USBPO United States Burning Plasma Organization

Report of the Burning Plasma Organization Panel on Planning for US Participation in ITER

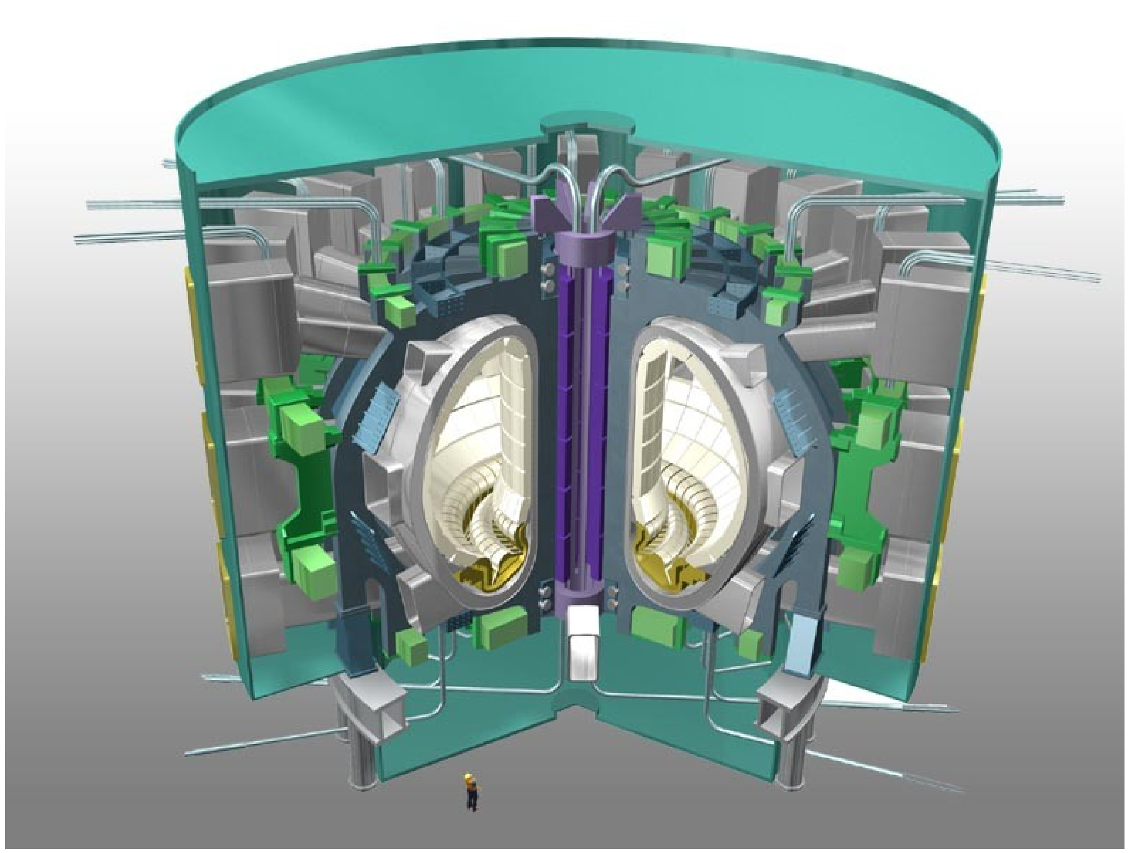

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Table of Contents

ITER, a collaboration of the world-wide fusion community, will produce the first magnetically confined plasma which is dominantly heated by fusion, opening the door to investigations of the burning plasma regime for magnetized plasmas. The overall goal for ITER is “to demonstrate the scientific and technological feasibility of fusion power for peaceful purposes”. During the ITER construction phase, it will be important for the US program to plan carefully for our ITER participation, to help ensure the success of the overall ITER program and to maximize our utilization and gain the most science that we can from ITER. In this context, this report attempts to answer three questions:

1. *What is the US research agenda for ITER?*
2. *How will ITER promote progress toward making fusion a reliable and affordable source of power, and how should this progress be assessed?*
3. *How does ITER relate to other elements of the US Fusion Energy Sciences program?*

The report is organized by six scientific themes: A) plasma macrostability; B) waves and energetic particles; C) multi-scale transport; D) plasma boundary interfaces; E) fusion engineering science; and F) integrated burning plasma science. Responses to the three questions for each theme are summarized here.

A. Macrostability

1. High level objectives in the macrostability science area are: to maximize fusion performance through understanding and control of instabilities that can limit plasma pressure or energy confinement in a burning plasma; to develop the scientific basis to understand and control instabilities that could limit the lifetimes of plasma facing materials.

2. ITER will access unique plasma regimes, not accessible in any present day device. With regard to stability, the most important of these are: the large Lundquist number, the small normalized gyro-radius, the significant, isotropic high energy ion population, and the significant effects of self-heating on the plasma pressure and current density profiles.

3. Existing magnetic confinement devices allow development and testing of the fundamental models for stability, which will ultimately be validated on ITER. They provide versatile, cost effective test beds for assessing new theories, control methods and stability issues. The theory and modeling program provides models of stability and control for the burning plasma regime, and will provide the necessary links from ITER to a fusion power plant.

B. Waves and Energetic Particles

1. Key ITER goals include: achieving high levels of performance of plasma heating and current drive systems essential to obtaining and maintaining high-gain plasmas; evaluating the technology of reliable RF antennas located at the edge of burning plasmas, to achieve reliable coupling and performance over long durations; validating the physics understanding plasma heating and current drive in fusion plasmas for multiple RF techniques and for NBI heating; validating self-consistent models for the dominant

effects of the fusion alpha particle population, including heating, instability drive, and fast particle transport.

2. ITER is the only planned fusion facility with dominant self-heating, offering the first opportunity to test actual alpha-particle production, transport and heating rates. ITER is uniquely positioned to evaluate the self-consistent non-linear effects of Alfvén wave turbulence on the alpha particle distribution. RF power will provide the majority of core heating required to initiate the fusion burn, and will be required to stabilize the ITER plasmas to access high performance regimes.

3. Highest priorities for US research in plasma wave heating and current drive for ITER include: improving understanding of RF coupling for Ion Cyclotron Heating (ICH) and Electron Cyclotron Heating (ECH); developing integrated models of RF propagation, absorption and energetic particle interaction; assessing auxiliary current drive and current profile control using waves in the Lower Hybrid range of frequencies; evaluating ECH/ECCD for stabilization of current-driven instabilities. For energetic particle science, high priority research includes: determining requirements and options for Alfvén wave diagnostics; understanding levels and effects of prompt loss of high energy alphas caused by magnetic ripple and by Alfvénic instabilities; evaluating the overall stability of ITER plasmas with energetic alphas, neutral beam ions, and RF-heated ions.

C. Multi-Scale Transport

1. Key objectives in transport research include: making the first studies of confinement properties in reactor-scale plasmas, across all accessible operational regimes, including baseline H-mode, small/no Edge Localized Mode (ELM) regimes, and advanced hybrid and non-inductive regimes; evaluating the non-linear effects on confinement and stability as the plasmas become predominantly self-heated; comparing turbulence and transport in reactor-scale plasmas with first-principles simulations; evaluating thermal stability at high Q.

2. Measures of success in resolving transport issues on ITER include: demonstrating that potentially detrimental transport effects which may arise in the burning plasma regime can either be avoided or controlled; demonstrating control of plasma conditions and fusion power production in a confined, burning plasma; demonstrating that first-principles computational models are capable of accurately predicting/reproducing the behavior of burning plasmas.

3. The development of experimental and diagnostic techniques, and of plasma scenarios in existing facilities, will be applied to ITER to improve prospects for success. The results will motivate new experimental studies in existing facilities to prepare for ITER operation. Critical transport related investigations that will advance during ITER construction include: transport in plasmas with dominant electron heating; transport barrier trigger mechanisms and evolution; bulk plasma rotation in the absence of external torque drives; verification and validation of plasma turbulence simulations.

D. Plasma-Boundary Interfaces

1. Key objectives in plasma boundary research for ITER include: developing models of the edge transport barrier region (the pedestal), benchmarked against existing

experimental data; developing, implementing and testing ELM control strategies; measuring and modeling Scrape Off Layer (SOL) parameters, including power widths at the midplane and divertor, and extrapolating to ITER; characterizing plasma wall interactions to guide the selection of plasma facing components for ITER; assessing plasma facing component performance on ITER.

2. ITER will provide a unique combination of low collisionality and high density reactor conditions in the H-mode pedestal and scrape-off layer, providing critical tests of transport and turbulence understanding. The ITER SOL is a unique test bed for detachment physics, with high opacity for both neutrals and radiation. ITER will provide important tests of our understanding of tritium retention and removal.

3. Near-term priorities for US plasma boundary research which contribute to ITER include: understanding pedestal structure and dynamics and ELMs, including the development and application of 2D transport and 3D MHD simulations; improving models for divertor detachment, and through iterative comparison with experiment, developing a predictive capability for the ITER scrape-off layer; comparing erosion, deposition and power handling for different candidate wall materials, including carbon and refractory metals.

E. Fusion Engineering Sciences

1. Priority US areas for fusion technology development from ITER include: valaidating the performance of power-plant scale superconducting magnets; developing and assessing steady-state actively cooled heating and current drive systems, operating in a high flux neutron environment; developing of fueling, pumping and fuel processing/tritium technologies; developing and applying of tools for real-time control of plasma parameters, including density, pressure, and current density, in the presence of strong self-heating with a significant population of high energy alphas; developing plasma diagnostics for the harsh neutron environment; developing nuclear monitoring and safety systems that will be needed for licensing of future fusion devices; evaluating the effects of tritium-breeding blanket modules on plasma operations.

2. ITER will provide a unique high neutron flux test-bed for assessing fusion technologies. Plasma support technologies developed for ITER should be directly applicable to future fusion devices including power plants. Information obtained during the construction, commissioning and operational phases of ITER will allow further optimization.

3. The majority of the engineering resources in the US Fusion Energy Sciences program are currently focused on developing and fielding plasma support technologies and plasma-facing components for ITER. Targeted testing of components, such as plasma facing divertor components, is carried out on the existing facilities. Diagnostic development is a significant element of the US program, and many of these developments have direct applicability to ITER.

F. Integrated Burning Plasma Science

1. The primary goal for US participation in ITER is to understand integrated burning plasmas which will require: producing and studying non-transient (pulse length >>

energy confinement time) high Q ($Q \geq 10$) D-T plasmas; accessing and optimizing advanced quasi-steady-state scenarios, including partially inductive hybrid and fully non-inductive scenarios; developing alternative operational scenarios which could lead to smaller or less technically demanding DEMO devices. Key issues involved in reaching these goals will include: accessing the H-mode in initial hydrogen and/or helium plasmas; the physics of confinement physics at a scale characterized by small ion gyro-radius normalized to machine size; understanding stability and plasma-wall interactions under burning conditions.

2. ITER is the only planned magnetic confinement experiment capable of producing high fusion gain, and thus represents a major step toward making fusion a practical energy source. Success on ITER will provide specific design information and confidence in key extrapolations to proceed to a next-step DEMO facility.

3. A major long-term goal of the US fusion energy sciences research program is development of a validated, comprehensive simulation capability. During the ITER construction phase, progress toward this goal will include integration of simulation capabilities through the Fusion Simulation Project to provide predictions for ITER which will ultimately be validated against ITER experimental results. This effort should contribute strongly to the ITER Organization's plans to develop integrated modeling capabilities during the ITER construction phase with the help of the ITER members' research programs.

Many important topics will be the subject of active research on existing and new tokamaks during ITER construction including: investigating hybrid scenarios and fully non-inductive scenarios with significant bootstrap current; developing methods and predictive capability for real-time current profile control using heating and current drive actuators; developing in medium-scale tokamaks reliable disruption avoidance and/or mitigation schemes and control of potentially adverse plasma-wall interactions.

The greatest lasting value of the ITER project will be the scientific knowledge gained, to help provide the predictive capability needed to design practical fusion reactors.

Chapter 1: Introduction

ITER represents the next frontier in the science of magnetic confinement plasma fusion, opening the door to the first investigations of the burning plasma regime for magnetized plasmas. The ITER project engages the world-wide fusion community, to build a large, deuterium-tritium burning tokamak which, in its first stage of operation, is planned to access plasmas with scientific Q, the ratio of fusion power to plasma heating power, of at least 10, for pulse lengths of 300 to 500 seconds. Under these conditions, the fusion power production will reach 500 million watts, and 2/3 of the plasma heating will be due to the fast alpha particles produced by the fusion reactions themselves. In a planned second stage of operation, ITER should explore quasi-steady-state advanced scenarios, with Q of at least 5, and pulse lengths of several thousand seconds. The overall goal for ITER is "to demonstrate the scientific and technological feasibility of fusion power for peaceful purposes".[1]

The ITER tokamak will be constructed in France, at the Cadarache nuclear research center, near Marseille. A conceptual rendering of the design is shown in figure 1.1. Seven major parties are participating in this joint international science project: U.S.A., the People's Republic of China, the European Union (represented by Euratom), India, Japan, the Republic of Korea and the Russian Federation. As the host, Euratom has agreed to support about 46% of the cost, while each of the other parties is committed to share equally in the balance. The international agreement for ITER was signed in Paris, France, on November 21, 2006.[3]

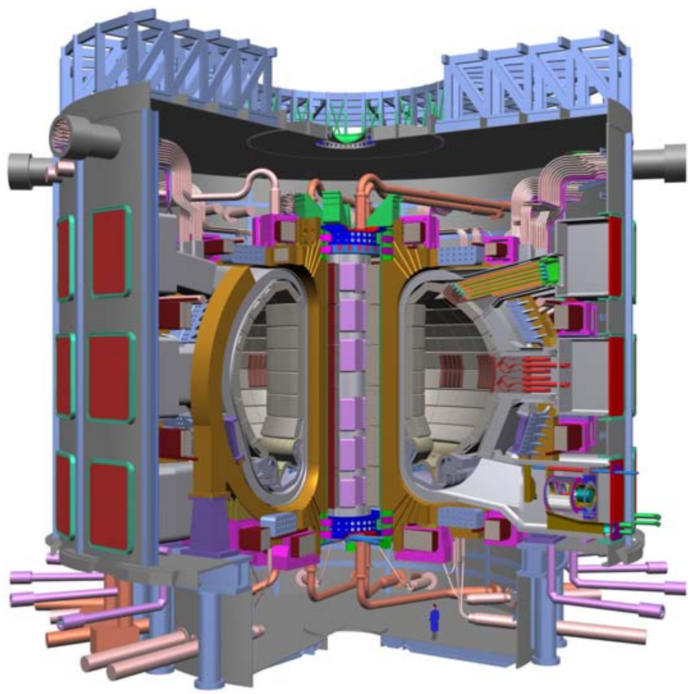

Figure 1.1. Artist's rendering of a cutaway view of the ITER tokamak.[2]

Responsibility for coordinating US scientific planning for ITER resides with the US Burning Plasma Organization (BPO).[4] An initial report on US planning for ITER research, *Planning for U.S. Fusion Community Participation in the ITER Program*, solicited by the DoE Office of Fusion Energies Sciences (OFES), was published by the

[1] ITER Objectives: www.iter.org/a/index_nav_1.htm
[2] www.iter.org/pics/iter8-high.jpg
[3] www.science.doe.gov/ofes/
[4] http://burningplasma.org/home.html

BPO in June, 2006.[5] As a follow-on to that activity, the BPO Council, again in concert with OFES, requested an updated study on the same topic. This report is the result of that request, and should be considered a current snap-shot of thinking within the US magnetic fusion community about those plans. We fully expect that these plans will be tuned and modified through the course of ITER construction, and then continue to evolve during ITER operation. As such, it can be expected that updates to this plan will be appropriate, perhaps with a cycle time of about 3 years.

By learning the science and technology lessons from ITER, and in parallel with its other research efforts, the US should be well placed to pursue the next step in pre-commercialization of fusion energy, a demonstration fusion power plant (DEMO), and be competitive with the world in ultimately producing fusion power plants. First operation of ITER is expected to begin after about a 10 year construction period; during this time, it will be important for the US program under OFES to plan carefully for our ITER participation, in order to maximize our utilization and gain the most science that we can from ITER. At the same time, we recognize that US efforts in Fusion Energy Sciences are imbedded in an ongoing large world program, not just on ITER, but also on broader related activities. A recently completed Fusion Energy Sciences Advisory Committee (FESAC) study, *Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan For Magnetic Fusion Energy*[6] spurred in part by US participation in ITER, has examined in detail the science and technology challenges that must be addressed on the path to prepare for design of a DEMO, including those that should be resolved through research on ITER.

A vigorous domestic Fusion Energy Sciences program is essential to create and maintain the next generations of plasma scientists. These are the people who will have the breadth of experience to reap the full scientific benefit from ITER, and exploit it in the development of fusion as a power source.

We have organized this report by six scientific themes: plasma macrostability; waves and energetic particles; multi-scale transport; plasma boundary interfaces; fusion engineering science; and integrated burning plasma science. For each theme, we have attempted to answer three questions:

1. What is the US research agenda for ITER?
2. How will ITER promote progress toward making fusion a reliable and affordable source of power, and how should this progress be assessed?
3. How does ITER relate to other elements of the US Fusion Energy Sciences program?

[5] http://burningplasma.org/ref/fp/EPAct_final_June09.pdf
[6] http://www.science.doe.gov/ofes/FESAC/Oct-2007/FESAC_Planning_Report.pdf

Chapter 2: Macrostability

Plasma stability is a critical issue for ITER or any other burning plasma device. The driving force for plasma instabilities originates in the plasma thermal energy, the magnetic energy associated with electrical currents in the plasma, and populations of very energetic particles such as the products of fusion reactions. Therefore, the stability of the plasma is intimately connected to its fusion performance.

Stable operation is crucial for reliable fusion power production. The consequences of instabilities include degradation of fusion power production, localized heating or electromagnetic forces on components near the plasma, and sudden termination of the hot plasma. Limits to plasma stability determine the maximum pressure of the confined plasma, and hence the fusion power that can be achieved. In addition, large-scale instabilities have the potential to limit the lifetime of the material surfaces facing the plasma. The stability properties of the plasma depend sensitively on its magnetic configuration and internal distribution of pressure and current density, so a detailed physical understanding is vital.

The 2005 FESAC Priorities Panel report posed a series of key questions for fusion science research, and the US goals for ITER research are well aligned with the three topical questions related to the Macroscopic Plasma Physics campaign:

T1. How does magnetic field structure impact fusion plasma confinement?

T2. What limits the maximum pressure that can be achieved in laboratory plasmas?

T3. How can external control and plasma self-organization be used to improve fusion performance?

These questions are the subjects of research in present devices, and will be further investigated in ITER with its unique features as the first magnetically confined burning plasma.

2.1 The US agenda for ITER in macrostability research

A long-term goal of ITER research is to establish the scientific and technical basis for understanding, prediction, and control of macroscopic stability in a burning plasma. The lasting value of ITER will be the scientific knowledge gained, which will not only add to the fundamental understanding of matter in the high-temperature plasma state but will also provide the predictive capability required to design future devices for producing fusion power. This knowledge must therefore take the form of validated theoretical models for the stability limits of the plasma, the consequences of the instabilities that occur at those limits, and control techniques for the avoidance, suppression, and mitigation of instabilities.

A specific objective is to maximize fusion performance through understanding and control of instabilities that limit the plasma pressure or energy confinement in a burning plasma. In a burning deuterium-tritium plasma, the fusion power increases rapidly with the plasma pressure; therefore improvement of pressure limits through control of the plasma configuration or direct suppression of instabilities translates directly into a gain in fusion power. Similarly, reduction of instability-driven loss of thermal energy and of fast ions produced by fusion reactions improves the ability of the plasma to heat itself and to retain heat.

A second, closely related objective is to develop the scientific basis to understand and control instabilities that may limit the lifetimes of materials that face the plasma. Instabilities driven by the strong gradients in pressure and electric current density that arise at the surface of the plasma can eject periodic, intense bursts of energy from the plasma edge, leading to erosion of plasma-facing surfaces. Global, long-wavelength instabilities may lead to a disruption of the plasma, accompanied by a large heat pulse to the wall and strong electromagnetic forces on nearby structures.

Fusion research over the last three decades has been very successful in understanding and predicting the instabilities described above, using increasingly sophisticated computer codes based on well-established magnetohydrodynamic (MHD) theory. Guided by these theories, present facilities have recently made much progress in controlling the plasma stability. Stability is maintained through modification of the confining magnetic field structure and the plasma's internal distribution of pressure, current density and rotation velocity, in order to avoid unstable conditions. Techniques also exist for direct suppression of some instabilities. In addition, results are promising in work toward mitigating the effects of disruptions, including heat loads, eddy-current forces on structures, and runaway electron production, when they do occur.

However, all of these techniques remain to be validated in a burning plasma environment. As discussed below, a burning plasma such as ITER differs in several important aspects from any existing fusion experiment. Therefore, an essential step on the way to a fusion power plant is to test our theoretical understanding and practical control techniques for plasma stability in an actual burning plasma. An additional challenge is to ensure that stability control techniques and plasma configurations optimized for high pressure are also consistent with steady-state operation.

2.2 The role and assessment of ITER in promoting progress in macrostability research toward making fusion a reliable and affordable source of power

ITER represents our first opportunity to investigate the physics of plasma stability in the environment of a burning plasma. Several important characteristics of ITER plasmas are unique to ITER. Some of these unique features result from its size, two to four times larger than any of the world's present "large" tokamaks. Others result from the fact that much of the plasma heating will be generated internally by fusion reactions, rather than from outside sources as in present tokamaks. These features cannot be duplicated in existing facilities.

- **Large Lundquist number**. The Lundquist number is a measure of how closely the plasma behaves like a fluid with perfect electrical conductivity; this distinction affects both the behavior of plasma instabilities and the choice of theoretical models. The Lundquist number in ITER will be at least ten times larger than in existing devices. This difference is potentially favorable for plasma stability in ITER, as a large Lundquist number generally results in slower growth of instabilities that require electrical resistivity in the plasma. In particular, the tendency of one instability to trigger another may be reduced.

- **Small ratio of ion gyration orbit size to plasma size**. In a magnetically confined plasma, the hot fuel ions move in small circular orbits. ITER's large size and strong magnetic field mean that these orbits will be smaller, in comparison to the size of the plasma, than in most existing devices. This difference is potentially unfavorable for ITER's stability, since the "averaging" effect of larger ion orbits tends to reduce instability.
- **Large, isotropic population of high-energy ions.** Fusion reactions in a burning deuterium-tritium plasma produce high-energy ions (alpha particles), which are trapped by the magnetic field and transfer their heat to the rest of the plasma. Existing fusion experiments are also often heated by energetic ions, accelerated in the plasma by electromagnetic waves or injected from outside. The high-energy ions in ITER will have energies more than ten times greater than those used to heat existing devices, allowing them to readily excite certain classes of high-frequency instabilities. Their orbits will also have a much more uniform distribution of directions than those created in existing devices, potentially altering their coupling to various instabilities. ITER's energetic ions are expected to help stabilize the "sawtooth" instability in the plasma core, an effect that may be favorable or unfavorable depending on whether the instability is eliminated or simply postponed.
- **Pressure profile largely determined by self-heating**. Much of the heating of ITER's plasmas will be generated internally by fusion reactions, in strong contrast to present externally heated plasmas. The pressure rise associated with this heating is favorable for fusion, but can also lead to instability if local pressure gradients within the plasma become too large. Operational control of the rate and location of heating will be much weaker in a burning plasma than in an externally heated plasma, posing challenges for maintaining stable operation that do not exist in present devices.
- **Current density profile partially determined by self-heating**. The pressure gradients in a high-temperature tokamak plasma generate a significant amount of electrical current within the plasma, which can alter the plasma's stability properties. As with the pressure profile, operational control of the internally generated current will be weaker in a burning plasma, posing challenges for maintaining stable operation that are greater than those in present devices.

The last two items are emblematic of the fact that a burning plasma will be more self-organized than most existing experiments. The pressure and current distributions will tend to evolve to a state that is determined to a significant extent by the fusion heating and the internal transport processes and less by external control (the plasma shape, and the small remaining amount of external heating and current drive). This tendency to self organization will probably be significant in ITER's baseline scenario, and will become very important in the steady-state scenario where higher pressure and a higher proportion of self-generated current are planned. Reliable operation of a burning plasma requires guiding that self-organized state to a configuration that is favorable for both power output and stability. ITER will be our first opportunity to learn to meet this challenge.

ITER also provides the first opportunity to test, in a realistic environment, the backup control systems needed for stable operation of a burning plasma. Reliable operation requires the capability to detect impending instabilities and take appropriate action to prevent them from growing. Such actions may include reduction of the plasma pressure, application of localized heating or current drive at the proper location in the plasma, direct magnetic control to oppose the instability, or a controlled shutdown of the plasma. The unique features of a burning plasma described above tend to reduce the effectiveness of external control. Therefore the detection and control systems for a fusion plasma will need to be based on well-validated theoretical models and carefully tested in ITER.

In summary, stable operation is essential in a power plant that must maintain a high fusion power output with maximum fusion power gain and minimum down time. ITER represents a critical stepping stone where the physics understanding and control techniques to ensure stability can be tested in a true fusion plasma.

Metrics for progress will consist of a series of steps to test both the scientific understanding of macroscopic stability in a burning plasma, and the application of that understanding to achieve reliable, stable operation in ITER. Initial steps will validate the underlying models of plasma stability in ITER's environment of large Lundquist number and small ion orbit size. Later steps will progress to demonstrations of stable operation with high fusion power. Completion of these steps will show that the scientific understanding has been achieved to allow the confident design and operation of a prototype power plant (DEMO). The necessary steps include:

- Validating models for stability limits and the evolution of instabilities in the absence of fusion heating, during ITER's initial phase of operation.
- Demonstrating reliable, real-time prediction and control of macroscopic instabilities in plasmas with moderate to high externally applied heating power.
- Validating models for stability limits and the evolution of instabilities in plasmas with moderate fusion heating, particularly the effect of fusion alpha particles.
- Demonstrating the capability of reliable, real-time prediction and control of macroscopic instabilities in a burning plasma.
- Achieving stable, disruption-free operation with high fusion gain in ITER's baseline scenario.
- Achieving stable, disruption-free operation with high fusion gain and a high degree of plasma self-organization in ITER's steady-state scenario.
- Demonstrating the validity of scientific modeling of MHD stability of burning plasma over the full range of ITER operational scenarios, including baseline and advanced scenarios, to inform future design of DEMO.

2.3 The relationship of ITER to the US Fusion Energy Sciences program in macrostability research

The other elements of the US Fusion Energy Sciences program will have crucial roles, not only in the preparation for ITER, but also in supporting ITER operation and in preparing for the steps beyond ITER. In general, the other elements of the US program will provide the flexibility to serve as prototypes for specific control methods or operating scenarios for ITER, the creativity to develop and test solutions for ITER beyond those now envisioned, the versatility to develop plasma configurations and operating scenarios for devices to follow ITER, and the fundamental plasma science that underlies ITER and all other fusion research.

- **Existing magnetic confinement devices**, including tokamaks, stellarators, and other innovative concepts, will allow development and testing of the fundamental theoretical models for plasma stability which will then be validated in the burning plasma regime by ITER.
- **Existing tokamaks** will allow testing of model-based techniques for prediction and control of plasma stability with direct applicability to ITER, but in more flexible and less costly operating environments.
- S**maller devices** (existing and future) will continue to provide versatile test beds for assessment of new theories, new stability control methods, and new stability issues that may be encountered by ITER. Smaller devices allow this to be done in a rapid, cost-effective, and low-consequence manner, before the results are applied to ITER.
- **Stellarator research** will clarify the benefits and costs of 3D plasma shaping, in parallel with ITER's operation. That knowledge, combined with the advances from ITER and other tokamaks, will influence the direction of fusion research toward DEMO.
- **The theory and modeling program** will provide models of stability and control in a burning plasma, to be validated in both ITER and smaller devices. These validated models are the ultimate product of US macrostability research in ITER, and provide the necessary link from ITER and the other elements of the US program to a fusion power plant.

Chapter 3: Waves and Energetic Particles

Magnetically confined plasmas support a wide variety of electromagnetic waves which can transport energy through the plasma and can interact with specific populations of plasma particles through resonances with particle motions. The motion of particles in the magnetic field can also create plasma waves which can then modify the particle orbits, affecting their confinement. Plasma waves also provide powerful ways to diagnose the characteristics of the plasma in which they propagate.

Powerful electromagnetic waves at radio frequencies (RF) will be coupled to the ITER plasma to heat the plasma to fusion temperatures. Electromagnetic waves can also be applied to maintain the electric current that must flow within the tokamak plasma for it to operate continuously. Lastly, application of RF waves can be used to control or suppress instabilities arising from unfavorable distributions of the current or pressure within the plasma. The necessary waves of different frequencies will be generated by high-power RF sources, transported to the ITER vessel through specialized transmission lines, and coupled to the plasma with antennas located near the plasma boundary. The high-power waves are absorbed by the electrons and/or ions in the ITER plasma, depending on their frequency.

Populations of energetic particles created by external sources or by fusion reactions in the plasma can heat the plasma but can also provide the free energy to drive plasma instabilities. In ITER, energetic particles in the form of directed beams of neutral hydrogen atoms will also be injected (neutral beam injection – NBI), to heat the plasma. NBI will also drive plasma current and plasma rotation.

The heating systems to be used on ITER are summarized in Table 3.1.

Table 3.1 Planned auxiliary heating systems for ITER

Type	Key purpose	Power (MW)	Availability
Electron cyclotron heating (ECH) at 170 GHz	Initial plasma formation	2	Desirable for Stage 1
Ion cyclotron heating (ICH) at 40 – 55 MHz	Core plasma heating to fusion temperature	20	Essential for Stage 1
Electron cyclotron heating (ECH) at 170 GHz	Heating and localized current drive for instability control	20	Essential for Stage 1
		20	Possible upgrade
Lower hybrid current drive (LHCD) at 5 GHz	Current drive for sustained operation	20	Possible upgrade
Neutral beam injection (NBI) at 0.5 – 1 MeV	Plasma heating and current drive	33	Essential for Stage 1
		16	Possible upgrade

Electron cyclotron heating makes use of the gyro-motion of electrons in the applied magnetic field of the tokamak. The gyro-frequency is proportional to the magnetic field strength (28 GHz per Tesla). Electrons in resonance with the launched wave can absorb the wave energy. This will occur in ITER when electron cyclotron waves at a frequency of 120 or 170 GHz illuminate a region of the plasma with a local magnetic field strength of 4.3 or 6.1 T. Electron cyclotron heating will generate and heat the ITER plasma discharges during their initial period of formation.

The resonance of the wave with the electron motion is also influenced by relativistic and Doppler effects. The latter enables electric current to be driven by electron cyclotron waves at a location in the plasma determined by the local magnetic field. Precise manipulation of the currents in the ITER plasma by electron cyclotron waves launched with steerable antennas will be applied to suppress electromagnetic instabilities known as neoclassical tearing modes which, if left unchecked, could lead to a major disruption and terminate the plasma discharge.

In a similar manner, lower frequency waves are launched to heat the ions of the plasmas, which, due to their heavier mass, exhibit a lower cyclotron frequency, typically in the range 10 – 100 MHz, than electrons in a given magnetic field. The physics of ion cyclotron wave heating is more complex than that for electron cyclotron waves, but previous work has demonstrated that both ions and electrons can be efficiently heated by this approach. Waves in this frequency range can also be used for limited tailoring of the current density profile, with possible application to advanced scenarios. The experimental results are backed by sophisticated calculations of ion cyclotron wave propagation and absorption. Figure 3.1 shows a simulation of the of the ion cyclotron wave propagation in

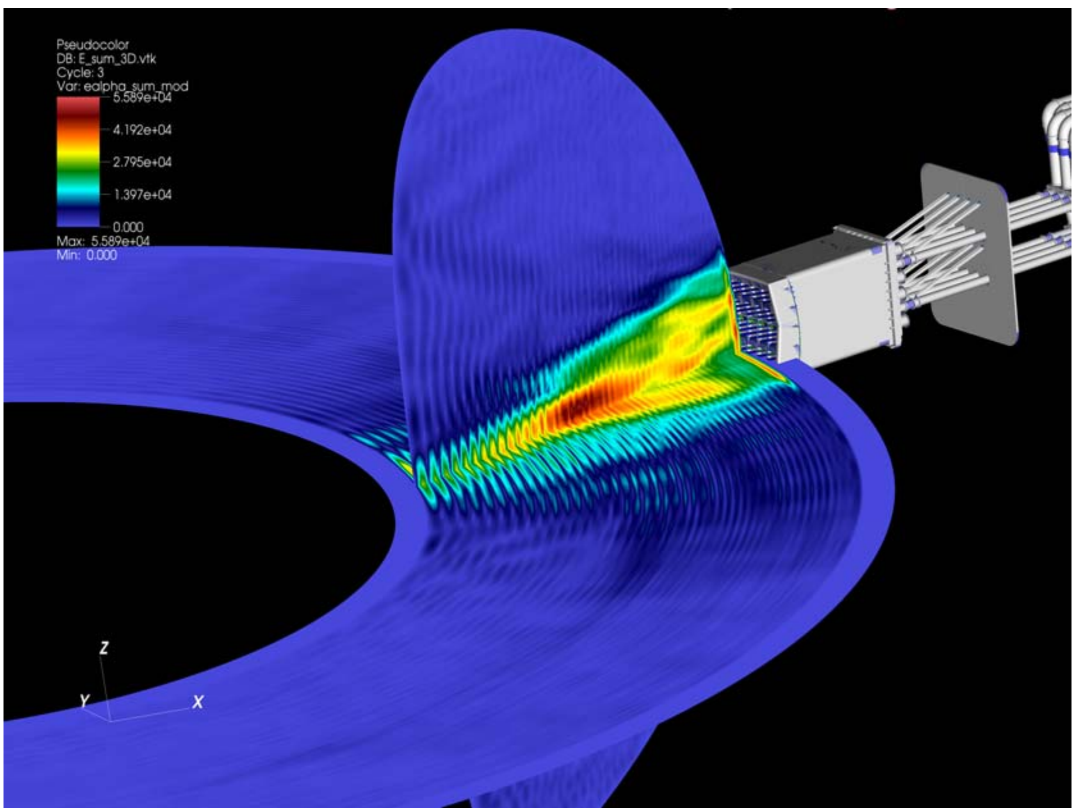

Fig. 3.1 Simulation with the AORSA code of ion cyclotron wave fields in ITER launched by antenna at right. The antenna is 2 m high.

ITER. As with electron cyclotron heating, this method allows good control of where heating by the wave takes place in the plasma. In particular, the frequency of the ion cyclotron waves may be chosen so as to directly heat the core of the ITER plasma.

ITER, like any tokamak, requires a substantial toroidal electric current to flow through the plasma to contain the plasma discharge. This plasma current may be generated transiently by induction, but with this method the pulse duration is inherently limited, so alternative non-inductive methods for current drive are needed to achieve steady-state operation. Although many wave-particle interactions in a plasma can be used to generate electric current, the most efficient means of driving current by RF methods is with waves in the lower-hybrid range of frequencies (intermediate between the electron and ion cyclotron frequencies). This technique is likely to be required in ITER both to sustain plasmas for long durations and to control the current density in the outer half of the plasmas for achieving advanced performance confinement scenarios. Tokamak concepts for DEMO and power-producing reactors assume operation in these advanced regimes.

Injection of neutral beams of energetic hydrogenic atoms has been the workhorse for heating high-performance toroidal plasmas during the last three decades. Through collisions with the plasma ions and electrons, the injected atoms are ionized, and give their energy to the background plasma. The neutral beams may also be used to generate net electric current and plasma flows. Both these ancillary benefits of NBI are routinely exploited to improve tokamak plasma confinement and stability. Because of its much larger size, ITER plans to use neutral beams of much higher energy, up to 1 MeV, for plasma heating and, even then, only a relatively small fraction of the injected beam power reaches the plasma core. This fact alone necessitates the simultaneous application of RF wave heating to heat the central plasma region to ignition temperature.

The primary scientific advance to be made by ITER beyond existing experiments is access to the burning plasma state in which the deuterium-tritium fusion reactions generate a substantial population in the plasma of very energetic, charged alpha-particles (helium nuclei). For ITER to achieve its goals as a self-heated fusion experiment, it is essential that the alpha particles remain within the plasma until most of their energy is transferred by collisions to the electrons and fuel ions in the plasma, in the same manner by which energetic neutral beams heat the plasma. Because the fusion-generated alpha particles are born with about 100 times the energy of the background plasma particles, they are likely to generate electromagnetic waves within the plasma by the inverse of the process in which injected waves are absorbed by plasma particles as described above. The most likely waves to be excited by energetic alpha-particles in ITER are the so-called Alfvén waves, with typical frequencies of hundreds of kHz. The computed wave amplitude structure of one typical Alfvén wave that could be excited in ITER is shown in figure 3.2. Generally, higher alpha particle pressures resulting from greater fusion reactivity are expected to lead to higher levels of Alfvén wave activity.

The turbulent spectrum of multiple Alfvén waves generated by the alpha particles is expected to disturb the regular confined orbits of the alpha particles within the ITER plasma to some extent. The significance of this is that the alpha particles can be distributed throughout the plasma by the action of the unstable waves more rapidly than simple collisional models would predict. The potential negative consequences are lower

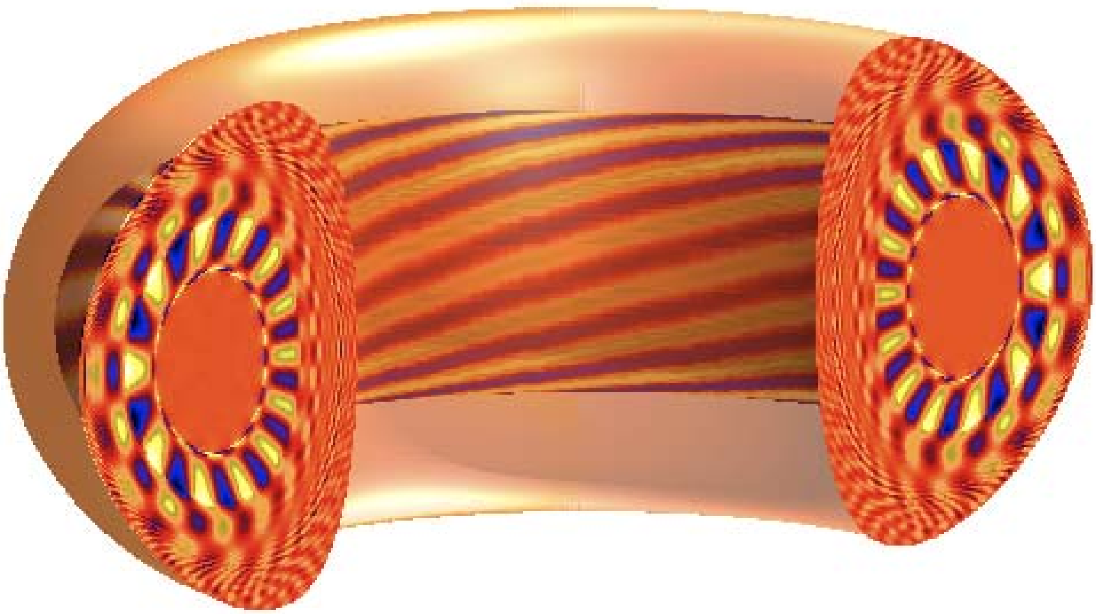

Fig. 3.2 Simulation[1] of the density perturbations associated with an Alfvén eigenmode in ITER with a toroidal mode number n = 10.

fusion gain due to reduced plasma heating by the alpha particles and, possibly, large localized losses of energetic alphas from the plasma capable of damaging the first wall and releasing undesired material into the plasma. The issue of excessive wall loading by prompt alpha particle losses is a high priority concern to ITER. Potential positive consequences of Alfvén wave activity are possible means of fusion burn regulation through active control of the relevant plasma instabilities, and increased fusion gain if the unstable waves damp predominantly on the ions of the background plasma rather than the electron population.

3.1 The US agenda for ITER in waves and energetic particles research

US researchers intend to gain from ITER the scientific knowledge and engineering experience necessary to proceed with the next steps towards eventual utilization of fusion power. This includes the development and experimental validation of physical models that will be increasingly relied upon to predict performance in a range of potential future facilities. Furthermore, a number of important technical assessments of the performance of key ITER systems will be made. In the context of this section, the key US goals are:

1. Achieving reliably high levels of performance of plasma heating and current drive systems essential for attaining and maintaining high-gain fusion plasmas.
2. Evaluating the need for maintenance, replacement, and further improvement of RF antennas located at the edge of fusion plasmas. The physical structures of ICH and LH antennas are required to be relatively far (≈10 cm) from the plasma boundary in ITER because of the high heat flux and nuclear environment. Achieving satisfactory coupling and performance under these conditions over the long durations of many ITER discharges is a critical US technical goal in this area.

3. Validating physics understanding of plasma heating and current drive in fusion plasmas by ECH, ICH, and LH RF and the NBI heating systems to a level sufficient for selection and design of DEMO. This validation would be achieved typically by comparing experiments with the results from multi-scale, linked computer codes based on first-principles physical models.
4. Validating self-consistent physical models for alpha particle heating and alpha particle loss rates to the walls in the presence of alpha-driven as well as other MHD instabilities.
5. Evaluation of sophisticated burn control by manipulation of Alfvén turbulence.

Regardless of which measurement systems the US is responsible for fielding on ITER, achieving these goals requires the successful implementation and operation of appropriate instrumentation to measure wave activity (launched and intrinsic) and correlated energetic particle behavior sufficient for rigorous testing and validation of physical models.

3.2 The role and assessment of ITER in promoting progress in waves and energetic particles research toward making fusion a reliable and affordable source of power

Results from the ITER experiment will foster advancement toward the goal of fusion energy in numerous ways. ITER is the only planned fusion facility with a sufficient level of fusion-produced alpha particles to provide majority self-heating. It offers the first opportunity to test actual alpha-particle production, diffusion, and plasma heating rates in reactor-like conditions. It is the only experiment in which the scale and impact of prompt alpha losses to the wall can be evaluated. Related to this, ITER will be uniquely positioned to evaluate the effect of self-consistent Alfvén wave turbulence on the distribution of alpha particles in the plasma in a range of operating scenarios. This knowledge is crucial to inform the next steps in burning plasma research, particularly with regard to magnetic configuration, parameters, and first wall/divertor materials.

The assessment of plasma heating and control by RF waves in ITER is key to determining the reliability and affordability of fusion. RF power will provide the major fraction of heat delivered to the core of the plasma to initiate and maintain the fusion burn. Knowledge of the actual efficiency and reliability of the RF systems to perform this task in the harsh fusion environment can only be obtained on ITER. In addition, RF power will be required to reliably stabilize the ITER plasma and allow it to access advanced performance scenarios with high fusion gain and near-steady-state operation through RF current-drive. The application of auxiliary power, including RF, determines the overall efficiency of a fusion power plant, and it is crucial to evaluate its multi-faceted performance on ITER to assess plans and opportunities for next-step fusion facilities.

Metrics in this area can be associated with the programmatic goals of ITER, including attaining a fusion gain Q=10, scientific accomplishments, such as successful predictive modeling of ICH wave propagation and absorption in ITER plasmas, and technical milestones, including the ability to couple a specific level of power into the plasmas. US metrics should also specifically include successful implementation and performance of the hardware contributions for which the US is responsible in the ITER agreement. In the area of waves and energetic particles, the current US plans include ICH and ECH transmission lines.

Metrics:

1. The degree and depth to which first principle-based scientific models, as documented in the scientific literature and other means, can accurately represent the behavior in ITER plasmas of:
   a) ICH wave propagation and heating;
   b) ICH antenna coupling, RF sheaths, and impurity generation;
   c) ECH-driven MHD instability suppression;
   d) Excitation of Alfvén wave spectra by fusion-generated alpha particles and ICH-generated energetic ions;
   e) Prompt alpha loss rates and location at a level that wall loading can be predicted;
   f) Fusion gain in presence of self-consistent level of Alfvén waves driven by energetic particles;
   g) Non-inductive current drive by ICH, ECH, NBI, and LH (if applied).
2. Achieving target levels of availability of key heating and current drive systems during various ITER program phases, consistent with maintenance needs.
3. Achieving prescribed power levels in all key heating systems.
4. Demonstrating that first wall power loading is acceptable in relation to concerns of instability driven prompt alpha losses.

3.34 The relationship of ITER to the US Fusion Energy Sciences program in waves and energetic particles research

The US fusion energy sciences program has a strong, long-term intellectual investment in RF heating, and can be considered among the leaders in ICH and ECH experiments and modeling. While the loss of TFTR in the mid-1990's ceded the opportunity for futher experimental studies of D-T generated energetic alpha particle behavior to JET, the US nonetheless fields a very strong effort in energetic particle-driven modes through modeling and studies of RF-driven and NBI fast particles on its existing devices. There is thus a very strong linkage of present US activities to the US goals for ITER, and a number of important preparatory activities for ITER and ITER-concurrent tasks are taking place.

In recent BPO planning activities, US experts have identified the key priorities for US domestic research in both the topical areas of wave and energetic particles with regard to US scientific participation on ITER. It is important to recognize that in all of these activities described below, there is a strong interaction between ongoing experiments on existing US facilities and state-of-the-art modeling of the results. The resulting quantitative understanding of the behavior of waves and energetic particles in the fusion plasma environment will continue to be incorporated into integrated modeling to simulate the behavior of the entire fusion plasma. Ultimately, the validated findings of the ITER experiment will guide the next steps of the US domestic fusion research program.

For the next ten years, the ordered priorities for US research in plasma wave heating in relationship to ITER are:

1. Improved understanding of the efficiency and reliability of the coupling of the RF power in the ICH range to the plasma with antennas located near the plasma edge. This includes the conceptual and experimental development of advanced, robust RF antennas, making use of existing facilities as necessary, to address challenges of maintaining good coupling efficiency in the harsh fusion environment.
2. Development of integrated models of RF propagation, absorption and interactions with energetic particles, coupling to models of transport and stability, and validation by comparison with experiments.
3. Continued evaluation of the variety of ICH heating and current-drive scenarios expected to be tested on ITER.
4. Assessment of the effectiveness of auxiliary current drive and current profile control in ITER conditions with injected waves at the lower hybrid frequency.
5. Evaluation of the ECH power and wave coupling requirements for stabilization of potentially serious current-driven instabilities in ITER.
6. Evaluation of power deposition profiles for ECH heating and current drive in ITER.

In the area of energetic particles, the ordered priorities are:

1. Determining the requirements and options for implementing measurement systems for the Alfvén waves associated with the fusion-produced energetic alpha particle population.
2. Understanding the level and effect of prompt losses of alpha particles to the wall caused by inhomogeneities (ripple) in the strength of the confining magnetic field. Because this is dependent on the underlying magnetic configuration of the ITER tokamak, the effect could have a fundamental influence on the efficiency of alpha particle heating in a number of different operating scenarios planned for ITER. Due to its potential impact on the longevity and performance of the first wall, the understanding of ripple losses could be a broad and fruitful interdisciplinary activity with US materials scientists.
3. Continued modeling of the expected level, spatial extent, and type of Alfvén mode excitation by fusion alphas, to be linked with other codes for larger scale fusion plasma simulation efforts.
4. Computationally evaluating the overall stability of ITER plasmas in the presence of energetic alpha particles, neutral beam ions, and RF-heated ions.

5. Investigating the nonlinear dynamics of waves driven unstable by energetic particles. This includes measuring and understanding the spatial transport caused by the instabilities. A key goal is to identify saturation mechanisms, including wave-wave couplings in the presence of multiple unstable modes and modification of the energetic-particle population.
6. Exploring the means to control the effect of waves that are driven unstable by the energetic particles. This may include the launching of additional waves to alter the stability threshold of dangerous modes or to mitigate their effect by modifying the nonlinear dynamics. Related to this approach is the testing of ideas to exploit energetic particle instabilities to improve plasma performance.

Chapter 4: Multi-Scale Transport in the Burning Plasma Regime

The fundamental Lawson criterion for fusion shows that reaching and sustaining the burning plasma state in ITER depends on adequate confinement of the fusion fuel particles and heat, as well as confinement of the highly energetic fusion helium particles until they transfer most of their energy into the other plasma constituents. Once this fusion energy has been transferred, the thermalized helium ash must then be exhausted at a rate that is high enough to prevent excessive fuel dilution in the plasma. Since bulk plasma flows, and the radial derivatives (shear) in those flows, can directly influence both macroscopic stability and the turbulence which drives energy and particle transport, the way in which momentum is transported in the plasma is also very important. The study of these confinement processes (energy, particles and momentum) is what is generally referred to as transport.

These transport processes are controlled in part by the generation of turbulence with a spatial scale that is much smaller than the plasma, and which is driven by the average plasma pressure gradient that necessarily exists between the burning plasma core and the cooler plasma edge. This small-scale turbulence interacts with larger-scale MHD and wave-particle phenomena in the plasma, thereby forming a system with multiple nonlinear feedback mechanisms occurring on multiple spatial and temporal scales. ITER will provide the first opportunity to study these processes in the burning plasma regime, and thus is a unique scientific tool to further our understanding of transport.

Achievement of the burning plasma regime in ITER requires plasma operation with adequate thermal confinement in a regime with low ratio of ion gyro-radius-to-system size, which is the relevant dimensionless regime. Since no other tokamak is of comparable size, ITER provides the first opportunity to explore this confinement regime, which is necessary for any burning plasma tokamak device. Thus, transport and confinement studies will be important elements of research on ITER from the outset and will continue through all phases of the ITER program.

4.1 The US agenda for ITER in transport research

US scientists will use ITER to study the unique elements of burning plasma transport that cannot be studied in any existing experiment, and will then use the resulting scientific understanding to validate first-principles transport simulations which are currently in development. Such tools will then allow US scientists to project the transport in, and performance of, future fusion energy systems with greatly increased confidence. Achievement of this goal requires that a number of technical objectives be demonstrated, including:

1. Making the first studies of confinement properties in reactor-scaled plasmas in all accessible operational regimes, including investigating the dependence of transport on operational parameters (dimensional scaling) and plasma parameters (non-dimensional scaling) using the different auxiliary heating methods available.
2. Characterizing confinement in ITER as burning conditions are approached in a variety of modes of operation, including the “baseline” H-mode scenario, regimes

with more benign, small ELMs ("grassy", Type V, etc.), non-ELMing regimes (EHO, EDA, edge stochastic, L-mode), hybrid modes, and reversed-shear modes.

3. Learning if these confinement properties change or develop new characteristics when the plasma becomes strongly self-heated, and determining the confinement properties of the energetic fusion by-products as their contribution to the total plasma pressure becomes significant.
4. Comparing experimentally determined turbulence and transport in a reactor-scaled device with simulations based on micro-turbulence theory to test our fundamental physics understanding of confinement.
5. Studying, for the first time, the feedback processes linking transport and confinement, self-heating, and overall plasma stability in a reactor-regime.
6. Measuring and characterizing the plasma response to perturbations in burning plasma conditions to project thermal stability of high-Q or ignited DT plasmas.

There is a planned staging for ITER operations, leading in the first phase to full D-T operation at Q≥10, and then in the second phase to long-pulse operation at Q≥5. Along the way, key transport issues will be explored across a variety of plasma regimes, including baseline H-mode, regimes with benign ELMS, hybrid regimes and advanced tokamak (AT) scenarios which can extrapolate to true steady-state.

For experiments in the non-burning (Q<1) phase, these issues include:

- What are the transport and turbulence characteristics in reactor-scaled plasmas (large ratio of machine size to gyroradius, ρ^* scaling)? How large does a burning plasma have to be?
- How does transport behave in plasmas dominated by electron heating?
- Can transport in the ion and electron channels be separated and studied as functions of dimensional and non-dimensional plasma parameters?
- Under what conditions can H-mode edge transport barriers be created? What is the H-mode pedestal pressure in ITER and is it sufficient to achieve Q=10 in the baseline scenario?
- Can internal transport barriers be created and sustained in steady-state to produce plasmas extrapolating to high performance?
- What drives intrinsic rotation? Will intrinsic rotation be high enough in ITER to stabilize resistive wall mode instabilities?

For all of these issues, differences between operation in hydrogen and helium compared to expectations for deuterium/tritium will need to be explored and understood.

Moving to the burning plasma regime (Q~10), additional questions must be answered:

- How do we use our understanding gained from the Q<1 studies to achieve burning plasma conditions in ITER?
- What determines the transport and confinement of alpha particles from their birth energy until they thermalize with the background plasma or are lost?

- How will transport affect helium ash confinement and removal?
- Do new transport phenomena appear in the burning plasma regime? Candidates include:

  – New feedback mechanisms, such as fast-alpha driven Alfvén modes, which in turn can have non-linear effects on particle transport, radial electric fields, sheared flows, turbulence, energy transport, and ultimately fusion power;

  – Effects of transport on the thermal stability of burning plasmas as Q is increased well above 1.

Achievement of these objectives requires that ITER develop and implement plasma diagnostic systems to make the research possible. The US scientific community should carefully consider how the currently planned set of ITER instrumentation and diagnostics could be used or should be modified to enable the necessary measurements to be made. Access to burning plasmas in ITER will provide the first opportunity for transport simulation validations in this regime. Because of currently expected limitations, particularly in the turbulence diagnostic set, it is likely that these studies will have to focus on transport validation, rather than assessing the success of detailed simulations of the underlying turbulence.

4.2 The role and assessment of ITER in promoting progress in transport research toward making fusion a reliable and affordable source of power

The issues discussed above form a key set of scientific questions that must be answered in order to proceed beyond ITER into the design and operation of a device that can demonstrate fusion energy. These issues cannot be adequately studied in existing experiments alone. Key measures of success in resolving issues related to transport on ITER will include:

1. Demonstrating that there are no large, unanticipated and detrimental transport effects that arise as the self-heated burning plasma regime is approached.
2. Demonstrating that it is possible to control the plasma conditions and fusion power production in a confined, self-heated burning plasma using existing and planned control mechanisms and measurement techniques.
3. Demonstrating that first-principles computational tools being developed to simulate transport and turbulence phenomena in burning plasmas are capable of accurately reproducing the actual behavior of burning plasmas.

The first two metrics are necessary for the achievement of the ITER design goals, *including* achievement of an energy multiplication factor, Q = 10, with 500 MW of fusion power produced for 300 to 500 seconds, and will be addressed during the period leading up to the demonstration of these burning plasma conditions. The third will provide confidence in extrapolating the results to any fusion reactor.

4.3 The relationship of ITER to the US Fusion Energy Sciences program in transport research

Existing elements of the FES program contribute to many of the key transport research questions discussed above. However, there are two elements – transport in reactor-scale plasmas, and transport in strongly self-heated plasmas – that cannot be directly addressed in existing experiments. On the other hand, theory and computational research are focusing on all of these issues, and these efforts are both shaping, and being shaped by, on-going experiments. However, because of the complexity of the problem, present-day computational simulations are incapable of providing direct, first-principles burning plasma simulations that include all relevant physics and all relevant spatial and temporal scales, and it is unlikely that this will change prior to the beginning of ITER operations. Thus, in some areas, ITER experiments will provide the first glimpse of the scientific opportunities and challenges that will exist in an energy-producing fusion plasma.

Experimental and measurement techniques and plasma scenarios developed through independent research programs in existing facilities will be applied to ITER both to improve its prospects for success and to develop these techniques on a larger scale to be ready for incorporation in a DEMO device. These results will also aid in interpreting simulation results; theory and simulation will, in turn, motivate new experimental studies on existing devices in preparation for ITER operations. We can also expect that existing experiments, with their extensive diagnostic capabilities, including for plasma turbulence, will continue to play an important role with respect to validation of the numerical simulations.

Critical transport related investigations that will be ongoing during ITER construction include:

- Transport in electron dominated regimes.
- Identification of conditions and trigger mechanisms for transport barrier formation, including the H-mode and internal transport barrier formation and evolution.
- Identification and understanding of the mechanisms responsible for bulk plasma rotation in the absence of explicit core torque drives.
- Verification and validation of plasma turbulence simulations using separate core and edge models.

Once ITER operations commence, the lessons learned in smaller scaled experiments, in theory, and in simulation will then be incorporated in the operational plans and experimental activities undertaken within ITER.

Chapter 5: Plasma Boundary Interfaces in the Burning Plasma Regime

One of the fundamental challenges of magnetic fusion is to isolate the burning core plasma, which is at a very high temperature (~100 million Kelvin), from its "room temperature" (~300 Kelvin) surroundings. While the helical magnetic fields in the core of a tokamak form an invisible, but very effective, "magnetic container" that confines the charged particles of the plasma, if even only a small fraction of the plasma escapes, its interaction with the material walls can have very significant consequences. There is a narrow transition region between the burning core plasma and the material walls, and this interface is the subject of this Chapter.

In current tokamaks, the boundary interface region is narrow, typically only a few percent of the minor radius. In the core plasma, the helical magnetic field lines lie on closed, nested surfaces; this magnetic structure confines the charged particles. However, at the boundary, the magnetic field lines become open, intersecting with material surfaces. Through collisions and MHD activity, particles can be lost from the confined core region and are transported to the open field line region in the boundary. Here they (mostly) flow quickly along the open field lines towards the special, armored wall that is called a divertor. During this transit along the open field lines, some particles are also transported outwards to the main chamber walls before they reach the divertor. Both in the divertor, and at the main-chamber walls, the plasma interacts with the material surfaces through complex surface physics and chemistry. For example, close to the wall there are multi-step atomic processes such as ionization, and multi-species ion and neutral transport. All of these processes depend on the details of the composition and geometry of the material surfaces.

Present tokamak experiments have established that the performance of the core plasma depends sensitively on the boundary interface – both the plasma conditions in this region and the material walls must be carefully designed. Understanding the dynamic physical processes of the boundary with sufficient clarity to predict the behavior of a burning plasma experiment is one of the greatest challenges of fusion science. The understanding of this region requires a well-coordinated effort between simulation and experiment that covers temporal scales from microseconds (e.g., atomic processes) to seconds (e.g., wall interactions), and spatial scales from microns (e.g. atomic processes and surface effects) to meters (vacuum vessel). In the edge, plasma parameters vary over very short distances, giving rise to large electric fields and affecting plasma transport and MHD stability.

For purposes of this discussion, the plasma-boundary interface can be naturally divided into four regions, distinguished by unique physical processes. As shown in figure 5.1, moving outward from the core, the four zones are: 1) the pedestal region which is at the edge of the confined plasma with closed field lines, 2) the Scrape Off Layer (SOL), which is the region of open field lines just outside of the core which connect to the divertor, 3) the Plasma-Wall Interaction zone (PWI), where the cooler edge plasma interacts directly with the wall, both in the main chamber and the divertor regions, and 4) the interactions within the plasma walls in the main chamber and divertor which are

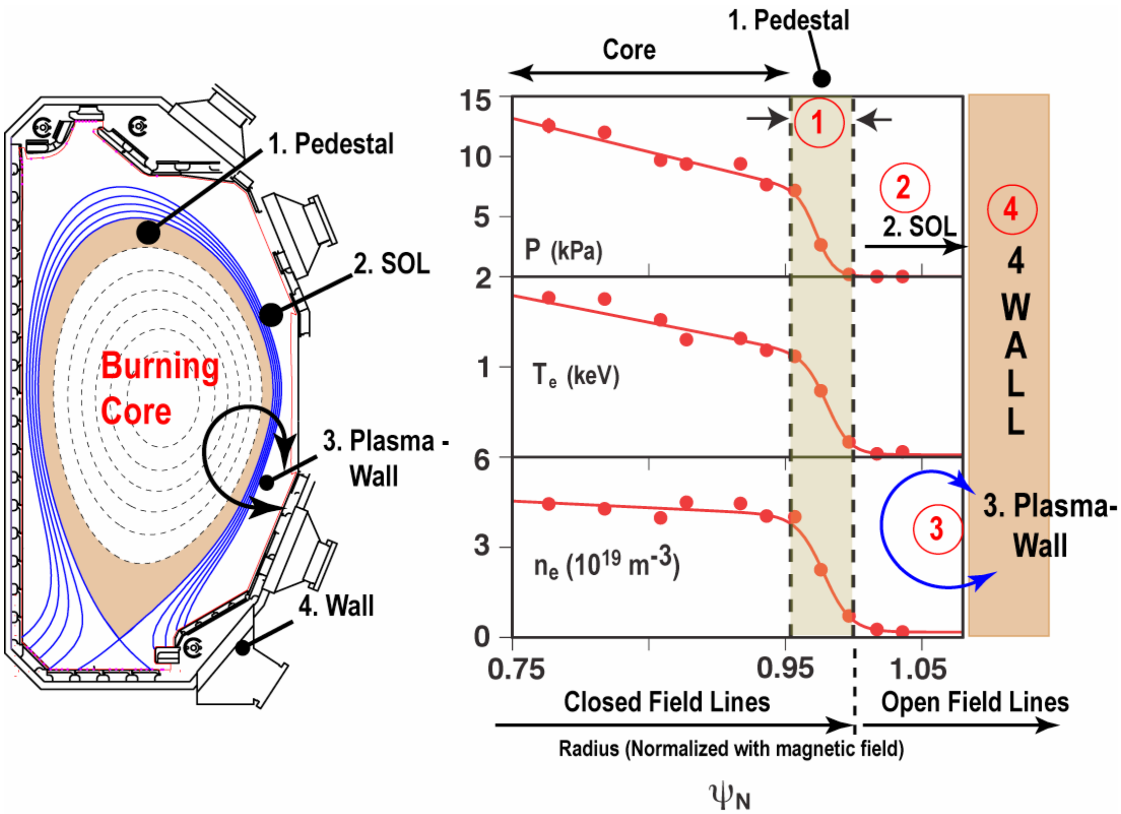

Fig. 5.1 The Plasma Boundary is naturally divided into four regions: 1) Pedestal (the outside of the core on closed field lines), 2) Scrape Off Layer (SOL) (open field lines), 3) Plasma Wall Interaction zone (PWI) where the weak SOL plasma interacts with the wall in the main chamber and divertor, and 4) Wall interactions, which depend on the properties of the material.

highly dependent on the material properties of the wall. These are the same regions identified as "research thrusts" in the 2005 FESAC priorities panel report.[7]

Present experiments and simulations have identified the important physics issues in each of these regions, which in turn motivate the US research agenda for ITER. In most cases, the ITER physics in the boundary region is an extension of that being studied in current US machines, and the current work naturally leads to a strong US role in the ITER science program.

5.1 The US agenda for ITER in plasma boundary interfaces research

5.1.1 Plasma Region 1: Physics of formation, structure, and stability of the pedestal

US Agenda: Develop models of the pedestal, benchmarked with existing experiments, that predict ITER pedestal physics – compare these with ITER data in H-mode.

In H-mode, a strong edge transport barrier isolates the hot core plasma from the SOL. The pedestal region typically extends over the outermost 3-10% of the volume of the confined plasma. The pressure at the top of this edge barrier, or "pedestal height" plays a

[7] http://www.ofes.fusion.doe.gov/more_html/FESAC/PP_Rpt_Apr05R.pdf

strong role in determining global confinement and overall fusion performance. The MHD stability of the edge barrier region is governed by peeling (current driven), ballooning (pressure driven), and coupled peeling-ballooning modes. While there has been extensive progress in the understanding of the physics of the pedestal, and current codes can adequately predict the maximum pedestal height given the width of the edge barrier, they cannot by themselves predict the width. The predicted fusion gain in ITER is expected to be a sensitive function of the pedestal height, so research in this area is a key component of the US research agenda. Developing empirical scalings of edge barrier width with machine size has been problematic, though there are recent encouraging developments.[8] Fully predictive models which predict both the pedestal height and the barrier width are in early stages of development. One such model combines peeling-ballooning stability calculations with a second constraint based on kinetic ballooning mode onset, and has had good initial success predicting pedestal height and width, but requires further development. Continued theoretical development of the nonlinear MHD evolution and pedestal transport is an ongoing part of the US research plan. This involves nonlinear calculations of MHD mode dynamics as well as kinetic and neoclassical simulations of plasma transport in the edge barrier and SOL region, and further development of simplified models. The ultimate goal is to use the experiments and modeling to be able to predict pedestal parameters, and thereby core fusion gain, for ITER and compare with ITER data. In present experiments, the spatial and temporal resolution of the edge current density measurements is barely adequate for detailed comparison with the computational models, and improvements in the resolution of density and temperature profiles, as well as continued development of edge turbulence diagnostics, would also be beneficial. Improving the quality of the edge measurements in ITER will be an important part of the US research agenda, and the ITER Motional Stark Effect (MSE), which measures current density profiles, is assigned to the US

US Agenda: Develop, implement and test ELM control strategies for ITER H-mode operation.

Present experiments have identified another key characteristic of the H-mode pedestal region – the presence of Edge Localized Modes or ELMs. This repetitive particle and power loading on the material walls causes additional erosion of material from the divertor surface. Depending on their magnitude and frequency, ELMs could limit the lifetime of the ITER walls. Recent studies[9] indicate that 0.5 MJ/m^2 per ELM may be the acceptable upper limit. For this reason, there is a strong US program in understanding ELM behavior, along with searching for small ELM and ELM-free regimes, such as the QH-Mode on DIII-D, and the EDA mode in C-Mod. Pellet pacing, using the repetitive injection of small pellets to induce less powerful, higher frequency ELMs, is another technique being explored. Recent experiments have shown that imposing resonant magnetic perturbations (RMP) can moderate or even eliminate ELMs. This result has stimulated both US experiments and modeling of RMP physics with coils both inside and outside the vacuum vessel.The US is a leader in the effort to add RMP coils for ELM control to the existing ITER design, as part of the 2008 ITER re-baselining activity. Thus,

[8] Snyder, et al., IAEA 2008
[9] Linke, J., et al., Proc. 13th International Conference on Fusion Materials, Nice, December, 2007

this is also a key component of the research agenda for current US experiments and is a key part of the US ITER agenda. The goal of the current design effort is to find a coil set that will both provide the RMP to control ELMs, and also control core MHD perturbations.

5.1.2 Plasma Region 2: Physics of plasma and impurity transport in the SOL

US Agenda: Identify the driving mechanisms for parallel and cross-field (perpendicular) transport in the SOL plasma. Measure, model, and "predict" the SOL width at the midplane and divertor in current machines and ITER.

In the Scrape Off Layer (SOL) region, particle and plasma heat are transported along and across open field lines; this competition between parallel and perpendicular transport sets the width of the heat flux channel at the divertor plate. It also determines the particle flux at the main chamber walls. Impurities transported out of the core along with those generated by interactions with the wall are transported along the open field lines in the SOL, and on this path they either re-renter the core or interact again with the main chamber or divertor walls. Current experiments have discovered a "detached divertor" mode of operation, where the heat flux is substantially reduced to the divertor plate by intrinsic impurity radiation. A cold, recombining plasma forms very close to the divertor plate, effectively isolating the divertor plate from the upstream SOL plasma. This mode was pioneered on the DIII-D and C-Mod machines in the US, has been obtained on nearly all major tokamaks and NSTX, and forms the basis of the ITER divertor design.[10] A key US diagnostic will be a system of six periscopes and camera systems to measure the IR and visible emissions of the divertor with full toroidal coverage. This will allow the US to compare the footprint of the power deposition on the divertor plate from existing machines with a burning plasma in ITER. A goal is to be able to predict the power footprint at the divertor on ITER.

Plasma fluid models have been used to model the SOL region and good progress has been made in comparisons with experimental data, particularly for non-detached divertor plasma conditions. Heat transport along field lines is reasonably well described by fluid transport equations with kinetic corrections. Detached conditions are more challenging, particularly when particle drifts are included. In addition, parallel plasma flows are found in the SOL, which often depend on the details of the magnetic field configuration and the rotation of the core plasma; the SOL flows, in turn, may also influence the core rotation. The cross-field heat and particle transport in experiments involves intermittent, bursty, plasma turbulence. Codes indicate that the transport should have a strong poloidal dependence. Fluid code calculations of the SOL with cross-field particle drifts do not adequately describe the experimental observations, and are a focus of current research.

An important US research agenda item for ITER is to develop numerical simulations (both interpretive and predictive) to relate the measured SOL profiles to the underlying physics of particle and energy transport. In interpretive modeling, a 2D or 3D fluid model with a Monte-Carlo neutral model is used to match the time-averaged plasma conditions measured by a large suite of diagnostics – using the best available models of classical physics. Discrepancies in the comparisons motivate improvements in the models. Areas

[10] A. Loarte, et al., 22nd IAEA Conference on Fusion Energy, Paper IT/P6-13, Geneva, 2008.

of strong sensitivity highlight the key physics processes that need to be studied carefully. By the time ITER operates, the US should have these models well in hand, and can then start comparisons in the new, burning plasma regime.

A second thrust is to develop first principles models of the pedestal and SOL plasmas, such as a kinetic pedestal model coupled to a fluid scrape-off model. This development will require careful comparisons of turbulence simulations with 2-D measurements of fluctuations and transport, resulting in any needed refinements to the model. These transport models will then be coupled with fluid codes that calculate plasma profiles and flows averaged over transport time scales. By the time ITER operates, the US should have made significant progress on developing these models, and can begin the comparisons with the ITER burning plasma regime.

5.1.3 Region 3: Plasma Wall Interactions (PWI) – erosion, deposition, power handling

US Agenda: Characterize and understand plasma wall interactions well enough to guide the selection (possibly staged) of plasma facing components for ITER. On ITER, assess the performance of different wall materials. Determine the suitability of these materials for reactor concepts.

Region 3 is at the interface of the SOL plasma and the solid material surfaces. From a burning plasma in ITER, the wall receives fluxes of a) particles with sufficient energy to cause wall erosion, b) radiation (UV, visible, x-ray) which can heat the material, and c) 14 MeV neutrons which can cause cause material damage. The wall provides sources of deuterium and tritium and impurity particles to the plasma; the wall can also trap tritium and reduce the amount of available plasma fuel. In ITER, the tritium inventory must be kept at or below a specified maximum, so the wall inventory must either be minimized or controlled with removal techniques.

The choice of wall materials for ITER has not yet been finalized, and is the subject of focused research both in the US and elsewhere. The US program is well poised to address the main wall materials issues: C-Mod currently uses solid all-metal molybdenum and tungsten walls, DIII-D has a carbon wall (baked to 300 C), and NSTX currently uses carbon (also baked to 300 C), but is also investigating the use of lithium walls. There are also smaller, complementary lithium programs on the Lithium Tokamak Experiment at PPPL and PISCES at UCSD. Research on the US facilities complements the other international machines and plays an important role in determining the sequence of wall materials in ITER. The various wall materials have different issues, and no single one is clearly ideal. The fusion program has a lot of experience with carbon, and if carefully conditioned (baking, plasma cleaning), the highest performance plasmas have been obtained in machines with carbon, partly because low-Z materials are less damaging to the core confinement. However, carbon erodes by both chemical and physical sputtering, and when this material redeposits on the wall, it can combine with tritium and locally trap large amounts of tritium. With strict limits imposed on the allowable tritium inventory in ITER, effective removal techniques would need to be developed for carbon. Metal walls can handle high heat fluxes, but metal impurities introduced into the core cause UV and x-ray radiation that can degrade the core plasma. The tritium retention issue, previously thought to be negligible for tungsten, is being re-examined by the C-Mod group. It has also been determined that the type of auxiliary heating affects impurity generation from the wall material. Specifically, ICRF heating in metal-wall machines can generate more impurities. In C-Mod, which uses high power density ICRF for the majority of plasma heating, highest plasma performance is generally obtained after coating the metal plasma facing components with a thin layer of boron which has low Z. ASDEX-U is also studying these issues using tungsten coated graphite PFC's, and JET is planning to convert to tungsten coated graphite as well.

A suitable diagnostic set will be required for resolving PWI issues in ITER, although surface analysis diagnostics are not currently a major component of the ITER baseline diagnostic. The development of innovative PWI diagnostics for ITER should be encouraged.

5.1.4 Region 4: Wall material and component research

US Agenda: Develop wall materials and components that are appropriate for high-power, long pulse experiments (active cooling) with high neutron fluence. These materials must be consistent with plasma blanket designs.

In Region 4, we are concerned with the influence of the plasma on the wall, and the efficient removal of the heat flowing from the plasma. Compared to existing devices, the stored energy in ITER will be at least 20 times greater and the pulse lengths will be minutes compared with seconds. The neutron fluence, which can significantly alter the material properties, will be much higher than in current experiments.

Carbon, beryllium and tungsten are candidate materials for ITER, and these may be introduced into ITER in a staged manner. Because of the high heat flux and long pulses in ITER, active cooling of the wall material will be required.

5.2 The role and assessment of ITER in promoting progress in plasma boundary interfaces research and its relationship to the US Fusion Energy Sciences program

In the Plasma Boundary area, there is a logical progression of issues and tasks starting with the existing experiments, leading through the startup and burning plasma stages of ITER. The culmination of this research will be a self-consistent concept for a reactor that generates reliable and affordable electrical power. In the area of PWI, which will be critical for a power-producing reactor, the issues, tasks, and assessments are most effectively addressed in the framework of the four edge regions developed above.

5.2.1 ITER pedestal physics: progress, US linkages, and assessment

For the baseline ITER H-mode scenario, the characteristics of the pedestal region critically affect the core confinement properties and the nature of the ELMs. In existing US machines, we are making high-resolution measurements of this region and developing computational models. During the ITER construction period, these computational models will become more sophisticated, adding kinetic effects and more fundamental calculations of edge plasma turbulence. Comparisons with plasma data will motivate improvements to the models. During ITER operation, the pedestal physics will be addressed in a new physics regime, and this may reveal other necessary additions to the suite of computational models. With the currently-envisioned ITER diagnostic set, the basic measurements of pedestal parameters (e.g., density and temperature) should be at about the same level as in current machines, allowing meaningful, but probably not comprehensive, comparisons with the models. At the minimum, it would be expected that the pedestal models used during ITER operations would be able to predict plasma parameters, even if certain parts of the physics need to be simplified or parameterized. A combination of experimental scaling with calculational models should be able to predict the pedestal parameters of a next-step machine beyond ITER, including a DEMO reactor.

5.2.2 ITER SOL physics: progress, US linkages, and assessment

In the SOL region, current experiments are developing more sophisticated diagnostics to measure the SOL plasma parameters, particularly the poloidal and toroidal flow patterns. It is expected that the ITER SOL turbulence measurements will be limited compared to those being developed and deployed on current US tokamaks. However, many of the current diagnostic techniques (e.g. edge probes) are not appropriate for either high power H-mode plasmas on existing machines or on ITER. New inventions in diagnostic techniques could allow better flow measurements in the SOL. At the divertor plate, the US will provide a dedicated IR-Visible diagnostic for ITER and this should ensure that heat flux and impurity radiation measurements can be made with even more detail than in current experiments. As in the case of the pedestal, both semi-empirical scalings and computational models will be developed for the SOL region. More fundamental treatment of turbulence with edge transport codes will be developed, and models will include kinetic effects. Again, discrepancies between the experimental measurements and the models will stimulate development the models. In ITER, divertor detachment will be tested in a new collisionality regime. ITER will show whether intrinsic impurity radiation can continue to create the desired cold, recombining plasma near the divertor plate because, for the first time, this plasma will be optically thick – at least for the ultraviolet hydrogen emission – and absorption of radiation by the plasma will be important. Calculating or predicting detachment regimes on ITER will challenge the modeling, and improvements will be necessary. At a minimum, a combination of empirical scalings and computational models should be able to “predict” the divertor heat flux, and whether the plasma is detached. The ratio of parallel to perpendicular SOL transport must be understood well enough to develop a prediction of the power scrape off length at the divertor plate. The SOL plasma dynamics will inform the selection of wall materials for the various phases of ITER operation, and for future machines.

5.2.3 Plasma Wall Interaction (PWI): progress, US linkages, and assessment

The goal in the area of PWI will be to develop a concept for the wall of a fusion reactor. Again, this will be a progression from existing machines to ITER. The US machines are studying refractory metal, carbon, and lithium walls in tokamak devices. Other effects can be simulated in laboratory experiments. The US fusion community has a competitive suite of surface diagnostics and computational models. Tritium removal techniques for both carbon and metal walls can be simulated. The world research, including the JET ITER-like wall experiment and ASDEX-U using tungsten coatings, will help guide the design of the ITER first wall. The current plan on ITER is to use carbon near the strike points, tungsten in lower heat flux regions of the divertor, and beryllium for the rest of the first wall.[11] Knowledge gained on ITER will help guide the wall choices for follow-on experiments, recognizing that a fusion DEMO will have much higher neutron fluences, and will operate with high temperature walls.

5.2.4 Wall material and component research: progress, US linkages, and assessment

In the area of wall material and component research, there will be less direct linkage between the ITER and the US tokamak program. Small subassemblies may be tested in existing machines, but they will not have the combined harsh conditions expected on ITER: neutrons, MHD forces, high particle fluxes, high heat fluxes, long pulse operation, and active cooling. The ITER test blanket modules will probably be the first of their kind, and will not have a precedent on any existing tokamak experiments. In this area, the US Virtual Laboratory for Technology (VLT), in concert with other testing facilities in the US and abroad, will have to develop the relevant technology for the wall modules, divertor cassettes, and any test blanket modules. The first semi-integrated test of these components will actually be in ITER which means that in this area there will be a larger extrapolation of knowledge from ITER to follow-on machines.

[11] R.J. Hawryluk, et al., Nucl. Fusion **49** (2009) 065012.

Chapter 6: Fusion Engineering Sciences

The fusion engineering sciences encompass:

a) Development and deployment of various plasma support technologies, *i.e.* the tools needed to create, confine, understand, and control burning plasmas. These include magnets, heating and current-drive sources and launchers, fueling systems, and radiation-hardened diagnostics for a burning plasma and associated control systems.

b) Material science and mechanical engineering of plasma facing components which operate under extreme conditions of high energy, particle, and neutron fluxes

c) Nuclear engineering for harnessing fusion power; including tritium production and management (fusion fuel cycle), converting fusion power to useful heat, and the properties of materials under intense neutron radiation.

Some aspects of these have been discussed in the preceding chapters; we focus here on engineering challenges.

Plasma support technologies are essential for ITER to achieve its scientific research and performance goals. The burning plasma environment of ITER will provide a wealth of information on the behavior of the plasma material interface and developing the technology of plasma facing components. ITER will also provide focused information on fusion nuclear engineering including tritium handling and reprocessing of hydrogen isotopes on a large scale, and deployment of extensive remote handling capability.

6.1 The US agenda for ITER in fusion engineering sciences research

6.1.1 Plasma Support Technologies

6.1.1.1 Superconducting magnets − ITER's ability to study burning plasmas for long time scales is primarily due to advances in superconducting magnet technology. ITER's superconducting magnet system will be the largest ever assembled, and is comparable to that expected for a fusion power plant. A wealth of information will be gathered during the construction and commissioning phases of ITER. Monitoring of the superconducting magnets during ITER operation will provide useful information on the performance of power-plant scale magnets.

6.1.1.2 Heating and current drive systems − Continued development of plasma heating and current drive systems will be a decisive factor in how far ITER can progress in advanced, long-pulse or even steady-state burning plasma research. Auxiliary heating systems must heat the ITER plasma to the burning plasma condition. To sustain the burning plasma, the spatial distribution of electrical current in the plasma must be controlled by the various neutral-beam and radio-frequency auxiliary systems. Experiments on ITER will advance our knowledge in many areas including coupling of heating and current drive power to the plasma, steady-state cooling of the associated components and the behavior of insulators in the neutron environment.

6.1.1.3 Particle fueling and exhaust – Because of its long pulse length, the ITER plasma must be continuously fueled during the discharge and the exhaust gases collected and sent for processing. Fueling methods will include gas injection (for edge fueling) and frozen pellets of deuterium and tritium (for deeper fueling). The neutral beams also provide some fueling, which, depending on particle confinement, could be in the range 5 – 10%. Fueling methods can have a major impact on the radial profile of plasma density. ITER will be a critical test of fueling methods for a burning plasma with power-plant scale.

6.1.1.4 Real-time Plasma Control – Careful control of plasma properties is essential for ITER to achieve its mission. Existing experiments have shown that plasma performance can be substantially improved through fine-scale control of plasma profiles, particularly the plasma current, pressure and rotation profiles. Real-time plasma control requires an integrated set of tools to manage the magnets that provide confining and shaping fields, plasma heating and current drive systems, and fueling and pumping systems for control of edge plasma conditions and particle exhaust. Experiments on ITER are essential for developing the tools needed for controlling a burning plasma with a large population of energetic alpha particles and exploring the degree of control that is achievable with limited external power input.

6.1.1.5 Plasma Diagnostics - Real-time diagnostics for plasma parameters are an essential part of the plasma control toolkit, providing crucial data on plasma response. Meeting new measurement requirements, adapting present technologies to the harsh ITER environment, and satisfying high reliability specifications are challenges requiring significant scientific and engineering innovation. Looking ahead to yet more robust and simpler systems for fusion power plants, ITER experience will help determine the reduced set of essential measurements and the associated durable technologies.

6.1.2 Plasma-facing components

ITER will be the first experiment to produce copious amounts of fusion power (500 MW). As much as 150 MW of thermal power will be exhausted from the plasma and must be removed through the plasma facing components. No experiment to date has had to exhaust this much power or deal with such high power and particle fluxes for long pulses. Thus, ITER poses new challenges both to develop plasma operating modes in the plasma edge to reduce particle and heat fluxes on the plasma components and to develop erosion-resistant plasma-facing surfaces capable of handling high heat fluxes and also suitable for use in the tritium environment.

In addition to the normal operation, plasma facing components should be able to handle a substantially higher heat flux (but for a short duration) during off-normal events such as edge-localized modes (ELMs) and disruptions. Tools for reliable mitigation of these events and their consequences also need to be developed and tested. Experiments on ITER are essential for developing plasma operating scenarios which minimize and mitigate these off-normal events.

Tritium retention and removal from plasma facing components are key challenges for ITER where the walls will not be operated at high bulk temperature.

6.1.3 Nuclear Engineering Science

As ITER will be the first experiment to produce copious amounts of fusion energy, the operation of its monitoring and safety systems will provide a wealth of information necessary for licensing of follow-up fusion devices.

ITER will incorporate a power-plant scale tritium management facility to separate tritium out of the exhaust from the plasma chamber and make it available for re-injection into the plasma. Operational experience of the ITER tritium plant will be crucial for the development of a fusion power plant.

A DT fusion power plant will utilize a blanket surrounding the plasma to breed tritium fuel from the fusion neutrons. While the ITER baseline design does not include a full breeding blanket, it does include ports for Test Blanket Modules (TBMs) to assess their performance in a fusion environment. However, due to the relatively low neutron fluence expected in the ITER base line scenario (compared to a commercial fusion reactor), the contributions of ITER to fusion nuclear engineering will, in this regard, be limited. Table 6.1 shows some comparisons between ITER and DEMO with regard to surface heat and neutron loading, as well as fluence.[12]

Table 6.1 Comparison of the main operating specifications for TBMs in ITER and for DEMO Breeding Blanket

Parameters (materials relevant)	ITER H phase Design (& Typical) Values	ITER D-T phase Design (& Typical) Values	DEMO Expected Values *(example)*
Surface heat flux on First Wall (MW/m^2)	0.3 (**0.15**)	0.5 (**0.27**)	**0.5**
Neutron load on First Wall (MW/m^2)	-	0.78 (**0.78**)	**2.5**
Pulse length (sec)	Up to **400**	400 /up to **3000**	**Quasi – continuous**
Duty cycle	**0.22**	**> 0.22**	1
Average neutron fluence on First Wall ($MW.yr/m^2$)	-	**0.1***(first 10 yr)* **up to 0.3** *(EOF)*	**7.5**

[12] L. Giancarli, V. Chuyanov, and ITER Test Blanket Working Group Members, *Scientific/Technical Merits and expected Outcomes of the TBM Programme*, September 2008

The TBMs in ITER can also provide information on the possible effects of blanket components in a reactor on plasma performance. Due to such concerns, the timing and priority of TBM installation is currently under discussion among the ITER parties.

6.2 The role and assessment of ITER in promoting progress in fusion engineering sciences toward making fusion a reliable and affordable source of power

Plasma support technologies developed for ITER are directly applicable to future fusion devices and fusion power plants. As such, information obtained during the construction and commissioning phases of ITER as well as continued monitoring of these critical systems will allow further optimization of these technologies to make them cheaper and/or higher performance as well improving their reliability.

Plasma-facing components envisioned for ITER baseline are not directly applicable to a fusion power plant which requires operation at higher neutron fluences and high temperature to convert plasma exhaust power into useful energy. ITER operation, however, can contribute substantially to developing high-performance plasma facing components for power plant application through: 1) Developing plasma operating modes with reduced heat/particle fluxes, 2) expanding our understanding of the plasma-material interface, and 3) development of methods to minimize and mitigate off-normal events. Assessment of components will improve our knowledge of the limits and performance of materials in the complex fusion environment, which cannot be fully duplicated in test facilities.

6.3 The relationship of ITER to the US Fusion Energy Sciences program in fusion engineering sciences

Most of the US fusion energy sciences engineering program at present is focused on developing and fielding plasma support technologies and plasma-facing components for ITER (particularly for the specific US contributions to ITER construction) and for other current and planned confinement experiments. For ITER, the present focus is to ensure on-time construction and successful commissioning and operation of ITER.

Chapter 7: Research Plan for the Integrated, Burning Plasma System

7.1 The US agenda for ITER in integrated, burning plasma research

It is anticipated that ITER will initially operate only with hydrogen and helium plasmas for a period of up to two years, during which the facility to handle tritium in the exhaust from the tokamak will be commissioned and the licensing process for proceeding to "nuclear" operation will be conducted. Even operation with pure deuterium plasmas will require the nuclear operating licence because of the expected activation of the tokamak structure and the production of tritium from D-D fusion reactions. While most of the research on the integrated, burning plasma system must await the introduction of full-scale deuterium-tritium (D-T) operation to the facility, there are several areas where research in the non-nuclear phase of operation will be both necessary to the ultimate success of ITER and of considerable interest for the physics that may be revealed, particularly because of the scale of the plasma and parameter regimes that can be accessed in ITER. It is also expected that some data from the initial experiments in ITER itself, particularly relating to the effects of plasma disruptions, will be needed to complete the licensing process.

7.1.1 Research in the Non-Nuclear Phase of ITER Operation

A prerequisite for proceeding to full-scale D-T operation is confirmation of the physics assumptions that underlie the design. Of particular importance is the size-scaling of confinement in the baseline operating mode, namely the ELMy H-mode. In several other respects, namely temperatures and densities and even D-T fusion power density, dimensional plasma parameters approaching those expected in ITER have already been produced in existing (*e.g.* JET and JT-60U) and former (*e.g.* TFTR) tokamaks, albeit in most cases using auxiliary heating deposited predominantly on the ions rather than on the electrons as will be the case with alpha-particle heating in a burning D-T plasma. Even in its initial phases of operation, ITER will be equipped with several heating schemes, ICH, ECH and some amount of negative-ion neutral-beam heating (NINBH), which will preferentially heat the electrons and thus provide a reasonable simulation of alpha-particle heating. Unlike most currently operating tokamaks, which rarely operate at their design specifications and often express their results in normalized parameters, the achievement of high D-T fusion power in ITER will require that it be run very close to its maximum magnetic field, current and heating power. Two important normalized parameters on which existing results are judged are the Troyon-normalized beta β_N and the H-factor of the confinement relative to an empirical scaling relation. Thus it will be important to confirm that the extrapolations which have been made to ITER from existing experiments based on those normalized parameters are indeed valid.

A difficulty for the initial operation of ITER with hydrogen is that the heating power expected to be available may not be adequate to achieve the H-mode at full magnetic field and current. In present tokamaks, the H-mode has a higher threshold heating power for hydrogen than for deuterium or tritium plasmas. It may be possible to operate ITER at reduced current and field to reduce the H-mode power threshold but, since both the ICH and ECH are resonant heating mechanisms, the choice of magnetic field and absorption schemes for ICH is limited. Operation with helium plasmas is also a possibility for the early non-nuclear phase and, in this regard, recent results from ASDEX and JET are encouraging although not yet definitive. The US therefore has an interest in studying and developing over the next decade in its existing devices alternative scenarios for achieving a productive non-nuclear phase for ITER, taking account of the constraints on its operation and on its plasma heating systems.

As discussed in Chapter 4 on transport, a fundamental physics interest in studying confinement scaling in ITER is its dependence on the normalized gyro-radius, ρ^*, defined as the ratio of the characteristic ion gyro-radius in the plasma to the characteristic dimension of the plasma cross-section, at the appropriate levels of plasma collisionality, β, MHD safety factor q, Mach speed of the plasma flow, ratio of ion to electron temperature and dimensionless plasma shape parameters. This issue of size scaling is a critical test of the success of our efforts to simulate fusion plasmas from first principles and thus our ability to project both to the full-scale D-T performance of ITER itself and also beyond ITER to a power-plant scale device, such as DEMO. While it is possible to study the ρ^* scaling of confinement in existing devices, it is not possible to do so while simultaneously matching the other important dimensionless parameters to those characteristic of a burning plasma in ITER. Furthermore, given the importance of the plasma-wall interactions which has become apparent in tokamak experiments over the past twenty years, it is important that the scaling of confinement to the ITER scale be carried out with the appropriate scale and even geometry governing neutral-atom interactions in the plasma edge.

Confirmation is also needed of the design assumptions regarding plasma stability to both current and pressure driven modes of instability (Chapter 2) and of plasma-wall interactions (Chapter 5). The management of heat loads on the divertor and other plasma-facing surfaces, both in quasi-steady-state and in transients, such as ELMs, and minor and major disruptions is critical to the success of burning-plasma research on ITER. While these studies would ideally be performed in conditions matching as closely as possible those expected in the burning-plasma phase, this will obviously not be possible with respect to steady-state power flow to the divertor since the auxiliary heating power cannot match the expected alpha-heating power in ITER. However, in the pre-D-T phases, the total plasma thermal energy, which is important for the effects of transient events, will be a significant fraction of that expected in full D-T because of the unfavorable scaling of energy confinement with heating power typically observed in H-mode plasmas.

The experiments undertaken in the non-nuclear phase should include operating for extended pulses, *i.e.* with stationary plasma conditions lasting much longer than the energy confinement time, insofar as such operation is possible with the heating systems and power available. These studies should be performed as early in the ITER research program as possible so that if they show that the reference high-Q D-T plasma scenario will pose a severe challenge, because either confinement or plasma stability are inadequate or plasma-wall interactions are unacceptable, then it would be possible to make modifications or upgrades to the capabilities of ITER to meet its objectives.

7.1.2 Research in the Nuclear Phase of ITER Operation

At the conclusion of the non-nuclear phase of operation, it is anticipated that the necessary data will be available and operational experience accumulated to seek and gain approval of the regulatory authorities to begin the nuclear phase of ITER operation, beginning with deuterium plasmas and progressing to D-T plasmas.

It is planned that the full auxiliary heating power from all systems will become available during the deuterium phase. With this increased power, routine operation in H-mode plasmas at full current and toroidal field should be achievable in deuterium, which would be expected to reach about 60% of the plasma thermal energy of the planned $Q = 10$ reference plasmas. There will be research interests in characterizing ELMs and their effects on divertor erosion and on the behavior of tritons produced in D-D fusion reactions, both their confinement in the plasma and their retention in the plasma-facing components. Tritium retention remains a critical issue for the subsequent D-T research program in ITER since high levels of retention could necessitate periodic replacement of some in-vessel components and thereby determine the long-term duty cycle of the facility. Another issue for planning the deuterium phase is activation of the tokamak structure and its impact on maintenance and installation activities. Some calculations indicate that activation during D-D operation will be high enough to require full remote handling. A more detailed look at this question would be very valuable, and could impact the overall plans for the operational phase leading up to full D-T operation.

The initial research objective of the D-T phase will be producing and studying a non-transient high-Q ($Q \geq 10$, timescale $\gg \tau_E$) D-T plasma state. In order to achieve this objective, it will be necessary to explore and optimize trade-offs between operational conditions and plasma characteristics necessary or desirable for high performance, *e.g.* confinement enhancement *vs.* stability limits *vs.* PFC and divertor survival. Methods for optimizing the D-T mixture and maximizing fuel burn-up, considering both fusion reactivity and possible effects of isotopic mass on confinement, will be of interest. Characterizing and developing methods to control the in-vessel accumulation of tritium will be critical to the long term D-T research program. The achievement of $Q \approx 10$ in ITER will provide the opportunity to investigate for the first time the effects of a significant population of energetic alpha particles on stability and confinement. Methods for avoiding operational limits during high-Q operation will need to be demonstrated or developed, including investigating and implementing techniques to achieve burn stability and control.

Once conditions for high fusion gain have been established for timescales comparable to the energy confinement time, the focus of research will move to extending operation to quasi-steady-state through a sequence of characteristic timescales, *viz.* the current relaxation, the wall thermal and, eventually, the wall loading timescales. This will require increasingly sophisticated control of many operational and internal plasma parameters simultaneously, including the external heating and current drive systems, the plasma equilibrium and divertor strike points, the plasma fueling and exhaust systems, the radiated power and divertor detachment, the current and pressure profiles to maximize the bootstrap current while maintaining MHD stability and, possibly, performing some degree of phase-space control on the energetic alpha-particle population to maintain burning and reduce the buildup of helium ash. While modeling, based on the results of dedicated experiments to be carried out on smaller scale tokamaks over the next decade, can guide the implementation of these control capabilities, their integration and application to a burning plasma, which will intrinsically have a non-linear response, will be a major research challenge. The goals of this research will be to explore and to demonstrate the possibility of achieving high operational duty factor at high fusion performance which will be needed for the subsequent phase of the ITER program involving test-scale tritium breeding and fusion energy extraction. The elements of this research phase will include developing strategies and techniques to minimize recirculating power, particularly pulsed power, and to minimize reliance on inductive current sustainment. These experiments will enable the identification of possible limitations on extending operation to true steady-state in a future power-producing reactor.

In addition to fulfilling its primary objectives through operation in its currently planned baseline plasma regime, an important element of the ITER research plan should be exploring and developing alternative operational scenarios that could achieve ignition in a smaller or less technically demanding DEMO device. While such research is more speculative, the potential benefits are high, so it is important that a fraction of the ITER research time be allocated to exploring possibilities that might improve the prospects for fusion energy production, either by reducing the unit size of a power plant or by decreasing its complexity. This is a role in which the US has excelled and it should continue to be an important element of the US research agenda for ITER.

7.2 The role and assessment of ITER in promoting progress in integrated, burning plasma research toward making fusion a reliable and affordable source of power

As the only planned experiment capable of producing high fusion gain with D-T fuel in a magnetically confined plasma, ITER represents a major and necessary step toward making fusion a power source. It will provide a unique opportunity for research on the integrated, burning plasma system. Through the stages and elements outlined in the research agenda above, it will provide specific design information and confidence in extrapolations to proceed to a next-step DEMO and/or a Materials Component Test Facility. Again, it should be pointed out that the prospects for fusion power could also be improved by demonstrating and developing in ITER techniques and scenarios which would reduce the scale and complexity of future devices designed for fusion power production, although this element of fusion energy research will probably not be a major part of the overall ITER research program.

In order to gauge progress towards the ultimate goal of fusion power, a sequence of research campaigns should be established for ITER, each including objectives specifically related to achieving reliable, controlled, sustained burning plasmas with high duty factor. To this end, the ITER Organization is currently preparing an ITER Research Plan. In this plan, the first version of which was completed in mid-2008, the sequence of research campaigns is determined by the planned development of facility capabilities as outlined above.

It should be anticipated that some upgrades to the ITER facility will be required during its lifetime to take advantage of what is learned at each stage in the research program. Some of these upgrades may be expected to be directed primarily towards the goal of fusion power. Examples would be improvements to the plasma facing materials and component design to provide improved performance and longevity in a prototypical reactor environment, increased input power if results indicate the necessity, and the implementation of advanced control methods which would allow longer plasma pulses at high fusion performance with reduced risk of deleterious off-normal events. Such advances would also allow ITER to make a more significant contribution to studying tritium breeding and the extraction of heat from a power producing reactor.

7.3 The relationship of ITER to the US Fusion Energy Sciences program in integrated, burning plasma research

A major, long-term goal of the US fusion energy research program is the development of a validated, comprehensive simulation capability for predicting the characteristics and performance of fusion reactor systems. To this end, the Office of Fusion Energy Sciences within the US Department of Energy has created a Fusion Simulation Project (FSP), involving researchers from many institutions, with the mission to develop a predictive capability for integrated modeling of magnetically confined burning plasmas. This modeling must eventually involve all the elements and processes which are in play in such a complex system, spanning wide ranges in dimensional and time scales, physical states and, indeed, sub-disciplines within physics (*e.g.* plasma, atomic, nuclear, materials), chemistry and engineering. For the FSP, ITER represents both an unprecedented opportunity and a formidable challenge. The opportunity arises because ITER plasmas will enter regimes where, for the first time, all plasma parameters, and the conditions in many ancillary systems too, will simultaneously overlap those expected in a power-producing DT fusion reactor. If the FSP proceeds as planned, it will, by the time ITER begins to produce and study self-heated plasmas, have produced *ab initio* predictions of its performance and plasma behavior. These predictions should be, as far as possible, tested through comparisons with experiments on facilities other than ITER; however, the first integrated tests with burning plasma will necessarily require comparisons with results from ITER itself. The challenge will be to include sufficient fundamental physics and other relevant science into the integrated models to make the simulations essentially independent of empirical assumptions and adjustable parameters so that we can be confident that further extrapolation to eventual reactors will be soundly based. Complementary to this ultimate test of the capability of physics-based simulations to predict the performance of the first burning plasmas, ITER itself will benefit from the FSP through the development of the capability to conduct "virtual experiments" to assess operational scenarios for ITER and to optimize the use of its precious operational time. The ITER Organization is fully aware of the essential role to be played by advanced plasma simulations (see, for example, "ITER Modelling Needs and Plans" presented by W.A. Houlberg on behalf of D. Campbell, ITER Fusion Science and Technology Department, presented to the US-Japan Workshop on Integrated Simulation of Fusion Plasmas, 29-31 January 2007, Oak Ridge, TN).[13] Already, in the present design and

[13] http://cswim.org/meetings/us-japan-2007/Houlberg_ITER_Modelling.pdf

construction phases of the project, simulation codes are being used to develop specifications and operational techniques for the plasma control system, diagnostics and ancillary systems. As ITER approaches and enters its operation phase, techniques to start up the plasma, apply auxiliary heating and fueling sources to achieve controlled high fusion power gain while maintaining plasma stability and avoiding operational limits will need to be investigated thoroughly and rehearsed prior to performing experiments.

Although the design of ITER is now well advanced, unresolved issues remain in some areas. Some of these issues have, in fact, been revealed by research conducted on existing tokamaks since the original design of ITER was undertaken. The success of ITER in meeting its objectives therefore depends on performing specific, directed research tasks on existing or future purpose-built facilities to address these issues. In addition, new techniques and plasma scenarios developed within independent research programs in existing facilities may eventually be applied to ITER both to improve its prospects for success and to develop these techniques on a larger scale to be ready for incorporation in a DEMO device. An example is the development of the "advanced tokamak" scenarios with strong non-inductive current drive, which were invented and have been developed since the original design basis for ITER was adopted.

Several topics of concern for ITER will be the subject of active research on existing and new tokamaks during the next decade prior to the start of research in ITER. The International Tokamak Physics Activity, conducted since 2002 under the auspices of the IAEA International Fusion Research Council (IFRC) and the IEA Fusion Power Coordinating Committee (FPCC) with participation of experts from all the ITER Parties, and which in 2008 has come under the auspices of ITER, has compiled a list of research tasks to address outstanding issues for the ITER design. While most of the high-priority tasks identified by the ITPA concern research within other topical areas, two in particular are relevant to the integrated burning plasma system, namely, 1) investigating hybrid scenarios for prolonged plasma operation and developing full current-drive plasmas with significant bootstrap current, and 2) developing methods and predictive capability for real-time current profile control using heating and CD actuators, in particular off-axis CD. The US is well positioned, both in terms of experimental facilities, either already equipped with or planning to install the necessary capabilities, and theoretical modeling codes to make major contributions to both these research tasks. The development in existing medium-scale tokamaks of disruption avoidance and/or mitigation schemes and the control of adverse plasma-wall interactions should also be of considerable benefit to the design and ultimate operation of ITER for achieving sustained high fusion performance.

Much of the research in preparation for ITER related to steady-state operation will probably best be undertaken in the medium-sized tokamaks EAST (ASIPP, China) and KSTAR (NFRI, Korea) which have superconducting magnets designed for very long plasma pulses. US collaboration on such facilities is expected to be important in gaining expertise in these areas. These devices will not be capable of operating in D-T plasmas, however, so the effects of the plasma self-heating and of the energetic alpha-particle population on maintaining steady-state conditions will not be assessed.

Research in ITER can be expected to have a major impact on the design of a power producing DEMO reactor. At the request of the US DOE, a panel organized by the Fusion Energy Sciences Advisory Committee (FESAC) prepared a report identifying gaps in our present knowledge for designing and building the step(s) beyond ITER and setting priorities for bridging those gaps. The report anticipates that many, though by no means all, of the current gaps will be filled by successful research on ITER. Foremost among the requirements identified for proceeding to DEMO is the achievement of "predictable, high-performance steady-state burning plasmas". Such research will clearly be centered on ITER in the foreseeable future. Several of the specific tasks assessed in the report as requiring major extrapolation from our current state of knowledge and needing substantial development, involve the integrated burning plasma state, including maintaining high fusion performance, power extraction, sustaining the D-T fuel cycle, managing off-normal events, controlling plasma-wall interactions, and the development and validation of predictive modeling for burning plasmas.