnets

Presentations at FESAC TEC Meeting June 20-22, 2017

Brunner — Developing a reactor power exhaust solution by testing advanced divertors in a compact, divertor test tokamak

[44] LaBombard, B., Marmar, E., Irby, J., Terry, J.L., Vieira, R., Wallace, G., Whyte, D.G., Wolfe, S., Wukitch, S., Baek, S., Beck, W., Bonoli, P., Brunner, D., Doody, J., Ellis, R., Ernst, D., Fiore, C., Freidberg, J.P., Golfinopoulos, T., Granetz, R., Greenwald, M., Hartwig, Z.S., Hubbard, A., Hughes, J.W., Hutchinson, I.H., Kessel, C., Kotschenreuther, M., Leccacorvi, R., Lin, Y., Lipschultz, B., Mahajan, S., Minervini, J., Mumgaard, R., Nygren, R., Parker, R., Poli, F., Porkolab, M., Reinke, M.L., Rice, J., Rognlien, T., Rowan, W., Shiraiwa, S., Terry, D., Theiler, C., Titus, P., Umansky, M., Valanju, P., Walk, J., White, A., Wilson, J.R., Wright, G., and Zweben, S.J., *"ADX: a high field, high power density, advanced divertor and RF tokamak,"* Nucl. Fusion **55** (2015) 053020. http://stacks.iop.org/0029-5515/55/i=5/a=053020

Abbreviations and Symbols

PMI	Plasma-Materials Interactions
DEMO	Demonstration power reactor or pilot plant
DTT	Divertor Test Tokamak
PFCs	Plasma Facing Components
AM	Advanced Manufacturing
UHTC	Ultra high temperature ceramics
MAX	Layered, hexagonal carbides and nitrides that have the general formula: $M_{n+1}AX_n$ where n = 1 to 3, M is an early transition metal, A is an A-group (mostly IIIA and IVA, or groups 13 and 14) element and X is either carbon and/or nitrogen.
HEA	High entropy alloys
LM	Liquid metal
MHD	Magnetohydrodynamic
Pedestal	Region of steep plasma pressure gradients at the edge of confined plasma
SOL	Scrape-off layer – plasma region on open magnetic field lines
Recycling	Plasma ions (hydrogen, deuterium, tritium) that impact wall surfaces are 'recycled' as neutrals at a rate that depends on the wall material and its level of hydrogenic saturation
ELMs	Edge localized modes – quasi-periodic bursts of hot, dense plasma into the SOL arising from instabilities in the Pedestal
LPTS	Linear Plasma Test Stand
HFS RF	High field side radio frequency
NBI	Neutral beam injection
ECCD	Electron cyclotron current drive
T_{et}	Divertor plate target electron temperature
T_{it}	Divertor plate target ion temperature
n_{et}	Divertor plate target electron density
$n_{//}$	Parallel index of refraction, $n_{//} = ck_{//}/\omega$. $k_{//}$ is component of parallel wavenumber along magnetic field.
$q_{//}$	"Upstream" parallel heat flux entering into the divertor region

Tritium Fuel Cycle

Description

For the deuterium-tritium fusion fuel cycle, the large quantities of tritium required must be produced in the fusion facility itself, and is most efficiently done with lithium bearing materials. Tritium is produced by fusion neutrons interacting with Li-7 and Li-6 isotopes. Surrounding the plasma are blanket structures that contain these lithium materials, either solid (e.g. Li_4SiO_4) or liquid (e.g. liquid metal $Li_{15.7}Pb_{84.3}$). Blankets have multiple simultaneous functions, such as absorbing neutron heating, provide neutron shielding, and breeding tritium, but here the focus is on the tritium aspects. Since tritium is radioactive and easily bonds with water or other biological molecules, it must be strictly controlled, with a facility releasing only very low quantities < 1 g/year [1-4]. This is in spite of generating and handling ~ 10's of kg's annually in a Fusion Nuclear Science Facility (FNSF) or ~ 100's of kg's annually in a commercial power plant, and injecting and exhausting ~10 times these amounts into and out of the plasma chamber. Tritium, being an isotope of hydrogen, can easily migrate through systems. Most importantly it can move right through solid materials by entering their matrix and diffusing. Since the temperatures associated with the fusion core (e.g. blankets, divertors), near core (vacuum vessel, cryostat) and even apparatus beyond the core (e.g. tritium extraction, heat exchanger) range from 300-700 ^oC, tritium diffusion is rapid and will lead to tritium moving throughout these zones. Highly precise behavior predictions, control, and accounting are required to maintain the plant tritium releases to the lowest tolerable levels. Description of the various issues related to tritium breeding, extraction, processing and handling are detailed below.

1) Breeder Materials Behavior in the Fusion Core

In the fusion core, where the tritium breeder resides, tritium will be produced either in a liquid or solid breeder material [5]. The primary liquid candidate is Li-Pb [6], which has lead as a neutron multiplier, and generally has an enriched Li-6 fraction relative to Li-7 (natural Li is 93% Li-7 and 7% Li-6). Primary solid candidates are Li_2TiO_3 or Li_4SiO_4, which are in the form of pebble beds [7] or cellular ceramics [8]. These require an additional neutron multiplier, such as $Be_{12}Ti$ [9], and can also require Li-6 enrichment. Tritium produced in the liquid mostly stays in the liquid but will also diffuse and enter surrounding materials as it flows through the blanket and out to an extraction apparatus and heat exchanger. For the solids, the tritium must diffuse out of the solid into the open pore spaces and then into a gas stream that takes it to an extraction apparatus.

The study of liquid metals in a magnetic field is complex, and understanding the breeder flow behavior in a magnetic field under heating, high temperatures, corrosion and mass transport, and gas production and transmutation has not been established. Liquid metal science challenges break into three main areas, with strong coupling among them: MHD thermo-fluid phenomena, liquid metal interaction chemistry and mass transport, and the electrical/thermal insulator required for the liquid metal breeder to be feasible [10]. The fluid flow structure of the liquid metal in a magnetic field will have 3D and non-steady features since this type of flow will not be fully developed, and is subject to a range of flow instabilities. In addition, the asymmetry of the heating and magnetic field lead to asymmetries in the flow and interactions with the conduit walls. The corrosion of a conduit wall can be up to 10x higher due to the magnetic field and its orientation, relative to the wall and flow direction [11,12]. The high operating temperatures aggravate these mechanisms. The flow channel insert material [13] provides its own challenges, since it must provide both electrical and thermal insulation while minimizing its own interaction with the liquid metal. These phenomena are made more complex by the presence of ionizing gamma and neutron radiation in the fusion environment. The liquid metal facility at UCLA, Maple, running Li-Pb is a critical inroad to developing the knowledge base for liquid metal breeders.

Even in the case of solid breeder materials, our understanding of their behavior is quite limited. High operating temperatures are required to guarantee tritium release, while excessively high temperature will lead to sintering (coalescence of the solid, removing porosity). Processes like these are aggravated by neutron irradiation. The solid breeder material will interact with its steel container, and it will be consumed as neutrons transmute lithium into He and tritium. The other constituents (e.g. Si, O, Ti) will undergo transmutations as well. The associated neutron multiplier, a beryllium compound, will also undergo transmutation/consumption as Be transforms into helium and neutrons. These solid breeders remain in the blanket of the fusion core for extended periods and their evolution is critical to maintaining a viable blanket, while liquid breeders are continuously flowing into and out of the blanket.

2) <u>Tritium Extraction from the Breeder or Purge Stream</u>
For liquid breeders, the breeder flows out of the blanket and fusion core to a tritium extraction apparatus. The most recent examination [14] of this is targeting group 5 elements on the periodic chart, which have high permeability for hydrogen and can serve as vacuum permeation windows, with potential to remove 80% of the tritium in a single pass according to simulations. The liquid metal flows past these window materials and tritium and deuterium adsorb onto the window and then move through the material to a vacuum where it is taken to processing. Challenges for this approach lie in the high temperatures required for fusion and possible low levels of impurities that can degrade the window material. Industrial hydrogen purifiers have been produced for lower temperature operation [15], but also show that oxidation can be controlled, and interlayers may stabilize the window materials at higher temperatures [16]. This approach has not been demonstrated even on small scale experimentally, and requires a dedicated activity to establish its feasibility.

For solid breeders a purge gas (usually helium) is used to gather the tritium diffusing out of the solid and transport it out of the blanket and fusion core. The tritium must be removed from the helium gas stream. Again, a vacuum permeation window may be the best option to isolate the tritium (and deuterium) from other impurities in the gas stream. Getters are well established for removing hydrogen, and many other materials, from gas streams, but they may not be sufficiently selective to isolate tritium. Tritium will also have to be recovered from all helium cooling flows from the fusion core, and if liquid metal plasma facing components are considered, then tritium must be extracted from these fluids, which may not be the same as the breeder fluid, requiring different methods.

3) <u>Tritium Behavior in the Fusion Core, Near Core, and Tritium Intensive Apparatus</u>
Tritium will migrate throughout the fusion core from its production in the breeder, and its introduction into the plasma chamber by the fueling system. Although the behavior of tritium in a fusion system is governed by physical chemistry at a basic level, the actual environment aggravates and complicates this tremendously. The experimental data on

various tritium properties used to calculate its behavior (e. g. diffusivity, solubility, and surface dissociation and recombination rate coefficients) have extremely large variations, due to practical system variations, such as the condition of a surface, or inherent difficulties in measuring very small amounts of non-radioactive hydrogen isotopes. The resulting impact on the amount of tritium that could be lost can be 50x, based on simulations to explore this impact [17]. The neutron irradiation environment will significantly aggravate properties, and likely generate synergies that must be understood to the extent possible, such as enhanced trapping of tritium in solid material due to damage or even the nanostructured particles introduced to enhance the material's radiation resistance. Multiple materials are present and in contact in the fusion core, and tritium will migrate through them and across these interfaces. Reliable tritium permeation barriers are not available in spite of decades of research to produce them [18], largely due to unknowns regarding their performance in neutron and gamma radiation and high temperature environments and over long times. Simultaneously, large amounts of tritium are injected into and exhausted from the plasma chamber (burn fraction will be only a few percent, at best), and some will be implanted in the first wall and divertor materials, will be trapped by eroded and re-deposited material, or adhere to dust. A comprehensive knowledge, and predictive capability, of the tritium behavior over a wide range of materials and environmental conditions is required to allow safe fusion systems to operate. Much of this uncertainty is associated with experiments that do not characterize the material or material surfaces sufficiently, experiments that operate at uncharacteristic hydrogen pressures, experiments that do not simulate the prototypical environments, and the ultimate aggravation that will be presented by irradiation on all these mechanisms. The US has had the deepest and most respected hydrogen research anywhere in the world, and is reflected by recent work by Causey [18] that identifies the many flaws and difficulties (complex material physics) in experiments performed over the years. The unique tritium-capable facilities at INL have been utilized by Japan for several years as part of collaborations, and the US is also uniquely qualified to pursue the complex issues associated with tritium in a fusion facility from the vast experience at LANL and SRNL (engaged in ITER tritium system design and operation).

4) <u>The Plasma Fueling and Exhaust of Tritium (and Deuterium) to/from Plasma Chamber</u>

A fusion facility will require that tritium and deuterium are injected into the plasma chamber to sustain the burning conditions, and that the unburnt fuel, reaction byproducts (He) and impurities are removed. Studies indicate [19,20] that the amount of injected fuel that is actually burned can be relatively low, and is determined by complex particle physics in the plasma, scrape-off layer (SOL) and divertor. Fueling the plasma can only be accomplished by pellet injection, since the efficiency of SOL gas penetrating the plasma is extremely low in ITER and future devices (minimizing recycling). In addition, the plasma and atomic physics in the divertor affect the accumulation of species there, and these things conspire to produce low tritium (and deuterium) consumption. The residence time of helium in the plasma, which leads to fuel dilution, and the residence time of tritium in the plasma, which leads to higher fuel burnup, are correlated. Significantly better understanding of the particle physics in a burning plasma device is required to maximize fuel burnup in a self-consistent way with exhaust, core plasma purity and pumping capability. If the tritium burnup fraction (fraction of injected fuel that is consumed in fusion reactions) is 10%, then the fueling/exhaust system is cycling 10x the amount of tritium consumed. This requires a significant inventory of tritium to be sustained in the fueling cycle. In the case of liquid metal plasma facing components, depending on the liquid metal, it can act as a getter (no recycling) or very similar to a solid (high recycling) and can influence the tritium fuel cycle.

5) Tritium Processing in the Fusion Facility

Once neutral particles are exhausted from the plasma chamber by pumping, they must be recovered, the hydrogen separated, hydrogen isotopes separated, and then sent back to the fueling system. It is possible to streamline this process if the hydrogen isotopes do not need to be separated before fueling, but this would compromise one's ability to precisely adjust the D and T injection levels, and would require careful measurement of isotope mix. ITER has provided a tremendous leap in tritium processing due its higher inventory and processing flow rates [21-24]. A key technology and safety challenge for fusion reactors is the quantity of tritium fuel being processed (2-3 kg for ITER tritium plant) and the rate at which this tritium must be processed (maximum 200 Pa-m^3/s for ITER) while at the same time minimizing tritium release to the environment during operation and under accident conditions. As illustrated in Day et al. [23], these challenges only grow in magnitude for a demonstration reactor (DEMO), where the inventory and processing rate are anticipated to increase by a factor of ~4 above ITER for a 2 GW fusion power device. The majority (~80%) of this tritium resides in the fuel processing plant's cryogenic isotope separation system (ISS) (~60%) and on the reactor's vacuum vessel (VV) cryopumps (20%). In addition it is uncertain if cryopumps will prove to be an effective VV pumping option for a steady state fusion reactor like DEMO. This uncertainty relates to possible reliability concerns for cryopumps given their transient mode of operation, i.e. cycled fuel loading and unloading modes.

A solution called the "Direct Internal Recycling" (DIR) approach has been proposed that has the potential for reducing the DEMO tritium processing plant size to that of ITER's, or 75% smaller. A key technology proposed for the DIR approach is called a "superpermeable" metal foil pump (MFP) [25-28]. The MFP is a steady state, high-temperature vacuum pump that works by directly extracting the unburnt hydrogen fuels from the plasma exhaust, instead of condensing them. Because this extracted fuel is free from plasma exhaust impurities, it can be sent directly to the reactor's fueling system for reinjection into the plasma instead of to the fuel processing plant. Another approach identified [29] is to continuously remove the hydrogen ice in the cryopump by scrapping the cryopanel, recovering the hydrogen which is then sent to the fueling system. This is based on the fact that cryopumps can differentially adsorb different materials, so that hydrogen can be isolated to specific panels.

Benefit

A fusion facility cannot function without a closed tritium fuel cycle, and this represents a fundamental feasibility issue for fusion power production. Tritium provides a difficult species for control, accounting, and safety, yet it is critical to the fusion fuel cycle. In order for fusion to realize its maximum potential for safe operation and benign environmental impacts, high fidelity understanding of all processes involving tritium is required. The tritium fuel cycle has a very broad footprint on any fusion facility, the breeding of tritium in the blanket surrounding the plasma, tritium burn fraction in the plasma, extraction efficiencies from the breeder and coolant streams, tritium processing time from plasma exhaust to fueling, tritium losses from and inventories in the fusion core, near-core and ex-core subsystems, and many more constitute a complex and interacting system. This is an essential capability for a fusion power plant, and so advances in these areas would bring a power plant to reality more quickly [30-31].

Current Status

Virtually all of the technologies related to the tritium fuel cycle are at low technical readiness, with widely varying parameters that describe tritium's migration through materials, across interfaces, and its retention in bulk solids and liquids, and retention and behavior in plasma facing materials. Extraction of tritium from breeder materials is still highly uncertain, and the development of tritium barriers has been largely unsuccessful. ITER has provided a strong step in tritium processing and the fueling/exhaust tritium loop, with higher amounts of tritium required in the future (relative to ITER). Breeder material behavior and interactions are still at a low level of understanding.

Programmatic Context

The ITER Test Blanket Module (TBM) program will be the first context where the full, integrated tritium fuel cycle environment is present. Although the amount of bred tritium to be handled is not significant, due to a small testing area and low plasma duty cycle, the tritium transport processes involved in the TBM program are representative phenomena of a DEMO FW/blanket tritium fuel cycle, including a D/T neutral ion flux implantation and consequent transport/permeation under prototypical tokamak plasma facing surface and operating conditions. However, the TBM program does not fall under the larger international ITER agreement; data from ITER testing will only be shared through "partnership." An approach could be for the US to seek supporting partnerships with two or three ITER TBM Leaders to gain access to and experience with substantial R&D results, nuclear design, instrumentation and control, safety and licensing processes, and integrated TBM testing in ITER H/D/T phases that will include all tokamak normal and off-normal operation conditions except significant neutron fluence. In such a partnership collaboration, for example, the US can contribute critical property data such as recombination coefficients, tritium diffusivity in PbLi, MHD mixed convection on tritium transport, etc., through small scale laboratory tritium experiments. It should be noted that there are very few applications for these technologies outside of fusion. A few areas of possible cooperation with Generation IV fission reactors are in lithium isotope separation (for higher concentrations of Li-6) and in tritium extraction techniques. Overall, international collaboration on the various aspects of the tritium fuel cycle and the accompanying areas of fusion nuclear materials, plasma facing materials, fusion nuclear science, and enabling technologies requires serious consideration.

Possible 15-year U.S. Research Agenda

Tritium/deuterium migration data is needed in appropriate materials, temperatures, partial pressures, surface conditions, multi-material environments, and plasma facing environments. These experiments move from basic to more integrated as the actual component and its environment are made prototypical, albeit without neutrons. Some testing with neutron irradiated samples is also required. Both solid and liquid metal breeder studies are required to understand their behavior and interactions, and requirements (e.g. insulator for LMs). Both non-nuclear and nuclear test can and should be pursued. Liquid metal loops are needed and ultimately an apparatus in HFIR could be developed. Tritium extraction schemes require R&D to establish their feasibility and optimization for the fusion environment. [32] Prototypical fluids are needed, with impurities, and at prototypical conditions is required. Exploration of approaches to fueling, exhaust and recovery processing are critical to making the tritium cycle more efficient. The tritium fuel cycle research will converge and culminate with a multi-function integrated tritium-breeding blanket that must endure the multi-physics environment of a fusion core. Apart from the tritium fuel cycle research thrust, the fusion nuclear materials and plasma facing materials areas would impact this development directly.

Research Directions Beyond 15-year Horizon

A Fusion Nuclear Science Facility (FNSF), ranging from a volumetric neutron source [33] to a FNSF [34] to a pilot plant [35] (net electricity), is the target beyond the 15-year time frame, and requires that the tritium related issues are explored and understood to a level sufficient to pursue such a device. All behavior cannot be established before the FNSF, particularly in the complex integrated environment of a fusion core, and the device will continue defining the fuel cycle requirements for a power plant. The FNSF (or similar) is the only component-level, integrated fusion facility that has been proposed internationally prior to DEMO, indicating a US strategy based on a break-in fusion nuclear step, followed by a US demonstration power plant.

Critical Objections and Advocates Response

This research can be delayed until we are ready to build a fusion nuclear device. This is generally untrue, since the R&D in the various tritium topics will require basic science and progressive experimentation toward prototypical conditions. This provides the technical basis to pursue a fusion nuclear device, and requires several years to complete. Postponing this R&D will generate a ~ 15-year delay to any decision to move on toward next steps in fusion. Feasibility demonstrations are needed to establish the credibility of approaches, and allow innovation and optimization to generate attractive solutions for the long term. Not to mention the licensing issues which could delay the construction of a power plant even if all technical obstacles are resolved.

We must choose among specific blanket concepts to make this research focused, be able to afford it, and have a reasonable timeframe. This is probably accurate; however strategies can be designed to avoid carrying several differing blanket concepts simply because they are immature in their technical readiness. Blanket concepts can be chosen by their simulated performance in a power plant, where attractive thermal conversion efficiency, tritium fuel self-sufficiency, simplicity, and long-term relevance are optimized. An attractive blanket concept can be "backed-down" by changing the most vulnerable aspects, the breeder for example. The Dual Coolant Lead-Lithium [36] blanket can be chosen as primary, with the Helium Cooled Lead Lithium [37] as first back-down where liquid metal effects are weakened while overall performance is compromised somewhat. Next a solid breeder can be envisioned as the next back-down, to eliminate the liquid metal breeder (and its complications) altogether, say with a Helium Cooled Pebble Bed [38] or Cellular Breeder concept. Due to the similar structural material and primary helium coolant, these blanket concepts can be carried more efficiently in the US program, or can involve international collaborators.

References

[1] D. Babineau, SRNL, ITER tritium loss targets, tritium handling systems: www-naweb.iaea.org/napc/physics/fec/fec2010/talks/itr_2-2.pdf
[2] Petti, Tritium Safety Issues, aries.ucsd.edu/LIB/MEETINGS/0103-ARIES-TTM/Petti.ppt
[3] Cadwallader, Tritium and Safety Issues for Power Plants, aries.ucsd.edu/ARIES/MEETINGS/1104/Cadwallader.ppt
[4] https://www.nrc.gov/reading-rm/doc-collections/fact-sheets/tritium-radiation-fs.html
[5] El-Guebaly, Breeding Potential for Candidate Breeders for US DEMO Power Plant, IEEE SOFE 1995: http://ieeexplore.ieee.org/stamp/stamp.jsp?arnumber=534441
[6] S. Malang et al, Fusion Sci Technol, **60**, (2011), 249.
[7] E. Proust et al, Fusion Eng Des, **16**, (1991), 73.
[8] B. Williams, 2017 *"Robust Cellular Solid Breeder Offering Potential for New Blanket Designs with High Tritium Breeding Ratio", FESAC-TEC whitepaper*
[9] H. Kawamura et al, Fusion Eng. Des., **61-62**, (2002), 391.
[10] N. Morley et al, Fusion Sci Technol, **47**, (2005), 488.
[11] M. Gazquez et al, J. Nucl. Mater., **465**, (2015), 633.
[12] R. Moreau et al, Fusion Eng. Des., **86**, (2011), 106.
[13] Malang, S. (1986). *Flow channel insert for the reduction of MHD pressure drop in liquid metal flow* (IAEA-TECDOC--373). International Atomic Energy Agency (IAEA).
[14] P. W. Humrickhouse and B. J. Merrill, Fusion Sci Technol, **68**, (2015), 295.
[15] V. Alimov et al, J. Hydorgen Energy, **36**, (2011), 7737.
[16] Edlund et al, J. Membrane Sci, **107**, (1995), 147.
[17] F. Franza et al, IEEE Trans Plas Sci, 42, (2014), 1951.; Tritium Transport Analysis in the HCBP-DEMO Blanket With the FUS-TPC Code, KIT report 7642, https://www.ksp.kit.edu/download/1000034375
[18] R. Causey et al, Tritium Barriers and Tritium Diffusion in Fusion Reactors; https://www.sciencedirect.com/science/article/pii/B9780080560335001166
[19] A. S. Kukushkin et al, J. Nucl. Mater., **415**, (2011), S497.
[20] A. Loarte, The Plasma Physics Aspects of the Tritium Burn Fraction in ITER, IAEA DEMO Workshop, KIT, December 2016.
[21] I. R. Cristescu *et al.*, 2007 *Nucl. Fusion* **47** 5458
[22] M. Glugla et al, Fusion Eng Des, **82**, (2007), 472: Gluglia and S. Willms, Lecture on Tritium: http://tritium.nifs.ac.jp/project/20/pdf/t2.pdf
[23] C. Day, 2004 *"Validated design of the ITER main vacuum pumping systems", 20th IAEA Fusion Energy Conference, paper IT/P3-17*
[24] B. J. Peter, and C. Day, 2017 *Fus. Eng. Des.,* http://dx.doi.org/10.1016/j.fusengdes.2017.05.124 at press
[25] M. Shimada, 2017 *"Superpermeable Metal Foil Pump for Increasing Tritium Burn Up Fraction for DEMO," TEC whitepaper*
[26] A. Güntherschulze, H. Betz, and H. Kleinwächter, 1939 *Z. Physik* **111** 657
[27] A. L. Livshits, 1976 *Sov. Phys. Tech. Phys.* **21** 187
[28] A. L. Livshits, 1997 *J. Nucl. Mater.* **241** 1203
[29] C. A. Foster and H. C. McCurdy, IEEE SOFE, 1993: http://ieeexplore.ieee.org/stamp/stamp.jsp?arnumber=518500 & Journal of Vacuum Science & Technology A: Vacuum, Surfaces, and Films **5**, 2558 (1987)
[30] B. Bornschein, C. Day, D. Demange, and T. Pinna, *Fus. Eng. Des.*, **88**, (2013), 466.
[31] T. Tanabe, *J. Nucl. Mater.*, **438**, (2013), S19.
[32] D. Demange et al, ISFNT, 2015, http://www.euro-fusionscipub.org/wp-content/uploads/2015/11/EFCP150616.pdf.
[33] M. Abdou et al, Fusion Eng. Des., **27**, (1995), 111.
[34] C. E. Kessel et al, Fusion Eng. Des., 2017, https://doi.org/10.1016/j.fusengdes.2017.05.081.

[35] J. E. Menard et al, Nucl. Fus., **51** (2011), 103014.
[36] S. Malang et al, Fusion Eng. Des., **14**, (1991), 373.
[37] A. Li Puma et al, Fusion Engr. Des., **81**, (2006), 469.
[38] S. Hermsmeyer et al, Fusion Engr. Des., **75-79**, (2005), 779.