Burning Plasma Physics

Technical Subgroup of the Magnetic Fusion Concepts Working Group

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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/ρ_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 ($L/\rho i$ scaling of microturbulence, effects on transport barriers…

- **Stability**
  Non-ideal MHD effects at high L/ρ_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/ρ_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_α=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, ρ^* 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 ($n\tau$ ~4 $x10^{20}$ 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

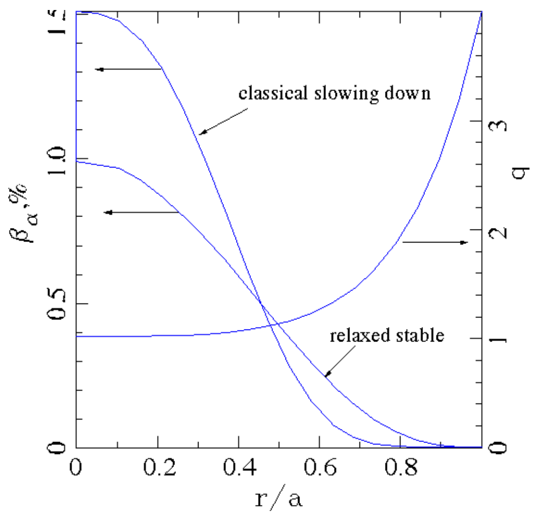

Fig. 2.1.1. Alpha particle slowing down beta profile vs. minor radius for a circular equilibrium with parameters typical of a compact high field plasma: B=10T, R=2.5m, a=0.5m, ne(0)=5x10[14]cm-3, T(0)=20keV, and a 50-50 d-t mixture. The $\beta\alpha$ profile is predicted to be unstable according to the HINST code. However, the relaxed profile is stable. (Figure courtesy of N.N. Gorelenkov).

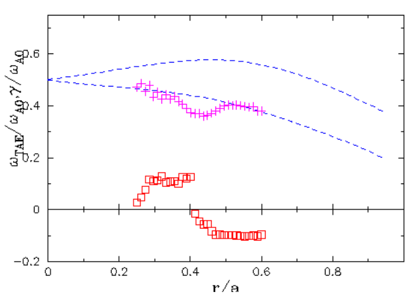

Figure 2.1.2 Calculated linear growth rate (squares) and frequency (crosses) indicate large (γ/ω~20%!) linear growth rates for the unstable profile in Fig. 2.1.1.

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 β^*, ρ^*,ν^*, 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 α-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 ρ^* in the absence of external flow drive. In large machines (small ρ^*), E_r shear may no longer offset the expected “gyro-Bohm” scaling of heat conductivity with ρ^*. If this theory is correct, this would unmask an unfavorable isotope scaling of confinement with ρ^*, 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 ρ^*, 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 β_{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 ρ/a which would extrapolate to lower beta in larger tokamak devices. However, the extrapolation to low ρ/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 ρ/a. The ρ/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 β_N
(2) Higher β_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/ρ_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.

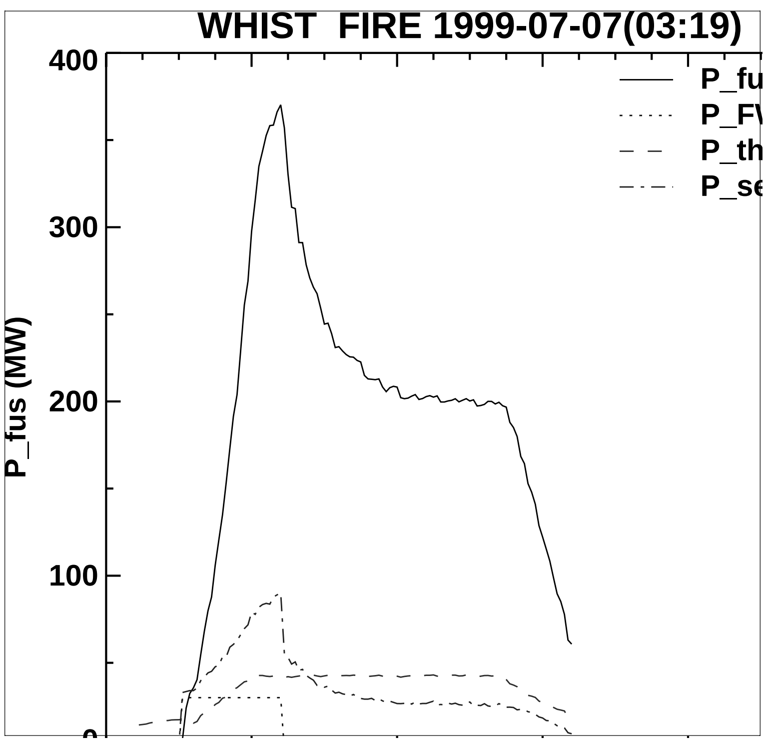

Fig. 2.5.1. FIRE example of start-up, burn and termination.

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 ρ* 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 , τ_{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 β_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 $P_\alpha / \left(P_{add} - P_{Fusion}\right) \sim 0.2$. 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.

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* 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 ρ^*/β_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 (δ) 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 β_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 β_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 = 83m³ δ = 0.22	**V = 106m³ δ = 0.57**		**Ref pulse 42976 V = 85m³ δ = 0.57**	**V = 106m³ 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
β_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 β 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 β_p.
• Low β_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)	$\leq$13 T
toroidal current (I_p)	$\leq$12 MA
mean poloidal field $B_p \sim I_p/5\sqrt{ab}$	$\leq$ 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_α = 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:

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* 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)
- τ_E, the plasma energy confinement time (~ 0.6s)
- τ_{He}, the confinement time of alpha ash, slowed down alpha particles (~ 5 τ_E ~ 3s)
- τ_{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 $Q \approx 10$ (full I_p and B_T)	τ_E (s)	τ_{He} (s)	τ_{cr} (s)	τ_{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 τ_E) would allow the pressure profile to come into equilibrium with alpha heating and allow the alpha ash to accumulate for ~ 3 τ_{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 $P_a / P_{heat} \geq 0.5$ ($Q \geq 5$) 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 $P_\alpha/P_{heat} \geq 0.8$ 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

* 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 $\kappa 95$, κX	1.68, 1.83	1.74, 1.92	1.6, 1.75
Ave δ triangularity, δX	0.43	0.49	0.35
$\langle T \rangle$ (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
β, β_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 β. 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 γ^* 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

BURNING PLASMA PHYSICS ISSUES IN A POSSIBLE JET UPGRADE

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1. Motivation

Record fusion yield (16MW) and fusion energy (22MJ) have been achieved in JET during the DTE1 campaign in 1997 [1], [2], respectively with an ELM-free H-mode and with an ELMy H-mode. Alpha heating has been observed [3] with alpha power in the range of 1.2MW. Significant fusion yield (up to 8MW) has been achieved with advanced scenarios [4]. These results have generated a large number of significant advances in the physics of fusion plasmas. It is felt that more significant burning plasma physics issues could be addressed in a JET upgrade. Increasing power capability would allow to access high confinement modes and to assess beta limits at full field. Increasing plasma volume would allow to increase plasma current and the fusion gain. Increasing plasma shaping would allow to operate at higher densities and to increase the ELM-free period of ELM-free H-modes. Increasing both plasma volume, plasma shaping and power capability would allow: i) to increase significantly JET performances, ii) to reduce errors in Next step extrapolation, iii) to operate at much higher values of fusion yield and alpha heating power and to tackle some burning plasma physics issues, which are needed to progress towards a fusion reactor.

2. Possible Upgrades in JET

JET is underpowered as compared to other machines such as Asdex-U, DIII-D and JT-60U. With the present power capability (16 to 18MW of Neutral Beam Injection (NBI) power and up to 10MW of Ion Cyclotron Resonance Heating (ICRH) power in ELMy plasmas), β_N values up to 1.3 and 2 have been achieved at a magnetic field of 3.4T, respectively in ELMy plasmas and in optimised shear plasma, while values considered for Next Step have to be at least 2.3. Also, it is necessary to access high confinement regimes (type I ELMs, ELM-free H-mode, Internal Transport Barriers) in order to optimise the fusion gain Q. At full field (up to 4T) it is estimated that up to 35 MW of power might be needed to produce high confinement ITBs.

Several options have been considered, but not yet decided, by the new EFDA JET sub-committee for some power upgrades in the period 2000-2002 in addition to the ICRH wide band matching system which might allow to increase the total combined power by 2 to 3MWs 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 to 7MW. 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

* These are personal views.

- an Electron Cyclotron Resonance Heating System (ECRH) making use of the recent technical developments and delivering 10MW in the 140GHz range;
- develop techniques allowing to increase the voltage handling of the ICRH antennae. If not successful, two additional antennae could be installed in the torus allowing to make full use of the RF power plant;
- in-situ ionising system in front of the Lower Hybrid Current Drive (LHCD) launcher allowing to increase the coupling and to make full use of the LHCD plant.

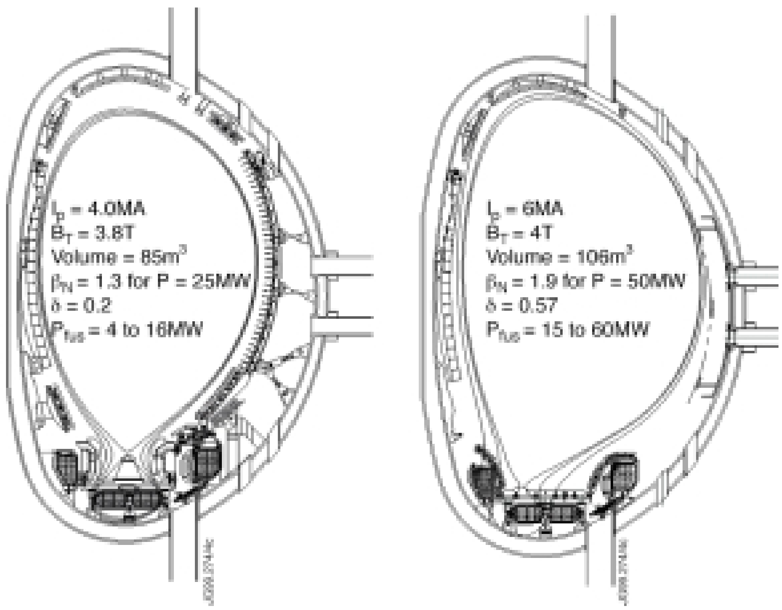

Fig. 1 Possible new JET upgrade configuration.

With the present divertor configuration, the plasma volume is limited to 80-85m^3, elongation (b/a) to 1.9, triangularity (δ) defined at the separatrix, to 0.35 and the plasma current to 4.5MA at 4T. Both in JET and in other machines, it has been found that beta increases with triangularity both in ELMy H-modes and in advanced scenarios. Moreover, the density normalised to the greenwald density can be significantly increased by increasing triangularity for a similar confinement [5]. Also, when δ is increased, the edge ballooning limit for MHD instabilities is increased and the time duration of an ELM-free H-mode is significantly prolonged. A configuration allowing to keep the divertor coils and to significantly increase the plasma volume and triangularity is shown in Fig. 1. It is to be noted that such a configuration is very flexible and large changes of elongation and triangularity are possible. A new divertor, using the existing coils and base structure, will have to be built.

3. Method of Extrapolation

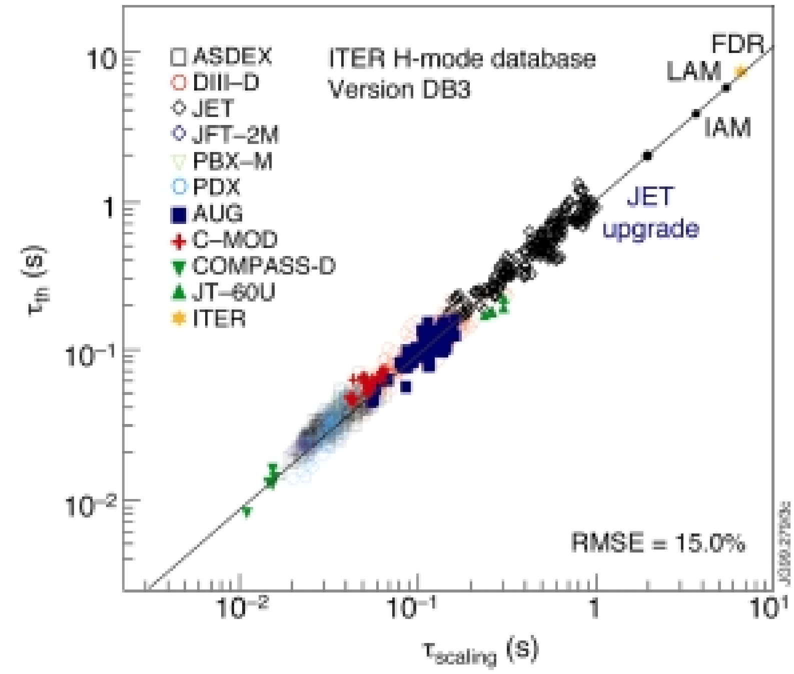

Fig. 2: JET upgrade in the ITER H-mode database.

Reference pulses have been taken from the JET database. ITER physics basis scaling laws have been used for extrapolation when available. If not, the own JET scaling has been used, for instance for the triangularity dependence, the Zeff dependence and for the advanced scenarios. In some cases, the transport modelling code JETTO has been used in a predictive way. The result is shown in Table 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 to operate at higher density still keeping a good confinement. Increasing power allows to operate at higher beta.

Extrapolation of the optimised shear scenarios is more difficult in the absence of established scaling laws. In JET, comparison of an ELMy H-mode with an optimised shear plasma at similar magnetic field (3.4T), plasma current (3.5MA) and additional power (25 to 28MW) shows an increase of β_N by a factor of 1.3 and a doubling of the fusion yield [8]. At 2.5 T, the threshold for ITB formation is ≈ 11 MW, and this increases to ≈ 19 MW of combined power for the ITB threshold at 4T. At 2.5 Tesla, the best results, in term of confinement, are produced with 24 MW combined power with a β_N~2.5. On this it is estimated that 36 to 40 MW would be needed to access good confinement regimes in optimized shear ITB plasmas at 4.0 T. Assuming the fusion yield can be taken as proportional to β_N^2, it would increase by a factor 1.7at 4.0 T. From extrapolations made in table I, a fusion gain of almost 1 with $P_{in} = 37MW$ and $\beta_N = 2.5$ with $P_{in} = 50MW$ could be achieved in a quasi steady-state advanced scenarios.

4. Burning Plasma Physics Issues

4.1 Heating by alpha particles and energetic particle stability effects

The alpha power might range from steady-state 4MW up to transient 14MW as compared to the transient 1.2MW 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

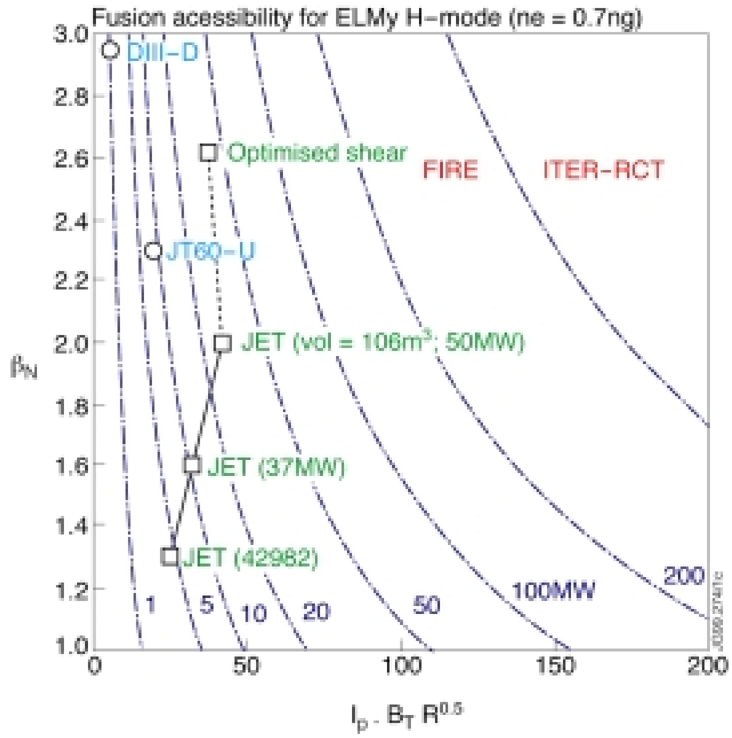

Fig. 3: Fusion accessibility for ELMy H-mode
at n = 0.7ng

can be done. Initial estimate of the TAE stability indicates that TAE modes still appears marginally stable. But since their growth rate increases with electron temperature, an ERCH system would allow stability studies. As well, the instability growth rate increases with q_o^2. Therefore optimised shear plasmas with ERCH will be an ideal tool to study energetic particle stability effect. A detailed estimate remains to be made.

4.2 Reactor regime core confinement

As shown in Fig. 2, a substantial reduction in extrapolation for Next Step devices can be achieved in a JET upgrade. This is also illustrated in a fusion accessibility domain shown in Fig. 3 where β_N is plotted against $I_p.B_t.R^{0.5}$ which is a measure of the fusion gain. It shows the substantial step in fusion and β_N capability as compared to today's experiments.

4.3 Beta limit studies at full field

Assessing beta limits at operational limits is obviously a key issue. Recently the importance of the ρ* not only on confinement by also on beta limits has been discussed [7], possibly linked to neo-classical tearing modes. In Fig. 4, various scans in density, magnetic field, power and plasma current have been made to define an operational space in a diagram ρ^* versus β_N. It shows that the gap between today's databases and the various options of ITER-RC can be filled. It is also to be noted that an ERCH system could allow to assess stabilising effects on neo-classical tearing modes in reactor relevant regimes.

4.4 Other aspects

Several other aspects of burning plasma physics issues can also be studied such as scaling of advanced scenarios with Internal Transport barriers (power dependence, confinement scaling, ρ^* dependence) and tritium transport issues. Helium retention and fuelling optimisation can also be studied in reactor relevant regimes. The installation on JET of a high field side pellet launcher is ongoing and, if successful, could be adapted to tritium operation.

Without more profound and costly modifications, the time duration of the high power pulse will be limited to 5-8 seconds. Therefore only the quasi steady-state aspects of high performance plasmas (MHD stable pressure and current profiles) can be studied.

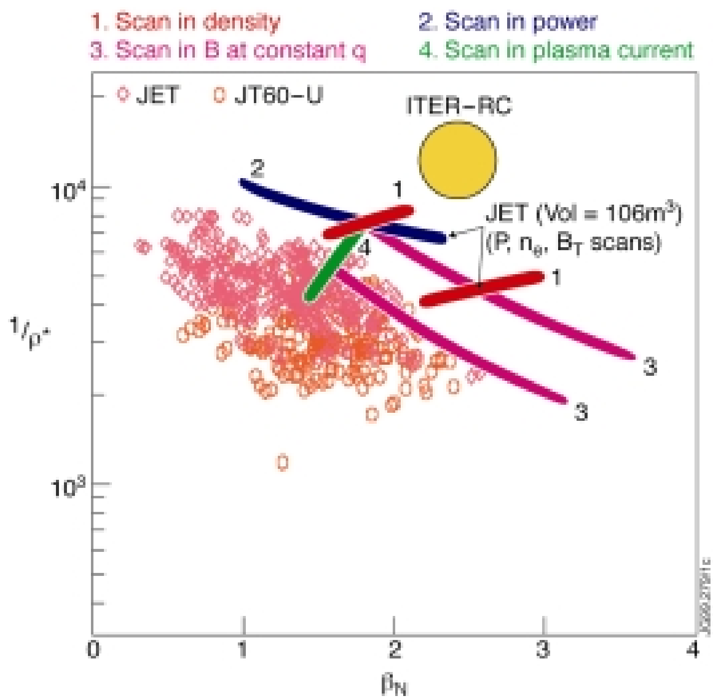

Fig. 4: ρ^*/β_N operational space for JET upgrade

5. Summary and Conclusion

JET performances can be significantly improved by increasing: i) the plasma volume (increase I_p, increase Q), ii) the plasma triangularity (higher density), iii) the additional power (up to 40MW to access high performance regimes, up to 50MW to assess beta limits).

Extrapolations have been made for the ELMy H-mode (steady-state), the Optimised Shear mode (steady) 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 particle ($0.5 < Q < 2$);
- energetic particles instabilities, in particular in high T_e plasmas obtained with ERCH and optimised shear plasmas;
- the domain ρ^*/β_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 to have flexibility with the divertor and to test 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.

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Table 1

	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
β_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

Critical Physics Issues for Ignition Experiments: Ignitor

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Abstract

The crucial physics issues related to fusion burning plasmas and potential fusion reactors can only be studied in a burning plasma experiment. The Ignitor experiment is designed to take the most conservative approach to the near term study of the physics of fusion burning plasmas, using an optimal combination of compact dimensions and high magnetic fields to support high plasma particle densities and high plasma currents. The values of its geometrical parameters, plasma current, and magnetic field have been chosen based on current knowledge of ignition physics, so that ignition is most likely to be achieved. This article presents the most important ideas behind its design.

1 Introduction

Demonstration of fusion ignition is a major scientific and technical goal for controlled fusion. Until the fundamental physics of fusion burning have been confirmed by experiment, the defining concepts for a fusion reactor must remain uncertain. Other factors would also have to be taken into account, such as the method for extracting fusion energy. Nevertheless, two major areas can be addressed in a near term ignition experiment. The *ignition process* will be similar for any magnetically confined, predominantly thermal plasma. Heating methods and control strategies for ignition, burning, and shutdown can also be established.

These three issues, demonstration of confined ignition, the physics of the ignition process, and heating and control of a burning plasma, are specifically addressed by the Ignitor experiment [1][2][3][4][5]. Its design has been driven more closely by physics considerations than that of any other existing design. The associated physics studies have gone beyond simple identification to include interaction of the physical processes involved in ignition. Ignitor is part of a line of research that began with the Alcator machine at MIT in the 1970's [6][7], 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 [8]. Based on present knowledge of fusion physics, high magnetic fields still offer the best path to achieving ignition, when both energetics and plasma stability are taken into consideration. The high field approach also allows a possible development path [9][10] to tritium-poor, relatively 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.

A large amount of work on the physics of ignition has been carried out over the course of the Ignitor design evolution. Much of it is generally applicable to ignition in a confined plasma, not only at high field. This article presents the basic physics that underlies the Ignitor design, including open questions. It starts with the physics questions that cannot be addressed in present experiments, then discusses the problem of attaining ignition and the advantages of high magnetic field, the dynamic nature of the ignition process and its relation to the initial current rise phase of a discharge, and other issues. Since these questions overlap in many ways, the Appendix summarizes the self-consistent characteristics of a high field ignition experiment.

2 Advances beyond present experiments

Even without strong assumptions on the possible form of a fusion reactor, it is clear that present experiments do not operate in plasma regimes close to those required for ignition. There are a number of discrepancies, of which one or more always apply:

1. The effective charge Z_{eff} is in general too high, compared to the limiting value for stable ignition, $Z_{eff} >$1.5–1.6 [1]. High Z_{eff} prevents ignition by allowing excessive radiation emission, so that the ideal ignition temperature is not attained. Although initially demonstrated for Ignitor, this Z_{eff} limit can be shown to be general. Exceeding this value requires large amounts of auxiliary heating power, operation near the β stability limit, and other conditions.

2. The central ion temperature is substantially higher than the electron temperature, $T_i > T_e$. A thermal burning plasma, will have $T_i \simeq T_e$ unless the temperature is very high. Fusion α's, and all other charged particles produced by fusion reactions, have relatively high energies in the MeV or multi-MeV range and therefore primarily heat the electrons by collisional slowing down. In present experiments, the ions used for neutral beam heating have relatively low energy on the order of 100 keV, and primarily heat the ions. In addition, they sustain a large fast ion population, due to the relatively long collisional slowing down times at the low plasma densities.

3. The α-particle slowing down time is long compared to the energy confinement time τ_E, while in an igniting plasma the time should be much shorter.

4. The effective electron collision time is short compared to the diamagnetic frequency, $\nu_e < \omega_*$. This is important for $m = 1$, $n = 1$ mode stability when the plasma poloidal beta β_p approaches the ideal MHD instability threshhold. Most present machines operate at low densities compared to an ignition experiment or reactor, where higher density is desirable to increase the fusion reaction rate and improve plasma purity. The collisionless reconnecting modes seen in present high temperature plasmas are relatively easy to stabilize, but those expected in ignition experiments as diverse as Ignitor and ITER will be at most semi-collisional, with $\nu_{ei} \gtrsim \omega_*$, and are expected to be more dangerous.

5. Present experiments have relatively low peak pressure. The ignition figure of merit $n_i(0)T_i(0)\tau_E$ requires a minimum absolute pressure with peak value $p_o \gtrsim 1.5 - 4$ MPa. For D-T fusion, $n_i(0)T_i(0)\tau_E \simeq 70$ (in units of $10^{20}\mathrm{m}^{-3}$, keV, sec). More accurate figures of merit, for example $n_H(0)T_e(0)\tau_E\epsilon_{sd}\epsilon_p F(T_e/T_i)$, could take into account the slowing-down-time of the fusion α's, through $\epsilon_{sd} = 1/(1 + \tau_{sd}/\tau_E)$, the plasma purity through $\epsilon_p = (5/4)/(1 + Z^2_{eff}/4)$, and the ratio of the electron and ion temperatures.

6. The sub-ignited D-T experiments performed so far have been ballooning unstable. High fusion yield discharges have been terminated consistently by plasma instabilities, TFTR by $n = 1$ kink-driven edge ballooning modes and JET also by MHD instability.

7. The known improved confinement regimes are transient and/or nonthermal (significant non-Maxwellian particle distributions). Improved confinement regimes tends to be associated with modified, transient q-profiles, while most high confinement experiments using NBI heating have a nonthermalized ion population due to the relatively low bulk-plasma densities.

One further criterion, that the electron-ion energy equilibration time be short, $\tau_{ei} < \tau_E$, is satisfied in present experiments. The remaining points must be addressed by a burning plasma experiment.

3 The ignition dilemma

The 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 $T_o \simeq 6$ keV for typical centrally peaked profiles), under conditions in which the fusion heating can continue to rise. The basic problem is that plasma confinement is still not understood well enough to predict performance reliably in these regimes. For ignition experiments, this creates a problem. To study ignition and true fusion burning, experiments must operate in regimes with high levels of fusion power relative to other inputs, e.g. $Q_\alpha \geq 2$, where $Q_\alpha \equiv P_{\alpha h}/(P_L - P_{\alpha h})$, P_L represents the total power loss from the plasma, and $P_{\alpha h}$ the fraction of the D-T fusion power in α-particles that actually heats the plasma. Effective plasma heating must also be provided to reach this state. (Recall that the plasma power balance is $dW/dt = P_{\alpha h} + P_{Aux} + P_{OH} - P_L$, where W is the plasma kinetic energy and P_{Aux} and P_{OH} the externally applied and ohmic heating powers, respectively. The definition of ignition used in the early Ignitor work was $P_{\alpha h} = P_L$, which when first reached in a time evolution sequence corresponds to an over-heated state with $dW/dt > 0$, where the temperature will first make an upward excursion before settling at a steady state level.) To guarantee ignition, experiments may also consider using relatively high levels of input power. This approach, however, leads to several difficulties, including that of ensuring plasma stability.

This uncertainty must be resolved by the design of an ignition experiment. 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. These are not the only considerations that favor a high field approach and a strong argument can be made that high magnetic field is the only real solution for the ignition dilemma (Section 4). High field introduces an interlocking set of requirements [1], which are summarized in the Appendix. The maximum value of the field and plasma current that can be generated and the length of time over which they can be sustained in a given magnetic configuration is thus a strong constraint on ignition capacity. Maximizing these values constitutes the major goal for the engineering design. For reference, 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.

4 Physics conditions to ensure ignition: High B_T

Many of the physics limitations and uncertainties regarding ignition (see also the Snowmass Burning Plasma report [11]) lead to the statement that "High magnetic field is the most advantageous approach to ignition using the present knowledge of the physics and technology of high temperature plasmas." This conclusion also emphasizes the importance of continuing technological

Table 1: Ignitor Reference Design Parameters

major radius	R_o	1.32 m
minor radius	$a \times b$	0.47×0.86 m
aspect ratio	A	2.8
elongation	κ	1.83
triangularity	δ	0.43
toroidal field	B_T	≤ 13 T
toroidal current	I_p	$\lesssim 12$ MA
mean poloidal field	$\overline{\overline{B}}_p \equiv I_p/5\sqrt{ab}$	≤ 3.75 T
poloidal current	I_θ	$\lesssim 9$ MA
edge safety factor ($I_p \simeq 11$ MA)	q_ψ	3.6
magnetic flux swing	$\Delta\phi$	36 Vs
plasma volume	V_o	≈ 10 $\mathbf{m^3}$
plasma surface	S_o	≈ 36 $\mathbf{m^2}$
auxiliary heating	P_{RF}	18–24 MW

Table 2: Effects of Different Density Profiles for Ignitor

Density Profile	Narrow	Reference	Broad	Broad*	Almost Flat**
n_{eo} (10^{20}m^{-3})	11	11	11	8.5	8.4
$n_{eo}/\langle n_e \rangle$	2.9	2.2	1.5	1.5	1.1
t_{IGN} (sec)	4.1	4.3	4.7	4.3	4.5
β_p	0.12	0.13	0.15	0.15	0.15
W (MJ)	10.7	11.7	13.4	12.6	13.7
T_{eo} (keV)	11.2	11.0	11.1	13.0	13.3
τ_E (sec)	0.61	0.66	0.70	0.68	0.74
P_{OH} (MW)	8.8	9.5	9.9	9.1	8.9
P_α (MW)	17.4	17.8	19.0	18.7	18.6
P_B (MW)	3.2	4.1	5.8	4.2	5.4
P_C (MW)	0.4	0.5	0.8	0.6	1.1
$Vol_{q=1}$ (%)	4.0	5.8	> 10	4.8†	10.2

*Lower density; $n_{eo} = 6.5 \times 10^{20}$m^{-3} at end of current ramp ($t = 3$ sec), increasing after.

**Lower peak density; optimum value is lower than this.

†Large low shear region for $q \simeq 1$.

P_α is fusion α-heating, B bremsstrahlung loss, C cyclotron radiation loss. $P_{Aux} \equiv 0$.

progress, such as the development of superconductors capable of sustaining fields of 20 T or more. In fact, experience with the Ignitor and other designs very strongly points to the conclusion "High magnetic field is the only possible approach to ignition at this time."

A simple argument shows why a high toroidal magnetic field is so indispensable. Consider the possible values of the edge safety factor q_ψ, at the required values of the central pressure and plasma β for ignition and for plasma stability. Since the actual q is a complex function of the plasma fields and shape, define an "engineering" $q_E = (5ab/R)(B_T/I_p)$, which satisfies $q_E \propto q_\psi$ and $q_E < q_\psi$. A corresponding "engineering" poloidal field can be defined by $\overline{\overline{B}}_p = I_p/(5\sqrt{ab}) = (\sqrt{ab}/R)(B_T/q_E)$. Plasma stability can be measured by the poloidal plasma beta, $\beta_p = 8\pi\langle p\rangle/\overline{B}_p^2$, where $\overline{B}_p$ is the actual average poloidal field. At ignition, the minimum central pressure p_o must be in the range of $1.5 \lesssim p_o \lesssim 4$ MPa for 50:50 D-T (1 MPa $\simeq$ 10 atm), because of the minimum limit on the ignition parameter $n_i(0)T_i(0)\tau_E \simeq 70 \times 10^{20}\text{m}^{-3}\text{keV} \cdot \text{sec}$.

There are two possible ignition regimes, at low and high q_ψ. At low edge $q_\psi \lesssim 3.3$ (an approximate value), the regions where $q < 1$ and $q < 2$ are large. Then large scale internal modes with dominant $m = 1$ and $m = 2$ harmonics, extending to r_1 and r_2 respectively, will exist unless β_p is also small. Since the volume average pressure $\langle p\rangle$ cannot be too low at ignition, the plasma stability requires a minimum B_T that depends on the critical $\beta_{p,crit}$ of the modes. Starting from the definition of q_E and using the definitions of $\overline{\overline{B}}_p$ and β_p gives the limit

$$B_T > q_E \frac{R}{\sqrt{ab}} \left(\frac{\overline{\overline{B}}_p}{\overline{B}_p}\right) \left(\frac{8\pi\langle p\rangle}{\beta_{p,crit}}\right)^{1/2} .$$

The definition of β_p in terms of inverse I_p^2 also shows why trying to increase q_ψ by lowering the current relative to a fixed B_T is a poor idea at low q_ψ.

At high $q_\psi > 5$, the plasma current must be relatively low, since

$$I_p = \left(\frac{\sqrt{ab}}{R}\right) \sqrt{ab}\, B_T \frac{5}{q_E}.$$

Assuming that the confinement time is $\tau_E \propto I_p$, as is typical of the L-mode, then larger values of the confinement improvement factor H over L-mode are required to reach ignition. In practice, H is observed to be limited to values of 2–3, so that the remaining factor $\sqrt{ab}\, B_T$ cannot be too small. In the opposite limit, if $\sqrt{ab}\, B_T$ is increased by expanding the radius, the average poloidal field $\overline{B}_p$ remains low as long as B_T is low, and β_p becomes large. Pressure-gradient-driven ballooning modes then become a problem.

Applying the actual values from experiment and theory shows that these criteria give fairly stringent practical limits on B_T. Intermediate values of $q_\psi \sim 4$ correspond to the least restrictive conditions and are the best choice for ignition. This is the Ignitor reference value. On the other hand, limits on the achievable B_T with present day magnets tend to force q_ψ somewhat lower in most high field designs (e.g., down to $\simeq 3.6$ in the Ignitor design, that uses normally conducting, cryogenically cooled magnets).

5 Ignition criteria: Natural density for ignition

One aspect of advantages due to high B_T can be illustrated by the limitations on time-dependent ignition. An idealized "natural density" for an ignition experiment can be defined as a measure of

Table 3: Natural Densities for Ignitor (volume-averaged)

	$\langle n_N \rangle$ (10^{20}m^{-3})	T_{io}/T_{eo} (keV)	R/a (m/m)	B_T (T)	I_p (MA)	$\overline{n}_G$ (10^{20}m^{-3})
Reference	5	$\lesssim$ 15/15	1.32/0.47	13	12	17.3
RevShear	3	17/19	1.32/0.47	12	7	10.1

the absolute ignition margin and the difficulty of achieving desired performance that includes cost and complexity.

The "natural density" n_N for D-T ignition in a given device is the density at which a pure ($Z_{eff} = 1$) 50:50 D-T plasma ignites most readily for the nominal machine parameters. It is a characteristic property of a given machine, i.e., the achievable plasma size and shape, magnetic field, plasma current, and auxiliary heating power, and it can also be defined for each operating scenario within a machine. Since there are maximum and minimum density limits on ignition in a given experiment, determined by a balance between radiation power loss, available heating power, and energy confinement (and other factors, see [23]), there is also the possibility that n_N may not exist for a given case. When it does, it indicates the best possible ignition performance for that device, since the required heating and plasma confinement at n_N are the absolute minimal requirement and every real plasma will be at least slightly contaminated and thereby suffer degraded performance. It provides a measure of the potential plasma performance at the design operating conditions, even though these may be very different from the ideal conditions used to determine n_N. The difficulty of achieving the desired operating parameters (cost, complexity, physics ignition margin) depends on the degree of improvement needed in the heating power, confinement, etc., over the ideal level, based on the expected degree of contamination (Z_{eff}), which is a sensitive function of density.

The natural density must be determined using, at a minimum, a 1 1/2 D transport simulation. Heating during the initial current ramp phase must be included. Unless the design specifies otherwise, the current ramp is chosen to have the minimum duration required to reach the design current I_p without developing nonmonotonic current density profiles J_ϕ, at the optimum programming of the time evolution of the plasma size, shaping, and ramp rates $\dot{I}_p$, $\dot{B}_T$, $\dot{n}$, etc. Given a standard thermal transport model, the combination of the minimum required enhancement factor over a standard confinement scaling that allows ignition and the minimum amount of external heating required (using an idealized heating profile) is determined. The optimum or expected density transport/fuelling or a given profile shape may be used. Sawtooth oscillations can be ignored, since they usually increase the difficulty of ignition.

Different operating regimes in Ignitor can be used as an example. Table 3 shows results for the volume-averaged n_N for the reference scenario at full field and full current and for a reversed shear, improved confinement regime at reduced current, both with relatively flat density profiles. The maximum confinement was constrained as far as possible to approximately H-mode, i.e., 2–3 times the ITER89-P L-mode scaling. The reference scenario, based on Refs. [1] and [2] and Table 2, shows ignition at low central temperatures, $T_{eo} \simeq T_{io} \sim 12$–$15$ keV, with confinement slightly above L-mode (ITER-89P) and ohmic or almost entirely ohmic heating. These results (actually obtained for $Z_{eff} \simeq 1.2$, but very similar to $Z_{eff} = 1$) are close to expected operating conditions. In comparison, the reversed shear case at 12 T and 7 MA [12], has approximately $\langle n_N \rangle \sim 3 \times 10^{20}$ at a maximum $P_{Aux} \simeq 8$ MW during the current ramp and $H = 2.5$–3 (again at $Z_{eff} \gtrsim 1.2$). The reversed shear field and current are fairly similar to the FIRE parameters, and the Ignitor result, along with its larger dimensions, implies a lower natural density for that machine, relative to the reference Ignitor.

At fixed maximum plasma size and shape, when the density rise occurs primarily during the current ramp, the natural density varies approximately proportionately to plasma current I_p. At varying minor radius and current, it varies roughly like the Greenwald density, line-average $\overline{n}_G = I_p/(\pi a^2)$, although with a somewhat weaker dependence in minor radius. This occurs because the density rise and the rate of current penetration are inter-dependent, the magnitude of the density affecting the local temperature for a given heating rate, and the local temperature in turn determining the resistive diffusion rate of the current. The relationship is less direct when substantial plasma fuelling occurs outside the current ramp.

6 The transient nature of the ignition process

In a contained burning plasma, the approach to ignition is a transient process, where both spatial and temporal effects are important [13]. The strong positive dependence of the fusion cross sections on the kinetic energy of the reactants also allows the possibility of a "thermonuclear instability" phase near the ignition point, where the plasma temperature and fusion power can rise rapidly.

For magnetically confined plasmas, the transient nature of the approach to ignition becomes more important because the most efficient approach to ignition is to use the initial phase of the discharge, when the plasma current is being raised to its final value (the current ramp phase), to heat the plasma towards ignition and to help control the development of the plasma profiles, in particular the toroidal current density J_ϕ, for plasma stability (initial current ramp studies [14], integration of heating and plasma stability effects for Ignitor [15][16][1]). An important constraint is the final edge safety factor q_ψ allowed by the plasma field, current, and shape. A great deal of work for the Ignitor has been done to confirm that this procedure can be effective and to study its limitations, e.g., [1][2][12][17]. Much of this work predated later successful control of the current ramp to produce improved confinement regimes (the early Ignitor work did not consider such regimes and actually imposed the condition that the the q-profile remain monotonically increasing toward the plasma edge; reversed shear and improved confinement was considered in [12].)

Understanding the transient approach to ignition is a complex problem, since a large number of independent or semi-independent time-varying parameters must be optimized. A numerical transport simulation model containing at least the radial (flux-surface) coordinate is required for quantitative results. The basic principles are clear, however, are clear.

For a high field experiment with a high plasma current, transient effects can be exploited to use ohmic heating to give a substantial boost toward ignition [16][1]. 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. The plasma loop voltage peaks at radii near the edge of the plasma, a region of relatively large volume (cf. the figures in [1]). Since the resistivity $\eta_\parallel \propto T_e^{-3/2}$, a relatively large ohmic heating power $I_p V_\parallel$ can be produced even when the central plasma temperature is high. The ohmic heating rises continuously during the current ramp, at a rate $\dot{P}_{OH} \propto \dot{I}_p$. For the Ignitor reference scenario, this can be on the order of 10 MW or more at the end of the current ramp, somewhat less at ignition, when roughly $P_\alpha \gtrsim 2P_{OH}$. Due to the high field and current, self-sustained burning states can be reached and maintained by the residual ohmic heating $P_{OH} \simeq 1-2$ MW at reduced levels of confinement, even if full ignition ($P_\alpha = P_{Loss}$) is never reached.

7 Confinement and Thermal Transport Models

The inability to predict even the global level of plasma transport (energy and particle confinement) for a given plasma configuration with a high degree of reliability is the single most troublesome question for the design of an ignition experiment. In all cases, the degree of extrapolation from the existing experimental database is enough to raise concerns as to its accuracy. High field experiments at high density require the least extrapolation, but still lie outside existing experimental data.

A number of important considerations for predicting transport and performance in ignition experiments exist. First, 0D (global, volume-integrated) steady-state models are not sufficient to predict ignition. At a minimum, 1 1/2D time-dependent transport simulations are needed because the energy balance is intimately tied to the plasma profiles (including q and current) and therefore to plasma stability [14][15][16][1]. A 0D steady state model gives only a rough idea of global power requirements for ignition. It gives a functional relationship between input power and loss, but does not predict the optimal point for operation, and says little about the achievability of a given operating point in practice. Transport simulation is needed for prediction and for real-time control.

Second, present widely accepted global scalings for energy confinement time are based on a set of criteria and an experimental database [18] that have been chosen to apply to a particular design, the ITER EDA [19], whose requirements are different from those of high field designs. One result is the ITER89-P scaling for the L-mode confinement time [20] that predicts that the energy confinement time τ_E degrades with the total heating power as roughly $\tau_{E,L} \propto P_H^{1/2}$, or even more strongly [22][18]. An important question is whether different selection criteria, more suited to high field ignition conditions, would yield different results.

In fact, such criteria can lead to different confinement predictions. A case can be made that the degradation of τ_E with the heating power P_H stops above a certain power level. This is the prediction of the Coppi-Daughton effective thermal diffusion coefficient [24][25]. A global τ_E dependence was initially derived from the observed behavior of β_p in Alcator C-Mod ohmic and RF-heated discharges [21], where for OH heating $\beta_p \simeq$ constant (0.25), while with additional ICRH, β_p increases linearly with P_{ICRH}. The resulting τ_E does not have a power law dependence on the plasma parameters, but an offset relation that suggests that the confinement ceases to degrade with heating power above a certain power level,

$$\tau_E \simeq 0.031 R q_E^{2/3} I_p \left(1 + f_3 \frac{I_p \mathcal{V}_o}{P_H}\right) \left(\frac{d_i}{a}\right)^{1/2} \left(\frac{\omega_{pe}}{\Omega_{ce}}\right)^{1/3} \tag{1}$$

in MA, MW, and mks units. Here the coefficient is $f_3 \simeq 1.4(r/4a)^{1/2}(R/20d_i)^{1/2}$, $d_i = c/\omega_{pi}$, ω_{pj} is the plasma frequency for species j, Ω_{cj} the gyrofrequency, q_E a characteristic safety factor parameter, $q_E = 2\pi a^2 \kappa B_T/(\mu_o I_p R)$, and $\mathcal{V}_o$ is an expression for a characteristic voltage, given below. All numerical coefficients were determined from Alcator C-Mod data. The resulting expression for τ_E was then shown to fit the global energy confinement times of a specific subset of the ITER L-mode and OH database (as it existed in 1997), with no additional free parameters. The subset, 1088 cases, was chosen to be more applicable to high density, high field experiments than the general ITER database. It consisted of all the datapoints satisfying

- OH or L-mode
- clean: $Z_{eff} < 2$
- $T_i \simeq T_e$: $0.7 < W_i/W_e < 1.3$

- mostly thermal: $W_{th}/W_{tot} > 0.7$
- steady state: $(dW/dt)/P_H < 0.1$.

Using a volume-averaged β_{p*} gave excellent results, with an RMS error of 13.1%, compared to 23.6 % for the ITER96 L-mode scaling [22] restricted to these cases. (The ITER96 scaling had a lower error than the original ITER89-P scaling.) Only 7 of the 14 machines represented in the full ITER database appear under these criteria.

A possible radial dependence for a thermal diffusion coefficient was also derived and shown to fit a wide variety of steady state ohmic and RF-heated discharges from Alcator C-Mod [25]

$$\begin{aligned}
\chi_{eff}^{CD97} &= \mathcal{V}_o^o \frac{I_\phi(\rho)}{n(\rho)T_e(\rho)} \mathcal{F}_D \frac{V_a}{\langle|\nabla V|^2\rangle} \qquad (2)\\
\mathcal{V}_o^o &= \left[\frac{\nu_*}{1+\nu_*} + \left(\frac{n_o^o}{n}\right)^{1/3}\right]\mathcal{V}_o \\
\mathcal{V}_o &= \alpha_v \frac{T_e}{e}\left(\omega_{pi}\frac{c^2}{\omega_{pe}^2}\frac{\nu_e}{v_{the}^2}\right)^{2/5}, \qquad \left(\frac{n_o^o}{n}\right)^{1/3} \equiv C_1 \left(\frac{\omega_{pi}}{\nu_e}\right)^{2/3}\left(\frac{c}{4\pi v_{the}}\right)^2 \frac{m_e}{m_i} \\
\mathcal{F}_D &= C_2\left(\frac{a}{d_i}\right)^{1/2} f_{C3}, \qquad f_{C3} = \begin{cases} C_3\left[\frac{10\beta_{p*}}{q_E^{2/3}}\left(\frac{\Omega_{ce}}{\omega_{pe}}\right)^{1/3} - \frac{R}{4a}A_i^{1/2}\right] & \text{if} > D \\ D & \text{otherwise}\end{cases}
\end{aligned}$$

using a poloidal beta based on the maximum interior pressure gradient $\beta_{p*} = 8\pi p_{e*}/\langle B_\theta^2\rangle$, $p_{e*} = \max(dp_e/d(\rho/a))$. The coefficients were derived from the previously determined global fits to Alcator C-Mod, $C_1 = 0.24$, $C_2 = 0.0833$, $C_3 = 1.7$, $D = 0.25$, and $\alpha_v = 0.18$. (Here V is a volume inside a flux surface, V_a within the entire plasma, $\nu_* = \nu_e(q_E R/v_{the})(R/a)^{3/2}$ is the trapped electron collisionality with ν_e the electron-electron collision time, v_{the} the electron thermal velocity, and A_i the average ion atomic mass.)

The radial form of a transport coefficient is important for predicting ignition, which is a strongly dynamic and non-local process. Even if global confinement is accurately described, a derived transport coefficient may give poor results. Numerical transport simulation consistently indicates that a coefficient that preserves temperature profile shape in some manner ("profile consistency" [26]) is required to fit many present-day experiments, especially their transient phases, as well as having a strong effect on ignition predictions. For example, the CD97 coefficient described above does not work as well for transient conditions, including Alcator C-Mod current ramps and ignition simulations, because the strong dependence on gradient in β_{p*} tends to produce an artificially steep gradient at mid-radius. A profile-consistent coefficient, such as the original CMG [27] scaled to match a desired global confinement, does much better, suggesting that the radial form of CD97 should be modified.

8 Transport and Control

8.1 Thermal transport questions

Other open questions about thermal transport in burning plasmas remain. For example, the heating of the plasma due to collisional slowing down of the charged particles produced in fusion reactions is isotropic in velocity space an axisymmetric in real space, with its magnitude centrally localized in the plasma. Does it then cause degradation of confinement time with increasing input power, $\tau_E \sim P_{Heat}^{1/2}$, in the same way as most existing methods of injected heating, which are anisotropic in velocity space, non-axisymmetric in space, and often concentrated off-axis? This empirical rate of degradation with power exerts perhaps the most crucial influence on current designs for ignition experiments and potential reactors. Evidence that some heating methods, such as ECH, do not degrade confinement in this way should also be explored.

A further consideration for fusion burning is that the electron thermal transport is important at higher density, unlike present-day lower density, mainly ion-heated experiments with $T_i < T_e$, that are dominated by the ion thermal transport due to toroidal ion temperature gradient (ITG) modes. Relatively little has been done for electron transport, even for global scalings. The theory of electron transport processes is poorly understood and the connection between fluctation and transport is much more difficult to simulate numerically than for ions.

8.2 Density profile control

The prediction and control of the density profile at high densities is another important transport and edge plasma physics problem for ignition experiments. 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, e.g., [2] and Table 2. (Note that the flattest profile case in Table 2 probably has a lower optimum density.)

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, is a subject of current investigation. Control of the plasma edge density during startup and steady state is also important, since it regulates the current penetration rate as well as being related to the edge temperature. A balance must be struck — high edge density improves impurity screening from the main plasma, but may be less beneficial for other processes, such as plasma heating and/or stability. (High edge densities result in relatively lower edge temperatures, which speed up the rate of the edge current penetration, resulting in lower central safety factors q and potential stability problems, as well as tending to reduce the central plasma temperature.)

8.3 Burn control

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 τ_P are generally longer than energy confinement times τ_E. Short-time-scale sensitivity to the fuel-ion particle source rate requires that the confinement τ_E be marginal relative to that needed to maintain the desired level of burning, or that the burning rate is high enough that a strong source of fuel ions is required to sustain it. For a reactor, economical operation dictates that τ_E be significantly above marginal, and a major goal of pre-reactor burning plasma experiments should be to increase this margin. A generalized form of burn control by specifying the concentration of tritium relative to deuterium in a discharge can always be used. Much better control is possible by operating in a slightly sub-ignited state that is driven by a small amount of externally supplied heating.

Emergency methods of burn control 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 can be studied.

8.4 Radiating cold mantle

Fusion plasmas deal with relatively large amounts of applied and self-generated power, which must all eventually exit the plasma to its surroundings. An important concern is to reduce the power loading on the physical walls. In the case of Ignitor, the first wall is covered by molybdenum tiles. Among the disadvantages of introducing a divertor is that it becomes a "hot spot" in the plasma-wall interaction system.

A potentially effective method for minimizing and distributing the power loading on the walls is to use a cold mantle of partially ionized plasma to surround the main plasma that is contained within the closed flux surfaces. It has been successfully demonstrated for non-burning plasmas in the RI-mode in TEXTOR [29]. There, impurity injection into the scrape-off layer (SOL) greatly increased impurity line radiation losses from the layer, allowing a relatively large part of the power put into the plasma to be radiated. This has the double advantage that radiation is much less damaging to material surfaces than particles and that since it is relatively evenly distributed throughout the SOL, so is the resulting load on the walls.

This method is particularly well suited to a high field, high density plasma, which can expect to have a relatively high plasma edge density with relatively low edge plasma and SOL temperature (e.g., Ignitor, based on data from the Alcator series of experiments [28]). High plasma edge densities confine outside impurities generated from the walls to the scrape off layer ("cold plasma blanket"[30], DIII-D VH mode [31][32]), while the low plasma edge temperatures allow the formation of a cold radiative mantle.

9 Low-neutron-yield fusion

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 releases 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. For example, Ignitor 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 [33] or somewhat less for thermal ^{3}He in a D-T plasma [34].

10 Summary

The major points driving the design of the Ignitor experiment can be summarized as follows:

- The crucial physics issues related to fusion burning plasmas and potential fusion reactors can only be studied in an experiment capable of approaching ignition.
- The Ignitor experiment takes the most conservative approach to the near term study of the physics of fusion burning plasmas, through an optimal combination of geometrical characteristics, plasma current, and magnetic field. This approach lends itself to important developments that include advanced fuel burning (low neutron yield, e.g., D-^{3}He).

- The Ignitor design has been strongly driven by the physics of ignition. A large amount of original and early work on the physics has been carried out during the design process, that is applicable to all magnetically confined burning experiments. This statement can also be extended to the engineering design of the machine and the technology solutions devised for it.

- High magnetic field, high density plasmas have the most favorable characteristics and expectations for ignition, and are the only ones that, given the present knowledge of plasma physics, allow this goal to be pursued realistically.

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A Requirements for a high field ignition experiment

This Appendix summarizes the set of characteristics required for a tight aspect ratio, high field ignition experiment [1]. These provide another way of looking at the physics and engineering requirements for such an experiment.

The requirement of high toroidal field B_T and compact size leads to an interlocking set of characteristics favorable for ignition. The combination of high field and compact dimensions, with significant vertical elongation $\kappa > 1$, allows a relatively large plasma current, toroidal current density, and poloidal magnetic field to be supported. (In Ignitor, the mean poloidal field is $\overline{\overline{B}}_p \leq$ 3.75 T. Also, there is a large paramagnetic current $I_\theta \simeq 9$ MA at the low β of ignition and this increases the central B_T by $\simeq 1$ T.)

High toroidal field supports a high plasma density with $n < n_G = I_p/\pi a^2$, where n_o can be correlated empirically with the ratio B_T/R or with the plasma current density. In Ignitor, densities $n_{eo} \simeq 10^{21}$ m^{-3} should be possible, based on the values of B_T/R obtained by Alcators A and C, FT and FTU, and TFTR. Alcator C obtained $n_o \simeq 2 \times 10^{21}$m^{-3} at $B_T = 12.5$ T. If the maximum density instead correlates with the volume–averaged current density, Ignitor's value of $\langle J_\phi \rangle \simeq 0.93$ kA/cm^2 should again allow $n_{eo} \simeq 10^{21}$ m^{-3} without difficulty.) Therefore, based on the required confinement $n_o\tau_E \simeq 4 \times 10^{20}$s/m^3 for ignition conditions $T_o \sim$ 12–15 keV for 50:50 D-T plasma, only a moderate energy confinement time $\tau_E \sim 0.4$ sec is required.

As a consequence, such plasmas have

- High levels of ohmic heating up to ignition [1] (P_{OH} is high due to highB_p).
- Good confinement of plasma energy and particles (since empirical scalings show that, approximately, $\tau_{E,L} \propto I_p$)
- Good confinement of fast fusion charged particles. ($I_p > 6$ MA will give good central confinement of D-T α-particles.)
- Low temperature ignition ($T_{eo} \simeq T_{io} \lesssim 15$ keV in Ignitor) at relatively low levels of fusion heating ($P_\alpha \lesssim 2P_{OH}$).
- Ignition at low β_p. Ideal MHD and long wavelength resistive $m = 1$ internal modes are expected to be stable, due to low $\beta_p \lesssim 1/4$ (the limit is $\beta_p \simeq 0.3$ for Ignitor [35]).
- Low fusion power and 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"[30], DIII-D VH mode [31][32]), where line radiation from them helps to evenly distribute the wall loading (RI-mode in TEXTOR [29])

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. This avoids serious degradation of confinement before the fusion α-heating regime is reached and allows the possibility that fusion heating may have better confinement characteristics than injected heating, since it is axisymmetric and isotropic like ohmic heating, but unlike injected heating.

- Divertors, which concentrate the thermal wall loading on small regions. Divertors require an expanded volume inside the toroidal field coils to accomodate the magnetic separatrices, the divertor, and the associated shaping coils. For high field designs, relatively small increases in the size of the coils and the major radius have serious consequences through the cascade of relations: larger $R \rightarrow$ lower $B_T/R \rightarrow$ lower n_e, lower $B_T \rightarrow$ lower I_p and P_{OH}, so that β_p is higher at ignition. The I_p is also lower for given B_T because the necessity of squeezing magnetic separatrices and the divertor inside the toroidal field coils reduces the plasma cross sectional area. Divertors introduce additional complexities in machine and magnet design, as well as operational risks associated with the presence of current carrying conductors in regions of high magnetic field.

- X-point configurations, which reduce the plasma cross-sectional area and current carrying capacity for a given toroidal magnet size and capacity. (In Ignitor, X-point configurations with single or double magnetic nulls can be produced for all or part of the discharge if necessary, with relatively little sacrifice in plasma and magnet parameters, i.e., somewhat smaller I_p, estimated as 10 MA for a single lower X-point, and more localized wall loading. The Ignitor X-points can be swept over regions of the wall to further distribute the power load.)

- Current drive to control q, which may be required to control central sawtooth oscillations at low edge q_ψ and/or high β_p.

Fusion Ignition Research Experiment (FIRE)

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Introduction

The major next steps in magnetic fusion energy need to address the following critical fusion plasma issues:

> Burning Plasma Physics - The achievement and understanding of self-heated plasmas with high gain that have characteristics similar to those expected in a fusion energy source, and
>
> Advanced Toroidal Physics - The achievement and understanding of sustained self-heated plasmas with characteristics (steady-state or high duty factor pulsed systems) similar to those expected in a competitive fusion system

The tokamak is technically ready to address these issues. The plasma performance and duration to study these issues are shown schematically in Fig.1 in terms of the natural time scale for the important plasma processes.

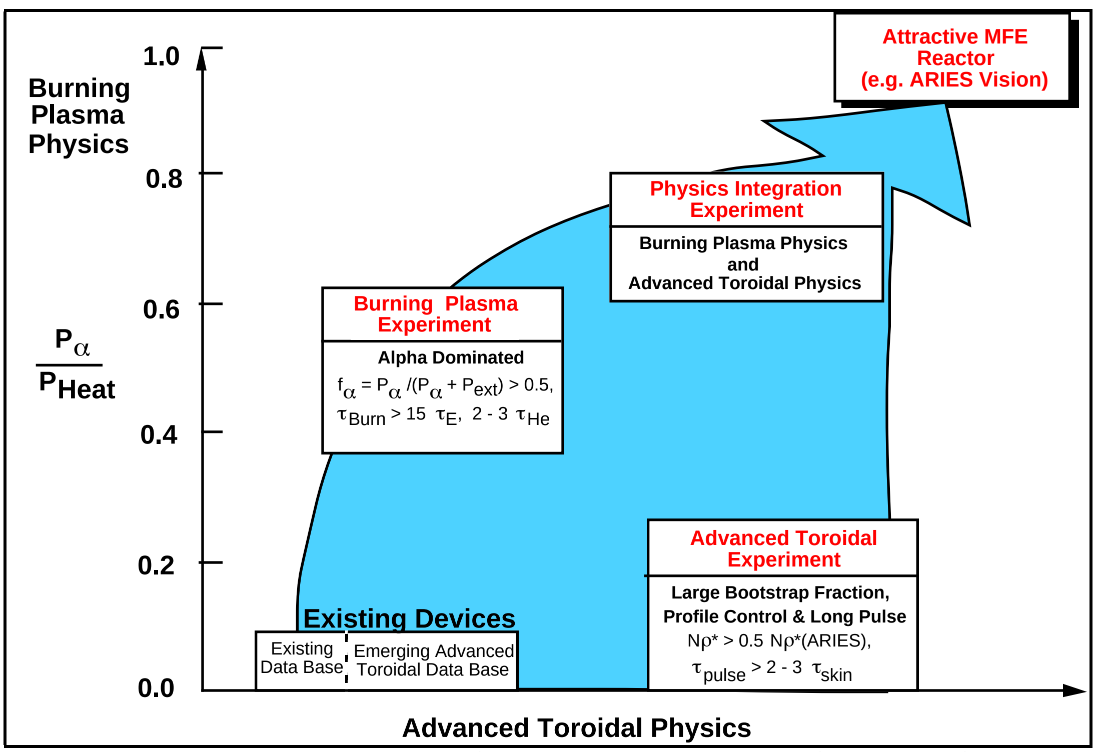

Fig. 1. The stepping stone approach for developing the science foundation for an attractive MFE system.

This report documents the results of a study to evaluate the capability of compact high field tokamaks to address the alpha-dominated burning plasma physics, long-pulse advanced-toroidal physics and fusion technology as part of a Modular Pathway to Magnetic Fusion

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Energy (MFE). The conclusion is that a compact high field tokamak utilizing LN cooled copper-alloy coils has the capability to address a major portion of both the Burning Plasma Experiment Step, the Advanced Toroidal Experiment Step and also has significant capability to integrate burning plasma physics with advanced toroidal physics. The device studied resembles CIT. The plasma configuration was drawn from TPX and is a ≈ 1/3 scale model of ARIES-RS. The size was constrained with the goal to achieve the most important physics goals at a construction cost of <$1B.

The Next Frontier in MFE Research - Exploration, optimization and understanding of alpha-dominated burning plasmas.
The attainment and control of a high Q plasma dominated by alpha heating is the single most important requirement for any approach to fusion power. Fusion gains Q ~ 20 are needed for an economical magnetic fusion reactor that is sustained at near steady-state conditions; at this Q value the alpha particles dominate the plasma dynamics, providing 80% of the plasma heating. The goal for the Next Step in Magnetic Fusion is to access sustained alpha dominated plasmas with alpha heating fractions more than an order of magnitude higher than present experiments.

The advanced tokamak, advanced stellarator and the spherical torus plan to have the bootstrap current, generated by gradients in the pressure profile, produce a large fraction of the current needed to define the stabilizing magnetic field. Since the alpha heating profile is directly linked to the pressure profile, this process becomes very non-linear in the alpha-dominated plasmas required for a fusion reactor based on an advanced tokamak. This coupling of advanced tokamak confinement and MHD stability physics with alpha-dominated plasmas is a key generic issue for the development of attractive toroidal magnetic reactors.

Fusion Ignition Research Experiment (FIRE)- A Next Step Option for MFE.
The mission of FIRE is to attain, explore, understand and optimize alpha-dominated plasmas that will provide the knowledge for 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. This report summarizes the first nine months of a study to evaluate the physics and engineering capabilities of a compact high field tokamak utilizing cryogenically-cooled copper-alloy coils to accomplish this mission.

FIRE Physics Objectives
The physics objectives of FIRE developed to satisfy the mission are to:

1. Determine and understand the conditions required to achieve alpha-dominated plasmas:
    - Energy confinement scaling with alpha-dominated heating
    - β-limits with alpha-dominated heating
    - Density limit scaling with alpha-dominated 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.

Alpha Heating Fraction, the metric for alpha-dominated burning plasmas
The alpha heating strength can be expressed in terms of $f_\alpha = P_\alpha / (P_\alpha + P_{ext})$ where P_α is the alpha heating and P_{ext} is the externally applied heating (ohmic + neutral beam + rf waves). The fraction of alpha heating, f_α, is plotted in Fig. 2 in terms of the ratio $n\tau_E / n\tau_E(Q = \infty)$. D-T experiments on TFTR and JET have measured small temperature increases in agreement with the expected alpha particle heating. The sustained D-T discharges on TFTR and JET had $Q \approx 0.2$ for ~ 10 energy confinement times with the alpha particles providing about 4% of the overall plasma heating. The vision of an MFE fusion reactor is ARIES-RS, an advanced tokamak with a Q = 25 which has $f_\alpha = 0.83$. The investigation of an alpha-dominated plasma can begin at $f_\alpha = 0.5$, and plasmas with $f_\alpha = 0.66$ to 0.83 would match the dynamics expected in the MFE reactor regime. Note that small reductions in confinement produce only small changes in the alpha heating fraction, while Q is sensitive to small changes in confinement, especially in the high Q regime.

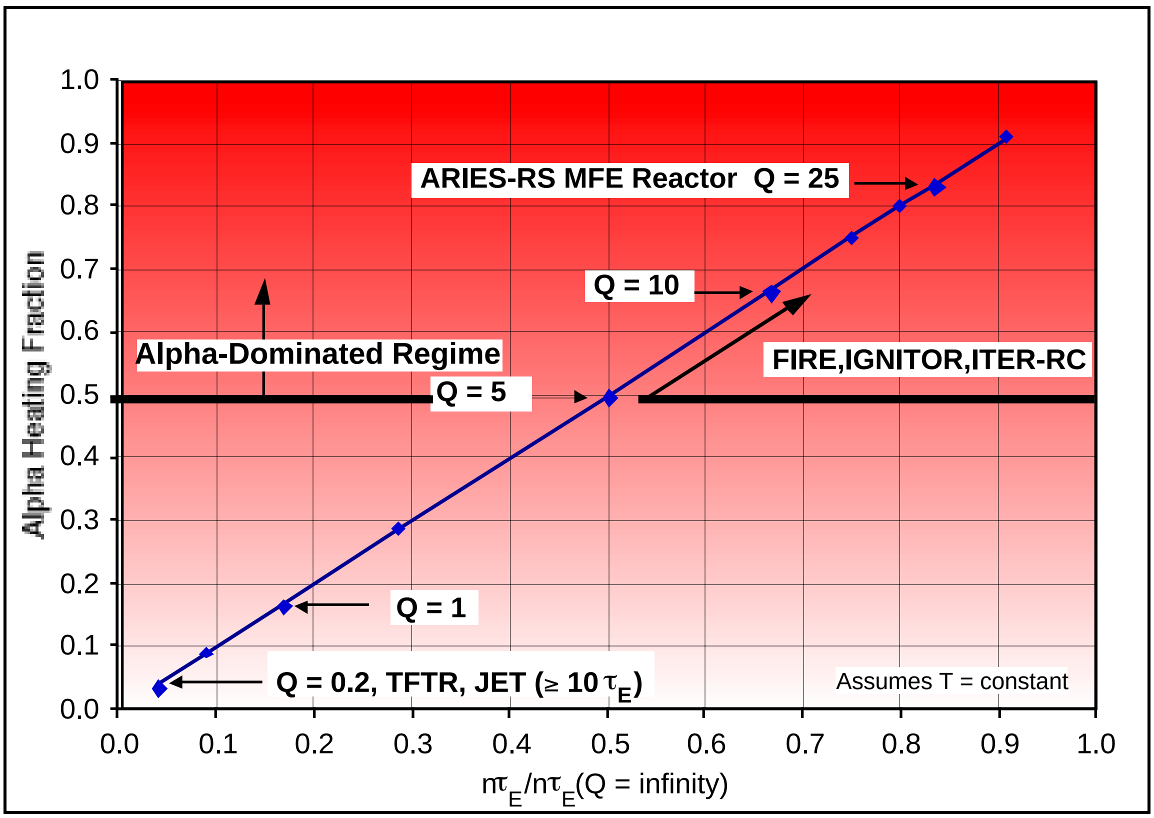

Fig. 2. Fraction of Alpha heating versus $n\tau_E / n\tau_E(Q = \infty)$ illustrating the alpha-dominated regime.

Choice of FIRE Plasma Performance Requirements.
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

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upgrades; it is not a demonstration of the scientific and technological feasibility of magnetic fusion. The strategy for the FIRE program is to have a first stage of burning plasma experiments aimed at accessing the alpha-dominated regime with a minimum f_α of ≥ 0.5 using 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. This capability is the natural starting point for an experimental campaign to study alpha-dominated plasmas and would be sufficient to accomplish a significant fraction of the stated objectives. The goal for the second stage of burning plasma experiments is to achieve strongly alpha-dominated plasmas with f_α = 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:

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

The characteristic time scales for plasma phenomena in FIRE plasmas 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 I.

Table I. Characteristic time scales for plasma phenomena in FIRE and ITER-RC.

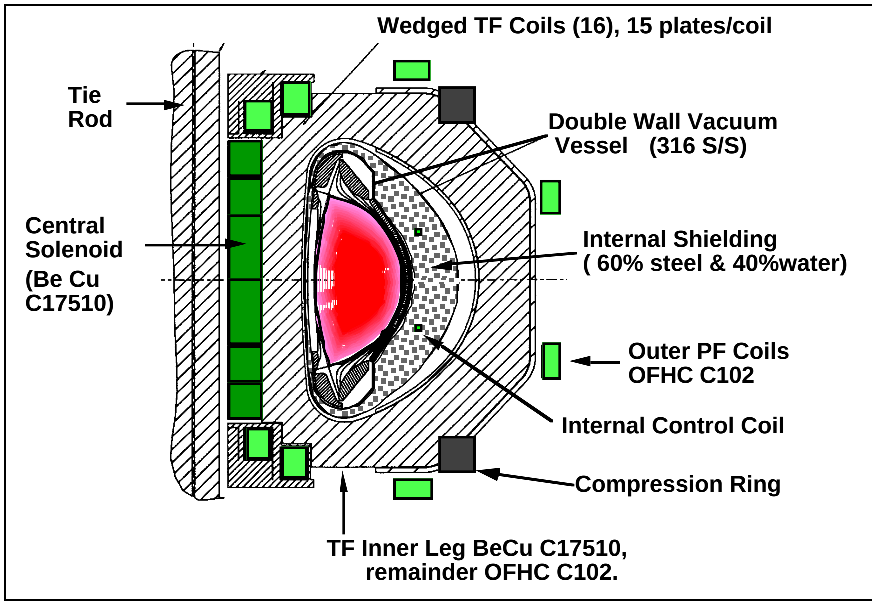
For Q ≈ 10	τ_E (s)	τ_{He} (s)	τ_{cr} (s)	τ_{burn} (s)
FIRE	0.6	3	~13	15
ITER-RC	2.5	7.5	~200	450

(Burn times are for inductively driven plasmas)

A FIRE plasma with a burn time of 15 s (~ 25 τ_E) would allow the pressure profile to come into equilibrium with alpha heating and allow the alpha ash to accumulate for ~ 5 τ_{He} . This pulse length would be sufficient to address Physics Objectives 1, 2, 3, and 5. A significant part of Physics Objective 4 could be accomplished using a current profile that is only partially redistributed with $\tau_{cr} \sim \tau_{burn}$ as in the full field FIRE. 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 ~3τ_{cr} would be sufficient to allow the bootstrap driven current in an advanced tokamak mode to come to within 5% of 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 ~20s at full toroidal field. If advanced tokamak physics improves confinement relative to ITER design guidelines by 25% and β 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 burn time to be increased to ~ 40s ~3τ_{cr}.

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. The capability of FIRE to carry out long-pulse non-burning plasmas experiments will be described in a later section. FIRE will have the flexibility to incorporate new innovations as the ongoing advanced tokamak program develops them. The parameters summarized in Fig. 3 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. A more extensive list of parameters and features is given in Appendix 1.



FIRE Design Goals

R =	2.0 m,
a =	0.525 m
B =	10T,(12T)*
W_{mag} =	3.7 GJ
I_p =	6.5 MA(7.7MA)*
P_{fusion}	~ 200 MW
Q ~	10
τ_E ~	0.6 s
τ_{skin}	~13 s
Burn Time	~ 18 s
Tokamak	~ $0.3 B
Facility	< $1.0 B

* Upgrade potential

Fig. 3. Cross-section view and design goals of the FIRE.

Capability for Alpha-Dominated Burning Plasma Experiments on FIRE

The technical basis for a compact high-field tokamak like FIRE has improved markedly since the CIT (R = 2.14 m) and BPX (R = 2.59 m) studies of 1989-91. Tokamak experiments (1989 -1999) have led to the development of a new scaling relation (e.g., ITER-98H) which predicts 1.3 times higher confinement than the 1989 CIT design assumption. Alcator C-Mod, which can be considered as a prototype of FIRE, has come into operation and demonstrated:

- Confinement of 1.4 times the 1989 CIT design assumptions, ~ 15% higher than the ITER - 98H scaling.
- High power density ICRF heating of high density shaped plasmas with a divertor.
- Detached divertor operation at high power density.

In addition, D-T experiments on TFTR and JET have shown that tritium can be handled safely in a laboratory fusion experiment. The D-T plasmas behaved roughly as expected with slight improvements in confinement for the very weak alpha heating conditions available. The behavior of the energetic alpha particles was in agreement with theoretical expectations.

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 <0.7 of the Greenwald density closer to those in the ITER confinement data base rather than the higher values assumed in the ITER performance projections. 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. The FIRE projections assume the ITER98 Elmy H-mode confinement scaling relation, $\beta_N \leq 2.5$ and the same H-mode power threshold formula as ITER. FIRE takes credit for lower impurity fractions (~ 3% Be) characteristic of high-density tokamak plasmas. In particular, FIRE assumes no significant high-Z impurities in the plasma core.

It is important to note that while these guidelines are quite useful for estimating the $n\tau_E T$ performance of **existing** tokamaks to within 30%, the guidelines are mainly empirical with a modest amount of theoretical understanding and can not accurately predict the performance of a Next Step Burning Plasma experiment much less a technology demonstration. An affordable flexible experiment with a performance capability about midway between today's tokamaks and a fusion reactor is needed to benchmark physics understanding and to serve as a Stepping Stone to a reactor.

The strategy of FIRE is to minimize the extrapolation in τ_E, the most uncertain quantity. The fusion gain is maximized by maximizing $n\tau_E$ at a plasma temperature of ~10 keV. Analysis of the power balance in the plasma, first done by J.D. Lawson, shows that $n\tau_E$ values of ~ 4 x 10^{20} m^{-3} s are required to achieve Q values ~ 10 for a D-T plasma with modest impurity contamination and typical profiles. The compact high field tokamaks (IGNITOR and FIRE) reduce the requirement on τ_E by operating at densities almost an order of magnitude higher than larger lower field devices such as ITER. Operating in the high-density regime $n_e(0) = 6.75 \times 10^{20}$ m^{-3} is a straightforward matter since Alcator C-Mod has already operated up to ~ $10^{21} m^{-3}$. For Q = 10, FIRE requires an energy confinement time, τ_E , of only ~ 0.6 s, which has been achieved in existing tokamaks such as JET, rather than the ~ 2.5 s required in the reduced size ITER or the 6 s required for ignition in ITER. The dimensionless confinement time, $B\tau_E$, is useful to quantify the extrapolation required in plasma energy confinement from present experiments to potential Next Step Options (Table II) for burning plasmas.

Table II. Extrapolation of dimensionless energy confinement time for potential Next Step Options.

	JET	FIRE (Q = 10)	ITER-RC (Q = 10)	ITER-EDA (Q = ∞)
$B\tau_E$ (T-s)	3	6	14	34

The extrapolation to Q ~ 10 conditions in FIRE is a factor of two beyond JET, while ITER-RC requires a factor of four extrapolation, and ITER-EDA required an extrapolation of ~11 to achieve the objective of ignition.

The plasma parameters for a nominal FIRE operating point were calculated using a zero dimensional model and the physics guidelines. The alpha heating fractions for FIRE and ITER-RC are illustrated in Fig. 4 under the assumptions of modestly peaked density profiles (triangles) and flat density profiles (crosses). The initial design point selected for FIRE satisfies all of the standard tokamak design guidelines needed to access the alpha dominated range with $P_\alpha / P_{heat} \geq 0.5$ ($Q \geq 5$). 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 $P_\alpha/P_{heat} \geq 0.8$ required in a fusion reactor.

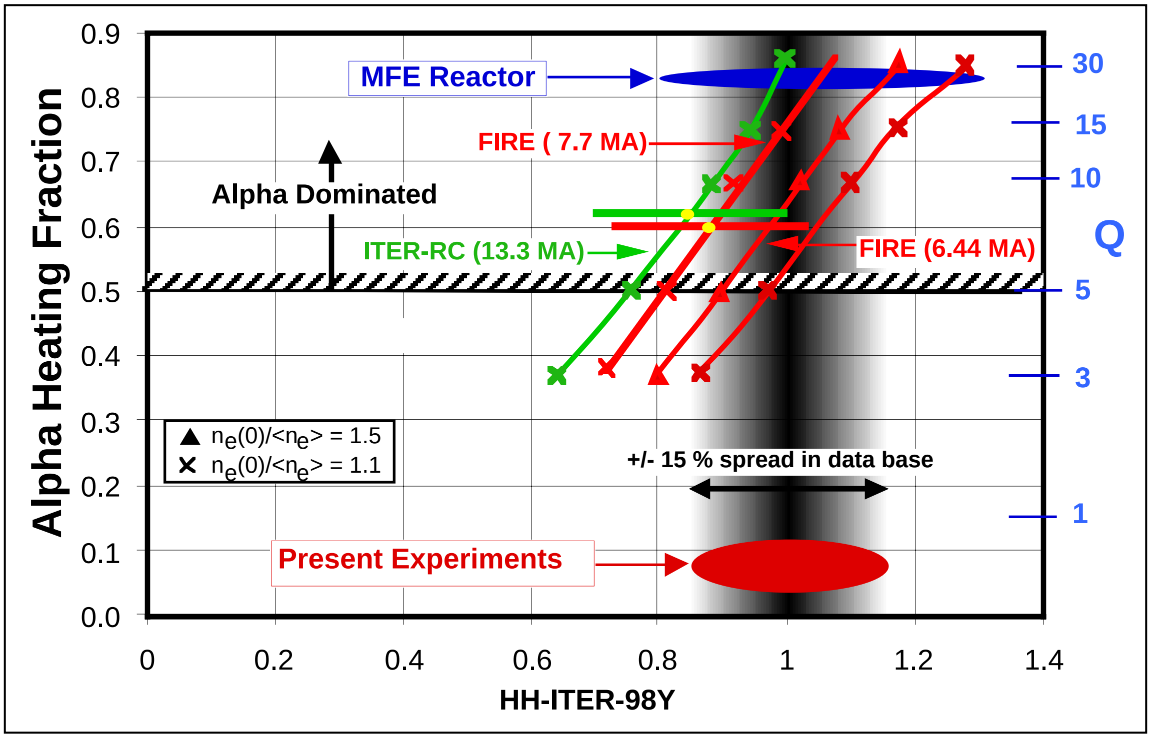


Fig. 4. Performance of FIRE and ITER-RC versus H-mode multiplier. HH = 1.0 is the center of present tokamak H-mode data base (ITER DB3). The triangles are for slightly peaked density profiles, $\alpha_n = 0.5$. The MFE reactor points are for ARIES-RS at densities ranging from 1.0 to 1.8 times the Greenwald density.

The performance projections (Fig. 5) indicate that FIRE is also capable of exploring strongly alpha-dominated regimes with $P_\alpha/P_{heat} \geq 0.66$ (Q = 10 to 30) if the relatively higher performance (H98 = 1.2) of the smaller compact high field tokamak, Alcator C-Mod, or the top end of the JET confinement results are obtained at burning plasma conditions in FIRE. During the next ten years the ongoing world wide advanced tokamak program is expected to provide additional improvements of at least ~25% in confinement and 50% in β. This capability would allow FIRE to explore "ignited" plasma conditions with $P_\alpha/P_{heat} \geq 0.8$ (Q up to 30) at reduced fields and longer pulses with burn duration comparable to several plasma current redistribution times.

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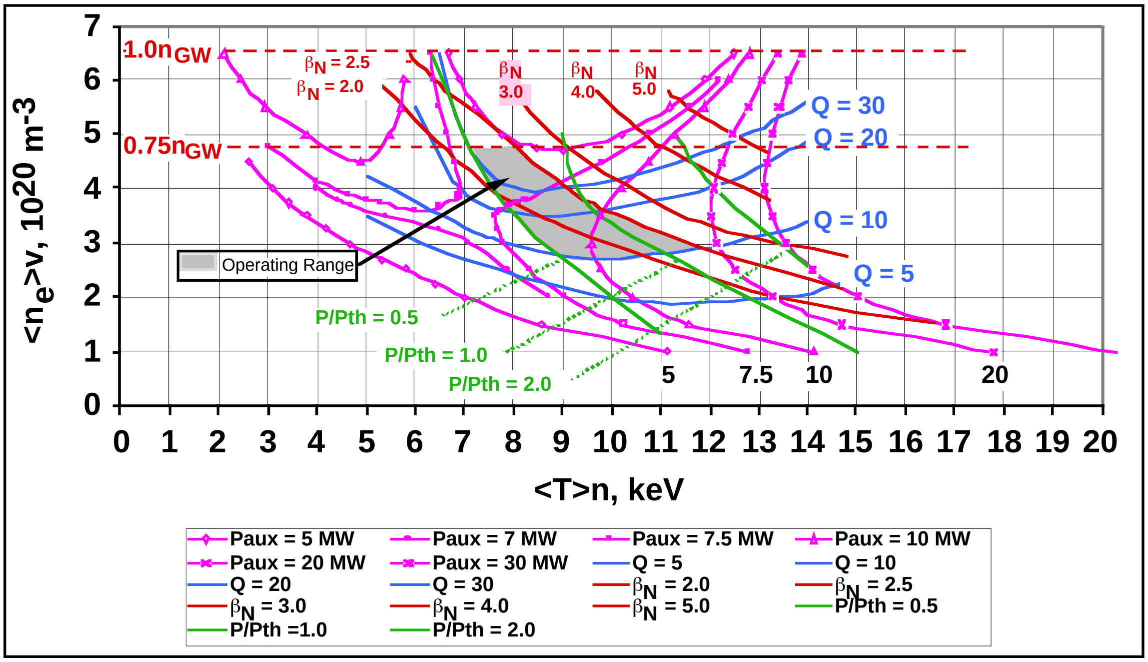

Fig. 5. High Gain (Q >10) FIRE operating range with an ITER-98 H-mode factor = 1.2, $\beta_N \leq 2.5$, $P_{heat} \geq 1.0\ P_{th}$ to access H-mode and $P_{heat} \geq 0.5\ P_{th}$ to remain in the H-mode while satisfying a density limit of $n/n_{GW} \leq 0.75$.

Success with FIRE would establish the basis for an advanced toroidal MFE reactor, namely a smaller and cheaper system obtained by introducing modest advanced features to make the tokamak even more attractive as a fusion reactor concept.

Capability for Long Pulse Advanced Tokamak Experiments in FIRE

The development, exploration and detailed understanding of high-confinement, high-fusion power-density and steady-state (high-duty-cycle) plasmas are needed to provide the basis for economically attractive applications of fusion power. FIRE has significant capability to address the "steady-state" advanced toroidal configuration initiative of the Modular Strategy using non-burning deuterium plasmas.

The issues to be addressed include:

1. Optimization of confinement and β using current and pressure profile control
2. Integration of high performance plasma with long pulse particle and power exhaust
3. Development of controls and techniques to avoid or mitigate the effects of disruptions

The objective of FIRE would be to extend these studies beyond the plasma performance and duration accessible in present tokamaks to values closer to those for a fusion plasma. The physics of a fusion plasma is characterized by the dimensionless parameters, ρ^* (normalized gyro-radius), ν^* (normalized collision frequency) and β. Existing tokamaks are able to replicate the ν^* and β for a fusion reactor plasma but not simultaneously ρ^*, which requires a plasma with a larger Ba. The important time scale for exploring advanced toroidal physics is $\tau_{cr,}$ the time for the plasma current profile to redistribute after a disturbance. The current redistribution time is often called the plasma current skin time. A plasma duration of ~ 3 τ_{cr} would be sufficient to allow the current profile to relax to within

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5% of the steady-state current profile, and would be sufficiently long to address long pulse advanced tokamak physics.

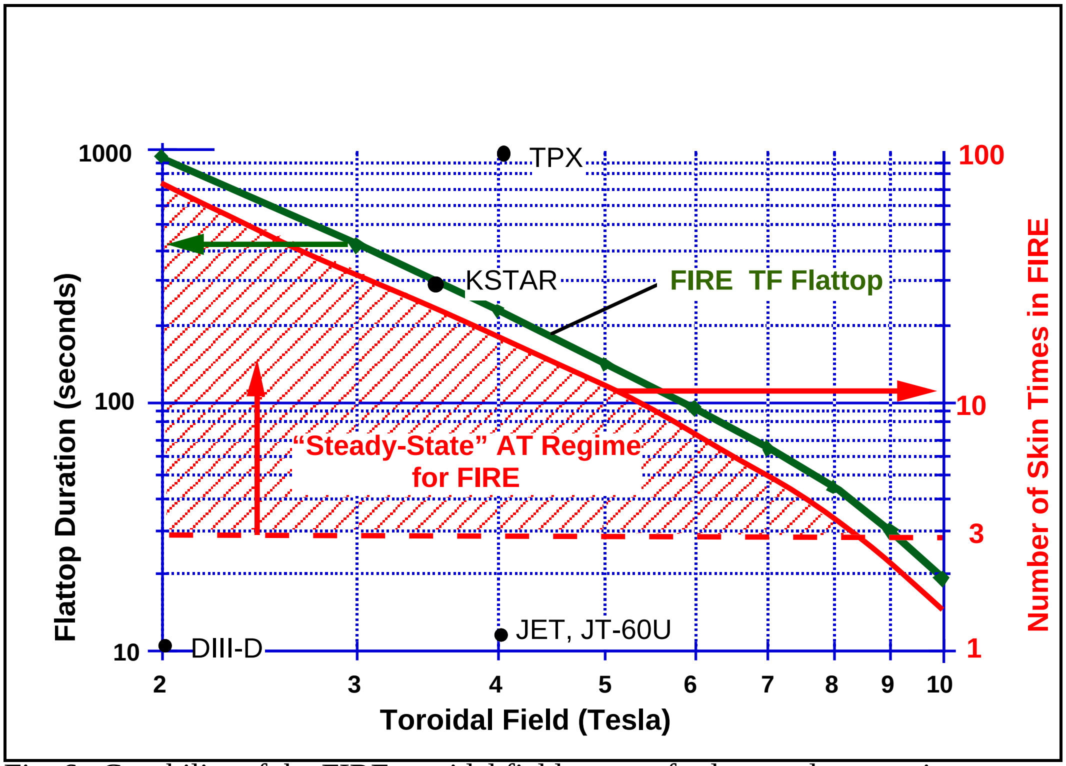

Fig. 6. Capability of the FIRE toroidal field system for long-pulse experiments.

The flattop of the magnetic field in FIRE increases rapidly as the magnetic field is reduced as shown in Fig. 6. In FIRE, the skin time is typically ~13 s at Q ~ 10 parameters due to the small minor radius, so pulses ~40s would allow the plasma current profile to approach within 5% of its equilibrium value. FIRE operated at 8T would be able to extend the range of long-pulse advanced tokamak physics studies a factor of two in ρ* beyond the capability of any existing shaped divertor tokamak or any under construction for "long-pulse" durations ≥ 3 τ_{cr}. The addition of lower hybrid in a later upgrade phase would be suitable for current drive to sustain the long pulse plasmas in FIRE. This capability would be within a factor of two of ρ* in ARIES-RS.

Table III. Comparison of Long Pulse Advanced Tokamak Experiments. (β/ν* = constant)

	DIII-D	JET	KSTAR	TPX	FIRE	ITER-RC	ARIES-RS
B(T)	2.2	2	3.5	4	8	5.5	8
Nρ† = 1/ρ*	0.25	0.27	0.25	0.3	0.45	1.05	1.0
Duration/τ_{cr}	1.7	4	7-100	>100	4	~16	∞

† normalized to 1/ρ* for ARIES-RS, ITER-RC pulse length 3,600 s in driven Q = 5 regime.

Integration of Long-Pulse Advanced Tokamak Physics and Alpha Dominated Burning Plasmas, the critical issue for Advanced Toroidal Configurations, can be explored on FIRE.

It is anticipated that over the next ten years the ongoing world-wide advanced tokamak program (total funding ~$1.2B) will develop the advanced tokamak understanding and techniques to improve confinement (e.g., H98) by 20% and β_N by at least 50%. An

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experiment, such as FIRE, could then exploit these techniques to extend the range of burning plasma experiments and begin to address the critical issue of integrating of advanced tokamak physics with alpha dominated plasmas.

Fusion Technology Experience from FIRE

FIRE would also make significant contributions to the Fusion Technology and Materials Initiative of the Modular Strategy. FIRE will produce reactor-like fusion power density and neutron wall loading, but neither FIRE nor ITER-RC would produce the neutron fluence needed to provide data on neutron damage to structural materials (Table IV). Nonetheless, some information could be obtained from tests involving blanket modules.

Table IV. Fusion neutron parameters for Next Step Options compared to a reactor.

	FIRE	ITER-RC	ARIES-RS
Fusion Power Density (MWm^{-3})	10	0.5	6
Neutron Wall Loading (MWm^{-2})	3	0.6	4
Neutron Fluence (MW y m^{-2}), lifetime	< 0.01	~1	120

The lifetime neutron production in FIRE will be limited to 5 TJ (3×10^{-3} MW-y m^{-2}) to reduce damage to insulators in the toroidal field coil. Consideration will also be given to the use of low activation materials in various components to gain experience for the follow-on Advanced Fusion Integration Facility in the Modular Strategy. Remote handling will be implemented to maintain and replace components inside the vacuum vessel. Shielding inside the double walled vacuum vessel would allow hands-on work in the region outside the TF magnets.

High speed tritium pellet injectors with vertical injection paths inside the magnetic axis and lower speed pellets guided toward the inboard mid-plane will be implemented to improve plasma performance and to reduce tritium inventory. FIRE will also develop and test materials for plasma facing components (W and Be) that are compatible with the fusion reactor requirement of low tritium retention. A fast low inventory tritium re-circulating system will be developed to minimize on-site tritium inventory.

The combination of high magnetic field and advanced tokamak plasma regimes could lead to very attractive fusion reactor concept with high fusion power density. This emphasizes the importance of developing the full capability of high temperature superconductors with high critical fields and high strength structural materials for fusion applications.

Status of FIRE Interim Engineering Study

Detailed 3-D stress analyses have been carried out including the effects of electromagnetic loads, and diffusion of the current and temperature in the presence of nuclear heating. The TF copper stress is ≤76% of the allowable stress and the temperature rise is within design allowables. The TF insulation shear stress is within allowables. A design configuration with a toroidal field flattop ≈ 18.5 s at 10 T or 12 s at 12 T has been developed.

The poloidal coils consist of a segmented OH solenoid with BeCu conductor and shaping coils using OFHC conductor. The poloidal coil stresses and temperatures are within the design allowables. The OH solenoid coil stress is only ~ 40% of the design allowable for BeCu. Higher conductivity copper alloys, with lower strength and cost, are being evaluated for the OH solenoid. The vacuum vessel is a double wall stainless steel vessel similar to that in the TPX and ITER designs. The inter space of the double wall will be

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filled with shielding material to reduce the neutron activation outside the toroidal coil envelope, thereby allowing hands-on maintenance in the region outside the TF coils.

The plasma facing components (PFC) must withstand very high power densities while minimizing tritium retention. For the interim design, carbon is excluded as a PFC material due to the high retention of tritium experienced in TFTR and JET. While schemes might be developed to mitigate the carbon tritium retention issue in FIRE, the mission of FIRE is to develop a material that would also be compatible with the requirements of a high duty cycle reactor. The leading candidate for the first wall material is Be with inertially cooled tiles in the main chamber and Be coating on copper backing plates for stabilizing plates. The power density on the outer divertor plate without peaking factors or the effects of elms is estimated to be ~ 25 MWm^{-2} without a detached or radiative divertor. Tungsten rods mounted on Cu backing plates capable of withstanding 25 MWm^{-2} have been developed for ITER and are the leading candidate for the FIRE divertor plates. The outer divertor plates will be actively cooled during the pulse while the inner plates are cooled between pulses. Plasma modeling is underway to ensure detached or radiative divertor operation so that there will be some margin to accommodate the effects on peaking, elms and disruptions. The engineering analyses of FIRE are evolving rapidly. The most up comprehensive information can be found at http://fire.pppl.gov.

Cost and Schedule Considerations for a Next Step Burning Plasma Experiment.

Cost estimates have not been done for FIRE except to note comparisons with comparably sized devices, CIT (R = 2.14m, 11T) and BPX-AT (R = 2.0m, 10T), which had costs estimated to be in the range of $700M for experiments located at the TFTR site with about $200M of site credits. Some of the physical parameters that drive the costs are toroidal field magnet energy, plasma volume and plasma surface area are compared for an existing large tokamak and for potential Next Step Options in the following Table V.

Table V. Comparison of some important cost drivers for burning plasma devices.

	JET	IGNITOR	FIRE	CIT	BPX	ITER-RC	ITER-EDA
TF Coil Technology	H_20 Cu	30 °K Cu	LN BeCu	LN BeCu	LN BeCu	Nb_3Sn	Nb_3Sn
TF Magnet Energy (GJ)	1.6	4	4	8	8	41	100
Plasma Volume (m^3)	85	11	18	35	65	740	2000
Plasma Surface (m^2)	147	36	60	85	130	640	1200
Cost Range ($B)	~0.5 FY84	~0.5	<1	~0.6 FY89	~1.5 FY92	~5	~10

The schedule to attain an MFE burning plasma test bed is an important consideration in the strategy and planning of the MFE program. Plans for the ITER-EDA required 15 years after the initiation of construction before D-T operation would be initiated; a 12 year construction period followed by three years of hydrogen and deuterium operation before D-T experiments. A similar time period would be required for ITER-RC. If ITER-RC construction were initiated in 2002, then D-T operation would begin ~ 2017. A small compact high field tokamak like IGNITOR or FIRE is expected to have a construction period of ~ 7 years followed by 3 years of hydrogen and deuterium operation based on previous schedule estimates for CIT and BPX. If construction were initiated on IGNITOR or FIRE in 2002, then D-T operation could begin in ~ 2012, which is ~ 5 years later than the initiation of similar inertial fusion burning plasma experiments using NIF.

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Siting Considerations for a Next Step Burning Plasma Option

The siting requirements for a FIRE/IGNITOR class device is much reduced relative to that required for an ITER class device, thereby significantly reducing the time and cost. The lowest cost Next Step Option of the FIRE/IGNITOR class would be to utilize an existing site such as TFTR or JET. However, these sites might limit later upgrades, and might preclude a natural progression of follow-on devices. Consideration should be given to the possibility of a unified site for the development of fusion which would support not only the development of tokamak burning plasma experiments, but tests of burning plasmas in other magnetic configurations as well as providing an infrastructure for plasma based fusion technology facilities.

The Modular Pathway for Magnetic Fusion Energy

The Next Steps in the Modular Strategy beyond the existing programs and facilities could include a compact high field tokamak(s) of the IGNITOR/FIRE class, a superconducting high performance advanced tokamak of the JT-60SU class and specialized fusion technology facilities to address the first wall problem for MFE. The fusion technology facilities might include intense neutron sources for materials damage studies and facilities to investigate the viability of renewable first wall concepts such as liquid surfaces facing the plasma. These facilities would feed into an Assessment of Magnetic Fusion milestone in ~ 2015, which if successful would lead to the construction of an Advanced Fusion Integration Facility with characteristics similar to those of an attractive MFE reactor. In this scenario a commercial fusion power plant could be operating midway through the 21st century. A more detailed description of the Modular Strategy can be found on the World Wide Web at <http://nso.ucsd.edu/chapter3.pdf>.

High Leverage Activities and Issues

The initial FIRE studies have shown that the power handling capability of the divertor and first wall limit the pulse length in FIRE. Continued 3-D modeling of power and particle handling with experimental benchmarks will be essential in defining the divertor design. Engineering studies are needed to develop a vacuum vessel design capable of handling reactor level neutron power densities. These are generic fusion issues that can be addressed on FIRE. Experimental and theoretical studies to clarify the scaling of the L to H-mode power threshold and the H to L mode power threshold for plasmas significantly below the Greenwald density would be important in optimizing the design for both standard regimes as well as advanced reverse shear regimes. Much more work is needed to develop self-consistent advanced tokamak modes of operation and to evaluate stability to alpha driven instabilities in advanced tokamak configurations. More comprehensive 1 1/2-D simulations of standard and advanced regimes including burn control techniques are needed.

Summary

Exploration, understanding and optimization of alpha-dominated plasmas is a critical issue for all approaches to fusion. The tokamak is the most cost-effective vehicle to investigate alpha-dominated plasma physics, and its coupling to advanced toroidal physics for MFE. The performance of a burning plasma depends sensitively on the details of confinement, β-limits, density limits and edge plasma conditions. This uncertainty can only be reduced by studying actual alpha-dominated plasmas in the laboratory in conjunction with other advanced toroidal experiments and improved numerical simulations. The compact high field tokamak offers the possibility of addressing the important alpha-dominated plasma issues, many of the long pulse advanced tokamak issues and beginning the integration of alpha-dominated plasmas with advanced toroidal physics in a ~$1B class facility.

Acknowledgements
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Appendix 1. 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 (=18.5 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	< 30 g, Category III Low Hazard Facility

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

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RC ITER: An Opportunity to Study Burning Plasmas and Develop Fusion Technology in a Reactor Relevant Device

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I. Introduction

During the last year of the ITER EDA, the ITER parties recognized that the cost of ITER, although in line with that estimated at the conclusion of the CDA, was nevertheless an insurmountable barrier to entering into a construction agreement. However, the Parties also recognized that the arguments leading to the formation of the ITER collaboration in 1986 were still valid and that the goals originally envisioned for ITER were not diminished in their validity. It was thus decided to attempt a redesign of the EDA device, in which the original goals and objectives would be retained as much as possible but with a cost objective of about half that of the EDA design. A number of names and acronyms have been used to refer to the redesign; in this paper it will be called Reduced Cost (RC) ITER.

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. Although there seems to be an emerging consensus toward selection of the IAM option, no official choice between these variants has yet been made, and the design and performance of both the IAM and LAM will be described below.

2. Objectives

As implied by the title of this paper, RC ITER has both scientific and technological objectives, and these are, as much as possible, in line with the objectives established for the EDA design. Specifically, with regard to *plasma performance*, the device should:

- 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.

In addition, the possibility of controlled ignition should not be precluded.

In regard to *engineering performance and testing* the device should:

- 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.

3. IAM and LAM Designs

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. NbTi conductor is used for the PF coils in both designs, except for the CS which is wound from $NbSn_3$ conductor. Cross sections of the two designs are shown in Figure 1 where it can be seen that the access in LAM is somewhat better than that in IAM.

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. Moreover, both IAM and LAM designs are responsive to concerns, raised particularly by members of the US physics community, regarding the performance and flexibility of the FDR device. Both designs have a segmented central solenoid which permits stronger shaping, and both designs achieve their reference performance with $n < n_{GW}$. Further, recent results obtained by the Edge Expert Group [1] have revealed a size scaling for the width of the edge pedestal that is more favorable for overall confinement projections than the poloidal gyroradius scaling proposed in [2]. And finally, more attention has been paid to the steady-state operation of RC ITER, including the incorporation of substantial current drive capability in an advanced tokamak mode, as the promise of these modes continues to be supported by results obtained by essentially every tokamak within the worldwide tokamak community.

	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
I_P(MA) (q_{95} = 3)	13.3	17	21
B_o (T)	5.51	4.23	5.68
Ignited/Burn Pulse Length (s)	450	450	1000
Elongation κ_{95}, κ_X	1.68, 1.83	1.74, 1.92	1.6, 1.75
Ave Triangularity δ_X	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
β, β_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.

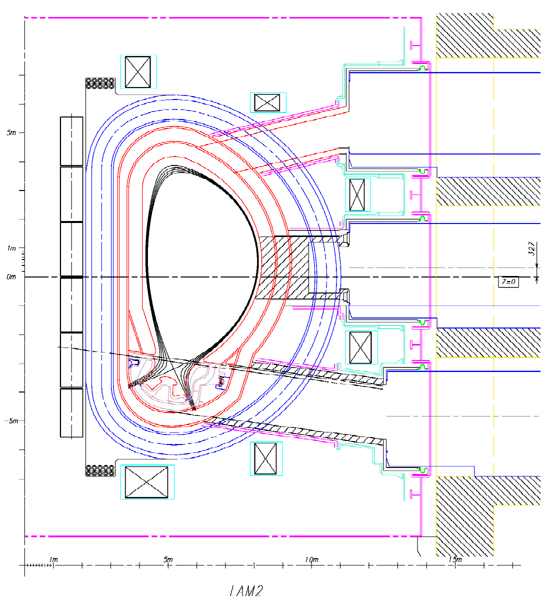

Fig 1a. Cross section of IAM design.

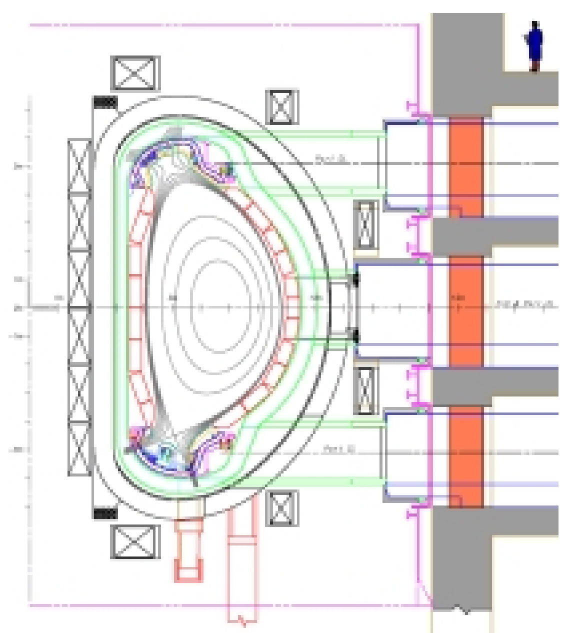

Fig 1b. Cross section of LAM design.

4. Operating Regimes and Performance Margin

4.1 Inductive performance

As shown in Figure 2a and 2b, both the IAM and LAM have reasonable margin in obtaining their baseline performance. The shaded area in the figures corresponds to the region of parameter space simultaneously below the Greenwald density and $\beta_n < 2.5$, but above the L-H transition scaling in the plane of fusion power vs. H_H, where H_H is the factor by which confinement exceeds IPB98(y,1) H-mode confinement scaling. Thus, within the 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. In particular, as shown in Figure 3, the window shrinks to essentially a point for IAM with nominal q=3 operation, while a small domain for ignition is predicted to exist for LAM even at q=3.

4.2 Non-inductive performance

Achieving a steady-state $Q \geq 5$ requires modest improvement in confinement and normalized β. Shown in Figure 4 are Q and β_n vs. the effective current drive power with H_H as a parameter. Here, the current drive efficiency nIR/P_{CD} is assumed to scale linearly with temperature, and γ^* is the current drive efficiency at T = 10 keV. In all cases the thick lines correspond to IAM and the thin to LAM. Also, in Figure 4b, the three cases are for H_H = 1.5, 1.25 and 1, as in Figure 4a. Thus, for example, with $\gamma^* = 0.2$ and P_{CD} = 70 MW, Q ~ 5 is possible with H_H = 1.25 and $\beta_n \sim 3.5$. Note that the current drive performance is slightly better in IAM than in LAM and that in both designs, advanced tokamak operation is required to achieve the steady-state Q =5 goal.

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.

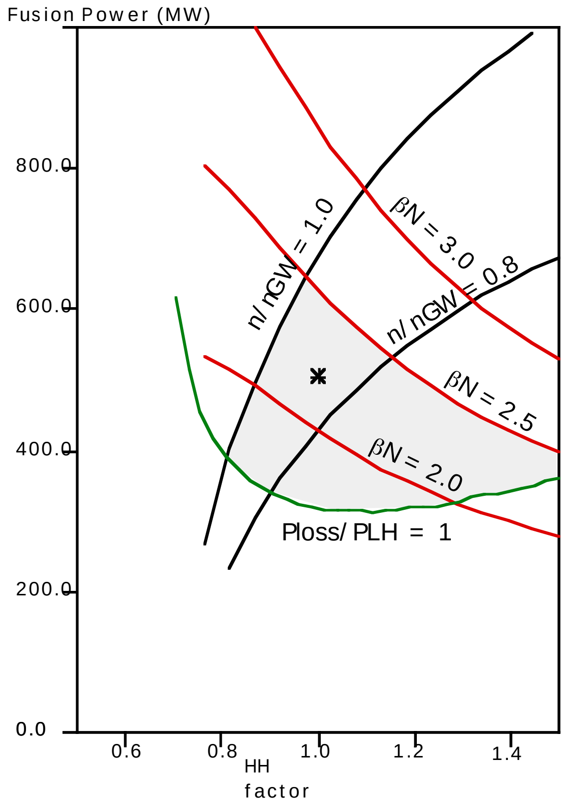

Fig 2a. IAM operating domain for Q=10.

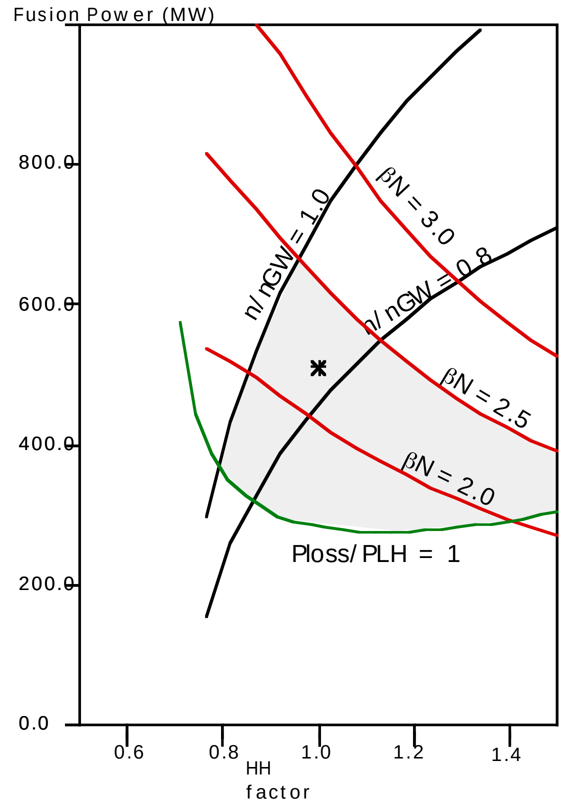

Fig 2b. LAM operating domain for Q=10.

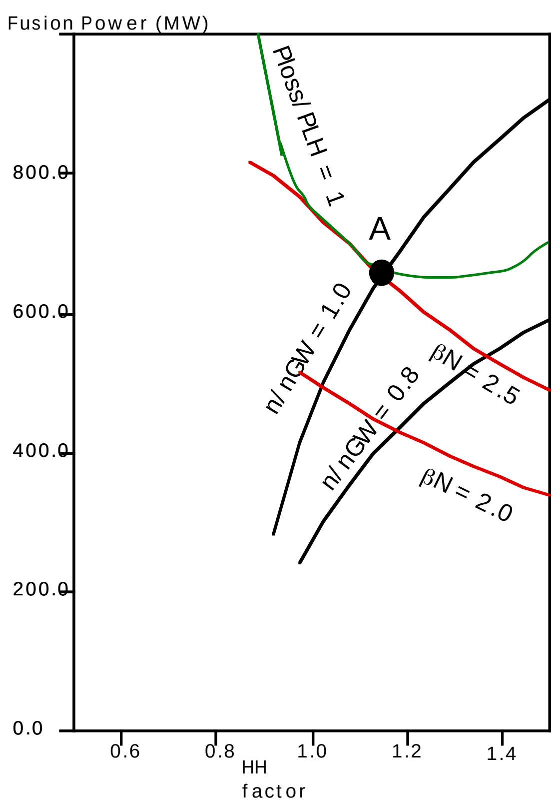

Fig. 3a. IAM ignited domain

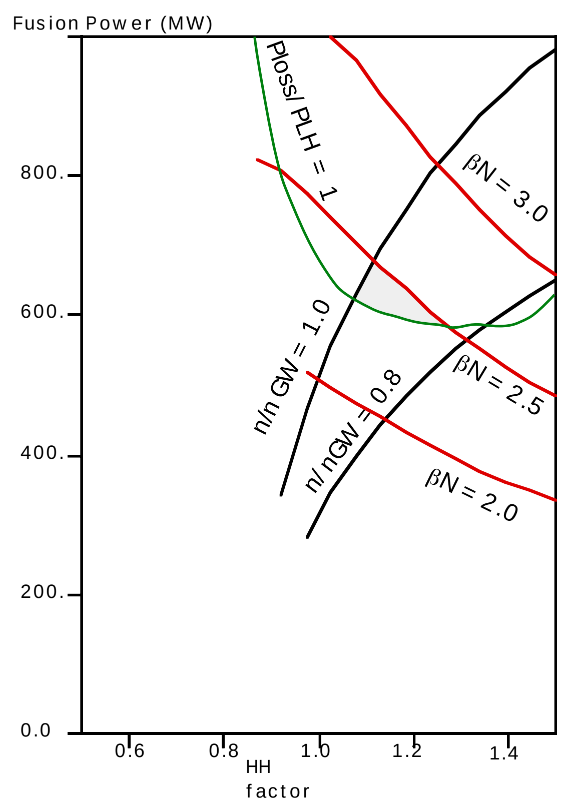

Fig. 3b. LAM ignited domain

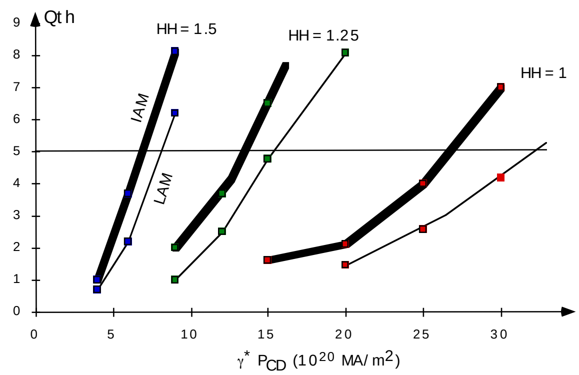

Fig4a. Q vs. effective current drive power with H-mode multiplier as parameter.

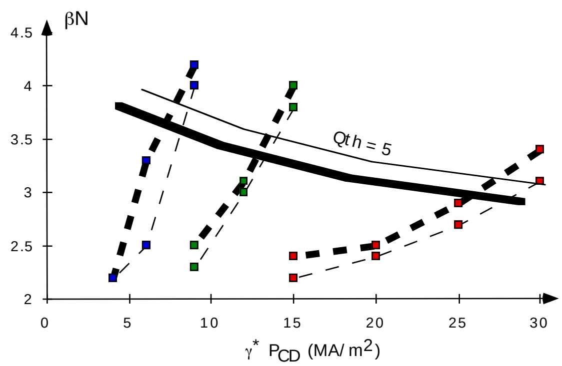

Fig4b. β_n vs. effective current drive power with H-mode multiplier as parameter. The three sets of dashed curves correspond from left to right to the H_H values in Fig 4a.

5. 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.

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&CD. The latter will be injected through two ports and can be made up of 40 MW of a single H&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.

As important as adequate H&CD is to a burning plasma experiment is implementation of a comprehensive set of state-of-the-art diagnostics. Extensive planning for the diagnostics has been done for RC ITER and a list of the diagnostics presently foreseen is presented in Table 2. 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. This point should be borne in mind when comparing a machine in the RC ITER class with lower cost, compact ignition experiments using copper coil technology.

Magnetic Diagnostics	Optical/IR Systems (Cont'd)	**Microwave Diagnostics**
Vessel wall sensors	Divertor Thomson Scattering	Electron Cyclotron Emission (ECE)
Divertor Magnetics	Toroidal Interferometric/ Polarimetric System	Main Plasma Reflectometer
Continuous Rogowski Coils	Polarimetric System (Poloidal Field Measurement)	Plasma Position Reflectometer
Diamagnetic Loop	Collective Scattering System	Divertor Reflectometer & Divertor ECA
Neutron Diagnostics	**Bolometric Systems**	Main Plasma Microwave Scattering
Radial Neutron Camera	Array For Main Plasma & Array For Divertor	Fast Wave Reflectometry
Vertical Neutron Camera	Spectroscopic and NPA Systems	**Plasma-Facing Components and Operational Diagnostics**
Micro-fission Chambers (In-Vessel)	Charge eXchange Recombination Spectroscopy (CXRS) based on DNB	IR/Visible Cameras
Neutron Flux Monitors (Ex-Vessel)	Motional Stark Effect (MSE): based on heating beam	Thermocouples
Radial Neutron & Gamma-Ray Spectrometer	H Alpha Spectroscopy	Pressure Gauges
Activation System (In-Vessel)	Main Plasma & Divertor Impurity Monitors	Residual Gas Analyzers
Lost Alpha Detectors	X-Ray Crystal Spectrometers	Hard X-Ray Monitor
Knock-on Tail Neutron Spectrometer	Visible Continuum Array	IR Thermography (Divertor)
Optical/IR Systems	Soft X-Ray Array	Langmuir Probes
Core Thomson Scattering	Neutral Particle Analyzers	**Diagnostic Neutral Beam**
Edge Thomson Scattering	Two Photon Ly-Alpha Fluorescence	
X-Point Thomson Scattering	Laser Induced Fluorescence	

Table 2. Diagnostic systems planned for RC ITER

6. Concluding remarks

A machine in the RC ITER class is the optimum way in which burning plasma physics can be fully and relevantly studied:

4 Simultaneous scaling of the dimensionless parameters ν^*, ρ^* and β to reactor-level values can most closely be achieved in a device of the RC ITER scale;

5 Burning plasma phenomena pertaining to steady-state AT physics can best be investigated in machines with essentially steady-state capability (and can only be adequately investigated in machines with T_{pulse} > L/R);

6 Sufficient access for diagnostics, auxiliary heating and current drive, and remote removal and installation of in-vessel components, is essential in order to carry out a meaningful burning plasma experimental program. Such access can be fully realized only in a device of the RC ITER scale.

The cost of either RC ITER option is presently foreseen to be about 55% of that of the FDR, or ~ 3 B$ (1989). Approximately 10% of the cost could be deferred until after the start of operations. A 15% partnership in constructing RC ITER in Europe or Japan would cost the US ~ 60 M$ per year (1999 dollars) over a 10 year period. Thus, participating in an RC ITER project as a partner would be by far the most cost effective way for the US to significantly advance its program in fusion science and technology in the area of burning plasmas.

The price paid for a substantial cost reduction in ITER is a lowering of the baseline performance from ignition to a gain of about 10. However, achieving a moderate Q, steady-state plasma is actually more useful for advancing the tokamak concept than achieving pulsed ignition or even very high Q. Since tokamaks require auxiliary power to drive a steady-state current, there is no possibility to realize an ignited, steady-state tokamak reactor. The best that can be envisioned, at present, is a steady-state tokamak with a Q of perhaps10-20 and, in that sense, RC ITER has precisely the right goal. Further, nearer term applications of fusion neutrons, such as burning weapons-grade Pu or spent fuel from fission reactors, do not require very high Q's to be competitive with other approaches, for example spallation neutron sources. In the relatively near term the fusion program would be better served by emphasizing applications requiring moderate Q and steady-state, rather than pure ignition which is not required, nor possible, for steady-state tokamak reactors. The RC ITER step is fully consistent with this paradigm shift and the US should vigorously support its construction as a participating partner at the earliest opportunity.

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