# Burning Plasma Physics

## Technical Subgroup of the Magnetic Fusion Concepts Working Group

## 1999 Fusion Summer Study


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# Contents

## 1. Introduction

The burning plasma subgroup of the magnetic fusion concepts group met extensively during the Fusion Summer Study in Snowmass Colorado. The charge of the burning plasma subgroup was to:

A. Articulate burning plasma issues inaccessible to study in present facilities.

B. Identify contributions that burning plasma experiments will make to the general development of plasma science and to the development of fusion energy.

C. Formulate research programs to resolve uncertainties in physics projections to burning plasma conditions.

D. Attain consensus that the goals and performance measures for a burning plasma experiment are accurately presented.

The discussions of the subgroup took place in the context of plenary speakers who urged the fusion community to continue to establish plans for a burning plasma experiment. The background to the burning plasma discussions, and indeed to the Snowmass meeting as a whole, was how to reach consensus on the future direction of our community after the long period of divisiveness which occurred over the ITER project. The ambitious goal set for the burning plasma subgroup, in the aftermath of the US withdrawal from ITER, was to arrive at a new consensus on the role and importance of a next step burning plasma experiment for the future development and sustainment of the fusion effort. After two weeks of discussions, they concluded nearly unanimously that the tokamak concept was technically ready to plan and perform a burning plasma experiment. This assessment was supported by an over eighty percent majority of the larger magnetic fusion concepts group in an open meeting towards the end of the workshop (Sec. 3).

There is however a clear difficulty of how the U.S. effort in supporting a burning plasma program can proceed in the near term. Though the burning plasma effort is seen as very important, there is no planned direction or prioritization of the way a burning plasma experiment will be supported in the United States. The burning plasma group addressed our future role in bringing such a burning plasma experiment to reality. The outcome of these discussions was a strong recommendation from the subgroup that the U.S. actively pursue opportunities for a burning plasma experiment, with options ranging from participation in an international project to taking leadership with a strong national design effort.

The assessment of technical readiness for a burning plasma tokamak experiment was based on so called conventional confinement regimes (L-mode, H-mode) with high-densities, divertors and low self-driven current. Although these developments have opened the way to the possibility of a controlled burning plasma, it is also clear that major obstacle exist to economic fusion energy production. Net fusion power production will almost certainly occur, however there will be many other uncertainties, including: the reliability of operation (MHD and transport control under steady state conditions), the development of power production as an economically competitive source, and the willingness to support the cost of the first experimental net power producing facility.

Present tokamak experiments have opened the way to several important new directions for improved performance. These advances include: steady state operation by combining bootstrap current with non-ohmic current drive (either RF or negative ion neutral beam driven) and operation with transport suppression using flow shear and q-profile reversal. To improve the ultimate tokamak performance, an essential aspect of research in a tokamak burning plasma experiment will be to address these advanced modes of operation.

The uncertainties in realizing optimal confinement properties with the tokamak underlie the universal acknowledgement that we must continue research in other less developed concepts, such as: (a) the stellarator, where unpredictable disruptions of the plasma may be easier to control; (b) the reversed field pinch and spheromak where a fusion plasma may be confined at a lower magnetic field but where much larger improvements in confinement characteristics need to be attained, and (c ) in alternate tokamak approaches such as the spherical torus (ST) that emphasize the advanced options noted above such as high beta and the possibility of $E_r$ shear suppressed turbulent transport. Developing concepts such as the "advanced tokamak" and the spherical torus, while recognized for their potential, still need to establish a database of operation sufficient for taking the step to a burning plasma experiment. By "Advanced Tokamak" we mean high-beta, high-bootstrap fraction tokamak relevant to steady-state operation, with the tools to explore active profile control and stabilization.

Given the uncertainties involved in whether any existing concept can eventually provide an economic source of power, the question arose as to how a burning plasma experiment based on the inductively driven tokamak concept could be justified at this time. The predominant view of the burning plasma group was that a burning plasma experiment, based on a conventional tokamak operating regime, needs to be planned in order to: (a) demonstrate the feasibility of a controlled plasma burn; (b) resolve transport, stability and other plasma science issues at large dimensionless scale ($a/\rho_i$) in a burning plasma regime; (c) develop methods of burn, profile and instability control relevant to high Q regimes which are also likely to be applicable to other MFE concepts; (d) access advanced modes of tokamak operation for concept improvement under burning plasma conditions. In addition, the achievement of a burning plasma regime may allow other nearer term applications to be developed such as the transmutation of nuclear fission waste.*

Presently, there are three proposals in development to demonstrate burning plasma operation. These are: RC/ITER, an international tokamak design with a divertor that

* See energy subgroup A report in these proceedings.

employs improved understanding of tokamak operation to reduce the cost and objectives of the original ITER proposal; IGNITOR which exploits the benefits that can be achieved with high magnetic field, high density and compactness, supported primarily by Italy; the FIRE proposal, a compact high field divertor design with strong shaping capability, being studied in the United States.[*] The magnetic fusion study group has agreed that all three design studies should continue, and the designs should implement features to carry out advanced tokamak experiments. Further, if the international governments agree to financially support the ITER proposal, it was agreed that the U.S. government should seek a partnership position. A proposed JET upgrade was also endorsed as it was recognized that the experiment could address alpha, MHD and confinement issues in a sub-burning regime ($Q<2$) on a time scale which bridged the gap between present experiments and a future burning plasma experiment. A brief description of these proposals is given in sec. 4, followed by longer contributed papers by the proponents. The experimental opportunities in particular largely reflect the views of the proposed projects or individuals associated with those projects. There was insufficient time at Snowmass to adequately evaluate the various projects against a common set of criteria. That type of evaluation can only be achieved by a much more thorough review that encompasses benefits, risks and cost.

The burning plasma group adopted the following resolutions with near unanimity.

[*] See sec. 4 for a discussion of the physics mission and design considerations of these proposals.

*Resolutions of the burning plasma subgroup*

A. On the question of justification for a burning plasma experiment, the following resolution was adopted unanimously.

**"The excitement of a magnetically-confined burning plasma experiment stems from the prospect of investigating and integrating frontier physics in the areas of energetic particles, transport, stability, and plasma control, in a relevant fusion energy regime. This is fundamental to the development of fusion energy.**

**Scientific understanding from a burning plasma experiment will benefit related confinement concepts, and technologies developed for and tested in such a facility will benefit nearly all approaches to magnetic fusion energy. "**

There was some discussion that the burning plasma experiment should be an attractive fusion energy device and not just relevant. However the majority chose to adhere to the word relevant (70%).

The issue of transferability and the entire statement regarding frontier physics was voted on and agreed to unanimously.

B. On the question of what constitutes frontier physics in a burning plasma experiment, the group agreed unanimously to the following.

FRONTIER PHYSICS TO INVESTIGATE AND INTEGRATE IN A SELF-HEATED PLASMA

- **Energetic Particles**
  Collective alpha-driven instabilities and associated alpha transport.

- **Transport**
  Transport physics at dimensionless parameters relevant to a reactor regime (L/ρi scaling of microturbulence, effects on transport barriers…

- **Stability**
  Non-ideal MHD effects at high $L/\rho_I^*$, resistive tearing modes, resistive wall modes, particle kinetic effects…

- **Plasma Control**
  Wide range of time-scales: feedback control, burn dynamics, current profile evolution

- **Boundary Physics**
  Power and particle handling, coupling to core

(*$L/\rho_i$ is the system size divided by the Larmor radius.)

C. On the issue of scientific transferability, the group agreed unanimously on the following statement.

*Scientific Transferability*

A well-diagnosed, flexible burning plasma experiment will address a broad range of scientific issues and enable development and validation of theoretical understanding applicable in varying degrees to other magnetic concepts

- Energetic particle density gradient driven instabilities

- Transport and burn control techniques

- Boundary Physics, power and particle handling issues

D. On the opportunities which the U.S. should pursue, the group accepted the following resolution:

*BURNING PLASMA OPPORTUNITIES*

- Burning plasma experiments are essential for the development of fusion. (All in favor)

- The tokamak is technically ready for a high gain burning plasma experiment (95% in favor)

- The US should actively seek opportunities to explore burning plasma physics by:

1. Pursuing burning plasma physics through collaboration on potential international facilities (e.g., JET Upgrade, IGNITOR and ITER-RC) (95% in favor)

2. Should the ITER construction proceed, the US seek a partnership position. (None opposed)

3. Continued design studies of moderate cost burning plasma experiments (e.g., FIRE) capable of exploring advanced regimes (80% for, 10% against, 10% abstain).

4. Exploiting the capability of existing and upgraded tokamaks to explore and develop advanced operating regimes suitable for burning plasma experiments. (None opposed).

## 2. Burning plasma physics issues.

In the last 45 years all controlled fusion experiments have studied plasma regimes dominated by external heating. This is true even for the TFTR and JET experiments that produced significant fusion power and would remain true even after the proposed JET upgrade experiment successfully reaches its goal of achieving Q = 2. Clearly, new and crucial regimes of plasma operation will be studied if the proposed burning plasma experiments achieve the confinement characteristics needed to obtain dominant plasma self-heating.

A near term burning plasma experiment will have to deal with a plasma that has achieved the thermal confinement properties that will allow the power produced by confined charged fusion products to approach the power removed by intrinsic plasma losses. The energy in the confined charged fusion products will then be transferred via collisional processes to the background plasma to help maintain the plasma fuel at the temperatures needed to sustain the fusion reaction, a process which can be called self-heating. If the self-heating power matches or exceeds the rate energy is lost from the plasma, the fusion system has achieved ignition where in principle it is not necessary to supply additional external power to sustain the plasma. If the self-heating power is somewhat less than the power loss, burning plasma conditions can be sustained by supplying additional external power.

The ratio of the fusion reaction power to the auxiliary (ohmic + neutral beam + radio frequency) heating power needed to sustain the plasma is often denoted by the fusion power gain Q. At ignition, Q equals infinity, however a value of at least 15 is probably needed for efficient energy production. The dynamics of a burning plasma may best be characterized by the ratio of the alpha heating power to the total heating power ($f_\alpha$=Q/(5+Q)). The alpha heating power exceeds the auxiliary heating power when Q>5, which may be taken as the lower end of the burning plasma regime.

A burning plasma experiment will open up new scientific and technological issues that are of interest for the entire field of fusion science. For example, all burning plasma concepts have to develop methods of controlling the burn and handling the exhaust power produced by the burning plasma. Such an experiment has the primary objective of investigating new physics processes arising from dominant self-heating by 3.5 MeV alpha particles born from the DT fusion reaction. These processes will assume new characteristics as compared to existing experiments due to:

(I) The presence of a large population of isotropic energetic particles near to or exceeding the Alfvén velocity,
(II) The extension of transport, MHD and other phenomena in present-day devices to dimensionless scales needed for accessing the burning plasma regime, and
(III) The close coupling expected between alpha particle heating, plasma confinement and MHD stability in an alpha dominated regime.

In case of (I) present day experiment gives rather optimistic preliminary results as experiments in several machines show that the energetic particle pressure can be above that expected in the burning regime, and frequently either Alfvén instabilities are not excited, or if excited do not spoil the energetic particle confinement. However, these results are incomplete as theory indicates that there are sensitive magnetic shear effects (particularly relevant to advanced tokamak mode scenarios) and the isotropy, effective collisionality, $\rho^*$ and ratio of the particle velocity to the Alfvén velocity of energetic particles is known to be an important factor in both the linear threshold and non-linear saturation of Alfvén waves driven by the energetic particles. There is still considerable uncertainty regarding the stability and influence of collective effects induced by an isotropic distribution of resonant alpha particles in a burning plasma regime - particularly at high plasma beta (such as in STs and ATs) - where perturbative linear theory breaks down. (C.f. Sec. 2.1) In (II), our present understanding is that all MFE concepts require a device size normalized to the ion gyroradius considerably larger than present experiments in order to attain the burning plasma regime. The scaling of fundamental transport, MHD and fast particle collective effects with device size or collisionality is a critical issue for investigation in a burning plasma experiment. In (III), the coupling of alpha heating to confinement and MHD stability expected in advanced operating regimes, together with the generic problem of controlling the plasma burn with low recirculating power (high $Q=P_\alpha/(P_{tot}-P_\alpha)$), represents a new frontier of investigation distinctly different from present externally driven experiments. The integration of all these issues needs to be addressed in the development of any MFE reactor concept. The strong consensus of the burning plasma group was that the science and technology of the field has progressed to the point where a burning plasma experiment is feasible and necessary in order to address these interrelated issues.

There was considerable discussion in our group on the extent to which a burning plasma experiment could address issues relevant to other MFE concepts. The essence of this discussion boiled down to whether any one burning plasma experiment could shorten the development path for an MFE reactor based on an alternative concept. It was the consensus of our group that progress in understanding burning plasma physics for any one MFE concept would aid in the development and validation of theoretical models applicable to other magnetic concepts. However, the transferability of knowledge gained from an experiment would strongly depend on flexibility of the device for investigating a range of operational regimes, as well as a comprehensive diagnostic set for advancing our basic physics understanding of burning plasma phenomena. It was noted that the development of an advanced diagnostics set, together with methods for plasma control under burning plasma conditions, constitute a highly transferable knowledge base for other MFE burning plasma experiments.

The physics issues for a burning plasma experiment were divided into five topical areas: Energetic Particles, Transport, Macroscopic Stability, Power and particle handling, Plasma Control and Integration. As issues were discussed, they were divided into three categorize:

(I) Issues that can be addressed by present experiments,

(II) Issues associated with new dimensionless scales needed to access the burning plasma regime, and
(III) Issues specific to the effects of alpha particles and self-heating.

The goal of our group was to reach consensus on both what constitutes physics issues unique to the burning plasma regime, and on the capability of candidate experiments to address these issues. It was the consensus of the group that the physics of transport and MHD stability at scales needed to access the burning plasma regime could not be fully resolved in present day facilities and should be an important part of the scientific mission of a burning plasma experiment. Also, there was strong agreement that alpha heating and burn control methods could not be simulated adequately in existing devices, so that a burning plasma experiment was considered essential in order to make progress in the development of burn control techniques. What follows is a discussion of important physics issues which can be addressed in a burning plasma experiment.

[Note: Discussion of physics issues to be addressed by specific machine proposals (JET-Upgrade, IGNITOR, FIRE, RC-ITER) will be presented in Section 4 on opportunities in burning plasmas. We should point out however that the JET-Upgrade, while not expected to approach dominant alpha heating, will enable the investigation of a range of alpha physics and alpha heating issues in the near term.]

### 2.1 Energetic Particles

*Generic Alpha Particle Heating Issues*

In the D-T fusion reaction (the one which is easiest to access and with the highest energy yields compared to any other fusion reaction) the fusion products are neutrons and alpha particles. The neutrons, which carry four times more fusion energy than the alphas, cannot be confined in the plasma, but can have its energy absorbed by solid (or perhaps liquid) walls with its energy, through thermal conversion, then used as a power source. Below we will assume that the fuel of the fusion reaction is D-T so that the charged fusion product is the alpha particle. Other fuels are of interest in more advanced fusion systems, such as D-3He. However, the confinement quality of the plasma required to access the self-heating regimes of advanced fuels is substantially higher than in D-T. Thus the demonstration that self-heating fractions above one half can be achieved D-T fuel is also a key to the possibility of demonstrating the feasibility of alternative fusion fuel cycles.

In order to operate in a burn mode it is necessary to be able to classically confine the alpha particle orbits, which are much larger than the background particle orbits. In tokamaks the rule of thumb is that plasma current needs approximately 3 MA to avoid significant prompt loss. In concepts like RFPs and FRCs, where the ratio of toroidal to poloidal magnetic is smaller than in a tokamak, the orbit widths are smaller as well, and therefore the plasma current required for alpha particle containment is somewhat smaller. Still one can roughly take 3 MA as the minimum plasma current in a variety of plasma concepts needed to contain the 3.5 MeV alpha particle of the D-T reaction. The plasma current is not crucial in a stellarator and instead one needs to translate to an equivalent magnetic flux that is needed to contain 3.5 MeV alpha particles (it should be noted that the three dimensional magnetic fields of a stellarator further complicates the classical confinement issue). However, single particle effects are generally well understood and calculable in arbitrary geometry and magnetic confinement concept. As such, single particle effects fell into the category of issues which could be addressed in present facilities.

Every fusion device begins with a start-up regime where heating power is dominated by external or possibly ohmic heating. In this regime it is necessary to attain a high enough temperature for the fusion reaction to be significant and for the charged energetic alpha particle population to heat the background plasma by transferring energy primarily to electrons, although some energy is delivered to the D-T fuel ions as well. If the overall energy confinement time is greater than the electron-ion energy exchange time, the background temperature of the D-T fuel will be close to the electron temperature. For conventional tokamak operating regimes, such as the L- and H-Mode, this is typical. If sufficiently good overall plasma confinement characteristics ($n\tau \sim 4 \times 10^{20}$ $sm^{-3}$) and central ion temperature (10-20 keV) are achieved, the fusion self-heating will be strong enough to dominate external and ohmic heating.

It may be necessary to feedback on the external heating rate so that the plasma is not over-heated once the self-heating becomes significant. If self-heating is not too dominant, feedback using variable external heating levels may be a means of controlling of the character of the achieved plasma. If ignition is attained, there will be an important dynamic process by which the plasma achieves its equilibrium state. There is the possibility of thermal instability where the plasma pressure could rise above plasma stability limits or that the plasma could expand beyond its physical containment regions. Thus evolution from the time self-heating sets in to the final quasi-steady state (that is ultimately achieved after the thermal instability saturates) is a sophisticated nonlinear problem. This problem can be further complicated by the possibility that MHD instabilities set in during the evolution of a thermal instability.

The “guidance” of plasma heating can be quite important. For example, it is known in tokamaks that confinement properties are related to the heating rate, a property that is also characteristic of other devices (e.g. stellarator). In such case, the plasma characteristics achieved can be quite sensitive to how the mixture of external and self-heating power is applied as the parameters of the plasma evolve. Often burning plasma scenarios attempt to achieve burning plasma characteristics with minimal applied external heating power to create a hot spot that will allow the propagation of the self-heating region. Such planning will have to be done taking into account the coupling of transport mechanisms of the background plasma with the total heating power and the possibility of induced energetic particle induced instabilities that will be discussed below.

*Destabilization Due to Energetic Particles*

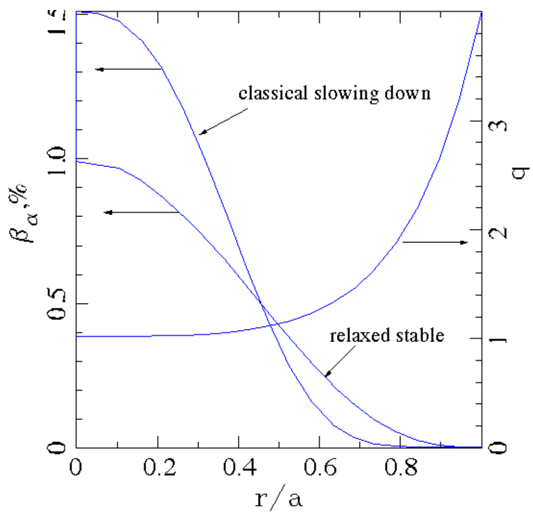


Fig. 2.1.1. Alpha particle slowing down beta profile vs. minor radius for a circular equilibrium with parameters typical of a compact high field plasma: B=10T, R=2.5m, a=0.5m, ne(0)=5x$10^{14}$cm-3, T(0)=20keV, and a 50-50 d-t mixture. The $\beta\alpha$ profile is predicted to be unstable according to the HINST code. However, the relaxed profile is stable. (Figure courtesy of N.N. Gorelenkov).

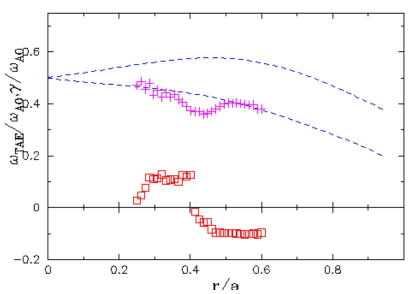


Figure 2.1.2 Calculated linear growth rate (squares) and frequency (crosses) indicate large ($\gamma/\omega$~20%!) linear growth rates for the unstable profile in Fig. 2.1.1.

In magnetically confined plasmas the dominant instability mechanism for a slowing down distribution of fast ions arises from the free energy of expansion of the fast ion profile. This mechanism is active if the phase velocity of a wave is less than the diamagnetic velocity of a particular species in plasma. As the diamagnetic velocity scales as the energy of the species, there will be a wide band of waves that that can be potentially destabilized by alpha particles. Every concept in magnetic fusion will need to account for this drive as it causes instability if the damping mechanisms of the background plasma are sufficiently small and the alpha particle beta is sufficiently high. When this instability is activated it causes a spatial expansion of the alpha particles.

The result of the instability may lead to an alteration of the heating profile, which can affect self-heating scenarios that are used to plan for a burn. Perhaps even more crucial is whether the instabilities cause alpha particle loss before their energy is absorbed by the plasma. This process could substantially change the fraction of self-heating. It may also introduce damaging plasma wall interactions, particular if it leads to a significant flux of ~3.5 MeV alpha particles on the wall.

There may be enough phase space available for the alpha particles to relax to a stable profile without significant alpha particle loss. Such relatively benign phenomena (as well phenomena which induce alpha particle loss) have been observed from energetic particle induced Alfvénic instabilities in tokamaks and stellarators. An example of how relaxation might be accommodated is shown in figure's 2.1.1 and 2.1.2. These figures show the results of an Alfvén instability calculation from the HINST code, using plasma parameters suitable for a high beta compact high field experiment (although these modes are relevant to AT regimes and STs as well). Observe that the nominal alpha particle distribution that forms from classical processes leads to a quite strong instability (called the Energetic Particle Mode, which is an Alfvén-like instability that exists only because energetic particles are present). As with the Toroidal Alfvén Eigenmode (TAE) the dominant drive for these modes comes from the passing alpha particles. These modes are relevant both for AT regimes and high beta ST plasmas. However, if the alpha particles are allowed to relax by spreading radially, the same number of alpha particles can remain in the machine and the distribution can still be stable. Still we cannot be sure that the distribution of alphas doesn't over-relax to cause significant direct loss. The extent to which these instabilities can be studied in present devices in regimes relevant to a burning plasma experiment is limited because of the expected difference in isotropy, parallel velocity, and relative size of the system to gyro-radius of the energetic particles.

Clearly more work is needed in order to quantitatively apply experimental and theoretical results to a burning plasma. There is much experimental data which has not been systematically studied, theoretical stability calculations still need further improvement and the nonlinear theory for understanding the effects of energetic particle instabilities is just beginning development. Further, it is important to note that empirical extrapolation of the results of intermediate (normalized) size experiments to burning plasma experiments will still leave uncertainty because the ratio of radial scale size to alpha particle orbit width in burning experiments will generally be appreciably larger than in existing intermediate size scale experiments. As a consequence there may be a lower excitation threshold for global diffusion of energetic particles in a burning plasma experiment than in intermediate size experiments. Also, there may be a turbulent “sea” of high-n modes which is not encountered in present intermediate size experiments.

Another important effect of energetic alpha particles is the modification of ideal and non-ideal MHD modes. This effect can be both stabilizing or destabilizing. As often the burning plasma regime will be close to instability threshold, the effect of energetic particles on the stability limits can be quite significant. When these effects are destabilizing, there will be a penalty in the accessible operating space. Even when the effects are stabilizing it is possible that one gets too much stabilization. This circumstance can arise if the new stability regimes allows the plasma to reach a new and previously inaccessible unstable configuration which then causes a more virulent relaxation process than would otherwise occurs. An observed example of this is the detrimental effects caused by the giant sawtooth instability, after the normal and mildly relaxing sawtooth instability was stabilized by energetic particles.

## 2.2 Transport

The radial transport of particles and heat is a crucial factor in determining whether a self-heated burning plasma can be attained and/or sustained. Transport, together with global stability, determine whether the fusion triple product (density)(temperature)(energy confinement time) will be sufficient for ignition in a given machine.

As issues were raised in the discussion of transport, the group categorized them into three areas. These were (1) issues that can be addressed by present experiments, (2) issues associated with the large scales typical of proposed burning-plasma experiments, and (3) issues specific to burning plasmas. Here system size is expressed as plasma minor radius normalized to ion gyroradius, a parameter which increases with magnetic field as well as plasma minor radius (large machines may be either compact, high field or physically large). In the following, we describe issues from categories (2) and (3) in accordance with the charge for this group.

The degree of confidence in extrapolating from present experiments (using dimensionless parameters $\beta^*$, $\rho^*$,$\nu^*$, $Z_{eff}$, etc.) to the new regimes required to achieve a high gain burn or ignition is a fundamental issue. This extrapolation should be reliable for the standard operating scenario of the device, and sufficient device flexibility is needed to access other advanced modes of operation where extrapolation is more speculative.

Significant effort has gone into developing models of plasma confinement, based on fundamental physics and exploiting the rapid growth of available computing power, to increase confidence in this extrapolation. Physics-based numerical models, while incomplete and augmented with empirical scalings where necessary, now reproduce existing confinement data in standard L- and H-Mode operating regimes with error comparable to or better than empirical scaling alone.[*]

Extrapolation from present experiments using empirical scaling or physics-based models is subject to the following fundamental sensitivity. The ratio of fusion reaction power to auxiliary heating power, Q, the important figure of merit for energy applications, is a sensitive function of the energy confinement time. In present designs the projected Q, which must exceed roughly 5 for self-heating to dominate over external heating, falls on the steep part of the curve. This places an emphasis on the accuracy of confinement projections. Present scatter in the empirical database allows for 30% variations in extrapolated confinement times, while present (incomplete) physics-based models reproduce global confinement times in the same L- and H-Mode database to within roughly 15% to 30% rms error. The accessibility and maintenance of H-mode regimes at large system size is another area of uncertainty. Even for relatively conventional H-Mode operation, there is at present a factor of two uncertainty in extrapolated H-mode threshold power for burning plasma experiments.

Present designs which expect to operate in the standard ELMy H-mode regime are projected to achieve Q ~ 10 using empirical scaling, assuming confinement times fall within roughly 10% of standard ELMy H-mode scaling H98P(y). While this represents a factor of 2 to 2.5 (in most cases) better confinement than predicted by ITER89P L-mode scaling, it is nominal H-mode confinement, routinely achieved in present experiments. Compact high magnetic field experiments such as IGNITOR, while perhaps not best described in relation to ITER L-mode scaling, requires a confinement time close to ITER89P projections [for density $\approx 10^{21}$ $m^{-3}$ and $T_i \approx 12$ keV]. For all proposals, if confinement were 30% worse than existing projections, Q would fall out of the range where heating by $\alpha$-particles dominate over auxiliary heating. The uncertainty in where exactly future experiments will fall on the Q vs. confinement curve is a fundamental issue. One of the major objectives of a burning plasma experiment is to benchmark

[*] See MFE transport and turbulence report in these proceedings.

existing model based and empirical projections to reactor scale plasmas. The issue of our technical readiness to proceed with a burning plasma experiment is discussed further in Section 3.

It was pointed out that we have reached the point where incomplete, theory-based models correctly reproduce certain features of experiments not described by present empirical scalings. These include the ability to simulate new enhanced confinement regimes, the dependence of confinement on edge influx, the varying strength of isotopic mass scaling in different confinement regimes, improved confinement with toroidal rotation, improved confinement with impurity injection and in radiative mantle discharges, profile stiffness and sensitivity to edge temperature, and weak or favorable scaling of confinement with heating power in some regimes. Given the success of theory based models in a range of regimes relevant to burning plasmas, there was agreement that some combination of empirical scaling and physics-based models should be used to make projections to burning-plasma experiments. However, because some physics-based models predict sufficient confinement to achieve dominant self heating while others do not, there is an obvious need for further work to resolve differences in the models.

The size scaling of the beneficial effects of $E_r$ shear on confinement is another important issue motivated by theory based models. Recent results on existing devices reveal the effect of radial electric field ($E_r$) shear in suppressing plasma turbulence, giving rise to a range of improved confinement regimes both with and without transport barriers. This becomes relevant in view of recent work which shows that $E_r$ shear can be a 30-100% effect in the bulk core of conventional L- and H-mode plasmas, and can explain the observed improved confinement with isotopic mass. With the inclusion of flow shear, the scaling of confinement with size becomes a subtle issue which can be tied to isotope scaling. Present theory suggests that the ratio of the shearing rate to the growth rate, describing turbulence suppression in the absence of transport barriers, scales with $\rho^*$ in the absence of external flow drive. In large machines (small $\rho^*$), $E_r$ shear may no longer offset the expected “gyro-Bohm” scaling of heat conductivity with $\rho^*$. If this theory is correct, this would unmask an unfavorable isotope scaling of confinement with $\rho^*$, giving an $A^{-0.5}$ dependence. The result would be a 12% reduction in confinement in 50:50 D-T mixtures relative to present deuterium experiments on which extrapolations are based. Accordingly, the favorable mass dependence, typical of empirical scalings used to project confinement times from present experiments, may be optimistic for large machines. It has not been conclusively determined in experiments whether or not transport has an explicit dependence on $\rho^*$, and the more subtle underlying role of $E_r$ shear may be responsible for this. Potential size scaling issues such as this provide important area of opportunity for a large scale experiment.

Theoretical notions on the size scaling of transport barriers, including the edge barrier characterizing standard H-mode operation, suggest that transport barriers may not scale well with machine size. Two separate issues come into play as described below.

First, attaining transport barriers through diamagnetically driven (intrinsic) $E_r$ shear may be more difficult in large machines because the ratio of the $E_r$ shearing rate to the growth rate is expected to scale with ρ*. This argument has generally been made in reference to H-Mode (edge), but is thought to apply to internal transport barriers as well. We point out, however, transport barriers may also be initiated by spontaneously generated (bipolar) zonal flows, or by steady turbulence-generated (Reynolds stress) flows. These potential mechanisms are little understood, leaving open the possibility that they scale more favorably with machine size. Once the transport barrier forms, it may well be sustained by diamagnetically driven flows, as follows. The ratio of the shearing rate from diamagnetically driven $E_r$ shear to electrostatic growth rate in a barrier would scale as (L/W)ρ*, where L is the system size and W is the transport barrier width, much smaller than L. As discussed in the next paragraph, the ratio W/L is expected to scale with ρ*, eliminating the adverse ρ* dependence in (L/W)ρ*. Accordingly, transport barriers, once formed, may be self-sustaining even in large machines because the criterion for turbulence suppression by intrinsic $E_r$ shear, i.e., the ratio of shearing rate to growth rate in the barrier, should have no size scaling (provided other necessary conditions are also met). Very little data exists to test these ideas in relation to size scaling.

Second, if transport barriers are attained, a small amount of evidence exists to suggest that their pressure gradients are limited by magnetohydrodynamic (ballooning) stability. Heuristically, their widths W may scale as several tens of gyroradii or with poloidal gryoradius, so that W/L ~ ρ*. A small amount of data from H-mode plasmas was presented at the meeting to support the possible scaling of pedestal pressure with the product of the critical pressure gradient for which ballooning modes go unstable, the poloidal gyroradius, and a function of plasma shape. The data show good correlation with this scaling, but the slope of the line is different for each machine. Data from ASDEX-U, C-MOD, DIII-D, JET, and JT-60U are all well-correlated, but the slope varies by a factor of roughly eight. This variation was suggested to arise from differences in edge-localized modes, differences in magnetic shear, and other factors. Accordingly, the pressure at the top of the barrier (or pedestal pressure) acquires a scaling with gyroradius, and the barrier width relative to the plasma cross-section would then scale with ρ*. This would amount to relatively small H-mode pedestals in large machines with the caveat that the large variation between existing machines is still not well understood. Because the global confinement is sensitive to edge temperature in some models, a ρ* scaling may significantly impact Q. Strong shaping (high triangularity), adopted in several designs, is generally recognized as increasing the pedestal height, most likely due to an increase in the threshold pressure gradient for onset of edge localized modes (ELMs).

For internal barriers supported by intrinsic equilibrium $E_r$ shear, the impact of a pessimistic scaling of height with ρ* may not be so great, and internal barriers could nevertheless result in significant improvements in core fusion power. Very little data is available from present experiments to support or contradict the expected scalings of transport barriers with size, however, leaving the question open for resolution by a future large-scale experiment.

Many tokamaks with neutral beam heating have consistently demonstrated enhanced confinement in regimes where the temperature of the deuterium (and, in JET and TFTR, tritium) fuel ions greatly exceeds that of the electrons. In some of these regimes, the higher ion temperature is due not only to dominant external heating of the ions, but to improved ion thermal confinement relative to that of the electrons. Empirically and theoretically, confinement improves with the ratio of ion to electron temperature. On the other hand, it is commonly believed that hot-ion operation is not compatible with self-heating by fusion alphas, which primarily heat the electrons. However, it has been shown [*e.g.* J.F. Clarke, Nuclear Fusion, **20** (1980) 563] that ignition with ion temperatures greater than the electron temperature is possible if the ions are sufficiently well confined. Such energetically favorable conditions are not restricted to low densities because both alpha-heating power and ion-electron coupling power vary as the square of the density. In addition, recent predictive studies [*e.g.* M. Bell, APS-DPP í98] suggest that hot-ion plasmas with Q ~ 10 may be obtained in tokamaks of modest size. Models for tokamak confinement based on intrinsic flow-shear suppression of ion temperature gradient driven turbulence would tend to support the view that such modes of operation are not limited to neutral beam heated plasmas. When this point was raised, some members of the group questioned the relevance of hot-ion plasmas. Their view was that high density is required to attain good plasma purity and reactivity, and that operation with hot ions would require much higher temperatures than foreseen necessary in certain high density experiments. Accordingly, general agreement was not reached on the relevance of hot-ion operation.

Another possibility would be to operate in configurations which access second stability to ion temperature gradient modes, by having large Shafranov shift, for example. Clearly the physics of enhanced confinement regimes and the role of spontaneous vs. externally induced flow shear needed to attain and sustain advanced confinement regimes represents a fascinating area for future investigation.

It is the consensus of our subgroup that a burning plasma experiment needs to access and investigate advanced confinement regimes in order to gain deeper understanding of the burning plasma state and to develop improved fusion energy concepts beyond the “standard” operating regime of the device. Indeed, such an experiment would help to motivate the development of transport control techniques in order to maintain advanced confinement regimes compatible with low recirculating power.

Some generic confinement issues for burning plasmas:

(i) Identification of self-consistent operating regimes with good confinement/stability/bootstrap fraction/bootstrap alignment/edge compatibility. (Bootstrap current optimization applies only to certain advanced modes of operation.)
(ii) Burn control techniques to maintain desired operating point and minimize large thermal excursions.
(iii) Transport control tools required to create and sustain steady state plasma with low recirculating power.
(iv) Exhaust requirements for minimizing accumulated He ash in the plasma core.
(v) Compatibility of advanced operating regimes with techniques for minimizing high heat loads (such as radiative mantle or radiative divertor methods).

(vi) Core edge coupling and the penetration and confinement of impurities generated by high wall/divertor heat loads.

Any one burning plasma experiment which can address these issues will provide an invaluable database for advancing our understanding of burning plasma issues in a range of concepts. However, in order to address these issues, conditions for dominant alpha self-heating are necessary, and at present these conditions are best determined for a tokamak burning plasma. In the long term these developments will help to reduce the degree of uncertainty in extrapolation to burning plasma experiments. However, current uncertainties in transport projections should not prevent us from taking the next step towards a tokamak burning plasma experiment, based on conventional operating regimes with advanced tokamak capability. Indeed, a burning plasma experiment was recognized as a way of making significant progress on outstanding confinement issues at large (dimensionless) scale. One of the challenges for a burning plasma experiment is to develop advanced diagnostics coupled with sufficient device flexibility to produce major advances in our understanding of turbulence and transport. Emphasis on a strong scientific mission for a burning plasma experiment is expected to benefit the entire fusion program.

### 2.3 Macroscopic stability

The development of plasma configuration which simultaneously achieves high-energy confinement at high plasma pressure is one of the driving objectives of the MFE program. These two requirements are essential for achieving fusion energy production in a sufficiently compact device for it to be economically viable. A range of advanced confinement regimes has been identified in tokamaks in which particle and energy confinement approach neoclassical levels in the plasma core, and highly insulating thermal barriers spontaneously occur. In spite of the fortuitous development in confinement physics, MFE configurations still face major obstacles to economic fusion energy production. The major limitation of most presently envisioned configurations are imposed by MHD stability limits which constrains plasma pressure and hence fusion power density. These limits tend to become more severe in enhanced confinement regimes with strong thermal transport barriers. MHD limits can result in profile relaxation or violent transient events (disruptions) which produce peak wall heat loads much greater than expected during normal operation.

The quest to overcome MHD limits on power density and disruptivity has spurred much innovation in fusion research. As with turbulence and transport, the study of MHD beta limits, and methods to maintain operation near those limits in any one configuration advances our fundamental understanding of MHD phenomena in many other configurations. Some generic macroscopic MHD issues for burning plasmas include:

1. Establish and maintain steady state equilibrium on a global resistive time scale sufficient for accessing regimes of dominant alpha heating.

2. Develop startup and shut down scenarios with dominant alpha self-heating which allow access to high gain operating regimes while avoiding disruptive instabilities.
3. Develop efficient methods for feedback control at high Q.
4. Identify the effect of alpha particles on plasma stability and their impact on plasma confinement and equilibrium profiles.

In this section we consider specific stability issues which can be addressed in a tokamak burning plasma experiment as it is the one configuration with the greatest near term potential for a burning plasma. Such a burning plasma experiment will most likely be designed to operate in a conventional confinement regime where the extrapolation from present experiments is most reliable. The integration of plasma self-heating with transport, MHD and burn control issues constitutes the primary mission for such a device. For a tokamak this will most likely be the ELMy H-mode regime (ITER-RC, FIRE) or some slightly enhanced L-mode regime such as IGNITOR. However, a well known criticism of the conventional tokamak is that its confinement and MHD stability limits makes economic power production and engineering design difficult. The challenge for advanced tokamaks is to operate at plasma pressure sufficient to maintain high fusion power and high bootstrap current for times longer than the skin current penetration time. It is thought that the combination of strong shaping, proper alignment of the pressure and current profiles, active feedback on instabilities and profile control to avoid stability limits, constitutes the set of tools which need to be developed in order to optimize the advanced tokamak reactor concept.

*Specific Issues for tokamak burning plasmas*

An important issue for the conventional H-mode or L-mode operating regime is the role of neoclassical tearing modes (NTM) in limiting plasma performance at high $\beta_{pol}$. The NTM mode is a metastable mode for beta values that exceed a minimum beta, i. e. if the minimum beta is exceeded; it is possible for an NTM to appear. However, this does not mean that an NTM will appear if the minimum beta is exceeded. The actual stability threshold depends upon island width threshold and seed island formation physics which are still lively topics for debate. A simple naive scaling of the minimum beta for instability to exist, would predict a scaling, with $\rho/a$ which would extrapolate to lower beta in larger tokamak devices. However, the extrapolation to low $\rho/a$ depends on the relative scaling between the critical beta and the seed island size; the naive scaling assumes this is fixed, but there is no basis for such an assumption. Also, reduced dynamical coupling between rational surfaces with increasing Lundquist number S is predicted to raise the critical beta at high S and low $\rho/a$. The $\rho/a$ scaling of the critical beta for unstable modes therefore remains an outstanding issue. This scaling is essentially empirical and is based on a limited parameter range which may by masking dependencies on other variables. There was some earlier theoretical work to support this scaling, but that theory is no longer widely supported and it remains an open question. The extrapolation in collisionality from existing devices is weak. It is doubtful that the scaling uncertainties can be satisfactorily resolved in current experiments. Nevertheless, this issue is being addressed on two fronts. For a conventional tokamak burning plasma, the discharge duration before an NTM is likely, which is essentially on the order of the current evolution time scale, can be optimized operationally as is done in most tokamaks - TFTR for example did this very successfully. In the longer term, research is ongoing at DIII-D and ASDEX-U on avoiding NTMs by eliminating seed islands, which has had some success, and by active control of the islands once they form using ECH and ECCD.

Disruption mitigation is also an active area of research because successful disruption avoidance or controlled plasma shut down can remove much of the engineering complexity from a tokamak reactor. A burning plasma experiment will be an invaluable tool for testing a range of techniques for disruption mitigation in a burning plasma environment.

Finally, various theories indicate that alpha particle modification of ideal MHD stability can have a major impact on plasma operation. The sawtooth instability in L-mode and H-mode plasmas can be stabilized by energetic particles to produce monster sawteeth. This is an important issue for several reasons. First the sawtooth event can generate seed islands which trigger the onset of NTMs. Second, the giant sawtooth can lead to large transient alpha particle and thermal plasma heat loads on the walls. Other instabilities are FLR modifications of ballooning modes at high temperatures and their resonant interaction with circulating or trapped alpha particles. These resonances typically have the effect of destabilizing the modes below the ideal stability limit when the diamagnetic frequency exceeds the mode frequency. These and other non-ideal effects relevant to fusion plasmas, their effect of thermal and fast particle confinement and role in determining plasma pressure profiles, will be major area of investigation in a burning plasma experiment.

*Advanced tokamak capability*

A fundamental issue for AT regimes is whether self consistent steady state high Q profiles can be maintained with high bootstrap fraction of the total current. Raising the plasma beta in order to raise self driven currents and fusion power density requires a combination of techniques from edge configuration control, to internal profile control and active feedback on the time scale of the instability. In the area of configuration control, it is widely accepted that high elongation and triangularity improve stability and plasma confinement (the latter through an elevation of pedestal temperatures). In particular, the gains from increased triangularity come from:

(1) Higher current and therefore higher beta for a given $\beta_N$
(2) Higher $\beta_N$
(3) A synergistic effect between higher beta and the gain in stability from broader pressure profiles
(4) Better bootstrap alignment.

The addition of stronger shaping has already been incorporated into the RC-ITER design in recognition of the need to access advanced operating regimes in a burning plasma experiment. However, other methods of profile control needed to avoid operational beta limits are only now being developed for existing experiments. Active feedback on the MHD instability is even more speculative at this stage, so that a burning tokamak experiment should not aim at advanced tokamak operation as its target (standard) discharge. The long term challenge is to develop profile control and MHD feedback techniques which raise beta limits in steady state using only a small fraction of the total power production. A burning plasma experiment would provide the motivation necessary to address a variety of control issues crucial for the development of viable fusion reactor concepts.

It is important to mention that the techniques developed for enhancing MHD stability (configuration, profile, MHD control) compatible with low recirculating power will have general applicability to other concepts even though the detailed physics may differ from one device to another. It is therefore important to make sure that a burning plasma experiment has sufficient flexibility to explore and develop the technology of plasma control required for a future fusion reactor.

## 2.4 Power and particle handling

A key issue generic to all MFE reactor concepts is the development of effective power and particle handling solutions at the plasma boundary compatible with the required confinement and macroscopic stability of the core plasma in order to maintain high fusion gain. The device closest to addressing these issues of core-edge coupling at the appropriate scale is the Tokamak. It was the consensus of the MFE Plasma Boundary and Particle Control working group that a reasonable basis exists for a steady-state tokamak divertor solution at high density (collisional edge, detached divertor).

Major issues remain for a viable divertor design for AT and alternate concepts. It is well recognized that present tokamaks cannot sustain improved (AT) modes of operation with high edge density, while low edge density presents severe problems for impurity accumulation and prevents detachment. Boundary control methods which maintain acceptable central impurity content *at low edge collisionality* is an important research area for investigation on a burning plasma experiment. Such methods developed for advanced confinement regimes on a tokamak burning plasma will be transferable to other advanced confinement concepts.

*Tritium retention*

Although the high edge density detached divertor solution is acceptable for ash control and power handling during ELMs, there are remaining concerns regarding the lifetime of the limiter to disruptions and the retention of tritium in carbon tiles designed to withstand the peak heat loads occurring during disruptions. Divertor solutions may exist using high-Z materials which would eliminate the tritium retention issue, but the major concern with these is their ability to withstand high transient heat loads.

Graphitic first wall materials are the simplest solution for the high heat flux during transient events since C does not melt. (C sublimates at high heat loads leaving behind a sound C substrate. This is unlike metals that can melt and leave areas of poor thermal conductivity unsuitable for subsequent discharges.) But, the problem of tritium retention due to codeposition with redeposited C presents a formidable challenge that must be overcome. ITER estimates result in 1kg of tritium trapped in the wall after 100 full-length DT discharges. The low retention assumed for these estimates has not been observed experimentally and retention could well be much higher. Removal of tritium from codeposited C is problematic both because of the thick codeposited layers that can result from normal operation and because the saturation of hydrogen in C is about 0.4 atomic fraction. (For comparison, metals do not produce the thick codeposited layers and the saturation is typically $10^{-6}$ atomic fraction.) The methods proven effective for removing tritium from C involve oxidation of the codeposited layer or physical removal, these are expensive to implement and may produce collateral damage.

The recent development of copper-backed W first wall materials by the ITER R&D program offer the promise of handling steady-state power loads of 25 $MW/m^2$ without the tritium retention problem posed by C. However, the ability of such W materials to withstand the high heat flux during transients without suffering damaging melting is yet unclear and may require disruption mitigation like killer pellets, liquid jets or large gas puffs. The investigation of disruption mitigation techniques with advanced divertor concepts compatible with conditions needed to sustain the plasma burn is one of the key areas of investigation for a burning plasma experiment.

### 2.5 Plasma Control and Physics Integration

The complex interplay between the plasma self-heating and the confinement and stability processes in a burning plasma gives rise to dynamics that may be appreciably different from those in externally controlled plasmas with dominant auxiliary heating. For example, not only will the fusion heating profile be dictated by the plasma profile but also the plasma will respond to the self-generated heating by modifying the profiles of temperature, density, current profile, and flows, in turn modifying the heating profile. Such self-heating dynamics constitute a new and essential area for scientific study and are primary motivations and justifications for the study of burning plasmas.

Control of self-heated plasmas will be more complex than in conventional auxiliary-heated tokamaks, particularly in advanced performance regimes. The self-heating due to the fusion alpha particles is not as flexibly controlled as auxiliary heating in present devices; this is particularly significant for the control of advanced performance plasmas, wherein the profiles of the plasma pressure, current density and flows strongly influence the confinement and stability of the plasma. Due to the lack of external control of the heating profile, control of the plasma pressure profile will likely be most optimally performed by external control of transport profiles, for example by injection of momentum to induce localized gradients of the plasma flow. In such high performance plasmas, transport barriers not only adjust the pressure profile, but the modified profiles influence the transport profile and the discharge stability. In addition, energetic fusion products can destabilize MHD modes, potentially decreasing the effectiveness of the alpha heating; conversely, energetic particles can also stabilize certain types of MHD activity, leading to prolonged periods of stability followed by larger relaxation oscillations that can challenge the sustainment of the plasma.

As an example of the revised dynamics of reactor-scale plasmas (before, during, and after fusion burn), consider the contrast between present-day tokamak plasmas which are heated by auxiliary systems and reactor-scale plasmas. The start-up phase of a reactor-scale tokamak plasma discharge (during which the plasma is initiated and evolved with strong external control and little self-heating) involves the growth of the plasma in a sequence that seeks to achieve confinement enhancement and avoid large-scale instabilities. Both of these processes involve extensions beyond the dimensionless size of present-day devices. For example, achieving enhanced performance by having sufficient power passing through the plasma edge (e.g., exceeding the H-Mode Power Threshold and causing a transition to a high confinement "H-mode") is challenging in burning plasma devices due to the scaling of the threshold with plasma size (ie, in terms of $a/\rho_i$). This demands start-up sequences that achieve the H-Mode at low density and subsequently increase the density at a rate such that the growing fusion power can support the edge transport barrier at the increased edge density. As another example, avoidance of locked-modes is another challenge for larger devices since locked modes are triggered by somewhat lower error fields in reactor-scale plasmas than in present tokamaks.

Similarly, following the fusion burn phase, the ramp-down of a reactor-scale plasma requires careful programming so as to avoid disruptive terminations. If rapid plasma terminations ("disruptions") occur in reactor-scale high-current plasmas, the magnetic flux change due to the current decay is sufficient to create avalanches of energetic electrons (runaway electrons for which the collisional drag is insufficient to stop acceleration of the electrons to multi-MeV energies). Such conversions of plasma current carried by thermal electrons to current carried by super-MeV electrons is a new phenomenon in reactor-scale multi-mega amp plasmas, necessitating new plasma termination control actions.

A simulation of start-up, burn and termination that exhibits both good and poor control features is shown in Fig. 2.5.1 for the FIRE design, and was presented during the Snowmass discussions on burning plasmas. The 30 MW of fast wave ICRH is applied as a square-wave during the current and density rise (flat-top current is reached at 5 s and the burn density a 10 s). Because the power across the separatrix exceeds the H-mode threshold, an L-H transition occurs during the rise and the improved confinement leads to ignition. In this simulation, the full auxiliary power is left on until 12 s, leading to a significant overshoot in the fusion power. During this overshoot beta limits are exceeded. Other cases show that the overshoot and beta limits can be avoided by carefully controlling the transition from auxiliary to fusion heating. How this transition proceeds is strongly dependent on both the global and local transport characteristics and would likely require feedback control on the auxiliary power. Because good helium ash pumping in the divertor is assumed in this simulation, the plasma eventually settles in to a

10 s quasi-steady burn producing 200 MW of fusion power. Without helium pumping the plasma quenches prior to the start of the ramp-down phase at 27.5 s. Because the heating is applied during the current rise, a hollow current profile with a large reverse shear region is maintained throughout the burn. Although no enhanced transport is assumed in the reverse shear region, this type of start-up would be an attractive prelude to a non-inductively driven advanced tokamak phase. At about 13 s the power across the separatrix falls below the L-H transition threshold. The H-L back transition can be avoided if there is at least ~35% hysteresis factor, as assumed in the example. Without the hysteresis, ~10-20 MW of auxiliary heating would have to be supplied to maintain the H-mode and a reasonable fusion power output. The termination consists of simultaneously decreasing the plasma density and current. Because the plasma is still hot as the current is ramped down, the core current remains nearly frozen and a large reverse current is generated in the plasma edge. This eventually leads to a loss of equilibrium in the simulation, and likely a disruption. Many possibilities exist to avoid this situation, but they require experimental investigation in burning plasmas. Thermal quench should precede the current ramp-down to reduce the resistive skin time and allow the current to be terminated in a controlled fashion. The thermal quench can be facilitated through an H-L back transition, density ramp-down, injection of strongly radiating impurities, or in the case of a plasma with resistive magnets, decompression through toroidal field ramp-down. All of these options would require feedback control because the timing and rates are functions of the plasma burn conditions that would vary considerably in a burning plasma experiment.

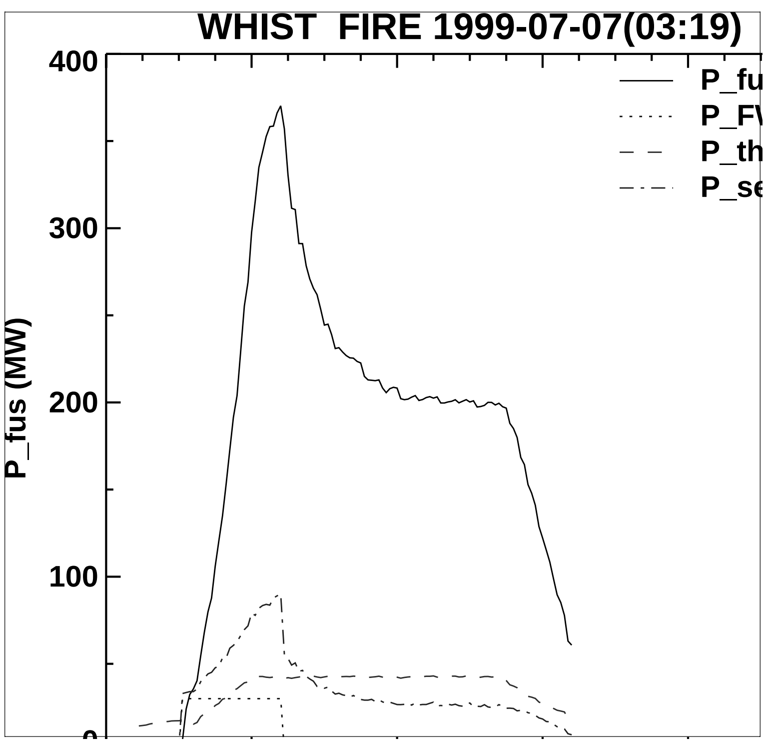


Fig. 2.5.1. FIRE example of start-up, burn and termination.

The new physical phenomena and the modifications of present-day plasma behavior (due to energetic particles, self-heating, and larger system size) will significantly change the dynamics of reactor-scale burning plasmas and lead to both scientific discoveries and integration challenges. Such new effects and their integration provide compelling motivations for the study of burning plasmas both for research and for achievement of the long-awaited state of a terrestrial burning plasma.

## 3. Technical Readiness for a Burning Plasma Experiment.

The tokamak is technically ready for a high-gain burning plasma experiment. There was extensive discussion of the challenges to a successful experiment. Nevertheless the great majority of the participants expressed their opinion in support of technical readiness.

### 3.1 Background and Process

As the most successful and most developed magnetic fusion configuration, the tokamak has been proposed as the vehicle by which to proceed to investigation of burning plasma physics. The ITER conceptual and engineering design activities, for example, have explored the details of one concept of a burning plasma experiment, indeed, an ambitious proposal that would include extensive technological as well as physics research objectives. The 1997 FESAC review of EDA-ITER, while pointing out a number of important technical concerns, recognized the major achievements that had been made in designing a buildable tokamak reactor experiment. The central objective of the EDA-ITER design was to achieve long pulse plasma operation at fusion-reactor conditions (high fusion gain).

Since a decision to fund EDA-ITER has not been forthcoming, current attention has focussed on identifying reduced mission, lower cost, paths to burning plasmas. The resulting hiatus has given an opportunity to consider again how well established is the tokamak's technical readiness to proceed. The Burning Plasma subgroup devoted considerable time to a discussion of this question and arranged for a plenary evening session to which all Snowmass participants were invited. Technical readiness is, of course, in large measure a matter of professional opinion, balancing risks, costs and benefits. Participants naturally had differing opinions on the relative weighting of these factors. After extensive discussion, a poll was taken of support for the statement prepared by the working subgroup **“The tokamak is technically ready for a high-gain burning plasma experiment. ”** The meeting voted 53 for, 12 against: an 81% majority in support of the statement. Other modified versions were explored in the interests of trying to find an even greater level of consensus, but while versions of the statement with additional caveats satisfied many of the dissenters, the revisions lost more of the original supporters than they gained of the original dissenters.

### 3.2 Key Challenges

The key problems faced by a burning plasma experiment that were discussed may be summarized as follows.

### 3.2.1 Confinement

The experimental confinement database must be extrapolated to predict the confinement of a next step burning plasma device. The extrapolation in confinement time to small $\rho^*$ is substantial for all the proposals and may be subject to additional uncertainties such as the dependence of confinement on normalized density $n/n_G$ (more relevant for modest field devices such as ITER) and the proximity of the proposed operating point to the H-L power threshold (more relevant for high field compact designs such as FIRE). Although IGNITOR does not face similar issues, it still requires some enhancement over L-mode confinement projections in order to achieve ignition. There is no absolute certainty that the level of confinement required for a particular Q value will be obtained for any of these devices. An important part of the scientific mission for a burning plasma experiment is to resolve these confinement issues at reactor relevant scales. This is an experiment, and the confinement performance is part of the experiment.

Nevertheless, the required confinement for each proposal is well known. It is generally within the upper range of established empirical confinement time scalings obtained from current experiments.

Part of the critique of EDA ITER was based on numerical transport models, some of which have a strong dependence on the edge pedestal temperature, together with a theory of how the H-mode pedestal height will scale. This pedestal scaling theory takes the height to be given by the product of a width that scales like the poloidal gyro radius and a slope that is given by ballooning theory stability. If this theory were correct, ITER would be predicted to have a rather low pedestal height and poor confinement. The experimental evidence on both the slope and width questions is contradictory: some supporting and some apparently inconsistent with the theory's predictions. The status of the theory is thus presently controversial. Moreover, the scaling is a sensitive function of triangularity, which is higher in the proposals now actively under consideration, in part as a response to this criticism. Some proponents of these pedestal concerns feel that their significance warrants qualification of the readiness statement. The majority, while acknowledging the importance of a fundamental understanding of the pedestal, do not feel the uncertainties warrant this qualification. Additional uncertainties also exist. One critical one is the threshold for obtaining improved confinement (H-mode). This topic is discussed extensively in sec. 2.2 on transport issues.

### 3.2.2 Disruptions

Tokamaks have to be built to withstand disruptions because one cannot guarantee disruptions will not happen. The stresses produced in a disruption increase with magnetic field but not specifically with size. Accommodating these stresses is an important problem of structural design but satisfactory solutions have been devised. A particular concern is the possibility of generation of energetic runaways during disruptions in large gyro-size (radius/larmor radius) devices. Mitigation via gas, or liquid jets is considered feasible.

3.2.3 Plasma Facing components

It is now considered unsatisfactory to use all carbon plasma facing components because of tritium retention. The alternatives pose problems for the extreme pulsed heat loads of disruptions and possibly ELMs because they melt rather than subliming. This might lead to runaway thermal distortion of the plasma facing components arising from progressive misalignment of melted regions. However, ITER has adopted a strategy with a mixture of materials that is believed to be satisfactory.The key uncertainty is then of the influence of erosion and tritium retention on divertor lifetime. ITER has therefore built in the ability for relatively rapid refurbishment of the divertor. Other proposals have not developed so detailed a solution, although they propose to use heavy metals. The issue of tritium retention in an extended period of DT operation remains an urgent problem for any burning plasma design that incorporates carbon in plasma facing components.

The dissipative divertor technique used to radiate a large fraction of the divertor heat load is proven only for relatively high plasma densities. Therefore, as detailed in the Edge Plasma subgroup report, technical readiness can only be regarded as established for this regime, not necessarily for a lower density regime that might be of interest for an Advanced Tokamak burning plasma experiment.

3.3 Metrics

The Burning Plasma group heard from the Metrics working subgroup who have developed a set of quantitative metrics for proceeding to the next development level, in the case of the tokamak, Fusion Development, which we interpret as the Burning Plasma step. We broadly agree with the metrics we saw. It appears that the conventional tokamak operating regimes satisfy those metrics, at least individually.

3.4 Advanced Tokamak

By “Advanced Tokamak” we mean high-beta, high-bootstrap fraction tokamak relevant to steady-state operation, with the tools to explore active profile control and stabilization. There is general agreement that this highly promising approach, under active development in existing facilities, is not yet technically ready to provide the basis for a burning plasma experiment. However, a burning plasma tokamak experiment must have the flexibility to be able to incorporate Advanced Tokamak operation, because if the present experiments are successful in demonstrating sustained AT operation, it is likely to become the optimum tokamak reactor concept. Copper-conductor tokamaks under consideration for a compact high-field burning plasma experiment, do not have the steady-state operational capability of superconducting designs with current drive capability, but can still be designed with sufficient pulse length to explore phenomena on a current relaxation time-scale critical to sustainment.

## 4. Burning Plasma Physics Opportunities

Numerous opportunities for pursuing burning plasma physics were identified. These could be categorized as:

- integrated modeling of physics effects that occur over a wide range of time scales
- near term experiments that push energetic alpha effects beyond those that have been achieved thus far (JET-Upgrade)
- intermediate to long term experiments based on relatively mature tokamak designs (IGNITOR, FIRE, and ITER-RC)
- confinement concepts other than the standard tokamak (DTST and MTF)

The experimental opportunities in particular largely reflect the views of the proposed projects or individuals associated with those projects. There was insufficient time at Snowmass to adequately evaluate the various projects against a common set of criteria. That type of evaluation can only be achieved by a much more thorough review that encompasses benefits, risks and cost.

### 4.1 Integrated Transport Modeling

Transport codes that couple time-dependent evolution of the one-dimensional radial (1-D) fluid particle, energy, magnetic flux, and possibility the momentum equations, with two-dimensional (2-D) MHD equilibria, provide the means for integrating many of the physic models into a more comprehensive, consistent treatment. These 1-1/2-D time dependent codes, are the primary tools for interpreting and predicting macroscopic toroidal plasma behavior. They are useful for:

- Scoping out the dynamics of access to attractive operating regimes

- Evaluating the capabilities of auxiliary heating, fueling and current drive systems to exploit these scenarios

- Identifying and avoiding the 'hurdles' of operation (e.g., density limits, tolerance to impurities, L-H transition, etc)

- Evaluating confinement times with consistent profiles

There are a number of similar codes available with varying emphases on different aspects of the plasma and employing various approximate (from empirical to a combination of empirical and theory-based) confinement models. In predicting the operating characteristics of a given proposed machine, they invariably show that the facility has a wide range of possible operation, similar to the capabilities of present experimental facilities. There are many additional control 'knobs' in these codes that incorporate diagnostic, feedback, and source characteristics that extend beyond the capabilities of any given device. Therefore, it takes a considerable number of cases to fully explore the range of possible operating conditions and these are invariably reduced to a very few reference cases for illustration in published reports. By the same token, there are aspects of plasma behavior observed in present facilities that are not fully explained by the existing theoretical models (e.g., internal and edge barrier formation and dynamics). Therefore, the simulations still cannot be viewed as an adequate substitute for experiments.

There are many operational aspects of burning plasmas that were identified by 1-1/2-D transport as being relevant to the performance evaluation of the various proposed facilities. The major challenge to fusion community is to enhance the physics basis of modeling codes for each of the component models so the predictive performance assessments of operating characteristics are more reliable. These were discussed in Section 4.2 and will not be repeated. Some additional operational characteristics of startup and shutdown are summarized here.

It was noted at Snowmass that the plasma current redistribution time , $\tau_{cr}$, would be long enough in burning plasmas so that reverse shear conditions could be generated without non-inductive current drive during the current ramp up phase. The generation of reverse shear conditions could follow the prescription used in most present experiments: heat during the current ramp to freeze in the current profile. In IGNITOR, the current relaxation time is a few times longer than the burn pulse length, while in FIRE the two times are about equal and in RC-ITER the burn pulse for inductive operation is about twice the current relaxation time. This means that startup conditions in these machines can be used to avoid sawtooth activity (at high edge q), at least for a substantial fraction of the burn pulse time. The long resistive time also enters into neoclassical MHD considerations. How these conditions interact with fast alpha dynamics and MHD stability considerations could be a major part of the research modeling development activities for any of the proposed devices.

Plasma termination was identified as another area needing significant development attention. With strong self-heating and reduced or non-existent external control of the plasma heating, the options for a controlled shutdown are reduced. Decompression, impurity pellet or gas injection, and fuel burnout have been proposed, but need further examination.

## 4.2 JET Upgrade*

### 4.2.1 Introduction

Record fusion power (16MW) and fusion energy yield (22MJ) have been achieved in JET during the DTE1 campaign in 1997 with an ELM-free H-mode and with an ELMy H-mode. Alpha heating has been observed with alpha power in the range of 1.2MW. Significant fusion yield (up to 8MW) has been achieved with advanced scenarios. However, it is felt that more significant burning plasma physics issues could be addressed in a JET upgrade.

JET is under-powered as compared to other machines such as ASDEX-U, DIII-D and JT-60U. With its present power capability [16–18 MW of Neutral Beam Injection (NBI) power and up to 10 MW of Ion Cyclotron Resonance Heating (ICRH) power in ELMy plasmas], JET has achieved $\beta_N$ values up to 1.3 and 2 at a magnetic field of 3.4T, respectively in ELMy and optimized shear plasmas.

JET performances can be significantly improved by increasing: i) the plasma volume (increase $I_p$, increase Q), ii) the plasma triangularity (higher density), and iii) the additional power (up to 40 MW to access high performance regimes, up to 50MW to assess beta limits). This section describes the objectives and design parameters of a possible JET upgrade.

### 4.2.2 Objectives of a JET upgrade

The main objectives of a JET upgrade are to:

- Increased power capability allowing access to high confinement modes and assessments of beta limits at full field.
- Increased plasma volume allowing increased plasma current and fusion gain.
- Increased plasma shaping allowing operation at higher densities and increases in the ELM-free period of ELM-free H-modes.

Increasing plasma volume, plasma shaping and power capability would allow: i) significantly increased JET performance, ii) reduced errors in Next Step extrapolation, iii) operation at much higher values of fusion yield and alpha heating power. These would allow JET to tackle some burning plasma physics issues, which are needed to progress towards a fusion reactor.

In a JET upgrade, the alpha power might range from steady-state 4 MW up to transient 14 MW as compared to the transient 1.2 MW in the alpha heating experiment of DTE1 where $P_\alpha / (P_{add} - P_{Fusion}) \sim 0.2$. Although the plasma will not be dominated by alpha heating since Q will reach, at best, 2 transiently, a much more complete assessment of the alpha heating can be done than in DTE1.

* For references and additional details see the section on contributed papers.

Extrapolations have been made for the ELMy H-mode (steady-state), the Optimized Shear mode (steady-state) and the ELM-free H-mode (transient). The presently achieved fusion yield in the JET DTE1 could be multiplied by a factor up to 4. The following burning plasma physics issues can be explored:

- substantial heating by alpha particles ($0.5 < Q < 2$);
- energetic particles instabilities, in particular in high $T_e$ plasmas obtained with ERCH and optimized shear plasmas;
- the domain $\rho^*/\beta_N$ can be significantly increased;
- beta limits at full field;
- extrapolation uncertainties for ITER scaling can be substantially reduced and scaling of advanced scenarios at full field can be done.

In addition, the remote handling capability allows flexibility with the divertor and to testing different choices of first wall material. Relatively modest upgrades of the JET facility would allow substantial progress in burning plasma physics issues in a time scale which is much shorter than the time required to build and operate a larger, more powerful tokamak such as ITER/RC.

### 4.2.3 Design parameters for a JET upgrade

Several options for power upgrades have been considered for the period 2000-2002, but not yet decided, by the new EFDA JET sub-committee. These are in addition to the ICRH wideband matching system which might allow an increase in the total combined power by 2–3 MW in the presence of ELMs. The first priority is to upgrade the 80kV power supply of one NBI box up to 130kV allowing the NBI power to be increased by 6–7 MW. Other upgrade options, which have not been considered, could include one or more of the following:

- a third positive (or negative) NBI box delivering 10 to 15MW;
- an Electron Cyclotron Resonance Heating System (ECRH) making use of the recent technical developments and delivering 10 MW in the 140 GHz range;
- developing techniques that allow the voltage handling of the ICRH antennae to increase. If not successful, two additional antennae could be installed in the torus allowing full use of the RF power plant;
- an in-situ ionizing system in front of the Lower Hybrid Current Drive (LHCD) launcher, allowing an increase in the coupling and making full use of the LHCD plant.

With the present divertor configuration, the plasma volume is limited to 80-85 $m^3$, elongation (b/a) to 1.9, triangularity ($\delta$) defined at the separatrix, to 0.35 and the plasma current to 4.5 MA at 4 T. In JET, as in other machines, it has been found that beta increases with triangularity both in ELMy H-modes and in advanced scenarios. Moreover, the density normalized to the Greenwald density can be significantly increased by increasing triangularity for a similar confinement.

Reference pulses have been taken from the JET database. The result is shown in Table 4.2.1 for the steady ELMy H-mode and for the transient ELM-free H-mode. It can be seen that the main effect of increasing the plasma volume is to increase the fusion gain Q. It can be shown that for similar β, q and ν*, $Q_{th}$ scales as $B^3 \times (a^3/R)^{5/4}$ assuming a gyro-Bohm scaling. Therefore, an increase of minor radius by 15% increases Q by 1.7. Increasing triangularity allows operation at higher density while still keeping a good confinement. Increasing power allows operation at higher beta.

Extrapolation of the optimized shear scenarios is more difficult in the absence of established scaling laws. In JET, comparison of an ELMy H-mode with an optimized shear plasmas at similar magnetic field (3.4 T), plasma current (3.5 MA) and additional power (25–28 MW) shows an increase of $\beta_N$ by a factor of 1.3 and a doubling of the fusion yield. Therefore, pending further development work, the increase in fusion yield can be taken as proportional to $\beta_N^2$, therefore the fusion yield increases by a factor 1.7. From extrapolations made in Table 4.2.1, a fusion gain of almost 1 with $P_{in} = 37$ MW and $\beta_N = 2.5$ with $P_{in} = 50$ MW could be achieved in a quasi steady-state advanced scenario.

*Table 4.2.1: Possible JET Upgrade Parameters*

| | **Steady-State ELMy H-mode** | | | **Transient ELM-free H-mode** | | |
|---|---|---|---|---|---|---|
| | **Ref pulse 42982 V = 83m$^3$ δ = 0.22** | **V = 106m$^3$ δ = 0.57** | | **Ref pulse 42976 V = 85m$^3$ δ = 0.57** | **V = 106m$^3$ δ = 0.57** | |
| $B_t$ (T) | 3.86 | 4 | 4 | 3.66 | 4 | 4 |
| $I_P$ (MA) | 3.27 | 6 | 6 | 4 | 6 | 6 |
| $P_{in}$ (MW) | 24.5 | 37 | 50 | 25.6 | 37 | 50 |
| $n/n_G$ | 0.56 | 0.7 | 0.7 | 0.29 | 0.5 | 0.5 |
| $T_{io}$ (keV) | 7.4 | 8.6 | 9.7 | 26 | 20 | 21.3 |
| $Z_{eff}$ | 2.4 | 1.8 | 2.0 | 2.6 | 1.75 | 1.9 |
| $\beta_N$ | 1.3 | 1.7 | 1.9 | 2.04 | 2.5 | 2.7 |
| $P_{Fus}^{th}$ (MW) | 1.65 | 15.2 | 15.8 | 9.5 | 63 | 64.7 |
| $P_{Fus}^{tot}$ (MW) | 4.4 | 21 | 21.9 | 16 | 71.7 | 73.5 |
| $Q_{tot}$ | 0.18 | 0.57 | 0.44 | 0.63 | 1.94 | 1.47 |

## 4.3 IGNITOR*

### 4.3.1 Introduction

IGNITOR is part of a line of research that began with the Alcator machine at MIT in the 1970's, which pioneered the high magnetic field approach to plasma magnetic confinement and has been continued by the Alcator C/C-Mod and the FT/FTU series of experiments. The idea for a high field D-T ignition experiment was formulated at about the same time. The high field approach also allows a possible development path to tritium-poor, low-neutron-production fusion, based on D-$^3$He or perhaps some form of "catalyzed" D-D reactions, which could yield a different kind of fusion reactor. This section outlines the objectives and design considerations of Ignitor.

### 4.3.2 Objectives of the IGNITOR experiment

*Approach to ignition*

For a high field experiment with a high plasma current, transient effects can be exploited to use ohmic heating to reach ignition. This is a major factor used in the Ignitor experiment. When the current ramp phase is considered, the plasma current is increased by adding "skin layers" of current to the outer surface of the plasma column that do not have time to diffuse inward.

*Density control*

The prediction and control of the density profile at high densities is another important transport and edge plasma physics problem to be addressed by IGNITOR. The basic shape of the density profile cannot be reliably predicted from present knowledge. Peaked density profiles are more favorable for ignition, although the level of degradation with flatter profiles is relatively small, as long as the total number of particles remains roughly the same. The question of the degree of profile control (peaking) by pellet injection, which translates to the question of the penetration of the pellet particles into the plasma, remains open. Edge density control during both startup and steady state is also important, since it regulates the current penetration rate as well as being related to the edge temperature.

*Burn control*

Development of burn control techniques is one of the major areas of investigation for Ignitor. Transport simulation readily demonstrates that precise time-dependent burn control through variation of the bulk ion density source is not possible in general, since particle confinement times are generally longer than the energy confinement time. Much better control is possible by operating in a slightly sub-ignited state that is driven by a small amount of externally supplied heating. This may be the preferred method for a

* For references and additional details see the section on contributed papers.

reactor and would be an important demonstration on the path to a reactor that could be performed in an ignition experiment.

Emergency methods of burn control to investigate in an ignition experiment include the firing of large “killer” pellets (Ar, Li, etc.) into the plasma to rapidly quench run-away ignition conditions and prevent or mitigate possible disruptions. This method has been adopted in Ignitor. The effects of introducing a large amount of impurities on ignition in the following discharges should also be studied.

Fusion reactions with low rates of neutron production (“advanced fuel” D-$^3$He or possibly D-D) may be a more attractive reaction for a reactor than the D-T reaction, which produces 80% of its energy in an energetic neutron. These reactions have their own set of problems, such as the source of quantities of $^3$He and the attainment of the higher plasma parameters required for burning. To begin to explore their possibilities, however, a D-T burning plasma experiment at high field is much closer to the required parameters than present-day tokamaks and would allow initial studies at the level of approximately 1 MW of power in charged particles from the D-$^3$He reaction cycle for beam-injected $^3$He or somewhat less for thermal $^3$He in a D-T plasma.

4.3.3 Design parameters for IGNITOR

Ignitor uses high toroidal magnetic field in a compact size, which allows ignition at relatively low fusion power levels and low plasma beta, with relatively strong ohmic heating. The practical goal of an ignition experiment is to reach the ideal ignition temperature at which fusion heating begins to dominate the bremsstrahlung radiation losses (peak temperature $T_o$~ 6 keV for typical centrally peaked profiles), under conditions in which the fusion heating can continue to rise.

The requirement of high toroidal field leads to an interlocking set of characteristics favorable for ignition. High field is most easily achieved at small major radius R, since the vacuum field varies approximately as $B_T$~1/R. Small R and tight aspect ratio leads to small overall size and thus lower cost. High toroidal field allows a relatively high plasma current, toroidal current density, and poloidal magnetic field to be supported. In Ignitor, the mean poloidal field is ≈3.75 T. Also, there is a large paramagnetic current ≈10 MA at the low $\beta$ of ignition and this increases the central $B_T$ by ≈1 T.

High toroidal field supports a high plasma density with $n<n_G=I_p/\pi a^2$, where $n_o$ is correlated empirically to $B_T/R$ or to current density. In Ignitor, densities $n_{eo}$ ~$10^{21}$ m$^{-3}$ should be possible, based on the $B_T/R$ obtained by Alcators A and C, FT and FTU, and TFTR. Alcator C obtained $n_o \approx 2.10^{21}$ m$^{-3}$ at $B_T$=12.5 T. If the maximum density instead correlates with the volume-averaged current density, this should allow $n_{eo} \approx 10^{21}$m$^{-3}$. Therefore, based on the required confinement for ignition for 50:50 D-T plasma, only a moderate energy confinement time $\tau_E \approx$ 0.4 sec is required.

As a consequence, such plasmas have:

• High levels of ohmic heating up to ignition ($P_{OH}$ is high due to high $B_p$).
• Good confinement of plasma energy and particles (empirical scalings indicate $\tau_{E,L} \sim I_p$)
• Good confinement of fast fusion alpha-particles. ($I_p > 6$ MA will give good central confinement.)
• Low temperature ignition ($T_{eo} \approx T_{io} < 15$ keV in Ignitor) at relatively low levels of fusion heating ($P_\alpha < 2P_{OH}$).
• Ignition at low $\beta_p$.
• Low $\beta_p$ reduces the required fusion power and the thermal wall loading
• Clean plasmas (since $Z_{eff}$ is a monotonically decreasing function of density).
• High plasma edge densities confine impurities to the scrape off layer ("cold plasma blanket"), as line radiation helps to evenly distribute the wall loading.

In addition, high field and the ability to ignite at low β gives the capacity for a broad range of operating conditions at less-than-maximum parameters.

These characteristics avoid or reduce the need for:

• Injected heating, except to control plasma stability, to extend the operating range, and as a backup to ignition.
• Access to H-mode.
• Current drive to control q-profile.
• Divertors, which concentrate the thermal wall loading on small regions.

IGNITOR uses high toroidal magnetic field in a compact size, which allows ignition at relatively low fusion power levels and low plasma beta, with relatively strong ohmic heating. The basic parameters of the Ignitor are given in Table 1. Flattop periods vary significantly with $B_T$, ranging from 4 sec at 13 T (reference value) to 10--15 sec at 9--10 T. Ignition scenarios at varying density are given in Table 2.

Table1: IGNITOR Reference Design Parameters

| | |
|---|---|
| major radius ($R_o$) | 1.32 m |
| minor radius (a, b) | 0.47m, 0.86m |
| aspect ratio (A) | 2.8 |
| elongation (κ ) | 1.83 |
| triangularity (δ) | 0.43 |
| toroidal field ($B_T$ ) | ≤13 T |
| toroidal current ($I_p$) | ≤12 MA |
| mean poloidal field $B_p \sim I_p/5\sqrt{ab}$ | ≤ 3.75 T |
| edge safety factor q | 3.6 |
| magnetic flux swing | 36 Vs |
| plasma volume | ~10 $m^3$ |
| plasma surface | ~36 $m^2$ |
| auxiliary heating $P_{RF}$ | 18--24 MW |

## 4 .4 Fusion Ignition Research Experiment (FIRE)*

### 4.4.1 Introduction

The mission of FIRE is to attain, explore, understand and optimize alpha-dominated plasmas that will provide the knowledge for the design of attractive MFE systems. The guiding design philosophy is that FIRE must have the capability and flexibility of studying and resolving the physics issues relevant to the design of a subsequent advanced integrated fusion facility. A major consideration is to accomplish this physics mission at the lowest possible cost, with a target cost <$1B. FIRE is a physics experiment to extend the frontiers of fusion plasma physics into previously unexplored parameter space using advanced capabilities and flexibility for later upgrades; it is not intended to be a demonstration of the scientific and technological feasibility of magnetic fusion.

### 4.4.2 Physics Objectives

The physics objectives of FIRE are to:

1. Determine and understand the conditions required to achieve alpha-dominated plasmas:
   - Energy confinement scaling with dominant-alpha heating
   - β-limits with dominant-alpha heating
   - Density limit scaling with dominant-alpha heating
2. Explore the dynamics of alpha-dominated plasmas using active control techniques.
3. Sustain alpha-dominated plasmas with high-power-density exhaust of plasma particles and energy and alpha ash exhaust in regimes suitable for future toroidal reactors.
4. Explore and understand alpha-dominated plasmas in advanced operating modes and configurations that have the potential to lead to attractive fusion applications.
5. Understand the effects of fast alpha particles on plasma behavior in relevant regimes.

*Phase I objectives:*

To access the alpha-dominated heating regime with a minimum self heating fraction of ≥ 0.5. This objective is based on projections from the middle of the present tokamak performance database. This would provide a test bed where alpha heating effects are easily observable, and the plasma dynamics could still be controlled externally.

*Phase II objectives:*

To achieve strongly alpha-dominated plasmas with self heating fraction $f_\alpha$ = 0.66 to 0.83. This level of performance is projected from the best results of the present tokamak performance database, or by a modest 20% improvement in confinement from employing advanced tokamak physics that is expected to be developed by the ongoing base tokamak program over the next 5 years.

The pulse length, or the burn time, is a very important consideration for any burning plasma experiment. The physics time scales of interest (with typical values for FIRE plasmas) are:

* For details and additional references see the section on contributed papers.

- $\tau_{\alpha S}$, the time needed for the alpha particle to transfer its energy to the plasma (~0.1s)
- $\tau_E$, the plasma energy confinement time (~ 0.6s)
- $\tau_{He}$, the confinement time of alpha ash, slowed down alpha particles (~ 5 $\tau_E$ ~ 3s)
- $\tau_{cr}$, the time for the plasma current profile to redistribute after a perturbation (~13 s)

It is important to recognize that the characteristic time scales for plasma phenomena in FIRE are significantly shorter than the corresponding time scales on ITER-RC due to the smaller size, higher density and somewhat lower plasma temperature as shown in Table 4.4.1.

**Table 4.4.1: Characteristic inductive plasma time scales in FIRE and ITER-RC.**

| For Q ≈ 10 (full $I_p$ and $B_T$) | $\tau_E$ (s) | $\tau_{He}$ (s) | $\tau_{cr}$ (s) | $\tau_{burn}$ (s) |
|---|---|---|---|---|
| FIRE | 0.6 | 3 | ~13 | 15 |
| ITER-RC | 2.5 | 7.5 | ~200 | 450 |

A FIRE plasma with a burn time of 10 s ( ~ 15 $\tau_E$) would allow the pressure profile to come into equilibrium with alpha heating and allow the alpha ash to accumulate for ~ 3 $\tau_{He}$. This pulse length would be sufficient to address Physics Objectives 1, 2, 3, and 5. A significant part of Physics Objective 4 could also be accomplished using a current profile that is only partially redistributed. In fact, it would be advantageous to establish a variety of plasma current profiles using current ramping as in present advanced tokamak experiments. A pulse length of ~30 s would be sufficient to allow the bootstrap driven current in an advanced tokamak mode to come into equilibrium. These pulse length requirements match the capabilities of liquid nitrogen (LN) cooled copper coils, which can be designed to allow a burn time of 10 to 20s at full toroidal field. If advanced tokamak physics improves confinement relative to ITER design guidelines by 25% and _ by 50%, then the toroidal field and plasma current can be reduced by 25% while maintaining high plasma performance (e.g., Q ~ 10). This small reduction in the field of the FIRE copper magnet cooled to LN temperatures would allow the magnetic flat top to be increased to 30 to 40s.

### 4.4.3 FIRE Device Parameters for Initial Evaluation

The FIRE plasma configuration is an extension of the advanced tokamak programs on DIII-D and Alcator C-Mod, and is a ≈ 1/3 scale model of ARIES-RS, the present vision for an advanced tokamak fusion reactor. The FIRE plasma has a size and shape very similar to the previously proposed advanced tokamak (TPX), with the added capability of high performance D-T operation. FIRE will have the flexibility to incorporate new innovations as the ongoing advanced tokamak program develops them. The parameters summarized in Table 4.4.2 were chosen as likely to achieve the FIRE mission at the lowest cost based on results of prior design studies for burning plasmas experiments (CIT, BPX and BPX-AT), as well as recent information from the ITER-EDA and ITER-RC design activities.

**Table 4.4.2: Basic Parameters and Features of FIRE**

| | |
|---|---|
| R, major radius | 2.0 m |
| a, minor radius | 0.525 m |

| | |
|---|---|
| κ95, plasma elongation at 95% flux surface | ~1.8 |
| δ95, plasma triangularity at 95% flux surface | ~0.4 |
| q95, plasma safety factor at 95% flux surface | >3 |
| Bt, toroidal magnetic field | 10 T with 16 coils, < 0.4% ripple @ OuterMP |
| Toroidal magnet energy | 3.7 GJ |
| Ip, plasma current | ~6.5 MA |
| Magnetic field flat top, burn time | ≥10 s ( =21 s at 10 T, Pfusion ~ 200 MW) |
| Pulse Repetition time | 2 hr |
| ICRF heating power, maximum | 30 MW |
| Neutral beam heating | None |
| Lower Hybrid Current Drive | None in baseline, upgrade for AT |
| Plasma Fueling | Pellet injection (≥2.5km/s vertical launch inside mag axis, possible guided slower speed pellets) |
| First wall materials | Be tiles, no carbon |
| First wall cooling | Inertial between pulses |
| Divertor configuration | Double null, fixed X point, detached mode |
| Divertor plate | W rods on Cu backing plate (ITER R&D) |
| Divertor plate cooling | Inner plate-inertial, outer plate active - water |
| Fusion Power | ~200 MW |
| Fusion Power Density (plasma) | ~10 MW m-3 |
| Neutron wall loading | ~ 3 MW m-2 |
| Lifetime Fusion Production | 5 TJ (BPX had 6.5 TJ) |
| Total pulses at full field/power | 3,000 (same as BPX), 30,000 at 2/3 Bt and Ip |
| Tritium site inventory | TBD (< 30 g) |

Possibility of upgrading to 12T and 7.7 MA with a 12 s flat top has been confirmed and is discussed in the Interim Engineering Report (http://fire.pppl.gov).

*Capability for Alpha-Dominated Burning Plasma Experiments on FIRE*

The plasma performance of FIRE is estimated using the guidelines similar to those used to project the performance of ITER. The primary considerations are the maximum density limit, plasma energy confinement, the maximum pressure (β) limit, the power threshold for accessing the high confinement mode (ELMy H-mode) and limitations imposed by impurities due either to alpha ash accumulation or impurities from the first wall and divertor. FIRE assumes an operating density relative to the Greenwald density close to those in the ITER confinement database. FIRE assumes a slightly more peaked density profile (identical to that used in the CIT and BPX projections) than ITER due to the potential for tritium pellet injection into a much smaller high-density modest temperature plasma. FIRE also assumes lower impurity fractions characteristic of high-density tokamak plasmas. In particular, FIRE assumes no significant high-Z impurities in the plasma core from the divertor. The initial design point selected for FIRE satisfies all of the standard tokamak design guidelines needed to access the alpha dominated range with $P_a$ / $P_{heat}$ ≥ 0.5 (Q ≥ 5) and to sustain these conditions for > 10 $\tau_E$. This represents more than an order of magnitude advance beyond the capability of TFTR/JET to study alpha driven physics, and would provide a checkpoint more than half way to the alpha heating fraction $P_\alpha/P_{heat}$ ≥ 0.8 required in a fusion reactor.

## 4.5 ITER-RC*

### 4.5.1 Introduction

During the last year of the ITER EDA, it was decided that a redesign was necessary in order to retain the retain the original goals and objectives as much as possible but with a cost objective of about half that of the original EDA design. Several basic design options, corresponding to different choices of aspect ratio, have been considered, namely a high aspect-ratio machine (HAM, A~3.5), one with intermediate aspect-ratio (IAM, A~3.26) and one with relatively low aspect-ratio (LAM, A~2.76). The HAM design has been abandoned owing to relatively poor access, lower shaping capability, higher cost and limited potential for electron cyclotron heating and current drive. In this discussion we focus on the IAM and LAM designs.

### 4.5.2 Objectives

A. Plasma performance objectives:

- Achieve extended burn in inductively driven plasmas with the ratio of fusion power to auxiliary heating power of at least 10 for a range of operating scenarios and with duration sufficient to achieve stationary conditions on the time scales characteristic of plasma processes;

- Aim at demonstrating steady state operation using non-inductive current drive with the ratio of fusion power to input power for current drive of at least 5.

B. Engineering performance and testing objectives:

- Demonstrate the availability and integration of technologies essential for a fusion reactor (such as superconducting magnets and remote maintenance);

- Test components for a reactor (such as systems to exhaust power and particles from the plasma);

- Test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency, the extraction of high-grade heat, and electricity generation.

Note that the only significant change from the EDA objectives is the replacement of the requirement to achieve ignition with the requirement to achieve a high gain Q ~ 10 burn, although the possibility of achieving ignition is still held out as being desirable. It is this reduction in required performance that allows substantial size and therefore cost reductions to be realized.

### 4.5.3 Design parameters: IAM and LAM

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* For details and additional references see the section on contributed papers.

The main parameters of the IAM and LAM are presented in Table 1 and compared with the corresponding parameters of the EDA device as described in the Final Design Report (FDR). Note that the IAM design has higher field and lower current than LAM, and has somewhat less shaping. IAM plasma shapes are limited to single null configurations, whereas LAM can be operated either with a single null or an up-down symmetric double null equilibrium. A feature of the LAM design is that the field at the TF coils is low enough to permit use of NbTi conductor throughout the coil. Both designs meet the objective of lowering the construction cost by about a factor-of-two below the cost of the FDR ITER, while offering a performance level consistent with the revised objectives given above.

| | IAM | LAM | FDR |
|---|---|---|---|
| R(m) | 6.20 | 6.45 | 8.14 |
| a(m) | 1.90 | 2.33 | 2.8 |
| Plasma Configuration | Single Null | Single or Double Null | Single Null |
| IP(MA) (q95 = 3) | 13.3 | 17 | 21 |
| Bo (T) | 5.51 | 4.23 | 5.68 |
| Ignited/Burn Pulse Length (s) | 450 | 450 | 1000 |
| Elongation κ95, κX | 1.68, 1.83 | 1.74, 1.92 | 1.6, 1.75 |
| Ave δ triangularity, δX | 0.43 | 0.49 | 0.35 |
| <T> (keV) | 10.5 | 10.8 | 12 |
| $\langle n_e \rangle$ ($10^{20}$ $m^{-3}$) | 0.83 | 0.83 | 1.0 |
| $\langle n_e \rangle / n_{GW}$ | 0.87 | 0.83 | 1.17 |
| $Z_{eff}$ | 1.9 | 2.0 | 1.8 |
| Fusion Power (MW) | 505 | 525 | 1500 |
| $\beta$, $\beta_n$ (%) | 2.86, 2.25 | 3.88, 2.25 | 3, 2.2 |
| Ave Neutron Wall Load ($MW/m^2$) | 0.6 | 0.5 | 1.0 |
| Number of TF Coils | 18 | 20 | 20 |

Table 1. Main parameters of IAM and LAM and comparison to the FDR design.

An initial installation of about 75 MW of auxiliary power is planned, with 33 MW coming from negative ion neutral beams and 40 MW from RF H and CD. The latter will be injected through two ports and can be made up of 40 MW of a single H and CD band chosen from ICRF, ECRF or LHRF, or two different 20 MW systems chosen from these three bands. Port allocation allows an additional 40 MW to be added; in addition, some upgrade of the NBI power may be possible. Thus, as an experiment of this magnitude demands, there is a high degree of flexibility in both the choice of H&CD schemes and the total H&CD power.

*Access and Diagnostics*

While not as impressive as the access in the FDR ITER design, the access in both RC ITER design variants is exceptional by standards of today's large tokamaks. For example, the 18 equatorial ports in IAM have cross-sectional dimensions of 1.74 x 2.2 $m^2$, while the 20 equatorial ports in LAM measure 1.5 x 2.2 $m^2$. Such generous access is required by the demands of auxiliary heating, diagnostics and blanket module testing.

Important to a burning plasma experiment is the implementation of a comprehensive set of state-of-the-art diagnostics. Extensive planning for the diagnostics has been done for RC ITER. Ports have been allocated for each of these diagnostics and detailed design work has been done for many of them at a fairly detailed level, including the machine interface. It should be emphasized that RC ITER is, above all, a physics experiment and, as with any experiment, its value in providing physics understanding is strongly dependent on the scope and depth of the diagnostic coverage.

*Inductive performance*

Both the IAM and LAM have reasonable margin in obtaining their baseline performance operating below the Greenwald density and $\beta_n < 2.5$, but above the L-H transition scaling. Within nominal constraints, Q = 10 can be obtained in both machines with confinement degraded to as low as 80% of that predicted by extrapolation of the IPB98(y,1) H-mode scaling. Higher Q performance for both machines is possible, although the operating window naturally shrinks. As required by the RC ITER objectives, the possibility of ignition is not precluded but requires some enhancement over the H-Mode confinement scaling projection.

*Non-inductive performance*

Achieving steady-state with $Q \geq 5$ requires improvement in confinement and normalized $\beta$. Current drive performance is slightly better in IAM than in LAM but in both designs advanced tokamak operation is required to achieve the steady-state Q =5 goal. Assuming the current drive efficiency $nIR/P_{CD}$ scales linearly with temperature, and $\gamma^*$ is the current drive efficiency at T = 10 keV, then for example, with $\gamma^* = 0.2$ and $P_{CD} = 70$ MW, Q ~ 5 is possible with $H_H = 1.25$ and $\beta_n \sim 3.5$.

An important parameter regarding steady-state operation is the pulse length capability normalized to the L/R time, the characteristic time for decay of the electric field in the plasma. For fully superconducting machines such as RC ITER, the pulse length can be made arbitrarily long providing there is sufficient cooling capability to cope with nuclear heating and incidental coil heating due to variations in the plasma control power. In RC ITER, steady-state pulse lengths of an hour or more are anticipated, corresponding to several L/R times. The ability to produce truly steady-state conditions reflects an important advantage that well-shielded superconducting machines enjoy over relatively short pulse and poorly shielded compact, copper burning-plasma experiments.

# Contributed Papers