International Collaboration Opportunities For the US Fusion Sciences Program

26 July 2011
Final Report

Report of the
International Collaboration Task Group
U.S. Burning Plasma Organization

1. Introduction

As the seven partner countries prepare for ITER and pursue their separate programs toward fusion energy, they are operating or constructing a wide spectrum of research and development facilities. These facilities provide opportunities for international research and possible US participation through collaboration. Development of a strategy in this context enables the US to build on these opportunities by determining which research areas it will pursue domestically, and which it will pursue by collaboration on international facilities.

The International Collaboration Task Group was formed to assess the opportunities to pursue US fusion research goals via international collaboration. It was chartered as a Task Group of the US Burning Plasma Organization in early 2009 with the membership listed in Appendix A, and the charge given in Appendix B. The membership was chosen to have a range of topical expertise and experience with international collaboration. The Task Group focused on the comprehensive set of issues identified by the "Research Needs for Magnetic Fusion Energy Sciences" report (ReNeW, 2009) and the opportunities to address them using collaboration on available or planned international facilities. This report is focused on fusion plasma confinement facilities. Subsequent updates will include fusion technology development facilities.

The Task Group collected information about the plans of the international facilities from their leaders, including recent presentations, white papers, studies, and planning documents. Additional information was provided at bilateral coordination meetings between the DOE and the EU, Japan, Republic of Korea, and Chinese program managers. Ideas for new collaborations on these facilities were requested and received from the leaders of US collaboration groups, and were gathered from the 2010 Field Work Proposal submissions. The documents used in this study are listed in Appendix C. Group members discussed the international program plans at the bilateral meetings, ITPA meetings, international conferences, and IEA implementing agreement meetings. Finally, some group members visited the AUG, EAST, JET, KSTAR, LHD, MAST, RFX, and W 7-X sites during the study to directly gather information and discuss opportunities.

The Task Group reviewed and discussed the ReNeW issues, the facility plans, and the opportunities in a large number of teleconferences. Sub-groups, as were available, met at the APS and IAEA conferences and other workshops for in-person discussions.

This report provides the Task Group's observations, findings, and conclusions. Section 2 discusses the general role of international collaboration in the US program. Section 3 identifies the ReNeW issues that cannot be fully addressed with present US capabilities, but can be addressed using international facilities. Section 4 analyzes the major international facilities to identify their potential technical impact, from the US perspective, and the opportunities for enhancing US collaboration. The timescales for scientific impact of the collaboration opportunities, and other conclusions are discussed in Section 5.

2. The Role of International Collaboration in the US Program

2.1 Motivation

The US fusion community has undertaken several studies, identifying the gaps and research priorities to prepare for ITER and the development of fusion energy systems. These culminated in the recent report "Research Needs for Magnetic Fusion Energy Sciences" (ReNeW, 2009), which documents the remaining research issues and possible approaches to resolving them. It was acknowledged that many of the remaining issues are not accessible by the present capabilities of US facilities. These issues can be addressed by constructing new US facilities, strengthening existing US facilities, or by collaboration on existing or planned international facilities. Continuing research on these issues is necessary to maintain and develop US expertise in preparation for burning plasmas and the development of fusion energy.

The US and other national fusion programs already engage in extensive international collaboration in order to conduct scientific studies on appropriate facilities, to corroborate and extend their results, and to exchange personnel. In addition, through international collaboration, multiple countries have combined their resources to address challenging issues not accessible to them separately. Through international collaboration, the world fusion program has designed and is constructing ITER, the first magnetically confined burning plasma experiment.

During the last 20 years, international investment in fusion research facilities has far outpaced US investment. Several foreign fusion programs are focused on developing fusion energy on a shorter timescale than the US presently envisions, and have broad programs with investments to advance their ability to confine steady-state and high pressure plasmas, control the plasma-wall interface, develop materials to withstand the fusion neutron flux, and engineer tritium breeding blankets. As a consequence, the facilities for best addressing many fusion energy science and technology issues are located outside the US or are under construction outside the US. The knowledge and experience generated in these facilities will be critical to fusion energy. Thus, to maintain a vital program, the US must balance between collaborating on these international facilities and developing domestic facilities targeting other critical issues.

When ITER begins experiments later in this decade, it will be the largest fusion research facility worldwide. It will provide unique capabilities to investigate and understand fusion-burning plasmas, addressing critical issues. ITER will operate as an international collaboration, in which the US will be one of seven partners. As a partner, the US has a limited voice and role in the management of ITER's construction and research. The US participation in ITER is expected to be the largest budget element in the US program. Thus, during this decade, the US fusion program will transition to incorporating the international collaborative efforts on ITER as a central part of the US national research strategy and planning. This transition has been foreseen since the US decision to rejoin ITER,.

For all these reasons, international collaboration will play a growing role in the US fusion program and should be included in planning the US fusion program strategy, similar to planning the programs for domestic research facilities.

2.2 Priorities and Methods for US International Collaboration

We identified criteria for comparing collaboration opportunities on international facilities should be

- Ability to address and resolve critical fusion research issues, as identified in the ReNeW and FESAC reports, which cannot be resolved on present US facilities;
- Potential for maintaining and developing key US competencies and capabilities in order to advance the US program strategy beyond the collaboration itself; and
- Utility in preparing the US researchers and program to participate in ITER and for further steps towards fusion energy.

These criteria will highlight collaborations which help close the gaps in knowledge needed to pursue fusion energy, strengthen the US capabilities to pursue fusion energy, and enable the US to capitalize on our investment in ITER.

In order to address US issues or involve US researchers, the US will need to make sufficient contributions to a collaboration to have an impact on the research directions of the foreign facility. Such contributions can include sharing personnel, codes, equipment, and results as appropriate for differing collaboration sizes and needs. To be effective, each collaboration should be cohesive and well-coordinated.

To ensure that a collaboration will achieve a challenging US goal requiring substantial effort, the US may need to make significant investments and commitments as a partner in one or more foreign facilities. This could involve taking responsibility for a portion of their program, providing research staffing, and/or providing equipment or other resources, resulting in having a direct voice in the program direction process. This is the strategy being pursued on ITER. Several foreign facilities have proposed to partner with the US in this way. In such a partnership, the US could have more confidence in accomplishing its goals.

In order for an international collaboration to succeed, there must be frequent and open communication between the collaborating parties. There must be a commitment by both parties to the success of the collaboration, in order to overcome intrinsic barriers due to cultural and linguistic differences, as well as institutional differences. To be effective, collaborating researchers need to have the rights and privileges of full research team members, including the ability to publish results, both at conferences and in refereed journals. The most successful collaborations are synergistic, providing benefits to both parties and mutual access to increased resources, capabilities, data, and resulting understanding. Such collaborations typically require a multi-year continuous commitment in order to climb learning curves, generate trusting relationships, and

produce significant new results. To prevent misunderstandings, it is essential that there be documented agreements on all aspects of the collaboration, including roles and responsibilities of the parties, the process for proposing ideas and experiments, access to data and results, approval of publications and talks, and authorship attribution.

2.3 Challenges of International Collaboration

Accomplishing US goals via collaboration can be far more challenging than on domestic facilities. Around the world, national fusion programs have different overall goals and constraints, resulting in different strategies, priorities, and budgets. For example, the different programs have differing balances between investigating fusion science and developing fusion energy technology and systems. When collaborating on a foreign facility, the host sets the overall facility direction and priorities, and a collaborator must find opportunities to pursue scientific goals within the constraints of those priorities.

Maintaining or developing US leadership in an area via international collaboration will require prior agreement with the host about the use of resources. It may be difficult to obtain agreement to take risks or explore novel directions before the ideas are proven elsewhere. Innovation and implementing game-changing approaches requires control over resources, and (from a US perspective) an interest in investigating new options related to US goals. Finally, unless there are timely US domestic activities to bring the collaboration results back to the US program, any US expertise developed or maintained may be ultimately lost.

In addition, successful international collaboration requires US personnel to spend substantial periods of time outside the US, integrating into and maintaining contact with the host research group. This can be a substantial burden and barrier, especially for reearchers with families. If US personnel relocate to join a foreign program, the barrier can reverse and they may never return.

3. ReNeW Issues For International Collaboration

The international fusion research facilities have key characteristics that are needed to address specific fusion science issues, but are not available in current US facilities. These include:

- Steady-state / very long-pulse operation, including use of superconducting coils,
- Large scale tokamaks and stellarators, providing dimensionless parameters closer to those of burning plasmas,
- DT plasmas,
- Other ITER-like characteristics (PFC material choices; superconducting PF coils with copper in-vessel control coils)
- Novel divertor geometries, both axisymmetric and 3-D.
- Actively cooled internal components (PFCs and launchers) at thermal equilibrium, including at high ambient temperature.
- Remote handling and maintenance of in-vessel components.

The ReNeW report identifies outstanding research and development issues that need to be addressed to prepare for ITER operation and for the design of fusion energy systems beyond ITER. ReNeW organized the issues into five themes and eighteen thrusts. In this section, we identify activities on international facilities, existing or under construction, to address the ReNeW issues. We do not include the issues that ITER will best address, since ITER is already part of the US program.

Theme 1 (Burning Plasmas in ITER)

Thrust 1 (Measurement techniques for burning plasmas)

1. Development of long-pulse / steady-state diagnostics, including strategies for maintaining baseline and calibration, reliability.
2. Development of long-pulse monitoring and protection systems
3. Validate fast-alpha particle diagnostics for burning plasmas
4. Development of reliable strategies for remote maintenance and calibration of diagnostics

Thrust 2 (Control transient events in burning plasmas)

1. Demonstration of stable operation, free of ELMs and disruptions, in ITER-relevant plasmas sustained for very long pulses.
2. Extrapolate techniques to avoid ELMs and disruptions or mitigate their effects to larger size, higher current plasmas, closer to burning plasma parameters
3. Long pulse control of limiting plasma instabilities at high normalized β

Thrust 3 (Role of alpha particles in burning plasmas)

1. Validate and test understanding of fast-ion instabilities in steady-state advanced scenarios at larger scale than present experiments, including methods to diagnose
2. Validate understanding of interaction between fast particles and global MHD instabilities in steady-state advanced scenarios
3. Assess impact of fast ion losses on first wall in steady state
4. Test methods to control alpha heating profile and alpha ash confinement

Thrust 4 (Qualify operational scenarios and physics basis for ITER)

1. Demonstrate ITER-like scenarios with DT-plasmas, including alpha-particle effects
2. Demonstrate ITER-like integrated scenarios (e.g., β, ν^*, and ρ^*, low disruptivity) with superconducting coils and ITER relevant H&CD methods for long pulse, including reliable control strategies.
3. Develop PFC cleaning and conditioning compatible with long-pulses.
4. Test ITER fueling and pumping strategies in the largest scale experiments accessible
5. Develop and test robust burn control strategies using simulation experiments with long-pulse.

Theme 2 (Creating predictable, high performance, steady state plasmas)

Thrust 5 (Expand limits for controlling and sustaining fusion plasmas)

1. Maintain the high β and bootstrap current levels (AT scenarios), suitable for high gain, for long pulse in the presence of fluctuations, restricting the diagnostics and actuators to those usable in a burning plasma environment. Demonstrate reliable operation without disruptions or ELMs. Establish the robustness of control required.
2. Determine the minimum diagnostic set and actuator set needed for high-gain long-pulse control
3. Develop and demonstrate long-pulse fueling and exhaust systems applicable to fusion plasmas.
4. Develop and demonstrate RF heating systems compatible with high neutron-fluence environments.

Thrust 6 (Predictive models)

1. Develop and validate models to predict evolution of long-pulse plasma confinement and evolution.
2. Use predictive models to improve long-pulse control systems

3. Predict and validate safe operating/control regimes for burning plasmas
4. Develop and validate models of burning plasmas in ITER

Thrust 7 (High temperature superconductors and magnet innovations)

1. Gather a database of reliability information on large superconducting systems. Identify and resolve issues that arise in practical use.

Thrust 8 (Integrated dynamics of self-heated and self-sustained burning plasmas)

1. Simulate effects of self-heating in high-gain equivalent long-pulse plasmas. Develop and demonstrate burn control.

Theme 3 (Taming Plasma-Material Interface)

Thrust 9: Unfold the physics of boundary layer plasmas.

1. Develop understanding of plasma material interaction (PMI) with prototypical PFC surfaces (for ITER and devices beyond ITER)
2. Explore innovative divertor configurations

Thrust 10: Decode and advance the science and technology of plasma-surface interactions.

1. Test the effects of fusion-energy (DEMO) level particle flux density, heat flux, density on PMI
2. Validate understanding of steady-state PMI at fusion energy-level particle flux density, heat flux, and plasma density
3. Understand the effect of high ambient temperature on physical chemistry and PMI with prototypical PFC materials.
4. Develop long-pulse high-power antenna structures for plasma heating and current drive, compatible with the burning plasma environment

Thrust 11: Improve power handling through engineering innovation.

1. Develop and demonstrate long-pulse refractory metal heat sinks at high temperature
2. Develop and demonstrate steady-state liquid metal (including lithium) PFCs
3. Assess the impact of neutrons/ions displacement damage on PFCs and PMI.

Thrust 12: Demonstrate an integrated solution for plasma-material interfaces compatible with an optimized core plasma.

1. Develop and test prototypical actively-cooled PFCs in an integrated long-pulse plasma environment. Document and iteratively improve characteristics (fuelling control, erosion & transport, fuel retention, dust production, etc.)

2. Develop control of plasma density with a long pulse divertor, compatible with technologically acceptable peak power removal.
3. Determine effect of high temperature walls on the plasma and PFC boundary, and core-plasma performance.

Theme 4 (Harnessing Fusion Power)

Thrust 13: Establish the science and technology for fusion power extraction and tritium sustainability.

1. Understand and test the impact of the fusion operating environment and conditions on components to capture the fusion power, breed tritium, and shield neutrons.

Thrust 14: Develop the material science and technology needed to harness fusion power.

1. Improve the performance of materials in the fusion neutron environment, including the ability to manufacture, engineer, and fabricate components.
2. Validate the materials and models for use in fusion applications, including interactions with the plasma and fusion neutron flux, as appropriate.

Thrust 15: Create integrated designs and models for attractive fusion power systems.

1. Conduct advanced design studies of integrated fusion energy systems to follow ITER.
2. Develop models of integrated fusion systems.

Theme 5 (Optimizing the Magnetic Configuration)

Thrust 16: Develop the spherical torus to advance fusion nuclear science.

1. Test use of EBW to initiate and raise the plasma current
2. Test control of ELMs using an array of in-vessel coils
3. Explore divertor design with extreme flux expansion

Thrust 17: Optimize steady-state, disruption-free toroidal confinement using 3-d magnetic shaping, emphasizing quasi-symmetry principles.

1. Develop and demonstrate steady-state operation with a 3D divertor, integrated with good core plasma confinement
2. Understand the effect of 3D non-symmetric shaping on confinement and operating limits; demonstrate adequate integrated performance at high-β.
3. Understand the confinement scaling for 3D systems at low ρ^*, low collisionality, high β, consistent with steady state
4. Demonstrate fueling and impurity control in long pulse, optimized configurations.

Thrust 18: Achieve high-performance toroidal confinement using minimal externally applied magnetic field.

1. Extend confinement scaling to higher current

4. Collaboration Opportunities on International Facilities

In this chapter we identify opportunities for enhanced US collaboration on specific international magnetic fusion facilities that can have a substantial impact on the US program. Emphasis is given to facilities where enhanced US collaboration has the highest priority, using the criteria in section 2.2, and will have a synergistic impact. The major confinement facilities having the most potential in this sense are EAST, JET, JT-60SA, KSTAR, and Wendelstein 7-X. Their major characteristics are summarized and compared in Table 1, along with other major foreign facilities. Five of the major international facilities, EAST, JT-60SA, KSTAR, LHD, and Wendelstein 7-X have superconducting coils and are designed for long-pulse steady-state operation with strong plasma shaping and sufficient heating power to access high-β, high confinement advanced regimes. The shared characteristics and opportunities of these long-pulse facilities are described in the next section.

The following sections discuss EAST, JET, JT-60SA, KSTAR, and Wendelstein 7-X in turn, identifying their

- Specific and unique capabilities;
- Technical importance to the US program, in terms of the ReNeW issues they can have the most impact on; and
- Opportunities for enhanced collaboration or partnership on their research program.

The ability of these experiments to address the ReNeW issues is summarized in Table 2. In identifying opportunities for enhanced collaboration, the existing US domestic research program was assumed to continue. There was no attempt to prioritize the collaboration opportunities relative to existing or proposed US domestic activities.

A final section discusses opportunities on AUG, LHD, MAST, and RFX, which offer less opportunity for research impact from expanded US collaboration. Smaller facilities with low heating power, such as SST-1 in India and QUEST in Japan, offer less capability to advance the US program and are not discussed.

Table 1. Characteristics and parameters for major international magnetic fusion experiments.

	JET	AUG	EAST	KSTAR	JT-60SA	MAST	LHD	W7-X	RFX
status	mature	mature	emerging	emerging	construction	mature	mature	construction	mature
R (m)	3.1	1.65	1.85	1.8	2.97	0.85	3.7	5.5	2
B (T)	4	3.3	3	3.5	2.25	0.55	3	3	
Ip (MA)	5	1.2	1.5	2	5.5	1.2	< 0.1	0	2
R / a	3	3.3	4.1	3.6	2.5	0.3		11	4.4
configuration	SN	SN (DN)	DN	DN	SN (DN)	DN/SN	3D, helical	3D, quasi-isodyn.	RFP
kappa, delta	1.8, 0.4	1.8, 0.5	1.9, 0.65	2, 0.8	1.9, 0.5	2.5, 0.6			1, 0
RMP max. mode #	n=2	n=4	-	n=2	n > 1	n=6	n=5, m=1	n=5, m=1	n =24, m=2
Heating (ult.) (MW)	51	34	31	28	41	10	36	20	
coil conductor	Cu	Cu	SC	SC	SC	Cu	SC	SC	Cu
pulse length (s)	20	10	1000	300	100	1	~3600	1800	~0.2
PFC material	W / Be	W on C	C (Ti) -> W	C	C	C	C	C	C

4.1 The superconducting long-pulse facilities – shared characteristics

A number of international superconducting (SC) tokamak and stellarator experiments are operating or under construction, including EAST, ITER, JT-60SA, KSTAR, LHD, and W7-X. The use of SC coils allows continuous magnetic fields with little energy loss. The biggest impact of continuous magnetic fields is that the plasma experiment duration can be greatly extended, in particular far longer than the ~1-10 seconds typical for US experiments with resistive coils. While superconductors allow continuous magnetic fields, the plasma duration can be limited by other aspects, including the maximum duration of the plasma heating and exhaust systems, and limits on the allowable number of emitted fusion neutrons. For SC tokamaks, the duration can also be limited by the ability to stably maintain and control the plasma pressure and current profile and equilibrium. Thus, it is important that SC experiments focus on sustaining high pressure operating scenarios, suitable for fusion energy.

The importance of the superconducting facilities with long-pulse capability is to enable experiments to test and understand the long-time behavior of the plasma confinement system. Some physical processes in fusion plasma experiments have very long characteristic times, particularly the current-profile relaxation time, the thermal equilibration time of the boundary wall, the gas saturation time of the boundary wall, and the time-scale for erosion and re-shaping of the wall. Since fusion energy systems will operate essentially in steady-state, it is necessary to understand the behavior of the system on these various long timescales. In addition, it is important to demonstrate that steady-state solutions exist and can be controlled, due to the highly non-linear plasma dynamics.

Essentially all of the superconducting facilities can address the following key issues for long-pulse fusion plasmas, which are a subset of the ReNeW issues listed in section 3.

- The critical issue of maintaining and controlling long-pulse high-performance plasma without disruptions or power-loss transients. Depending on the magnetic configuration, this appears in **Thrust 2, Thrust 5, Thrust 16,** or **Thrust 17**.
- **Thrust 1**: Develop diagnostics for steady-state plasmas, including strategies for maintaining calibration and baseline; develop techniques for steady-state operational protection.
- **Thrust 4:** Develop PFC cleaning and conditioning methods compatible with long-pulse.
- **Thrust 7:** Gather a reliability database of large superconducting fusion systems.
- **Thrust 8**: Simulate the dynamics of high-gain burning plasmas and validate burn control strategies in non-burning plasmas at lower performance levels.
- **Thrust 10**: Steady-state plasma-surface interactions, including dust production and hydrogenic gas retention. Development of long-pulse RF-heating antenna structures, compatible with the burning plasma environment.

- **Thrust 12**: Integration of an actively cooled divertor and plasma-facing components with a steady-state, high performance plasma, including control of plasma impurities. Measure long-pulse hydrogenic gas retention, PFC erosion and dust production. Demonstrating control of plasma fueling and exhaust in steady-state.

Other issues can only be addressed by facilities with specific capabilities and are discussed in the following sections.

4.2 EAST

http://202.127.204.25/asipp-english/index.html

EAST is a DIII-D-sized tokamak operated at Academica Sinica Institute of Plasma Physics (ASIPP) in Hefei, China. The key long-term goal of EAST is long pulse advanced tokamak, fully non-inductive operation, with a target pulse length of 400 s and possible extension up to 1000 s. EAST has superconducting PF and TF coils and a pair of in-vessel copper coils (IVC's) for vertical stability control, a configuration similar to ITER. The current EAST divertor are actively cooled carbon with an ITER-like vertical target. The main chamber PFCs have been changed from actively cooled carbon to molybdenum. The PFCs are conditioned using 250C bake, between-shot RF, and lithium coatings. In 2014, EAST plans to upgrade their divertor to tungsten (sprayed on chromium-copper). This will be upgraded for operation at 400C in 2017. EAST is also exploring the use of liquid lithium PFCs. The 2014 upgrade will also add an array of internal coils for MHD stabilization up to $n \leq 3$. The EAST program will have ~10 MW (source) of LHCD and ICRF heating power in 2012, and an aggressive upgrade program to add NBI and ECH reaching a total heating power (long pulse) of 20 MW in 2013 and 30 MW in 2015.

EAST has operated since mid-2006 and is producing plasma pulses lasting longer than 100 s, with single and double null shaped plasmas and elongations above 1.9, H-mode confinement, and LHCD and ICRF heating each in the 1 MW range.

EAST is part of an extensive program at ASIPP and elsewhere to establish the basis for fusion energy development in China. This includes a technology development program preparing components for ITER and for development of a fusion pilot-plant in the 2020s. As part of this, they are participating in the ITER TBM program, and are designing, modeling, and testing blanket module technologies, including operating a hot lithium-lead loop.

Technical Importance of EAST Collaboration

EAST offers the near-term availability (2013) of a long pulse steady-state superconducting tokamak with high heating power and a configuration similar to that of ITER. Collaboration on EAST would provide opportunities for the US to

- Study ITER-specific long-pulse physics and control scenarios, possibly including interaction with test-blanket-modules;

- Study hot tungsten divertor and wall operation in the near term, with potential for in-situ surface science and real-time PFC diagnosis;
- Assess advanced scenarios developed on US experiments on a long pulse facility to test their steady state non-inductive sustainment and understand their operating limits.

Many of these critical research areas are aligned with the thrusts from the ReNeW strategic planning activity. Under **Theme I**, EAST enables steady state diagnostic development and testing (**Thrust 1**), including development of maintenance and calibration methods. Under **Thrust 4,** EAST enables long pulse study of ITER scenarios with RF-heating and limited central fueling, coupled with sufficiently long pulses for equilibrated wall operation**.**

EAST is also well-aligned with needs identified under **Theme 2** and **Thrust 5**. These include demonstration and control of disruption-free, steady state operation with equilibrated walls, physics understanding of current, pressure, and rotation profile evolution in long pulse, investigation of the effects of lithium coatings of PFC's in steady state discharges, particle balance studies in a long pulse, hot wall, diverted device, including fueling and exhaust technologies, and active surface modification to compare with test stand studies.

The capabilities of EAST will contribute to several thrusts in **Theme 3**. Fuel retention in the plasma-facing components and fuel accountability are central to **Thrust 12**, and EAST will naturally address this issue in long-pulse diverted configurations.

Finally, EAST and ASIPP will address issues in **Theme 4**, especially **Thrust 13**. They are actively developing dual-coolant lithium-lead (DCLL) breeding blanket modules, including preparing for the ITER TBM program, and are already operating liquid PbLi testing loops as part of their R&D. ASIPP has expressed interest in international collaboration on the design of next step facilities for developing fusion energy.

Opportunities and Timescales for Enhanced US Collaboration on EAST

Opportunities for US collaborations on EAST have already been identified in many areas. Ongoing US collaborations include control development and research with the EAST real-time plasma control system (derived from the DIII-D PCS), long-pulse operational scenario development, lithium wall coating technology, many diagnostic collaborations, materials studies and design of materials testing systems, and several RF system technology and plasma coupling collaborations. EAST data is remotely accessed from the US routinely, facilitated by a data-mirroring site already established in the US. The EAST team has demonstrated its interest and ability to collaborate openly with the US. The large number of students and staff working on EAST is a significant resource that can be leveraged by US collaborators.

There are many opportunities for enhanced US collaboration on EAST beyond existing efforts. EAST offers a near-term platform for long pulse implementation and study of ITER scenarios and control solutions extending those developed on existing facilities,

including in the US. Advanced algorithms for long pulse control through the PCS will be clearly needed, and are an area in which the US presently has leadership. Programmatic goals to operate in advanced tokamak regimes beyond the no-wall limit can build on US expertise. This will require RMP/RWM coils, planned as an upgrade in ~ 2014, offering opportunities for collaboration on the coil design and experiments to demonstrate robust control in long-pulse. The US could propose to lead an effort to avoid and mitigate disruptions in long-pulse, including advanced regimes. EAST has requested that the US collaborate on the planned ECH upgrade, which offers opportunities for US technology and physics involvement, including gyrotrons, launchers, and transmission line hardware. There may also be an opportunity to collaborate on long-pulse, compact RF launchers.

The US could also establish a substantial collaboration with EAST developing long-pulse prototypical first-walls and their interaction with the plasma. The collaboration on lithium PFC technology with NSTX/LTX should be strengthened, particularly in developing designs to circulate liquid lithium to PFCs for long-pulses. A US-EAST collaboration on long pulse discharges with hot metallic walls and active surface modification could build on results from US test stands and planned experiments on Alcator C-Mod with a 500C tungsten divertor. US expertise could help accelerate design and development of high temperature metallic PFC's and enhance the productivity of research on long-pulse plasma-wall interaction, fuel retention, particle balance and pellet fueling, and the effect of lithium wall coatings. The long pulse may produce measurable changes in surface properties that exceed diagnostic thresholds for measurement in a single pulse. The US could contribute novel diagnostics for the edge plasma and PFC surfaces to help unfolding the boundary and surface physics.

The ongoing collaborations will be strengthened as the data bandwidth and quality-of-service are improved. The collaboration on improved safety in fusion facilities should continue.

4.3 The Joint European Torus (JET)

http://www.jet.efda.org/

JET is the largest operating tokamak experiment in the world with the highest heating power, and is the only magnetic fusion experiment currently able to use tritium. JET has operated since 1983 and is a mature research facility and program, studying a broad range of tokamak physics and technology issues. JET has the same plasma shape and single-null divertor configuration as ITER, and its current program is focused on preparing for ITER. JET has a number of unique characteristics in the world program:

- Plasma size and normalized plasma size (ρ^*) closest to ITER, providing confinement and stability data closest to reactor scale, and reducing the extrapolation step to ITER and reactors
- Highest plasma current of all tokamaks, allowing it to investigate and test disruption detection, dynamics, and mitigation closest to reactor scale.

- Plasma facing components (PFCs) composed of tungsten (divertor) and beryllium, similar to the first wall design of ITER, to test its integration with prototypical plasmas
- Ability to use deuterium-tritium (DT) plasmas, producing a significant alpha-particle population at low fusion gain Q~1.
- An extensive set of remote maintenance tools, for maintaining and installing in-vessel components.
- Ability to continuously vary the ripple of the magnetic field.
- Capability to operate with the vessel and all internal components heated as high as ~320 C.

JET uses copper coils and has a plasma pulse duration limited to ~20 seconds.

JET research will focus on the effects of the tungsten / beryllium ITER-like wall through 2013. EFDA will propose to complete these experiments by operating with deuterium-tritium plasmas in 2014 or 2015. Any JET operation past 2015 may require significant international partnership including funding.

Technical importance of JET collaboration

Collaboration on JET provides the best opportunity for the US to qualify operating scenarios and support the physics basis for ITER experiments, ReNeW **(Thrust 4)** in an integrated plasma environment including DT, ITER-like PFC materials, plasma parameters closest to ITER, and sub-burning levels of fusion alpha particles. In this way, JET provides a platform for investigating many of the (**Theme 1)** issues, preparing for extension to ITER, and an integrated environment to assess the scalability of some methods to control transient events (**Thrust 2)**, including disruptions and ELMs. JET DT experiments will provide the only new data before ITER on alpha-particle driven instabilities, transport, and loss (**Thrust 3),** including interaction of the alpha-particles with turbulence. JET can also explore novel strategies for controlling the alpha heating profile. The DT experiments can be used to test alpha-particle diagnostics for burning plasmas (**Thrust 1)** and strategies for controlling the alpha-heating profile.

JET will provide the only integrated data for understanding the plasma boundary layer and wall interactions (**Theme 3)** with the set of PFC materials to be used in ITER. In addition, JET provides important data on how the power scrape-off thickness varies with plasma size (**Thrust 9)**. The JET experiments will test the compatibility of an ITER-like wall with high performance core plasma scenarios for ITER, (**Thrust 12)**. Integrated results will be obtained with respect to erosion and redeposition, dust production, tritium retention, impurity control and response to transients. JET provides an opportunity to examine the performance of bulk-tungsten components in the divertor (**Thrust 10).** JET can study the effects of elevated PFC ambient temperature up to ~320 C. However, the PFCs are not actively cooled so the PFC temperatures do not reach steady-state during the plasma pulse. The effect of the time-varying temperature may be difficult to understand.

JET data play an important role for validating predictive models (**Thrust 6)**, particularly data on the scaling of plasma confinement, stability, boundary layer physics, and

transients with normalized gyroradius (ρ^*) in collisionless plasma, DT effects, and overall plasma size. Incorporating these effects into validated models will be necessary for developing US strategies to control and exploit ITER.

Collaboration on JET could also enable progress by the US on other issues, including simulation of high fusion gain in lower-performance plasmas to validate control strategies for ITER scenarios (**Thrust 8**) and improving the ICRF and LH antenna coupling to ITER prototypical plasmas and the understanding of the plasma-antenna interaction (**Thrusts 9** and **10**).

Opportunities for enhanced US collaboration on JET

The US fusion research community has collaborated in JET experiments for more than 20 years, both directly and through the ITPA joint experiment process. US researchers from many institutions have participated in JET experiments, and a few US researchers are stationed at JET full-time. The US has contributed diagnostics, analysis codes, an RF antenna, and a pellet injector to JET as part of these collaborations. Experiments proposed by US researchers must be formally sponsored through one of the European EFDA-associated laboratories, since the US is not a formal member of JET.

In 2010, Europe invited the US to join as a partner in JET after the completion of its current program in 2015. The role of a US partnership is subject to negotiation and could include the US taking lead responsibility for part of the JET program, enhancing JET's technical capabilities, and/or helping support continued JET operation. As a partner, the US would have representatives on the JET Council and scientific committees with a voice on decisions.

JET is uniquely capable of research on critical and high priority topics, significantly reducing risks and uncertainties for ITER and subsequent burning plasmas. However, JET's full impact may be constrained by availability of resources within Europe. The highest impact crucial topics for JET collaboration, where the US participation could provide synergy include:

- Validate ITER's chosen ELM suppression or mitigation strategies in ITER prototype scenarios, with ITER-like walls and dimensionless parameters (as close as possible), to reduce the risk of extrapolating to ITER.
- Assess the ITER disruption mitigation strategies at the highest plasma current, largest plasma size, for scaling to ITER and beyond.
- Prototype all ITER plasma operating scenarios in DT with ITER-like walls and ITER-like heating and current-drive.
- Validate the understanding of alpha-particle instability thresholds and alpha particle transport/loss for ITER prototypical scenarios operated in DT, and a range of q-profiles.
- Resolve and document the plasma profiles in the scrape off layer and pedestal, to provide validation data at JET's scale for edge models.

In collaborating on these topics, especially at a partnership level, the US could enhance JET's capabilities in several areas, e.g. ELM suppression systems, ITER-like disruption mitigation systems, flexible heating and current drive system, and specific diagnostics. However, there may be practical limitations on JETs ability to prototype ITER strategies, such as disruption mitigation or ELM suppression, due to constraints from existing hardware.

In 2010, a joint US-EU group studied the feasibility of adding in-vessel resonant magnetic perturbation (RMP) coils to JET to control ELMs and other MHD instabilities, building on DIII-D experiments. Such coils could be a US investment, as part of partnership in JET. It concluded that such coils are feasible to construct and install on JET, and that they would be scientifically useful for exploring the effects of magnetic perturbations on ELM stability. However, the proposed JET coils produce a different perturbation spectrum than on the DIII-D experiments and from the proposed ITER RMP coils, due to constraints from existing JET systems. The present theoretical understanding of the effects of RMP coils is incomplete and cannot assure that the proposed JET coils will conclusively assess the effectiveness of the ITER RMP coils. Thus, they might not reduce the risk of ELMs on ITER. In order to be installed in time for use in 2015, detailed design and R&D for the JET coils would have to start immediately.

A joint Russian-EU group studied the feasibility of adding 10 MW of electron cyclotron heating and current drive (ECH / ECCD) to JET to control the current profile for access to advanced, higher pressure regimes and to stabilize sawteeth and neoclassical tearing instabilities. The US could provide parts or all of such a system as part of a partnership in JET. This would provide JET with all of the ITER heating and current drive methods, for developing and testing the ITER plasma scenarios and control strategies, in combination with its other ITER-like characteristics. The study group concluded that such a ECH upgrade was feasible and should enable JET to prototype and develop the ITER steady-state scenarios at the largest scale available, in DT.

4.4 JT-60SA

http://www.jt60sa.org/

Introduction and Special Characteristics

As part of the ITER-site agreement, Japan and the EU entered into an agreement called the Broader Approach. One element of that agreement is a "satellite" tokamak facility to be sited in Japan at the JAEA Naka Center. Both Japan and the EU are partners in the design, construction, and eventual exploitation of this tokamak, called JT-60SA (Super Advanced). The tokamak assembly will be completely new, but will reuse the site infrastructure including the auxiliary heating and current drive systems and diagnostics from JT-60U. This should bring the project into operation in fusion-relevant regimes more rapidly than would be possible in a completely new facility. The goal is to start operation in 2016, well in advance of the ITER first plasma. However, high-performance

operation will require deuterium plasma which may be limited until full remote handling capability is available in 2021.

JT-60SA will have superconducting toroidal and poloidal magnetic field coils and significant remote handling capability to facilitate the study of stationary high-performance plasmas. Since a fundamental mission is to support ITER, single-null divertor operation will be the starting configuration. However, the machine design provides for eventual full double-null divertor operation to explore steady-state advanced tokamak scenarios for burning plasma devices beyond ITER. Internal coils for feedback control of MHD instabilities are integrated into the design to facilitate operation above the no-wall MHD β-limit. The control coil design is more DEMO relevant than those on present or planned experiments, consistent with the long-range mission to explore steady-state operation relevant to power plant designs. The dominant heating will be from neutral beam injection, including the negative-ion based system developed on JT-60U, which is prototypical in many aspects of the system planned for ITER. Pulse lengths will be limited by the energy capacity of the auxiliary heating systems and ultimately by the neutron budget of the site. The first-wall heat handling will use water-cooled graphite, which has the advantage of robustness allowing exploration of parameter space at the cost of not testing the physics and technology of metallic or high ambient temperature first-wall solutions.

Technical Importance of JT-60SA collaboration

The parameter space accessible to JT-60SA lies between that of present US tokamaks (DIII-D and Alcator C-Mod) and future burning plasma experiments such as ITER. JT-60SA is the only device that can extend the high-performance double-null divertor scenarios developed in the US to larger scale and toward burning plasmas. In the event JET is completed and closed, JT-60SA will fill the same role for extending single-null divertor scenarios until ITER operates. The presence of multiple facilities at different physical sizes and different dimensionless parameters provides crucial data for validating models, which should lead to increased confidence in projections for future fusion systems (relevant to **Thrust 4** and **Thrust 6**).

The long-pulse capability is primarily of use to demonstrate fully stationary solutions where the inductive current is completely equilibrated. In addition, the long pulses allow exploration of issues regarding control of the operational point (**Thrust 5** and **Thrust 8**) and the impact of feedback control of instabilities with magnetic coils on the rest of the facility. The coil design extends present understanding and technology towards that needed for powerplants (**Thrusts 2** and **16**).

The 500 keV negative-ion neutral beam will enable study of fast-ion dynamics and may facilitate the development of fast-ion diagnostics (**Thrust 1**). The interaction of these high velocity ions (similar to fusion alphas) with MHD instabilities such as Alfven eigenmodes and the resistive wall mode will be of considerable interest (**Thrusts 2** and **5**).

Issues regarding integrated advanced scenario sensitivity to wall conditions (**Thrust 12**) and information on carbon migration and retention of hydrogen in carbon will certainly be gained (**Thrust 9**); however, the low wall operating temperature appears to be a significant difference from any future application of graphite for the PFCs. The temperature dependence of the relevant physical processes of importance (erosion and co-deposition) is strong enough to limit the utility of the information gained. Double-null operation in JT-60SA will provide information on size scaling, which is unavailable from any other present or planned experiments. The graphite divertor has the advantage of relatively robust steady-state power handling, but the projected global power density (P/S, P/R) is significantly smaller than for reactor scenarios (**Thrust 12)**. These issues, combined with questions about the adequacy of the pumping, may compromise the range over which integrated core-boundary power handling solutions can be extended.

Opportunities for collaboration on JT-60SA

At present, the US does not have any formal collaboration with the JT-60SA project. A US collaboration on the technology of the negative-ion based neutral beam injector is continuing from the JT-60U project, and will contribute to both ITER and JT-60SA.

Since it is still early in the JT-60SA project, there may be many opportunities for participation and partnership. The Broader Approach agreement between Japan and the EU includes a procedure allowing other ITER partners to join. It should be expected that access at a scale where JT-60SA is a key part of the US fusion energy science strategy will require a commensurate level of investment. Areas of investment that would both enhance or accelerate reaching the project goals and match US interest and expertise are discussed here.

Of the total heating portfolio planned for JT-60SA, 34 MW of the 41 MW total are from neutral beam injection. Looking toward fusion energy production, where the dominant heating source must be fusion alphas, electron cyclotron heating mimics more closely than neutral beam injection the heating effects of fusion alphas. This is because electron cyclotron heating deposits energy directly to the electrons without any corresponding fueling or torque input. In addition, based on previous experiments, steady-state high-performance tokamak operation requires off-axis current drive, which can be supplied by electron cyclotron current drive or lower-hybrid current drive. Contributions toward enhancing the JT-60SA electron cyclotron system, up to supplying a complete system to double the JT-60SA system (~10 MW at the source) or a lower hybrid system should be considered, and the required flexibility evaluated. This would build on the US expertise in electron-cyclotron and lower-hybrid physics and technology, and steady-state tokamak scenario development.

The US could propose to supply the control system for the tokamak, both hardware and software. This would build on extensive US experience with digital control systems and MHD instability control, extending our expertise in the direction required for fusion energy production.

Finally, US membership in JT-60SA may require support for construction and operations, or taking responsibility for supply and operation of specific systems needed to ensure the success of the project. For example, the US could supply the remote handling equipment needed in order to move more quickly to high-performance operation. Similarly, the US could supply components to accelerate the installation of the upper divertor, speeding progress in exploring the advanced scenarios of greatest interest for steady-state high-performance operation. Hardware developed by the US for ITER obligations may also be of interest to JT-60SA, which would leverage existing US investments.

4.5 Korea Superconducting Tokamak Advanced Research (KSTAR)

http://kstar.nfri.re.kr

KSTAR is a new superconducting tokamak located in the Republic of Korea, similar in size to DIII-D but higher aspect ratio. The mission of KSTAR is to develop high performance steady-state physics operation and technology essential for ITER and fusion reactor development in Korea. KSTAR has superconducting PF and TF coils (Nb_3Sn and NbTi), copper stabilizing plates, and an array of normal-conductor in-vessel control coils (IVCC) for fast vertical position and MHD stability control for $n \leq 2$. The array has control coils at three poloidal locations, similar to ITER, providing control of the helical structure, and will be used to assess control of ELMs in 2011. KSTAR has inertial carbon PFCs, which will be upgraded for active cooling in 2011-2012. A divertor cryopump is planned for 2012. The PFC bake temperature will be upgraded from 200C (present) to 350C in 2011. KSTAR will have 8.5 MW (source) of heating power (NB, ICH, ECH, LHCD) at the end of its Operations Phase I in 2012, and will increase this to ~20 MW in 2018 and ~30 MW in 2023. During Operations Phase II (2013 – 2017), the experiments will focus on extending plasma operation to 300 sec. duration and preparing for ITER experiments. Operations Phase III (2018 – 2022) will focus on high performance and advanced scenarios, with a target of steady-state operation at twice the no-wall beta-limit at full field and current.

KSTAR has operated since 2008, producing limited, single null, and double null shaped plasmas up to 7s duration, H-mode operation, and NB heating up to 1.5 MW.

KTAR is a central part of the fusion energy development program in Korea. Its role as a test-bed for ITER physics and technology development and assessment is strongly emphasized and integrated into its planning. It is viewed as a key step to establishing the basis for construction of a future DEMO in Korea.

Technical importance of KSTAR collaboration

KSTAR will offer the ability to study long pulse steady-state superconducting tokamak plasmas with high heating power and a configuration similar to that of ITER. Collaboration on KSTAR will provide opportunities for the US to study ITER-prototypical long-pulse plasma confinement, stability, and control strategies (**Theme 1**). KSTAR collaboration can also enable the US to extend advanced scenarios to long pulse, testing compatibility with steady state, non-inductive operation and determining control

and operating limits. This will address the ReNeW issues in **Theme 2 (Thrust 5)**, including the control of disruption-free, steady state operation with thermally equilibrated walls. This will include use of the IVCC to control rotation, to stabilize wall modes to access high-β, and to control ELMs. Understanding the robustness of control and verifying the stability and confinement physics over wall equilibration timescales are critical knowledge for the US program. Several aspects of fast particle effects can be addressed (**Thrust 3**), albeit without an alpha particle population, including the validation of fast-ion instability theory in steady-state advanced scenarios, along with methods to diagnose these modes. Related effects of the fast particle distribution on RWM stability in long-pulse plasmas could be addressed. The impact of fast ion losses in steady state on first wall components could be assessed. ITER-like integrated scenarios with superconducting coils, and ITER relevant heating and current drive methods for long pulse could be demonstrated and qualified (**Thrust 4**). The main aspect of operation in this context is sufficiently long pulse to reach wall equilibration. The related physics is far reaching, including control of radiation fraction, impurities, and detachment, and the maintenance of operating points at high beta and bootstrap current in the presence of fluctuations.

KSTAR will provide information on a number of steady-state related issues discussed in Section 4.1, in common with other superconducting facilities. These include development of reliable diagnostics for steady-state plasmas (**Thrust 1**), simulating the dynamics of high gain plasmas (**Thrust 8**), steady-state plasma-material interactions and heating (**Thrust 10**), and integration of high performance steady-state plasmas with a divertor exhaust systems, including impurity and fueling control (**Thrust 12**). However, the low wall operating temperature and use of carbon will limit the impact of the PMI and divertor studies.

Opportunities for enhanced US collaboration on KSTAR

The US has collaborated on KSTAR since its design phase. Ongoing US collaborations include development of the control system, NBI and ECH heating, a number of diagnostics, equilibrium and stability analysis, scenario development, and future stability control techniques. Experimental proposals are submitted in a research forum format similar to those conducted at US fusion facilities, and a program advisory committee with international representation helps steer the project. A US-KSTAR bilateral collaboration meeting is held yearly, which coordinates US research activities with KSTAR developments and guides plans for future research.

There are a many opportunities for enhancing the US collaboration on KSTAR, including:

- Upgrading the power supplies and control system for the IVCC for combined control of radial and vertical position and MHD instabilities;
- Developing methods to predict, avoid, and mitigate disruptions, and characterize resulting reliability in advanced, steady-state scenarios;
- Adding more heating and current drive power, to improve control of the outer current profile and allow earlier investigation of high-performance steady-state plasmas;

- Augmenting the edge and in-situ first wall diagnostics, to better understand the steady-state plasma boundary and plasma-surface interactions;
- Upgrading the divertor and first wall to materials prototypical of future energy systems;
- Upgrading the divertor and first wall to operate at high ambient temperature, if possible.
- Upgrade the in-vessel coils to control higher-n field perturbations for MHD control, if possible.

In order to improve the effectiveness of US-KSTAR collaborative research, improvements to the present data access capabilities from the US are needed. This should include mechanisms for rapid, routine access to KSTAR data by US-based researchers, such as much higher network bandwidth or enabling data-mirroring sites in the US.

4.6 Wendelstein 7-X (W 7-X)

http://www.ipp.mpg.de/ippcms/eng/for/projekte/w7x/index.html

W 7-X is a large stellarator using superconducting coils, under construction in Greifswald, Germany at the Max Planck Institute for Plasma Physics (MP-IPP). The facility is designed for plasma durations of at least 30 minutes. The three-dimensional plasma shape has been numerically optimized to provide good plasma confinement, and MHD stability at high-β with good flux surface quality, based on theoretical models. The W 7-X optimization uses the 'quasi-isodynamic' principle, which minimizes drift-orbit widths and the bootstrap and Pfirsch-Schluter plasma currents. W7-X has a configuration optimized for plasma confinement, and is expected to give access to high temperature high-β plasmas with low ρ^*, and low collisionality. The experiment includes a helical-island divertor, for controlling particle and heat exhaust. The coil system is designed to provide shaping flexibility around the optimized design configuration. W 7-X will be much more capable than any 3D experiment in the US.

The first W 7-X physics experimental campaign is expected to start in 2015 with 8 MW of ECH and 7 MW of neutral beam heating and an initial set of diagnostics including plasma profile measurements. The initial divertor will not be actively cooled, which will limit the pulse length to ~10 seconds. The second campaign is planned to start in 2018, after installation of an actively cooled divertor, allowing steady-state operation. At that time, they plan to have 10 MW of ECH and 10 MW of neutral beam heating. The divertor designs use carbon plates. Replacing the plates with tungsten is being considered for a later upgrade.

Technical importance of Wendelstein 7-X collaboration

Collaboration on Wendelstein 7-X provides the only opportunity for the US to experimentally study steady-state, disruption-free toroidal confinement using optimized 3D magnetic shaping with fusion-relevant plasma parameters, addressing many of the stellarator research requirements in ReNeW **Theme 5** and issues in **Thrust 17**. In

particular, W 7-X will be the only experiment with optimized 3D shaping available to test whether ion confinement can be improved at low collisionality and high-β. It will provide crucial information on the operating limits for high-β optimized stellarators, and the limiting mechanisms. However, W 7-X will not provide information on quasi-symmetric configurations, which may lead to reduced aspect ratio (and costs) for optimized stellarators. In addition, confinement in quasi-symmetric plasmas is closely related to confinement in tokamaks, allowing a shared understanding and easier integration with ITER results.

Wendelstein 7-X will develop the understanding of 3D divertors at high power and steady-state, together with LHD. This will contribute to our understanding of the physics of the plasma boundary layer, **Thrust 9**, including 3D magnetic field effects. This understanding will enable the design of the plasma and divertor shape to control heat and particle exhaust, and impurity generation.

In general, W 7-X will provide the data to develop and validate predictive models of fusion plasmas in optimized 3D magnetic configurations, **Thrust 6**.

Through all of these activities, collaboration on W 7-X provides the earliest opportunity for the US to use significant 3D magnetic shaping to create predictable, high performance, steady-state plasmas (**Theme 2**) in an integrated experiment. Since stellarators typically require much less active control than tokamaks, the W 7-X experiments will help determine the minimum number of diagnostics and feed-back actuators required to control long-pulse high pressure plasmas.

Collaboration on W 7-X will also enable progress by the US program on developing and understanding steady-state related issues, in common with other superconducting facilities as discussed in Section 4.1. These include issues related to diagnostics for steady-state plasmas (**Thrust 1**), simulating the dynamics of high gain plasmas (**Thrust 8**), steady-state plasma-material interactions and heating (**Thrust 10**), and integration of high performance steady-state plasmas with a divertor exhaust systems, including impurity and fueling control (**Thrust 12**).

Opportunities for enhanced US Collaboration on W 7X

The MP-IPP has invited the US to become a partner in the exploitation of W-7X, together with other countries (primarily in the EU). They proposed that the US take responsibility for key parts of the W-7X program, supply part of the professional scientific team, and US have membership on the W-7X scientific steering committees and council. A series of US / MP-IPP meetings were held to identify areas where the US could make significant contributions.

In FY2010, the US DOE funded a multi-institution collaboration with W 7-X. This collaboration is focused on a coordinated set of tasks, preparing specific hardware for installation on W 7-X and developing key analysis capabilities for W 7-X operation, including:

- Design and construction of a toroidal array of external trim coils

- Preparation of magnetic-equilibrium analysis methods and codes
- IR cameras, for monitoring internal surface temperatures
- Design of a specific high heat-flux divertor component
- Strategies for control of the divertor

These tasks were chosen in coordination with the overall W 7-X program and build on recognized US strengths. They are viewed by MP-IPP as significant contributions to the W 7-X program, establishing a foundation for strong US collaboration. This level of activity is less than that requested by the MP-IPP for full US partnership in W 7-X, but these actions may enable a later decision to enter into full partnership on W 7-X exploitation. If the US maintains responsibility for these areas in the W 7-X research program, these investments may provide opportunities for US leadership of parts of the W 7X program, preserving US strengths.

The US stellarator community identified additional candidate topics where the US could take significant responsibility, as part of a increased US effort to enter into W 7-X partnership. These include:

- Predictive models of plasma confinement in 3D magnetic configurations and experimental analysis
- Diagnostics and analysis to understand the plasma-divertor interaction
- Detailed plasma diagnostics, for example of plasma turbulence and fluctuations
- Plasma control and discharge optimization
- Long-pulse fueling control

4.7 Summary of EAST, JET, JT-60SA, KSTAR and W7-X Opportunities for US Collaboration

From the above sections 4.2 – 4.6, EAST, JET, JT-60SA, KSTAR, and W 7-X present different opportunities for US collaboration. This is summarized in Table 2 in terms of their expected ability to make progress on the ReNeW issues identified in Section 3 as being generally within the reach of international collaboration. This summary is based on the development plans for each facility, which are contingent on funding and successful implementation.

Table 2. Capability of international facilities to address ReNeW issues in Section 3, organized by Thrust.

Available < 5 years	Available < 10 years	Under consideration

	EAST	JET	JT-60SA	KSTAR	W7-X
Th 1: Measurement techniques for burning plasma					
Long-pulse steady-state diagnostics					
Long-pulse machine protection					
Develop DEMO prototypical alpha diagnostics					
Remote maintenance and calibration of diagnostics					
Th 2: Control of transient events in burning plasma					
Long-pulse ITER prototype regime, without ELMs & disruptions					
Test extrapolation of ELM & disruption avoidance and mitigation to larger scale					
Long pulse instability control at high beta-N					
Th 3: Role of Alpha particles in burning plasma					
Validate understanding of effect of fast-ion instabilities in steady-state scenarios, at larger scale					
Understand interactions between fast-ion and global MHD instabilities in steady-state scenarios					
Impact of fast on losses on first wall in steady scenarios					
Test control of alpha heating profile					
Th 4: Qualify operating scenarios and physics for ITER					
ITER scenarios with DT, including alpha particles (with ITER-like PFCs)					
ITER scenarios with superconducting coils, relevant H&CD methods, long pulse					
PFC cleaning for long-pulse					
Test ITER fueling and pumping at largest scale					
Development of burn control strategies (sim.)					
Th 5: Expand limits for controlling and sustaining fusion plasmas (tokamaks)					
Maintain AT scenarios suitable for high fusion power gain for long pulse, restricting diagnostics & actuators					
Determine min. diagnostic & actuator set needed for high fusion power gain long pulse					
Develop & demonstrate long-pulse fueling & exhaust systems					
Th 6: Develop predictive models					
Develop & validate models of long-pulse plasma confinement					
Use predictive models to improve control sys.					
Predict & validate safe operating/control regions					
Develop & validate models of ITER and BP					

	EAST	JET	JT-60SA	KSTAR	W7-X
Th 7: Exploit high temperature superconductor & magnet innovation					
Develop database of reliability for superconducting systems					
Th 8: Integrated dynamics of self-heating and self-sustained burning plasma					
Simulate self-heating in high-gain equiv., long-pulse burning plasmas; demonstrate simulated burn control.					
Th 9: Physics of boundary layer plasmas					
PMI with prototypical PFC surfaces					
Innovative divertor configurations					
Th 10: Science & Technology of plasma-surface interactions					
Transient PMI at demo-level fluxes					
Steady-state PMI at demo-level fluxes					
Steady-state PMI, including erosion effects, with prototypical surfaces					
High ambient temperature effect on PMI and physical chemistry					
Long-pulse RF antenna development					
Th 11: Improve power handling thru innovation					
Refractory metal heat sinks at high temp					
Liquid metal (including lithium) PFCs					
Neutron damage effects on PFCs					
Th 12: Demonstrate integrated solution for PMI compatible with optimized core plasma					
Integrated testing of prototypical actively cooled PFCs					
Particle & fueling control with long pulse divertor					
Effect of high temperature on plasma/PFC integration					
Th 13: Science & technology for power extraction & tritium sustainability					
Impact of fusion operating environment on tritium breeding blankets					
Th 14: Develop material science & technology to harness fusion power					
Improved materials for the fusion neutron environment					
Validate models of materials for use in fusion applications, including interaction with plasma and neutrons.					
Th 15: Create integrated designs & models for attractive fusion power systems					
Advanced design studies of integrated fusion energy systems					
Develop models of integrated fusion systems.					

	EAST	JET	JT-60SA	KSTAR	W7-X
Th 16: Develop the spherical torus to advance fusion nuclear science					
Startup using EBW					
ELM control & mitigation using internal coils					
Explore extreme divertor flux expansion					
Th 17: Steady-state, disruption free confinement with 3D shaping, emphasizing quasi-symmetry					
Steady-state 3D divertor; integration with good core confinement					
Effect of 3D shaping on confinement and operating limits					
Stellarator confinement at low rho*, low collisionality, high beta, consistent with steady state					
Fueling and impurity control in long pulse, optimized configurations					
Th 18: Achieve high performance confinement with minimal external magnetic field					
Extend confinement scaling to higher current					

4.7 Additional collaboration opportunities (AUG, LHD, MAST, RFX)

AUG, LHD, MAST, and RFX host valuable ongoing collaborations with a number of US researchers and institutions. However, they offer less opportunity for expanded US collaboration and partnership than the previous group of facilities. In part, this is because they are all mature, well-established facilities and research programs, and are each largely self-sufficient. Thus, US contributions are likely to have only a minor effect on their capabilities or accomplishments. Some of these facilities have similar characteristics as current US domestic facilities, and offer additional capabilities for the US program only in connection with a few specific differences. Here, we briefly discuss the characteristics of these facilities and their opportunities for US collaboration.

4.7.1 ASDEX-Upgrade (AUG, Germany)

http://www.ipp.mpg.de/ippcms/eng/for/projekte/asdex/ziele/index.html

AUG is a medium scale, short-pulse tokamak, heated by neutral beams, ECH, and ICRF with a plasma shape similar to ITER. The PFCs and divertor on AUG are coated with a layer of tungsten, providing experimental data on integrating ITER-like scenarios with a tungsten wall. AUG is developing methods to control plasma interaction with the metallic wall, including controlling plasma radiative losses via impurity doping and design of ICRF launchers to minimize edge electric fields. AUG is adding a new set of in-vessel magnetic coils to stabilize and control ELMs using resonant magnetic perturbations with $n \leq 4$, and to access higher-β by stabilizing resistive wall modes. These coils provide opportunities for new collaborations by US researchers on

controlling such transient events, **Thrust 2**. This capability will allow AUG to test the compatibility of advanced tokamak scenarios with metallic walls. AUG invites and facilitates collaboration by researchers around the world community, with an open planning and decision process.

4.7.2 Large Helical Device (LHD, Japan)

http://www.lhd.nifs.ac.jp/en/

LHD is currently the largest stellarator in the world. It uses superconducting coils, and has operated with plasmas lasting an hour. It is dominantly heated by neutral beams and ICRF. LHD has achieved the highest temperature, average pressure, and peak pressure of any stellarator. LHD has a broad research program on all aspects of plasma confinement and heating. However, its magnetic configuration has regions with a magnetic hill, limiting stability, and has significant magnetic ripple, which limits ion confinement. LHD continues to increase its heating power and diagnostics, and is installing a new closed helical divertor, to control steady-state power and particle exhaust.

Ongoing US collaborations are investigating issues in **Thrust 17**, including the mechanisms limiting the maximum β-value, core and edge MHD instabilities, reconstruction of the magnetic equilibrium, and turbulence modeling. The US is installing new diagnostics on LHD to measure ion temperatures and flows, and is preparing to operate them. LHD has invited the US to increase the level of collaboration. A US collaboration investigating the boundary layer physics with the new helical divertor, including diagnostics and modeling, would help make near-term progress on **Thrusts 9** and **17**. In addition, collaboration with LHD can contribute to addressing many of the steady-state related issues, in common with other superconducting facilities as discussed in Section 4.1.

4.7.3 Mega Ampere Spherical Tokamak (MAST, UK)

http://www.fusion.org.uk/MAST.aspx

MAST is a spherical torus (ST) experiment with similar size and capabilities as the NSTX experiment in the US. It is being upgraded to increased plasma current (2 MA), toroidal magnetic field (0.8 T), and heating power (7.5 MW of neutral beams). While these overall parameters are similar to those for NSTX-U, other key capabilities and approaches are complementary. In particular, MAST-U will have (i) a carbon-faced divertor with very high flux expansion (super-X) and a cryo-pump, (ii) electron Bernstein wave heating (2 MW), and (iii) internal non-axisymmetric coils to control for MHD instabilities ($n \leq 6$ with the lower coil array, $n \leq 3$ with the lower array).

MAST research dominantly contributes to **Thrust 16**, which targets ST-specific issues, and develops predictive understanding of toroidal confinement and stability (**Thrust 6**). An ongoing US collaboration has contributed the EBW gyrotron system and is investigating solenoid-free plasma startup and ramp-up. The new super-X divertor will explore ways to reduce and control the heat flux density to the divertor plates (**Thrust 9**).

Refractory metal PFCs are under consideration for the expanded divertor (**Thrust 11).** The non-axisymmetric coil array will provide more control over the 3D perturbation spectrum than present experiments, for improved understanding of the physics of ELM control (**Thrust 2**). These new capabilities provide additional opportunities for US collaboration, to apply US expertise and to contrast the results with complementary US experiments.

4.7.4 RFX-mod (Italy)
http://www.igi.pd.cnr.it/wwwexp/index.html

RFX is the largest reversed field pinch (RFP) and has the highest plasma current. It uses a unique array of saddle coils surrounding the plasma for active feedback stabilization of the MHD instabilities. This allows RFX to operate with a relatively thin stabilizing conducting shell, unique for an RFP. The high plasma current gives access to plasmas with larger Lundquist numbers, which is important for understanding RFP confinement scaling, **Thrust 18**. RFX can also be operated as a tokamak at low current, to compare the two magnetic configurations. The RFX program is currently focused on understanding the tendency of the plasma to self-organize into a helical configuration with reduced magnetic turbulence and fluctuations. US researchers collaborate on RFX experiments, and have applied codes developed for stellarators to analyze and understand the helical RFX plasmas.

With a substantial upgrade of the RFX power supplies and other systems, it may be possible to significantly increase the RFX plasma current, evaluate methods to sustain the current, and assess the impact of sustainment techniques on RFP plasma confinement, which are high priority RFP issues. Such an upgrade program could motivate an increased US collaboration.

5. Timescales and Findings.

The technical capabilities of the international facilities will develop over the next decade, which will enable their research focus to evolve and address the research and development issues. At the highest level, the planned evolution is depicted in Figure 1. Here, high power is defined (somewhat arbitrarily) by availability of 20MW of total heating power (source).

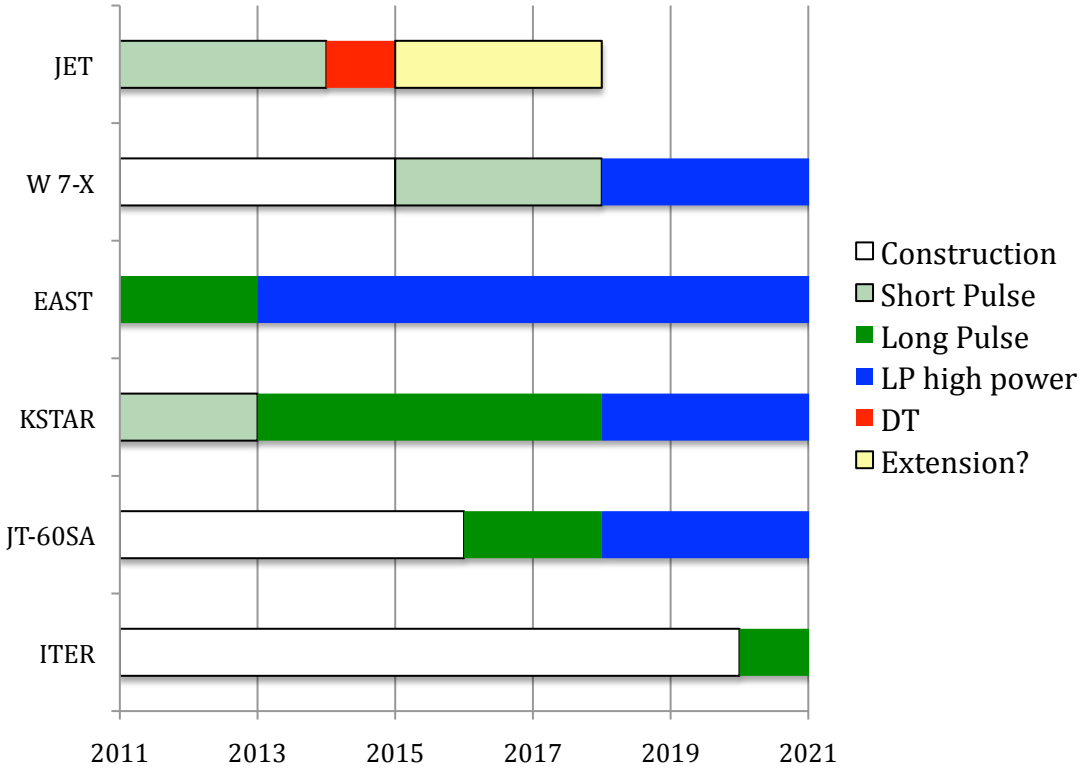

Figure 1. Planned facility capabilities and focus versus calendar year.

The highest impact opportunities were identified and discussed in Section 4, and are displayed in Table 2. The time-phasing in Figure 1 guides emphasis between the opportunities. **The Task Group has identified the following opportunitites for significantly enhanced collaboration, ordered by the need for immediate decisions:**

- **JET**: to prototype ITER operating scenarios with DT plasmas and ITER-like walls at the largest available scale, to make ITER operation more efficient. The US should seek to reduce risks and uncertainties for ITER, e.g., by validating at the JET scale techniques chosen by ITER to suppress ELMs and avoid and mitigate disruptions.

- **W 7-X**: to study and assess steady-state, disruption-free high-performance confinement in a large-scale optimized stellarator, including operating limits and compatibility with divertors.

- **EAST**: to study long-pulse plasma-wall interaction with prototypical metallic walls (including tungsten and/or lithium), and the effects of high ambient temperatures. EAST, KSTAR, JT-60SA, and W7-X will also study long-pulse plasma-wall interaction with water-cooled graphite PFCs.

- **EAST** and **KSTAR**, followed by **JT-60SA**: To study and assess steady-state, high-performance confinement in tokamaks, including operating limits, ability to operate disruption-free, and compatibility with divertors. EAST will have the earliest integrated capability. The three facilities will allow comparison of the effects of different mixtures of heating and current-drive techniques. JT-60SA will be crucial to extend steady-state, high-performance tokamak scenarios to larger scale.

These opportunities for enhanced collaboration were identified based on the current programs and plans for both the international and domestic fusion experimental facilities. There was no attempt to prioritize the opportunities relative to the ongoing US research activities.

There are a number of ongoing collaborations with international facilities targeting specific issues. The budgets for the ongoing collaborations have been decreased significantly, imperiling past US commitments and investments. The US program needs to follow through and complete existing commitments, as a basis for future agreements.

The opportunities for enhanced collaborations build upon international capabilities that are not available in US domestic facilities. In the first three cases, the capabilities are unique. These identified collaborations can have high impact because:

1. They address high priority issues,
2. US engagement can have a significant impact on the success of the research or timeliness,
3. They can preserve and extend US research expertise.

In order to achieve this impact, the US must commit suitable resources and personnel to these international programs, viewing them as key strategic elements of the US program, as discussed in section 2. This will prototype the relationship the US program will have with ITER. In order for the enhanced collaborations to succeed, appropriate task agreements with the host country and organization must be negotiated, including resource commitments, research access by US researchers, data access, and publication rules. The US has already been invited to become a partner in the JET and W7-X facilities. EAST and KSTAR have invited the US to strengthen the collaborations, including the design and planning for future steps beyond ITER. For JT-60SA, a formal application to join the Broader Approach must be submitted and approved. To implement the enhanced collaborations, the US should enter into appropriate negotiations to enhance the

relationships with the facility hosts. The timescales for negotiation are driven by the planned evolution of the facilities, see Fig. 1.

- If the US chooses to partner in JET, negotiations should start immediately.
- To participate in the aggressive EAST development program, the US should seek to strengthen its collaboration quickly with dedicated resources.
- While W 7-X and JT-60SA are still under construction, the US should negotiate relationships including a basis for US participation and partnership, and to help provide resources for project success. This has already started for W 7-X, but must be further strengthened to enable substantial impact for the US program.
- The impact of the KSTAR collaboration can be enhanced with increased funding.

As discussed in section 2, accomplishing US goals via collaboration will be more challenging than using domestic facilities. It will be more challenging for the program planning and execution, for the US institutions, and for the individual researchers. It is less flexible, because the required commitments become international agreements. It will also require substantial leadership to guide US personnel and institutions to conduct a substantial portion of their research outside the US, and to provide the resources needed to succeed.

However, in order for the US fusion science program to prepare for ITER, make progress on key fusion issues, and remain at the forefront of the field, it needs access to capabilities not available in current domestic facilities. This is especially true for issues related to stellarators and superconducting long-pulse tokamaks. To obtain access to these capabilities, the US program must either invest in more capable domestic facilities or significantly enhance its collaboration with international facilities.

Appendix A
Membership

Dr. David Humphreys	General Atomics
Dr. Charles Kessel (resigned to lead FNS Pathways Assessment)	PPPL
Dr. Timothy Luce	General Atomics
Dr. Stanley Milora	ORNL
Dr. Steven Sabbagh	Columbia Univ.
Prof. Dennis Whyte	MIT
Dr. Michael Zarnstorff (Chair)	PPPL

Appendix B
Charge

Evaluate and prioritize the opportunities for US collaboration on EAST, KSTAR, JET, JT60SA, LHD, W-7X and other major international facilities to prepare for US participation on ITER and to address the issues and gaps discussed in the recent report "Issues, Gaps, and Opportunities: Towards a Long-Range Strategic Plan for Magnetic Fusion Energy", DOE-SC-0102.

Appendix C
Documents and Resources

The Task Group made use of papers and presentations at the meetings coordinating the US bilateral agreements with China, Japan, Rep. of Korea, and the EU, the presentations at various IAEA Fusion Energy Conferences, APS/DPP Conferences, IEA implementing agreement meetings, ITPA meetings, advisory committee meetings occurring in 2008 – 2011, and other documents and presentations as available. These include:

EAST

- Presentations at the US-PRC Bilateral Meeting in Geneva, Switz., 16 Oct. 2008
- Presentations at the EAST International Advisory Committee meetings, 2009 and 2011.
- "Report on the 1st Meeting of the Joint Working Group on Safety for the US-PRC Coordinating Committee of Fusion Energy", 9-14 Dec. 2009
- "EAST Diagnostics Capability in 2010", L.Q. Hu, 15 March 2010
- "Present State and Future Plan for EAST", J. Li, March 2010, San Diego
- "Recent Progress in System Development and Experiments on EAST, B. Wan, 4th US-China Workshop, 5-7 May 2010, Wuhan
- "EAST Tokamak", Y.X. Wan, 14 Feb. 2011, San Diego.

JET

- Presentations at the US-EU bilateral coordinating committee meetings, 14 Oct. 2008 (Geneva) and 16 March 2010 (Washington D.C.)
- "Diagnostic Developments for the operation of JET with an ITER-like wall", A. Murari
- "New Developments in the Diagnostics for the Fusion Products on JET in preparation for ITER", A. Murari
- Presentations on the ECH and RMP feasibility studies to the EFDA STAC AHG (8-9 June 2010), and the final report of the EFDA STAC AHG (23 Sept. 2010)
- Presentations at the JET Science Meeting of 8 Nov. 2010 on a JET DT program, including talks by H. Weisen et al, G. Sips et al., and C. Challis et al.

JT-60SA

- The Broader Approach Agreement, http://eur-lex.europa.eu/LexUriServ/site/en/oj/2007/l_246/l_24620070921en00340046.pdf
- "Guidelines on participation of other ITER Parties in Broader Approach Activities", ITER Council 2nd meeting, 17-18 June 2008.
- "JT-60SA, ITER Satellite Tokamak Program of JA-EU", J. Bucalossi, 31 Aug. 2009, Karlsruhe.
- Presentations at the US-Japan ESM, 24 March 2010, Washington D.C.
- "Recent Progress of the JT-60SA Project", T. Fujita, 11 Feb. 2011, San Diego.

KSTAR

- "Summary of Results of KSTAR Workshop", 15-16 April 2009, San Diego, presentations: http://fusion.gat.com/conferences/kstar09/agenda.php
- "Simulations of KSTAR high performance steady-state operation scenarios", Y.-S. Na et al, Nucl. Fusion 49 (2009) 115018.
- "Status and Plan or Fusion Research in Korea", G.-S. Lee, 2 Dec. 2009, Washington D.C.
- "Status of the KSTAR Program", Y.-K. Oh, May 2010
- Presentations at the NFRI External Review, 23-25 August 2010.
- Presentations at the KSTAR Conference and KSTAR-US Workshop, Feb. 2011, Daejeon
- Presentations at the KSTAR PAC meeting, 2011.
- "Past, Present, and Future of Fusion Energy Development in Korea", G.S. Lee, 9 March 2011, Princeton.

LHD

- "2008 Status of NIFS and Universities in Japan", O. Motojima, 16 Oct. 2008, US-JA CCFE Meeting
- Presentations at the US-Japan ESM, 24 March 2010, Washington
- "Invitation to Joint Experiment on LHD", H. Yamada, 7th CWGM, 30 June 2010

MAST

- Presentations at the MAST PAC meeting, 6-8 Sept. 2010.

RFX-mod

- "RFX-mod and the International RFP Programme", P. Martin, 19 March 2009, Princeton

W 7-X

- "Cooperation on Wendelstein 7-X", T. Klinger, 16 March 2010, Washington D.C.
- "Physics programme for initial operation of Wendelstein 7-X", H.-S. Bosch et al., Contributions to Plasma Physics, 2010.

US Program

- "Research Needs for Magnetic Fusion Energy Sciences", 2009.
- "Report of the FESAC Toroidal Alternates Panel", FESAC, 2008.
- "Priorities, Gaps, and Opportunities, Towards a Long-Range Strategic Plan for Magnetic Fusion Energy", FESAC, 2007.

We thank the leaders of the international facilities for many useful discussions about their plans and programs.

We thank the following individuals for helpful suggestions for enhanced US international collaboration opportunities and perspectives: P. Gohil, R. Goldston, H. Guo, J. Harris, R.

Hawryluk, D. Hillis, E. Marmar, G.H. Neilson, R. Nygren, J. Sarff, C. Skinner, and C. Wong. The 2010 Field Work Proposals were also used to identify activities and opportunities.

Appendix D
Abbreviations, Acronyms, and Names

APS – American Physical Society
ASIPP - Academica Sinica Institute of Plasma Physics
AUG – ASDEX-Upgrade, tokamak experiment (Germany)
C-mod – Alcator C-mod, tokamak experiment (US)
DEMO – DEMOnstration fusion power plant
DIII-D – tokamak experiment (US)
DT – deuterium & tritium
EAST – tokamak experiment (China)
EBW – Electron Bernstein Wave
ECH – Electron Cyclotron wave Heating
ECCD – Electron Cyclotron wave Current Drie
ELMs – Edge localized modes, instabilities causing bursts of plasma lost to the wall.
EFDA – European Fusion Development Agreement
F4E – Fusion for Energy, the European Domestic Agency for building ITER
FESAC – Fuson Energy Sciences Advisory Committee, DOE Office of Science
IAEA – International Atomic Energy Authority
ICRF – Ion Cyclotron wave Range of Frequency heating
ITER – tokamak to study burning plasma (Cadarache, France; China, EU, India, Japan, Rep. Korea, Russia, US)
ITPA – International Tokamak Physics Activity
IVC – In-vessel coil
JAEA – Japanese Atomic Energy Authority
JET – Joint European Torus, tokamak experiment (EU)
JT-60SA – Japan Torus 60 Super Advanced, tokamak (Japan)
KSTAR – Korea Superconducting Tokamak Advanced Research, tokamak (Rep. Korea)
LHCD – Lower Hybrid wave Current Drive
LHD – Large Helical Device, stellarator experiment (Japan)
LTX – Lithium Tokamak experiment, tokamak (US)
MAST – Mega-Ampere Spherical Tokamak (UK)
MHD – magnetohydrodynamic
MP-IPP – Max Planck/Institute for Plasma Physics
NBI – Neutral Beam Injection
NSTX – National Spherical Torus Experiment, tokamak (US)
PCS – Plasma control system
PF – poloidal field
PFC – Plasma facing component
PMI – Plasma Material Interaction
ReNeW – Research Needs for Magnetic Fusion Energy Sciences (US)
RF – Radio frequency
RFX – Reversed Field Experiment, RFP experiment (Italy)

RMP – Resonant magnetic perturbation
RWM – Resistive wall mode
SC – superconducting
TF – toroidal field
W 7-X – Wendelstein 7-X stellarator (Germany)

β – beta, dimensionless pressure: the ratio of the plasma pressure to the magnetic field pressure
ν^* – dimensionless collisionality: the ratio of the parallel scale-length of the magnetic field to the collisional mean-free path
ρ^* – normalized gyroradius or inverse dimensionless scale size: the ratio of a characteristic gyroradius to the system scale-size, typically the minor radius of the torus
n – Fourier toroidal mode number, typically for an instability or an external perturbation