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FUSION ENERGY DEVELOPMENT APPLICATIONS UTILIZING THE SPHERICAL TOKAMAK AND ASSOCIATED RESEARCH NEEDS AND TOOLS

J.E. MENARD, R. MAJESKI, M. ONO
Princeton Plasma Physics Laboratory
Princeton, NJ USA
Email: jmenard@pppl.gov

N.N. BAKHAREV, V.K. GUSEV
Ioffe Institute
Saint Petersburg, Russia

M. GRYAZNEVICH, D. KINGHAM, S. MCNAMARA, P. THOMAS
Tokamak Energy Ltd
Oxfordshire, United Kingdom

K. HANADA
Kyushu University
Fukuoka, Japan

J. HARRISON, B. LLOYD
Culham Centre for Fusion Energy (CCFE)
Oxfordshire, United Kingdom

Y.S. HWANG
Seoul National University
Seoul, South Korea

B. LIPSCHULTZ, H. WILSON (also CCFE)
University of York
York, United Kingdom

Y. NAGAYAMA
Nihon University
Tokyo, Japan

Y. ONO, Y. TAKASE
The University of Tokyo
Tokyo, Japan

M. REINKE
Oak Ridge National Laboratory
Oak Ridge, TN, USA

K. TOBITA
National Institutes for Quantum and Radiological Science and Technology
Aomori, Japan

Z. GAO
Tsinghua University
Beijing, China

F. ALLADIO
ENEA
Rome, Italy

R.J. FONCK
University of Wisconsin-Madison
Madison, WI, USA

Abstract

The spherical torus/tokamak (ST) is a potentially attractive configuration for narrowing scientific and technical gaps to a fusion demonstration power plant and as a more compact and/or modular fusion power source. Due to a reduced plasma surface area to volume ratio, the ST configuration offers the potential to access high power exhaust heat fluxes and high neutron wall loading in devices of modest size. Further, due to increased ability to utilize magnetic pressure to confine a given plasma pressure, the ST may also offer fusion solutions with reduced magnet mass and increased mass power density. Potential fusion development and energy production applications of the ST include: divertor and first-wall heat-flux mitigation research and development (including tests of advanced divertor configurations and liquid metals), neutron sources with low-to-moderate fusion gain (Q=0.1-1) for hybrid applications, neutron sources with moderate gain (Q=1-5) for fusion nuclear science and component testing including breeding blanket development, higher gain (Q=5-10) systems for electricity break-even in a pilot plant, and high gain (Q > 10) modular (P_{elec} = 100-500MWe) and larger-scale (P_{elec} = 500-1000MWe) fusion power plants. Details of these applications including major scientific and technology needs and gaps for fusion energy development are described in this paper. Motivations for utilizing the ST for these applications and remaining physics and technological gaps to be addressed to realize the application are also discussed.

1. INTRODUCTION

The spectrum of scientific and technological gaps that must be closed to achieve practical fusion energy has been extensively documented in detailed reports both within the U.S. [1] and internationally [2,3,4,5]. A common barrier to narrowing or closing these gaps is the scale and cost of fusion facilities needed to address these gaps. For tokamaks, this motivates consideration of alternatives to the conventional aspect ratio tokamak. The low-aspect-ratio spherical torus/tokamak (ST) is a potentially attractive configuration for narrowing scientific and technical gaps to a fusion demonstration power plant and as a more compact and/or modular fusion power source in its own right. A distinguishing feature of the ST is a higher ratio of plasma pressure to externally applied magnetic field pressure and the corresponding potential for reduced mass and cost of the toroidal field magnets and fusion core in general. Due to the reduced plasma surface area to volume ratio, the ST configuration offers the potential to access high power exhaust heat fluxes and high neutron wall loading in devices of modest size. Potential challenges for the ST approach include the necessity to generate/ramp-up and sustain the plasma current non-inductively. Further, the physics basis – in particular, for energy confinement – is not as well established as it is in conventional tokamaks. However, several upcoming experiments aim to significantly expand the ST physics basis and provide more confidence in projecting the performance of next-step ST facilities. In the sections that follow, details of a range of ST applications are described, including major scientific and technology needs and gaps for fusion energy development. Further, motivations for utilizing the ST configuration for these applications and remaining physics and technological gaps to be addressed to realize the application are also discussed.

2. FUSION DEVELOPMENT APPLICATIONS OF THE ST

Fusion development paths and device applications can be organized according to both performance gaps and projected fusion performance / fusion gain. In this paper, ST applications Sections are organized as follows:

- 2.1 Developing solutions for the plasma-material-interface (PMI) challenge
- 2.2 Fusion-fission hybrid systems
- 2.3 Developing fusion components capable of withstanding high fusion neutron flux and fluence
- 2.4 Demonstrating electricity break-even from a pure fusion system
- 2.5 Electricity production in modular fusion power plants
- 2.6 Electricity production in larger-scale fusion power plants

NOTE: The work described here is a survey of potential ST applications and inclusion of a particular application. This does not necessarily represent endorsement of that application by all co-authors.

2.1. Developing solutions for the plasma-material-interface (PMI) challenge

In order for the ST to support the discussed fusion development applications, a set of key experimental milestones need to be achieved in the area of boundary science, primarily using NSTX-U [6,7], MAST-U [8,9,10], and ST40 over the next decade. These provide the existence proofs that are necessary but not sufficient for future operational scenarios They will provide both important information to scope new engineering designs using empirical models and generate data to support validation of modeling and simulation tools. Unordered for priority they are:

2.1.1. Develop a Consistent Empirical Heat Flux Width Scaling Across STs

Predicting the parallel heat flux, $q_{\|} = P_{SOL}B/2\pi R\lambda_q B_p$, enables designs of plasma facing components (PFCs) and sets boundary conditions for heat flux mitigation scenarios. Conventional aspect ratio devices have given consistent results when overlapping in engineering parameters, and recently Alcator C-Mod has confirmed the narrowing of the heat flux width out to B_p ~ 1.3 T [11]. For STs, there are presently multiple predictions which differ from the multi-machine scaling: $\lambda_{q,MM}[mm] = 1.35\varepsilon^{0.42}R_{geo}^{0.04}B_{pol,omp}^{-0.92}P_{SOL}^{-0.02}$ (Table 3, Regression #15 in Reference [12]). MAST found a much weaker poloidal field dependence and a stronger dependence on power into the scrape-off layer for H-modes, (Table 2 and Regression #2 in Reference [13]): $\lambda_{q,MAST}[mm] = 1.84(\pm 0.48)B_{pol,omp}^{-0.68(\pm 0.14)}P_{SOL}^{0.18(\pm 0.07)}$. In contrast, NSTX as well as Globus-M predicted a stronger variation with plasma current than is seen on conventional aspect ratio devices: $\lambda_{q,NSTX}[mm] = 9.1 I_p^{-1.6\pm 0.1}$ [14] and $\lambda_{q,Globus\text{-}M}$ [mm] ~ $I_p^{-1.4+/-0.2}$ [15]. Moreover, a further narrowing during operation with lithium evaporation was observed NSTX. When estimated for cases for the full operating parameters of NSTX-U the scalings give $\lambda_{q,MM} = 2.0\ mm$, $\lambda_{q,MAST} = 4.1\ mm$, $\lambda_{q,NSTX} = 3.0\ mm$. Such a disagreement, achievable within both MAST-U and NSTX-U operating space, will be observable within resolution of diagnostic sets, and experiments to explore heat flux width scalings are identified as high priority areas for early device operations. The ST40 tokamak, designed to operate up to B_p ~ 1T, will also make critical contributions in confirming how strong the narrowing is of λ_q with poloidal field, reducing the step between near-term machines and those listed in Table 1 supporting fusion energy development.

2.1.2. Combine High Performance Pedestals with Exhaust Solutions at High Parallel Heat Flux in the ST

The present class of STs are currently right on the border of necessitating heat flux mitigation to support operation. For double null, NSTX-U 2 MA, 1 T, 10 MW P_{NBI} scenarios, PFCs are being designed sustain 5 second flattops [6] using a combination of poloidal flux expansion and strike-point sweeping, assuming onlyonly 30% radiated power. ST40, while at high heat flux, is short pulse allowing for inertially cooled PFCs and MAST-U's large divertor surface area and flexible divertor PF coil set would allow, in absence of radiation, a large surface area for sweeping. Facilities used in fusion development applications will increase in power density and pulse length, necessitating increased radiated power in the core and boundary and active cooling, assuming a paradigm of solid material PFCs. Thus what is a presently a physics research topic becomes an existential operational issue for the next step. Strategies to address this could include using the early commissioning phase where the learning is done *in-situ* (much like planned ITER low-field operations) or an interim facility to demonstrate core-edge integration in the ST.

Fortunately, the new generation of STs will be able to investigate all conventional and advanced divertor scenarios via the combined and coordinated exploitation of NSTX-U and MAST-U. MAST-U will provide the fusion community with the first demonstration of a baffled and pumped divertor with high toroidal flux expansion (e.g the Super-X divertor). In contrast the baseline design of NSTX-U will utilize an open divertor and flat horizontal (vertical) plates for its outer (inner) divertor, but be able to operate at higher power and pedestal pressures. Both have sufficient inner poloidal field coils to support high poloidal flux expansion (e.g. X-divertor) and higher order poloidal field nulls (e.g. Snowflake divertors). Parallel heat fluxes are expected to reach 100's of MW/m^2, where volumetric losses by impurities are necessary. A key integration optimization will be the ultimate performance comparison of MAST-U and NSTX-U under pronounced and complete detachment, stimulated by extrinsic impurity seeding. MAST-U is expected to have superior control of detachment due to toroidal flux expansion, but may require low-δ equilibria which have reduced core performance. A key question for ST operation is what is the effect of X-point MARFE regimes on the pedestal and confinement, especially for open divertor geometries like NSTX-U. For STs, there are sufficient differences in magnetic geometry, such as larger elongation, and aspects of pedestal stability physics that require dedicated ST investigations beyond what has recently been achieved on conventional aspect ratio devices [16]. Additionally, NSTX-U will be able to operate P_{NBI} ~ 15 MW for $\Delta t < 2$ seconds. This will allow the ST to access the high P/P_{LH} regime that is necessary to investigate combined high-Z (core) and low-Z (edge) radiative exhaust. Significant progress is possible with existing devicesfor core/pedestal demonstrations to define the divertor boundary conditions. Until these studies are complete, we cannot be sure of the feasibility of radiative exhaust in future fusion energy development facilities.

2.1.3. Establish Sustained, High-Pressure ST Operation Without Carbon Plasma Facing Components

For the upgrades of MAST and NSTX, the decision to continue with graphite PFCs while knowing it ultimately is not reactor-relevant, has been made to balance the wider mission goals. Achieving the lowest collisionality is critical to validate the confinement scaling [17,18,19,20,21] and resistive wall mode physics [22]. Based on experience from JET and ASDEX Upgrade, we expect the integration of high-Z PFCs into ST H-mode scenarios to be challenging. Nevertheless, this is a key step towards fusion energy development for STs, so after high impact core/pedestal experiments have been completed, further upgrades of NSTX-U and/or MAST-U should explore large-scale replacement of PFCs with non-carbon alternatives. While there are STs which presently operate with metal walls, such as QUEST, the goal is to assess the integration of PFCs with the highest performance scenarios. The goal of ‘non-carbon’ is intentionally vague at this point, as there is not an established material solution that is capable of supporting all possible ST missions. Paradigms which use solid walls like tungsten will need to demonstrate high-Z particle control. This could be problematic for STs where neoclassical impurity transport is observed in the core [23] and may result in on-axis accumulation. Core electron heating tools that work to mitigate this on high aspect ratio devices are more limited in lower-field STs. Using laser ablation or high-Z gas injection, carbon machines can explore the fundamental impurity transport physics, and some initial insights into tokamak operation with high-Z PFCs could also be realized on ST40.

A possible alternative to high-Z PFCs with a radiative edge is to employ liquid lithium PFCs. The effects of low recycling PFCs on core confinement and SOL power flow can undergo initial exploration on LTX-β, and thorough testing on NSTX-U. Engineering development of flowing sytems can use test stands to develop flowing liquid lithium divertor targets and other plasma facing surfaces which could then be deployed toroidal devices for further. The ST would benefit from raising the technical readiness level for any tokamak-based testing of liquid metal (LM) solution, however deployment in a future ST-based reactor would benefit from testing in a present/near-term ST in order to test specifics of physics and engineering integration. Such an integration device would ideally have sufficiently long pulse to study LM flow startup, the effects of plasma startup/ramp-up on LM flow, LM surface stability during plasma instabilities, effective extraction of any implanted D from the LM inventory, and be able to demonstrate stable shutdown of the plasma and LM flow.

2.2. Fusion-fission hybrid systems

Development of hybrid fusion-fission systems systems [24,25,26,27] may sufficiently accelerate commercial usage of nuclear fusion. Hybrids can be used to generate electricity or breed nuclear fuel for LWRs. It is also possible to use hybrids for nuclear waste transmutation; however it’s not easy for them to compete in this field with fast-neutron reactors. Requirements for tokamak driver in hybrid system are much lower than for pure fusion tokamak considered for MFE applications. Desired discharge parameters, except stationary operation mode, were already reached at conventional tokamaks, e.g. JET or TFTR [28]. Such machines can be regarded as neutron source prototypes. However, keeping in mind required high availability of fusion neutron source driver for commercial usage, one must provide steady or quasi-steady state tokamak operation, which is impossible without application of superconducting technology. This will increase construction cost of huge conventional tokamak-hybrid driver by more than one billion dollars, which makes its construction unrealistic in the current conditions of world-wide fusion research funding.

On the other hand, STs have demonstrated significant progress in plasma performance during the last two decades and the second generation of STs is under construction: NSTX-U [6,7], MAST-U [8,9], Globus-M2 [29]. Prospects for plasma performance in these machines are positive, predicting low collisionality, high density, and sub-fusion temperatures. The time-scale of commercial fusion development makes achieving Q~1 conditions in the next-generation ST feasible and makes utilization of an ST-based neutron driver in the hybrid fusion-fission system potentially attractive due to compactness and possibly lower relative cost.

Fuel breeding is the easiest way to demonstrate commercial availability of fusion through combination with fission. Electric and T self-sufficiency is desirable but not necessary for demonstration of a working prototype. In hybrid breeders thermonuclear 14 MeV DT neutrons are used to produce 233U from 232Th or 239Pu from 238U. At present, the tokamak is the most promising solution for the 14 MeV neutron driver, generating neutrons in DT fusion reaction. Requirements for plasma performance in such a tokamak-driver are lower than that for pure fusion. A beam-plasma concept [30] may further simplify requirements. In contrast to having fusion reactions from a thermonuclear plasma dominate, neutrons in a beam-plasma-based tokamak are generated mostly due to fusion of fast ions, arising from NBI, with “slow” ions of non-thermonuclear target plasma. To potentially reduce the cost of the hybrid breeder, the ST may be used as a driver due to lower construction and operation cost and relatively small size.

Despite the fact that many requirements for beam-plasma ST drivers are lower than for pure fusion STs, several specific problems exist. As was mentioned above, to achieve high neutron rate in beam-plasma device, high Te should be provided, which is problematic in present STs due to anomalous electron transport. According to existing data [17, 20, 31], in future STs with higher toroidal magnetic field anomalous electron transport should be significantly reduced. However, applicability of these scaling for higher values of toroidal magnetic field should be verified in the new 1T STs: MAST-U [8], NSTX-U [6] and Globus-M2 [29]. Also in the future, high-B_T STs (including ST40) direct electron heating with ECR technology may be used [32].

Another class of problems is related to compatibility to the reliable fission technologies (such as simultaneous Tritium and nuclear fuel production) and hybrid system technologies (such as integration of fission blanket into fusion core or fusion blanket with fission reactor, and licensing). Development of the hybrid system technologies, which requires cooperation of two communities, may be facilitated by ST-based driver development. Compact ST configuration provides high neutron flux density in a quasi-cylindrical geometry, which allows a simple linkage between ST driver and nuclear blanket/fission reactor. Differentiation of fusion and fission parts not only makes possible electromagnetic and mechanical decoupling of the tokamak and blanket/fission reactor, but also decoupling of neutronic and thermos-hydraulic design for nuclear licensing.

Simplification of the hybrid system design may be further improved by adopting a molten salt reactor (MSR) [33] as a fission counterpart. Liquid MSR fuel can link the fusion blanket and fission reactor. Separation of the fusion blanket and the fission reactor allows significant relaxation of the blanket engineering design since it works only for transmutation without generation of the significant neutron and heat load, while a separate MSR generates major part of the heat and neutrons. Furthermore, MSR can generate significant amount of Tritium if an appropriate salt is used, leading to less severe requirements for Tritium breeding.

2.3. Developing fusion components capable of withstanding high fusion neutron flux and fluence

2.3.1. Component Test Facility – United Kingdom

The Component Test Facility, or CTF [34], has been designed to be a compact fusion neutron source for developing fusion materials and technologies (see Figure 1). A design philosophy was that the reliance on the technologies it was testing should be minimized, and therefore it should not need to breed tritium in order to operate; nevertheless it should demonstrate a tritium breeding capability. This constrains the tritium burnt to be obtainable from anticipated global markets, less than 1kg per year, and therefore constrains the number of fusion neutrons produced. To achieve the required neutron flux of at least $1MWm^{-2}$ for testing then drives one to a small device, with limited space for shielding the centre column. The design which was developed has a major radius of 81cm, aspect ratio of 1.55, TF rod current of 10.5MA and a plasma current of 6.5MA. A low density is employed to enhance current drive efficiency, 21% of Greenwald, and a modest β_N=3.5 avoids the challenges of the resistive wall mode, for example. This then ensures that for an assumed 30% availability (the design is steady state) and the derived fusion power of 35MW (including significant beam-thermal), a fluence equivalent to about 1dpa per year on the mid-plane can be achieved.

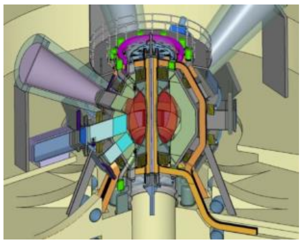

FIG. 1. ST-based Component Test Facility (CTF)

The compact design means there is limited room for neutron shielding, and therefore a decision was made to select copper for the TF conductor. With a relatively thin shield, the centre column is protected from most neutrons, but is nevertheless expected to become embrittled and develop an increased resistivity. These are taken into account in the design, in particular the maximum TF rod current allowed. Remote maintenance of the facility is enabled via a removable centre column that can be lowered into a hot cell below the device.

CTF is designed to operate in steady state, with neutral beams providing the baseline current drive capability – 10MW at 200keV (or 20MW at 150keV) on axis, and 34MW at 150keV for off-axis current. An alternative option for the on-axis current drive is 20MW of ECRH – 160GHz O-mode, for example. To handle this heating power, and that from the alphas, requires a novel exhaust scenario. An innovative cascading pebble design was developed, and some testing of the concept was performed [35]. Components testing is designed to be via “test blanket modules” that can be inserted into the CDT via ports. This provides a total testing area of approaching $11m^2$, with each module having an area of either $0.88m^2$ or $0.61m^2$.

2.3.2. ST Fusion Nuclear Science Facility – United States

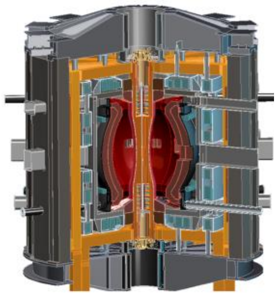

FIG. 2. ST Fusion Nuclear Science Facility (FNSF)

While there are several different definitions of the U.S. fusion nuclear science facility (FNSF) mission, the FNSF mission [36,37,38] is typically similar to that of a CTF but extends the device capability to include higher tritium breeding ratio (TBR) including TBR $\geq$ 1 to close the tritium fuel cycle. For larger devices with higher fusion power, tritium self-sufficiency is essential due to the large T consumption and limited supply of readily available tritium. Recent U.S. FNSF design activities have focused on developing self-consistent designs (see Figure 2) that include: (1) a blanket system capable of tritium breeding ratio TBR $\approx$ 1, (2) a poloidal field coil set supporting high elongation and triangularity for a range of internal inductance and normalized beta values, (3) a long-legged/super-X divertor which substantially reduces projected peak divertor heat-flux and has all outboard poloidal field coils outside the vacuum chamber and as superconducting to reduce power consumption, and (4) a vertical maintenance scheme in which blanket structures and the centerstack can be removed independently. Special attention has been paid to maximizing TBR in the presence of wall penetrations from plasma heating and from test-blanket modules and material-testing modules. Negative neutral beam injection (NNBI) heating and current drive with E_{NBI} = 0.5-0.7MeV is calculated to effectively support full non-inductive operation including non-inductive current overdrive ramp-up starting from initial plasma current levels as low as 2 MA. The NNBI blanket penetrations do reduce the TBR, but if NBI power fluxes as high as 50 MW/m^2 through blanket apertures can be supported, tangential injection and breeding at the back of the blanket are both computed to help reduce the impact of the NBI penetrations on TBR. Device major radius scans have also been carried out, and at A = 1.7, the plasma geometric major radius threshold for tritium self-sufficiency in a Cu-TF ST-FNSF is found to be approximately 1.7m. A smaller R_0 = 1 m ST-FNSF device has TBR $\approx$ 0.9 which is below unity but substantially reduces T consumption relative to not breeding. Shielding calculations indicate that the vacuum vessel, TF coils, outboard PF coils, and most or all of the divertor PF coils can be lifetime components for both R_0 = 1m and 1.7m Cu-TF ST-FNSF devices and can support a neutron fluence mission of 6 MWy/m^2.

2.4. Demonstrating electricity break-even from a pure fusion system

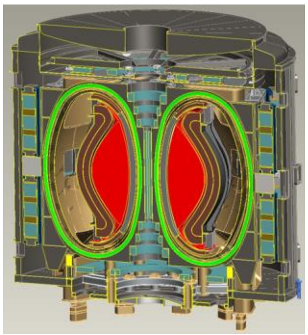

FIG. 3. Low-A HTS Pilot Plant

In the U.S., the Pilot Plant [39,38] has been proposed as a potentially attractive next-step towards fusion commercialization with a primary mission to generatesmall net electricity in as compact a facility as possible and in a configuration scalable to a full-size power plant. Further, the U.S. Pilot Plant commonly also aims to incorporate the mission of high neutron fluence and tritium self-sufficiency as adopted for the FNSF. Building on the TF and PF coil layouts found to be optimal for the Cu-TF ST-FNSF configuration described in Section 2.3, net-electricity producing pilot plants utilizing high-temperature-superconducting (HTS) TF magnets have been systematically studied as a function of aspect ratio and inboard shielding thickness. HTS TF magnets have been explored due to the prospect of very high current density combined with high field enabling consideration of lower aspect ratio solutions than normally considered. In particular, for low-A devices to be attractive at the R_0 = 3 m scale, high HTS winding pack current densities 40–70 MA/m^2 and peak fields up to 18 T are needed to provide space for shielding. To achieve peak outboard neutron fluence > 6 MWy/m^2 for the FNSF mission, approximately 60 cm of inboard WC-equivalent shield is needed to reduce radiation damage to the HTS TF magnets to acceptable levels. For shields in this thickness range, R_0 = 3 m is a favorable plasma major radius size for achieving Q_{eng} > 1 for a wide range of aspect ratios and shielding thicknesses. Lower aspect ratios with A = 1.6 to 2 are found to maximize the fusion power per unit TF coil volume, while higher A minimizes blanket volume for otherwise similar overall fusion performance. Figure 3 shows the cross-section of an A=2, R_0=3m device projected to be capable of 6 MWy/m^2 (peak), Q_{eng} > 1 (P_{net} = 0-100 MWe), TBR $\approx$ 1 (assuming a thin inboard breeding blanket), and a configuration with significantly reduced TF magnet volume (relative to conventional aspect ratio) which could help reduce overall magnet cost if/once large-scale HTS magnets have been developed and HTS tape production costs have been reduced.

2.5. Electricity production in modular fusion power plants

In the modular approach to the ST path, the economically feasible fusion power plant will consist of several ST modules, where the physics and technology are developed for a single compact module, making the development path cheaper and quicker. The economics is based on an assumption that for a modular-based plant, many auxiliaries are shared between modules, and a regular necessary maintenance is set in a module-to-module way, providing high availability of the power plant. The minimal size of a single module is determined by the physics and technologies developed up to date. The cost estimate for a modular power plant is based on the selected set of parameters for a single module and the necessary number of the modules in an economically feasible power plant. The cost of electricity (CoE), dependence of the CoE on the neutron wall loading and the effect of a module reservation are examined in detail in [40]. While the final cost of the new approach may be not significantly cheaper than for the mainstream approach, the development path has many advantages, can be affordable and much faster, so we could develop ST modules quickly to the point where they are reliable – and low-risk, with funding supported by the combination of private and public sources.

It was shown in [41] and further expanded on in [42] that for steady-state tokamaks operating at fixed fractions of the Greenwald density and beta limits that the fusion energy gain, Q_{fus}, is primarily determined by the absolute level of the fusion power, P_{fus}, and the energy confinement, and is only weakly dependent on the device size. The level of fusion power required for high gain operation differs substantially depending on the energy confinement time scaling assumed. Two qualitatively different sets of scalings have been derived from the ITER ELMy H-mode database: the conventional IPB98y2 scaling and an alternative set of beta-independent scalings. Both sets of scalings approximate the data to a similar degree of accuracy, however the IPB98y2 scaling has a negative beta-dependence, $\sim\beta^{-0.8}$, which contradicts the null dependence observed in several dedicated experiments [43,44,45]. Recent experiments on JET found that at high collisionality confinement has an inverse beta dependence but becomes beta-independent at low collisionality, the region where future reactors are envisaged to operate [46]. If the confinement of future devices is approximately beta-independent, as the experiments suggest, then the fusion power required for high gain operation is substantially lower than that predicted for IPB98y2 like confinement. In ST reactor concept studies, engineering and technology considerations often limit the device performance, such as those related to power handling and neutron shielding. The ability to achieve high gain operation at lower fusion powers alleviates these concerns. These findings, coupled with the recent advances in high temperature superconductor (HTS) manufacture and technology mean that relatively compact reactors based on high field spherical tokamaks have the potential to deliver the performance required for commercial applications.

Being relatively compact and having a modest power output makes such reactors amenable to modular deployment. In a modular plant, multiple reactors could be used to generate GW scale output or, if it proves economical, plants with a few reactor units and, therefore, lower net power output, could be locally distributed.
In a modular plant, reliability and availability will be improved due to not being reliant on a single power unit. Reservation of one module will allow for fully off-line maintenance without negatively impacting availability and studies have shown that up to 100% availability can be achieved depending on the number of modules and the duration of scheduled maintenance periods [40].

The engineering design and licensing costs can be shared between many modules and the modular approach does not require a multi-GW scale demonstration to validate the concept and determine the optimal commercial design. Further systems, such as heating and current drive – the demand for which is maximum during start-up – and remote handling can be shared between multiple modules, allowing for further potential cost savings. The modular approach is considered in detail in [40]. It concludes that the modular concept combines advantages of the "economy of scale" for the conventional part of the plant and the "economy of mass production" for the fusion core, which does not lead to an increase of the cost of the fusion energy. Perhaps the most significant characteristic of the modular approach is the potential for faster and cheaper development made possible by the reduced unit size.

2.6. Electricity production in larger-scale fusion power plants

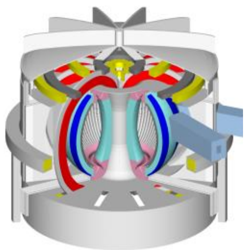

FIG. 4. Supercoducting low-A reactor VECTOR.

As an extension of the ST path, low aspect ratio power reactors with superconducting coils are proposed. Figure 4 shows VECTOR [47] (design parameters updated afterward) resulting from the quest for an ultimate compact reactor on a premise that advanced technology for high heat removal from the divertor and first wall is developed. In the conceptual design of VECTOR, a high-field ST was pursued adopting superconducting TF coils and excluding CS and breeding blanket on the inboard side. Installation of neutron and gamma shield to protect the superconducting coils on the inboard increases A compared with usual ST reactors, leading to an optimal design regime of A = 2.3-2.6. Using a Bi-2212 conductor, TF coils are designed to produce B_T of 5 T at the plasma axis. It is worth noting that VECTOR reduces n_e/n_{GW} requirement (= 0.9) offering an advantage over conventional tokamak reactors. Due to its compactness, the reactor has advantages in radioactive waste management and CO_2 emission as well as construction cost [48]. Based on the finding that a reactor weight is strongly dependent on the CS size [49], SlimCS was designed to mitigate physical and engineering difficulties as a DEMO concept based on fusion technologies foreseeable in near future [50]. In particular, the adoption of a reduced size CS facilitates plasma current ramp-up and plasma shaping, and the average neutron wall loading (3 MWm^{-2}) reduces engineering difficulty in breeding blanket design. SlimCS is envisaged in an extension of an advanced operation regime with low-A and high β_N in JT-60SA [51].

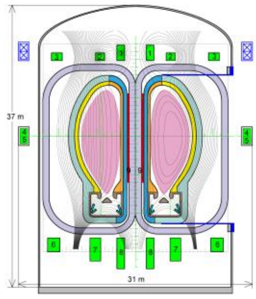

FIG. 5. Conceptual design and equilibrium of JUST.

Major issues of fusion power plant are as follows: (1) first-wall neutron damage, (2) high bootstrap fraction, (3) cost, (4) heat load to divertor, (5) minimizing T leakage in the power generation. The ST has advantages to solve the first three issues, namely: a large port to replace the first wall, intrinsically high bootstrap fraction, and high beta. However, a disadvantage of the ST reactor is small space for the superconducting magnet. Considering these issues, an 800 MWe fusion power plant named JUST (Japanese Universities' Super Tokamak reactor) has been designed [52]. JUST is a superconducting magnet machine with a thick radiation shield in the center stack and a liquid lithium divertor installed to mitigate the heat load problem. The conceptual design and equilibrium of JUST are shown in Figure 5. The height of the reactor is 35m and the diameter is 31m.

3. PRESENT AND NEAR-TERM ST FACILITIES IN SUPPORT OF FUSION APPLICATIONS

Several reviews of world-wide ST facilities and research progress have recently been completed [53,54] and are not repeated here. A selection of relatively new and/or unique mid-scale ST facilities is briefly summarized below to provide background and context for other parts of this paper.

3.1 ST experiments in United Kingdom - ST40

ST40 is a high field, spherical tokamak designed, constructed, and operated by Tokamak Energy Ltd, a privately funded UK company, founded in 2009 as a spin out from the Culham Centre for Fusion Energy (CCFE). The design parameters of ST40 are: R_0=0.4-0.6m, A=1.6-1.8, κ=2.5, B_{tor}=3T, I_p=2MA, $t_{pulse} \leq$ 3s, and 2MW NBI heating. A further 2MW of RF heating is also currently under consideration. The next experimental campaign is scheduled to begin in early 2019, with full performance operations planned for 2020.

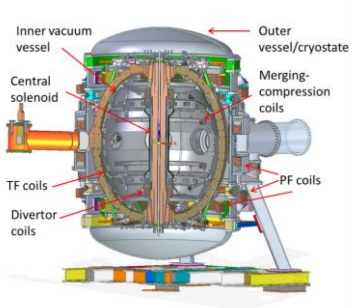

FIG. 6. ST40 3-D model

An engineering model of ST40 is shown in Figure 6. ST40 has an inner vacuum chamber (IVC) providing the ultra-high vacuum required for plasma operations and an outer vacuum chamber (OVC) that acts as both a cryostat to allow the TF and PF coils to be cooled to liquid nitrogen temperatures and as a mechanical support structure. The toroidal field coil has 24 turns and is comprised of a centre column containing 24 twisted wedges, each with a toroidal angular displacement of 15° along their length to provide continuity of the TF circuit, and 24 return limbs grouped in 8 packs of 3. A small central solenoid is wound around the TF wedges and will be used to sustain the plasma current flat-top. The return limbs are electrically connected to the centre column wedges using copper-copper pressed joints and mechanically isolated using copper foil flexibles. The out of plane electromagnetic loads, resulting from interactions with the TF and PF coils, are reacted to the OVC by a system of pre-tensioned carbon bands. These bands provide high stiffness and strength but with minimal thermal conduction.

The primary goal of ST40 is to extend the high field spherical tokamak physics basis and to reduce the uncertainty associated with predicting the performance of future ST devices. To this end, research is focused on: expanding the ST confinement time database, especially towards low collisionality [17,20]; evaluating various non-inductive start-up methods, including merging compression [55], and EBW assisted start-up [56]; developing scenarios with limited or no NB heating; and testing reactor relevant divertor solutions, such as those employing a liquid lithium strike surface.

3.2 ST experiments in the USA – LTX-β

The upgrade to the Lithium Tokamak Experiment (LTX-β) is a low aspect ratio tokamak with design parameters R0=0.4 m, A=1.6, kappa=1.6, B_{tor} =0.34T, I_p < 0.2 MA, t_{pulse} < 0.1 sec., and 0.7 MW NBI. A modest (< 100 kW) ECH capability will be added in late 2019 or early 2020. LTX-β has now begun operating, with neutral beam heating to be available in late 2018. The plasma is not diverted, but is wall-limited on a lithium coated high-Z liner. The liner can be heated to 300 °C during tokamak operations, to provide a full liquid lithium wall surrounding the plasma, at PFC temperatures relevant to reactor operations with liquid lithium walls. LTX-β is shown in Figure 7. The goal of the LTX-β research program is to extend investigations of the effect of very low global recycling on the core and SOL of a low aspect ratio tokamak, and to test the feasibility of using liquid lithium walls as PFCs for fusion reactors. Low global recycling with low-Z lithium walls eliminates temperature gradients from the core plasma [57], which has important consequences for neoclassical confinement, and strongly modifies the SOL [58], which (especially for the ST) to both broaden and reduce the divertor power deposition profile.

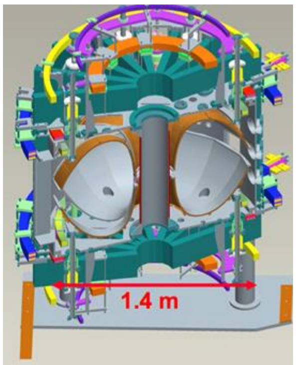

FIG. 7. LTX-β elevation view

3.3 ST experiments in the Russian Federation – Globus M2

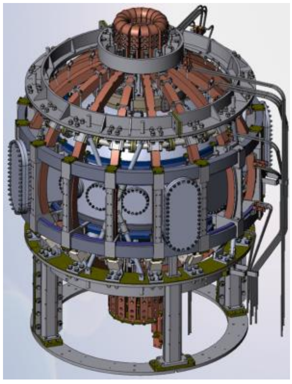

FIG. 8. Globus-M2 3D model

Globus-M2 (see Figure 8) with R=36 cm, A=1.5, B_T=1T, I_p=500kA, $t_{pulse} \le 1$ s, 2MW NBI, 0.5MW ICRH and LHCD [29,15,31] is considered as the next step on the line of compact beam-plasma STs due to high auxiliary heating power density (up to 5 MW/m^3), strongly anisotropic plasma ($P_{fast}/P_{thermal}$ ~ 30%), and high normalized Larmor radius ($\rho^*=\rho/a$). The scientific program of Globus-M2 will continue the Globus-M program in conditions of low collisionality, including experiments on ICRH, LHCD, NBCD and heating, plasma gun fueling, non-inductive start-up, energy and particle confinement study. Full-scale plasma experiments are scheduled for 2018. From the engineering point of view the machine is designed to withstand higher (more than 6 times) electromagnetic and thermal loads compared to the Globus-M values. Special solution, minimizing stresses in the central column with the help of the flexible contact joints, was chosen. Maximal toroidal displacement of the TF coil in the working pulse was limited to tolerable value with the help of the additional upper support ring and cross-pieces, translating forces from upper ring to the lower one. Final ANSYS analysis of full 3D model confirms maintenance free cycle of the electromagnetic system operation of 5000 pulses.

Construction of the next generation Globus-3 ST after Globus-M2 shutdown (~10 years of active research) is a suggested step towards development of the low cost, compact, high auxiliary power beam-plasma ST driver for the hybrid fusion-fission system. This machine (see Table 1), as a non-nuclear proof-of-concept prototype of the future neutron source for hybrid system, should demonstrate achievability of the required stationary discharge with the 0.2 MW/m^2 equivalent neutron flux density, necessary for hybrid neutron sources. Transport simulations with ASTRA code showed, that desirable T_e ~ 5 keV at 0.1 of Greenwald limit may be obtained at ~6 MW NBI. Simulations were made for a compact device with the R ~ 1 m, B_T ~ 3 T. However these parameters may be significantly changed in the light of the results from Globus-M2, NSTX-U, MAST-U, ST40 and superconducting technology progress. The research program of this machine will focus on stationary technologies and the main problems of the compact beam-plasma STs, such as electron confinement and heating and development of the noninductive CD and divertor, required for the stationary (high duty cycle) operation. Possible future DT-upgrade is under consideration.

3.4 ST experiments in Japan

3.4.1 Plasma Material Interaction (PMI) studies - QUEST

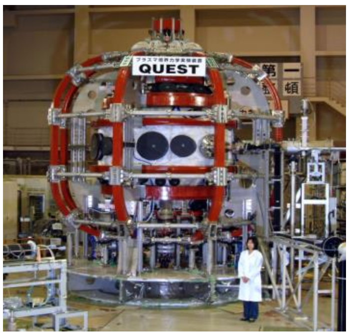

FIG. 9. QUEST device

Integrated simulations of 1.5-3 GW thermal output tokamak-based fusion power plant in Japan show that the fuel burnup fraction is approximately 1-5% [59,60], meaning that the remaining 95-99% of D-T fuel must be either pumped out or stored in plasma and plasma-facing walls (PFWs). Fuel particles are continuously circulating in plasma, scrape-off layer and PFWs and it was found that most of fuel particles are dynamically or statically retained in the PFWs, depending on the wall material property and temperature. This fuel particle balance may play an important role in the performance of future fusion power plants via recycling, tritium inventory and permeation; however, the balance during steady state operation is poorly understood due to a lack of proper PMI information and supporting experimental discharges of sufficient duration. Recently, a discharge up to 1 h 55 m was achieved in the Q-shu University Experiment with Steady State Spherical Tokamak [QUEST: (major radius, R~0.64 m, minor radius, a~0.4 m, toroidal magnetic field, $B_T \leq 0.25$ T at R= 0.64 m)] (see Figure 9) with all-metal and temperature-controllable PFWs. The dynamic retention from metal PFWs during long duration discharges closely matches the predictions of a previously proposed H barrier model [61,62] that indicates significant contribution of plasma induced deposition layer on PFWs. This transport barrier has also been observed in tritium (T) [63], and the described characteristics are common for hydrogen isotopes. The transport barrier will be useful for control particle balance in future power plants.

3.4.2 Plasma formation and ramp-up experiments

Reactor studies indicate the possibility of realizing a compact fusion reactor by complete elimination (VECTOR [47]) or substantial reduction (Slim CS: [49,50]) of the central solenoid (CS). But it is not clear how much reduction of the CS is acceptable. CS-less or limited-CS operation remains to be demonstrated. Several ST devices in Japan [54] are investigating the possibility of RF waves (LHW in TST-2, ECW/EBW in LATE and QUEST) to form the ST plasma and ramp up the plasma current. Highly reliable start-up and ramp-up to substantial levels of plasma current (quarter to half of the nominal Ip obtainable by Ohmic operation) have already been demonstrated. Ramp-up towards the full Ip capability is being planned using combinations of waves with different frequencies. Other methods of start-up such as CHI (HIST, QUEST), plasma merging (TS-3, TS-4, UTST), and AC Ohmic operation (TST-2) are also being investigated.

4. COMPARATIVE ASSESSMENT OF PERFORMANCE REQUIREMENTS FOR ST APPLICATIONS

Table 1 shows a parameter set for next-step ST devices ranging from a small (as indicated by plasma major radius R_0) fusion-fission hybrid, component test facility (CTF) and fusion nuclear science facility (FNSF), pilot plant, modular power plant, and full-scale power plants with electricity generation up to 1 GWe.

Parameter	FNS-ST [Kurchatov 2011]	ST-CTF [CCFE 2008]	Compact hybrid FNS / Globus-3 [Ioffe 2018]	ST-FNSF [R=1m, PPPL 2016]	ST-FNSF [R=1.2m, ORNL 2009]	ST-FNSF [R=1.7m, PPPL 2016]	ST-E1 Modular Power Plant [Tokamak Energy 2018]	Low-A HTS Pilot Plant [PPPL 2016]	JUST [Japan 2012]	VECTOR [Japan 2002]	SlimCS [Japan 2007]
Device - Scenario	FNS-ST	ST-CTF-CCFE	Globus3-IOFFE	ST-FNSF-PPPL-1m	ST-FNSF-ORNL-1.2m	ST-FNSF-PPPL-1.7m	ST-E1-TE	HTS-Pilot-PPPL	JUST	VECTOR	SlimCS
Aspect ratio	1.66	1.55	1.9	1.7	1.5	1.7	1.8	2	1.8	2.3	2.6
Major radius R_0 [m]	0.5	0.81	1	1	1.2	1.7	2	3	4.5	3.2	5.5
Minor radius [m]	0.30	0.52	0.53	0.59	0.80	1.00	1.11	1.50	2.50	1.39	2.12
Plasma elongation κ	2.75	2.4	2.5	2.75	3.1	2.75	3	2.5	2.5	2.35	2
Plasma triangularity δ	0.5	0.4	0.7	0.5	0.4	0.5	0.5	0.6	0.35	0.5	0.35
Plasma current [MA]	1.5	6.5	3.5	7.2	8.2	11.5	10	12	18	14.6	16.7
Toroidal field at R_0 [T]	1.50	2.60	3.00	3.00	2.18	3.00	4.00	4.00	2.36	5.00	6.00
Normalized current I_P/aB_T	3.32	4.78	2.22	4.08	4.70	3.83	2.25	2.00	3.05	2.10	1.32
Toroidal beta β_T [%]	17	16	6	18	18	16.5	11	8	22	12.5	5.7
Normalized beta β_N	5.12	3.34	2.71	4.41	3.83	4.30	4.89	4.00	7.21	5.96	4.33
Cylindrical safety factor q*	3.88	2.28	4.30	3.09	3.76	3.28	6.17	4.53	3.30	3.38	3.65
Internal inductance l_i	0.7	0.6	0.7	0.55	0.5	0.55	0.5	0.6	0.7	0.7	0.7
Bootstrap fraction	0.3	0.4	0.3	0.81	0.49	0.76	0.98	0.7	0.99	0.78	0.75
External current drive (CD) fraction	0.70	0.60	0.70	0.19	0.51	0.24	0.02	0.30	0.01	0.22	0.25
Non-inductive CD fraction	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
Greenwald fraction	0.2	0.24	0.1	0.8	0.3	0.8		0.8	1.5	0.83	0.98
Fast ion fraction W_{fast} / W_{tot}	0.4	0.1	0.5	0.09	0.24	0.1		0.1	0.1	0.23	0.22
H-mode multiplier H_{98}	1	1.3	1	1.25	1.5	1.25		1.8	1.8	1.44	1.3
Aux heating & CD power - NBI [MW]	10	44	15	60	31	85	0	50	2	60.4	100
Aux heating & CD power - RF [MW]	5	0	6	0	0	0	12	0	0	0	0
Aux heating & CD power [MW]	15	44	21	60	31	85	12	50	2	60.4	100
Volume [m^3]	1.85	8.0	10.4	14.1	34.9	69.4	140	253	1056	221	767
Total plasma stored energy W [MJ]	0.42	5.2	3.4	13.7	17.8	61.5	148	194	773	412	940
Fusion power [MW]	0.5	35	50	60	75	174	420	500	1900	2503	2950
Fusion Gain Q_{DT}	0.03	0.8	2.4	1.0	2.4	2.0	35	10	950	41	30
Net electric output [MWe]	0	0	0	0	0	0	125	100	800	1000	1000

TABLE 1. Performance parameters for future ST-based fusion research facilities and power plants.

Parameter	QUEST achieved	QUEST goal	Globus-M achieved	Globus-M2 goal	ST40 goal [Programs 1-3]	ST40 goal [Future Programs]	MAST achieved [High stored energy]	MAST-U goal [High non-inductive]	MAST-U goal [High stored energy]	NSTX achieved [High stored energy]	NSTX-U goal [100% non-inductive, H_{98}=1]	NSTX-U goal [100% non-inductive, H_{ST}=1, high-power]	NSTX-U goal [High stored energy, high power]
Device - Achieved/Goal - Scenario	QUEST-A	QUEST-G	GlobusM-A	GlobusM2-G	ST40-G-P13	ST40-G-FP	MAST-A-HSE	MASTU-G-HNI	MASTU-G-HSE	NSTX-A-HSE	NSTXU-G-HNI-H98	NSTXU-G-HNI-HST-HP	NSTXU-G-HSE-HP
Aspect ratio	1.7	1.7	1.5	1.5	1.7	1.7	1.3	1.56	1.56	1.45	1.78	1.73	1.7
Major radius R_0 [m]	0.68	0.68	0.34	0.34	0.4	0.4	0.85	0.82	0.82	0.89	0.94	0.94	0.94
Minor radius [m]	0.40	0.40	0.23	0.23	0.24	0.24	0.65	0.53	0.53	0.61	0.53	0.54	0.55
Plasma elongation κ	1.2	2.5	2	2	2.5	2.5	2.1	2.5	2.5	2.5	2.78	2.76	2.75
Plasma triangularity δ	0.2	0.68	0.5	0.3	0.35	0.35	0.5	0.5	0.5	0.6	0.5	0.5	0.5
Plasma current [MA]	0.01	0.3	0.25	0.5	2	2	1.2	1	2	1.33	0.87	1.4	2
Toroidal field at R_0 [T]	0.133	0.25	0.50	1.00	3.0	3.0	0.52	0.78	0.78	0.48	1.00	1.00	1.00
Normalized current I_P/aB_T	0.19	3.00	2.21	2.21	2.83	2.83	3.53	2.44	4.88	4.51	1.65	2.58	3.62
Toroidal beta β_T [%]	0.1	10	5.5	10	4.5	4.5	11.5	9.5	18	25	6.6	15.5	20
Normalized beta β_N	0.53	3.33	2.5	4.5	1.6	1.6	3.3	3.9	3.7	5.5	4.0	6.0	5.5
Cylindrical safety factor q*	19.09	3.55	3.78	3.78	3.76	3.76	2.95	4.76	2.38	2.77	7.44	4.83	3.48
Internal inductance l_i	0.7	0.7	0.7	0.7	0.7	0.7	0.7	0.7	0.7	0.45	0.6	0.67	0.5
Bootstrap fraction	0.01	0.2	0.25	0.37	0.35	0.35	0.2	0.17		0.35	0.67	0.67	0.48
External current drive (CD) fraction							0.15	0.71		0.15	0.33	0.33	0.18
Non-inductive CD fraction							0.35	0.88	0.47	0.50	1.00	1.00	0.66
Greenwald fraction	0.3	0.5	1.00	0.30	0.31	0.31	0.40	0.23	0.24	0.80	0.72	0.71	0.70
Fast ion fraction W_{fast} / W_{tot}	0	0.1	0.05	0.2	?	?	0.4	0.6	0.3	0.15	0.26	0.18	0.09
H-mode multiplier H_{98}		1.2	1.3	1.3	1.4	1.4	1	1	1	1.2	1	1.3	1.16
Aux heating & CD power - NBI [MW]	0	2	1	2	2	2	3	7.5	7.5	6.3	10.2	15.6	15.6
Aux heating & CD power - RF [MW]	0.05	1	0.2	1	0	2	0	0	0	0	0	0	0
Aux heating & CD power [MW]	0.05	3.00	1.20	3.00	2.00	4.00	3.00	7.50	7.50	6.3	10.2	15.6	15.6
Volume [m^3]	2.37	4.08	0.54	0.54	0.83	0.83	11.8	8.5	8.5	12.6	11.6	11.8	11.7
Total plasma stored energy W [MJ]	2.50E-05	0.015	0.004	0.03	0.20	0.20	0.22	0.29	0.56	0.43	0.46	1.10	1.40

TABLE 2. Physics and operational parameters for medium-scale present and near-term ST-based research facilities.

Table 2 shows a preliminary listing of ST physics and operational parameters for medium-scale present and near-term ST research facilities. Figure 10 shows an example comparison of β_N versus f_{BS} for steady-state next-step STs and projected high non-inductive current drive fraction scenarios (yellow symbols) in MAST-U and NSTX-U. As seen in the Figure, MAST-U will explore lower bootstrap fraction (and lower density – see Table 2) scenarios with a larger fraction of neutral beam current drive, while NSTX-U plans to explore higher density and higher bootstrap fraction scenarios. Together, both devices will approximately span the projected operating space for FNS-ST, Globus-3, ST-CTF/FNSFs, and super-conducting pilot plants and reactors VECTOR/SlimCS with the exception of the JUST reactor concept. This finding indicates that very high bootstrap fraction and normalized β scenarios will need to be explored in NSTX-U to establish the physics basis for advanced reactor scenarios with very small external current drive. Additional performance comparisons will be described at IAEA-FEC 2018 and in a more comprehensive future publication.

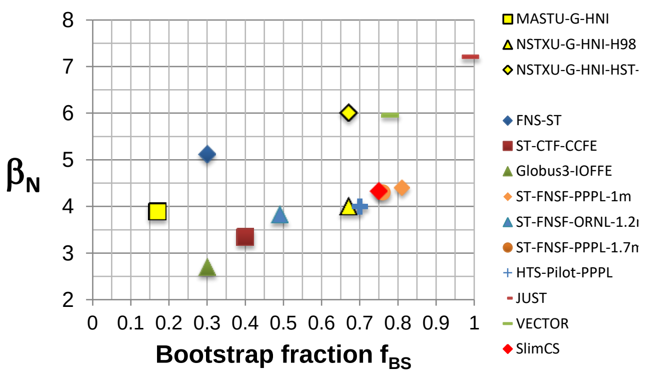

FIG. 10. β_N versus f_{BS} for steady-state next-step STs and projected MAST-U and NSTX-U high non-inductive scenario performance.

5. SUMMARY

The spherical torus/tokamak (ST) is potentially attractive for a range of fusion energy development applications and associated research including: developing solutions for the plasma-material-interface (PMI) challenge, fusion-fission hybrid systems, developing fusion components capable of withstanding high fusion neutron flux and fluence, electricity break-even from a pure fusion system, and electricity production in modular and full-scale fusion power plants. ST concepts have previously been developed (with varying degrees of analysis and design) for each of these applications and a preliminary survey of projected performance for next-step ST applications has been carried out. Further, a survey of the performance projections for present and near-term ST devices is underway enabling a systematic assessment of gaps to next-step performance. Future work will identify specific gaps for a range of physics and technological performance parameters.

ACKNOWLEDGEMENTS

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