2002 Fusion Summer Study

REPORT

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1.0 Executive Summary

The 2002 Fusion Summer Study was conducted from July 8-19, 2002, in Snowmass, Colorado, and carried out a critical assessment of major next-steps in the fusion energy sciences program in both Magnetic Fusion Energy (MFE) and Inertial Fusion Energy (IFE). The conclusions of this study were based on analysis led by over 60 conveners working with hundreds of members of the fusion energy sciences community extending over 8 months. This effort culminated in two weeks of intense discussion by over 250 US and 30 foreign fusion physicists and engineers present at the 2002 Fusion Summer Study. The objectives of the Fusion Summer Study were three-fold:

- Review the scientific issues in burning plasmas, address the relation of burning plasma in tokamaks to innovative MFE confinement concepts, and address the relation of ignition in IFE to integrated research facilities.
- Provide a forum for critical discussion and review of proposed MFE burning plasma experiments (IGNITOR, FIRE, and ITER) and assess the scientific and technological research opportunities and prospective benefits of these approaches to the study of burning plasmas.
- Provide a forum for the IFE community to present plans for prospective integrated research facilities, assess the present status of the technical base for each, and establish a timetable and technical progress necessary to proceed for each.

In the MFE program, the world is now at a major decision point: to go forward with exploration of a burning plasma, opening up the possibility of discoveries in a plasma dominated by self-heating from fusion reactions and filling this crucial and now missing element in the MFE program.

In the IFE program, the decision to construct a burning plasma experiment has already been made. The National Nuclear Security Administration is currently building the National Ignition Facility (NIF) at Lawrence Livermore National Laboratory. The NIF, and other facilities worldwide are expected to provide the needed data on inertial fusion burning plasmas. The IFE questions examined at the Fusion Summer Study revolve about the pace of development of the additional sciences and technologies needed for power production.

1.1 Magnetic Fusion Energy

Fusion energy shows great promise to contribute to securing the energy future of humanity. The science that underlies this quest is at the frontier of the physics of complex systems and provides the basis for understanding the behavior of high temperature plasmas. Grounded in recent excellent progress in the study of magnetically confined plasmas, the world is now at a major decision point: to go forward with exploration of a burning plasma, opening up the possibility of discoveries in a plasma dominated by self-heating from fusion reactions.

This exciting next step to explore burning plasmas is an essential element in the Fusion Energy Science Program whose mission is to “Advance plasma science, fusion science and fusion technology—the knowledge base needed for an economically and environmentally attractive fusion energy source.” The study of burning plasmas will be carried out as part of a program that includes advancing fundamental understanding of the underlying physics and technology, theory and computational simulation, and optimization of magnetic confinement configurations.

The participants of the 2002 Fusion Summer Study developed major conclusions regarding the opportunities for exploration and discovery in the field of magnetically confined burning plasmas. The principal conclusions are summarized below:

1. **The study of burning plasmas, in which self-heating from fusion reactions dominates plasma behavior, is at the frontier of magnetic fusion energy science. The next major step in magnetic fusion research should be a burning plasma program, which is essential to the science focus and energy goal of fusion research.**

The study of burning plasmas is a crucial and missing element in the fusion energy sciences program. It will make a large step forward in demonstrating magnetic fusion as a source of practical fusion energy for several applications, *e.g*., electric power generation and hydrogen production.

The tokamak is now at the stage of scientific maturity that we are ready to undertake the essential step of burning plasma research. Present experimental facilities cannot achieve the conditions necessary for a burning plasma. A new experimental facility is required to address the important scientific issues in the burning plasma regime. The conditions needed to study the key physics phenomena expected in the burning plasma state have been identified.

Burning plasmas afford unique opportunities to explore, for the first time in the laboratory, high-temperature-plasma behavior in the regime of strong self-heating. Production of a strongly self-heated fusion plasma will allow the discovery and study of a number of new phenomena. These include the effects of energetic, fusion-produced alpha particles on plasma stability and turbulence; the strong, nonlinear coupling that will occur between fusion alpha particles, the pressure driven current, turbulent transport, MHD stability, and boundary-plasma behavior. Specific issues of stability, control, and propagation of the fusion burn and fusion ignition transient phenomena would be addressed.

Recent physics advances in tokamak research, aimed at steady-state and high performance, demonstrate the potential to significantly increase the economic attractiveness of the tokamak. Therefore, Advanced Tokamak (AT) research capability is highly desirable in any burning plasma experiment option.

Physics and technology learned in a tokamak-based burning plasma would be transferable to other configurations. Scientific flexibility, excellent diagnostics, and close coupling to theory and simulation are critical features of a program in burning plasmas. Such a program would contribute significantly to the physics basis for fusion energy systems based on the tokamak and other toroidal configurations. The experience gained in burning plasma diagnostics, essential to obtaining data to advance fusion plasma science, will be highly applicable to burning plasmas in other magnetic configurations.

2. The three experiments proposed to achieve burning plasma operation range from compact, high field, copper magnet devices to a reactor-scale superconducting-magnet device. These approaches address a spectrum of both physics and fusion technology, and vary widely in overall mission, schedule and cost.

The following mission statements were provided by the proposing teams:

IGNITOR is a facility whose mission is to achieve fusion ignition conditions in deuterium-tritium plasmas for a duration that exceeds the intrinsic plasma physics time scales. It utilizes high-field copper magnets to achieve a self-heated plasma for pulse lengths comparable to the current redistribution time. IGNITOR will study the physics of the ignition process and alpha particle confinement as well as the heating and control of a plasma subject to thermonuclear instability.

FIRE is a facility whose mission is to attain, explore, understand and optimize magnetically confined fusion-dominated plasmas. FIRE would study burning plasma physics in conventional regimes with Q of about 10 and high-beta advanced tokamak regimes with Q of about 5 under quasi-stationary conditions. FIRE employs a plasma configuration with strong plasma shaping, double-null poloidal divertors, reactor level plasma exhaust power densities and pulsed cryogenically cooled copper coils as a reduced cost approach to achieve this mission.

The overall objective of ITER is to demonstrate the scientific and technological feasibility of fusion energy. ITER would accomplish this objective by demonstrating controlled ignition and extended burn of deuterium-tritium plasmas, with steady-state as an ultimate goal, by demonstrating technologies essential to a reactor in an integrated system, and by performing integrated testing of the high heat flux and nuclear components required to utilize fusion energy for practical purposes.

Construction schedules were reported as 5 years for IGNITOR, 6 years for FIRE, and 9 years for ITER. FIRE is not at the same level of readiness as ITER and IGNITOR and will require some additional time to be ready for construction. ITER must complete international negotiations and agreement before construction can commence.

Cost information was obtained from the ITER and FIRE teams and was assessed within the limited resources available for the Snowmass work. All costs were converted to 2002-US dollars. ITER assumes an international cost-sharing approach while FIRE costs are estimated as a US project.

- The purpose of the ITER cost information is to provide accurate estimates of the relative "value" of all the tasks necessary for construction to facilitate international negotiations on task sharing. The cost information is based on a large engineering effort (about 1000 professional person years {PPY}) and a large R&D effort (about $900M) with prototypes of all key components. Also, the ITER cost information (about 85 procurement packages) is based on input from the industries in all the parties. The estimate of the ITER total "value", when converted to 2002 US dollars, is about $5 billion. The actual cost estimate is to be developed by each party using its own procedures, including the use of contingency. Thus, the ITER cost information does not included explicit contingency.
- The US will need to carefully estimate the cost of any potential contributions to ITER. These estimates should include adequate contingency and any additional required R&D to mitigate against potential cost increases.
- The estimate for FIRE is about $1.2 B including about a 25% contingency. It is based on an advanced pre-conceptual design using in-house and some vendor estimates. However, substantial further engineering is needed as well as some supporting R&D.
- As an Italian project, IGNITOR has been designed in detail with supporting R&D. It has a detailed cost estimate that is confidential for business purposes and was not made available to the assessment team.

3. IGNITOR, FIRE, and ITER would enable studies of the physics of burning plasma, advance fusion technology, and contribute to the development of fusion energy. The contributions of the three approaches would differ considerably.

- **IGNITOR offers an opportunity for the early study of burning plasmas aiming at ignition for about one current redistribution period.**
- **FIRE offers an opportunity for the study of burning plasma physics in conventional and advanced tokamak configurations under quasi-stationary conditions (several current redistribution time periods) and would contribute to plasma technology.**
- **ITER offers an opportunity for the study of burning plasma physics in conventional and advanced tokamak configurations for long durations (many current redistribution time periods) with steady state as the ultimate goal, and would contribute to the development and integration of plasma and fusion technology.**

The three candidate burning plasma devices would contribute a number of key benefits, i.e., capabilities for studies of the physics and technology of burning plasmas (under the assumption that each facility will achieve its proposed performance).

Common benefits from all three candidate burning plasma devices include the following:

Physics

- Strongly-coupled physics issues of equilibrium, stability, transport, wave-particle interactions, fast ion physics, and boundary physics in the regime of dominant self-heating.

Technology

- Plasma support technologies (heating, fuel delivery, exhaust, plasma-facing components, and magnets) will benefit most because parameters and plasma conditions will be close to those required for power production.
- Nuclear technologies (remote handling, vacuum vessel, blankets, safety and materials) will advance as a result of the experience of operating in a nuclear environment. The level of benefit will depend on tritium inventory, pulse length, duty factor, and lifetime fluence.

<u>Key benefits from IGNITOR are the following:</u>

Physics

- Capability to address the science of self-heated plasmas in a reactor-relevant regime of small ρ^* (many Larmor orbits) for globally MHD-stable plasmas at low β_N (normalized plasma pressure).
- Capability to study sawtooth stability at low beta with isotropic alpha particles and self-consistent pressure profile determined by dominant alpha heating.

Technology

- Development of high-field copper magnets with advanced structural features, including bucking and wedging and magnetic press.
- Development of high-frequency RF antennas for wave heating in a burning plasma environment.

<u>Key benefits from FIRE are the following:</u>

Physics

- Capability to address the science of self-heated plasmas in reactor-relevant regimes of small ρ^* (many Larmor orbits) and high β_N (normalized plasma pressure) with a large fraction of non-inductive current sustained for up to a few current relaxation times.
- Exploration of high self-driven current regimes with strong shaping and active MHD stability control.
- Study of removal of helium ash and impurities with exhaust pumping.

Technology

- Development of electrical insulation for high-field pulsed copper magnets in a high neutron fluence environment.
- Development of high heat flux plasma-facing components with steady-state heat removal capability (tungsten/beryllium).

<u>Key benefits from ITER are the following:</u>

Physics

- Capability to address the science of self-heated plasmas in reactor-relevant regimes of small ρ^* (many Larmor orbits) and high β_N (plasma pressure), and with the capability of full non-inductive current drive sustained in near steady state conditions.
- Exploration of high self-driven current regimes with a flexible array of heating, current drive, and rotational drive systems.
- Exploration of alpha particle-driven instabilities in a reactor-relevant range of temperatures.
- Investigation of temperature control and removal of helium ash and impurities with strong exhaust pumping.

Technology

- Integration of steady-state reactor-relevant fusion technology: large-scale high-field superconducting magnets; long-pulse high-heat-load plasma-facing components; control systems; heating systems.
- Testing of blanket modules for breeding tritium.

4. There are no outstanding engineering-feasibility issues to prevent the successful design and fabrication of any of the three options. However, the three approaches are at different levels of design and R&D.

There is confidence that ITER and FIRE will achieve burning plasma performance in H–mode based on an extensive experimental database. IGNITOR would achieve similar performance if it either obtains H–mode confinement or an enhancement over the standard tokamak L–mode. However, the likelihood of achieving these enhancements remains an unresolved issue between the assessors and the IGNITOR team.

The three options are at very different stages of engineering development.

- ITER and IGNITOR have well-developed engineering designs.
- ITER has been supported by a comprehensive R&D program. Also, ITER has demonstrated full-scale prototypes for essentially all major components of the fusion core and their maintenance.
- FIRE is at the advanced pre-conceptual design level. It has benefited from previous R&D for CIT/BPX/IGNITOR and, most recently, from ITER R&D.
- IGNITOR has carried out R&D and built full-size prototypes for essentially all major components.

Projections for the three options are based on present understanding of tokamak physics.

- Based on 0D and 1.5D modeling, all three devices have baseline scenarios which appear capable of reaching Q [illegible]5 with the advocates' assumptions. ITER and FIRE scenarios are based on standard ELMing H–mode and are reasonable extrapolations from the existing database.
- IGNITOR's baseline scenarios, based on cold edged L–mode, depend on a combination of enhanced energy confinement and/or density peaking. An unresolved issue arose as to whether an adequate database exists (proposers) or does not exist (assessors) for assessing confinement projections in the proposed IGNITOR operational modes: L–mode limiter or H–mode with x-point(s) near the wall. Further research and demonstration discharges are recommended.
- More accurate prediction of fusion performance of the three devices is not currently possible due to known uncertainties in the transport models. An ongoing effort within the base fusion science program is underway to improve the projections through increased understanding of transport.
- Each device presents a reasonable set of advanced scenarios based on present understanding. ITER and FIRE have moderate- and strong-shaping respectively and the control tool set needed to address the issues of high beta and steady-state related to Advanced Tokamak regimes. FIRE has the capability to sustain these regimes for one to three current redistribution times, while ITER has the capability to sustain these regimes for up to 3000s allowing near steady-state operation. IGNITOR presents credible advanced performance scenarios using current ramps and intense heating to produce internal transport barriers on a transient basis.

A number of issues have been identified and are documented in the body of the report. For example, on ITER and FIRE, the predicted ELM-power loads are at the upper boundary of acceptable energy deposition; ELM-control and amelioration is needed. On FIRE, control of the neoclassical tearing mode by lower hybrid current drive is not sufficiently validated. Also, FIRE has a concern about radiation damage of magnet insulators. On ITER, tritium retention is a concern with carbon-based divertor materials. These issues are the subjects of continuing R&D.

5. The development path to realize fusion power as a practical energy source includes four major scientific elements:

- **Fundamental understanding of the underlying science and technology, and optimization of magnetic configurations**
- **Plasma physics research in a burning plasma experiment**
- **High performance, steady-state operation**
- **Development of low-activation materials and fusion technologies**

A diversified and integrated portfolio consisting of advanced tokamak, ICCs, and theory/simulation is needed to achieve the necessary predictive capability. A burning plasma experiment should be flexible and well-diagnosed in order to provide fundamental understanding.

Fusion power technologies are a pace-setting element of fusion development. Development of fusion power technologies requires:

- A strong base program including testing of components in a non-nuclear environment as well as fission reactors.
- A materials program including an intense neutron source to develop and qualify low-activation materials.
- A Component Test Facility for integration and test of power technologies in fusion environment.

An international tokamak research program centered around ITER and including these national performance-extension devices has the highest chance of success in exploring burning plasma physics in steady-state. ITER will provide valuable data on integration of power-plant relevant plasma support technologies. Assuming successful outcome (demonstration of high-performance AT burning plasma), an ITER-based development path would lead to the shortest development time to a demonstration power plant.

A FIRE-based development plan reduces initial facility investment costs and allows optimization of experiments for separable missions. It is a lower risk option, as it requires “smaller” extrapolation in physics and technology basis. Assuming a successful outcome, a FIRE-based development path provides further optimization before integration steps, allowing a more advanced and/or less costly integration step to follow.

IGNITOR allows early demonstration of an important fusion milestone, burning plasmas with a low initial facility investment cost. Because of its short pulse length, IGNITOR cannot thoroughly investigate burn control and/or advanced tokamak modes. IGNITOR could be an element of a portfolio of experiments supporting ITER-based or FIRE-based development scenarios.

6. A strong base science and technology program is needed to advance essential fusion science and technology and to participate effectively in, and to benefit from, the burning plasma effort. In particular, the development path for innovative confinement configurations would benefit from research on a tokamak-based burning plasma experiment.

It has been a much-affirmed premise of the current fusion energy program that a strong base program forms a foundation for the field. The base program develops a broad array of underlying fusion physics and technology, and provides the knowledge base to optimize the magnetic configuration for plasma confinement. The science associated with burning plasma science requires a major step beyond the base program. The science associated with a significant variety of other critical, fundamental issues constitutes the base program.

The base program is also essential to the successful and full exploitation of a burning plasma effort. United States participation in a burning plasma experiment clearly requires a cadre of fusion physicists and engineers. In addition tokamak experiments are needed to contribute to the database that helps guide and influence a burning plasma experiment. For the US to benefit fully from a burning plasma experiment requires not only experimentalists and engineers, but also theorists and computational scientists who can interpret the results, and generalize them for application to future tokamak experiments and non-tokamak configurations.

The development of innovative confinement configurations would benefit from a burning plasma experiment based on the tokamak configuration. Research in innovative configurations is essential for the broad development of fusion science and for the evolution of an optimal approach to fusion energy. The results from a tokamak burning plasma experiment will be sufficiently generic to accelerate the development of other toroidal fusion configurations. The tokamak shares many physics features with the spectrum of toroidal configurations, including nonaxisymmetric tori (the stellarator family), axisymmetric tori with safety factor $q > 1$ (including advanced tokamaks and spherical tokamaks), and axisymmetric tori with $q < 1$ (including the reversed field pinch, spheromak, and field reversed configurations). The behavior of alpha particles in these configurations is expected to have features in common, so that tokamak results can influence research in other configurations.

There are many geometric differences between a tokamak and these neighboring configurations; however, if the results from a tokamak burning plasma experiment are understood at the level of fundamental physics, then these results can be transferred through theory and computation. This transferability is expected to apply to the classical confinement of alpha particles, alpha-generated instabilities, the effect of alpha particles on existing instabilities, the effect of turbulence and MHD instabilities on alpha confinement, and aspects of burn control. Clearly, the transferability is largest for configurations that are geometrically closest to the tokamak. However, nearly all physics results obtained in the tokamak configuration have had influence on the large family of toroidal configurations, and it seems clear that this influence will extend to results from tokamak burning plasma experiments.

The technological information learned from a tokamak burning plasma experiment will strongly apply to other configurations. Areas of technology transfer include superconducting magnets, plasma facing components, fueling, heating sources, blankets and remote handling.

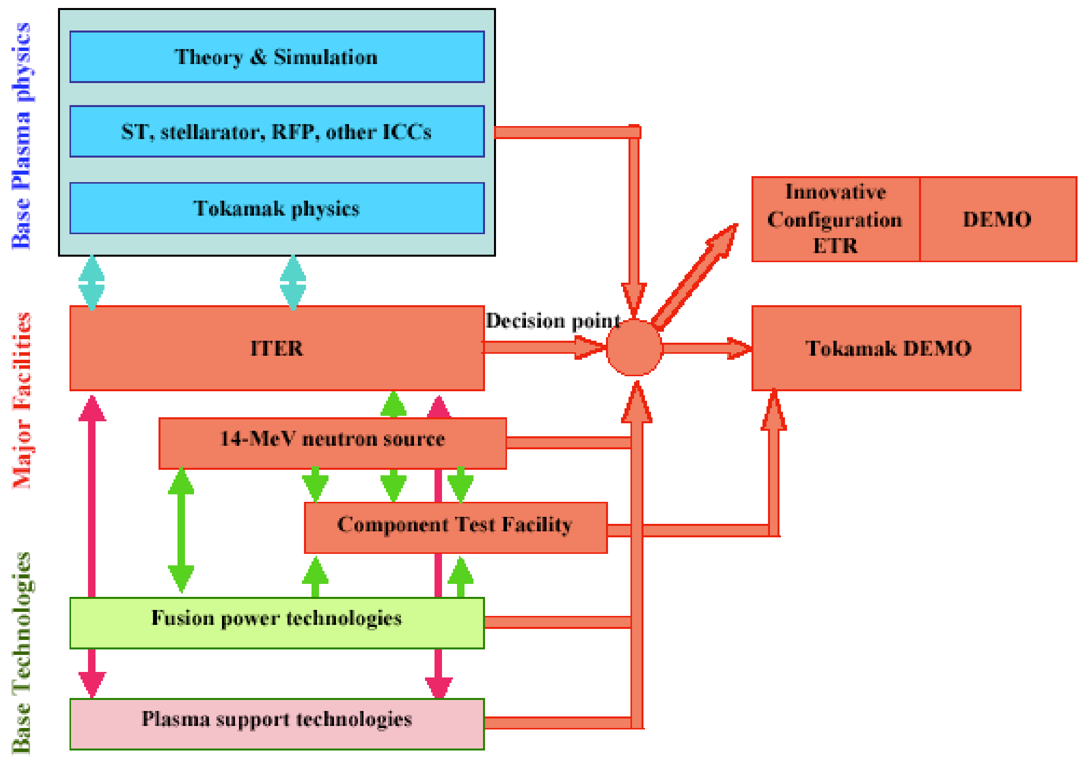


Figure 1.1.1. Schematic of development path based on ITER-class burning plasma experiment.

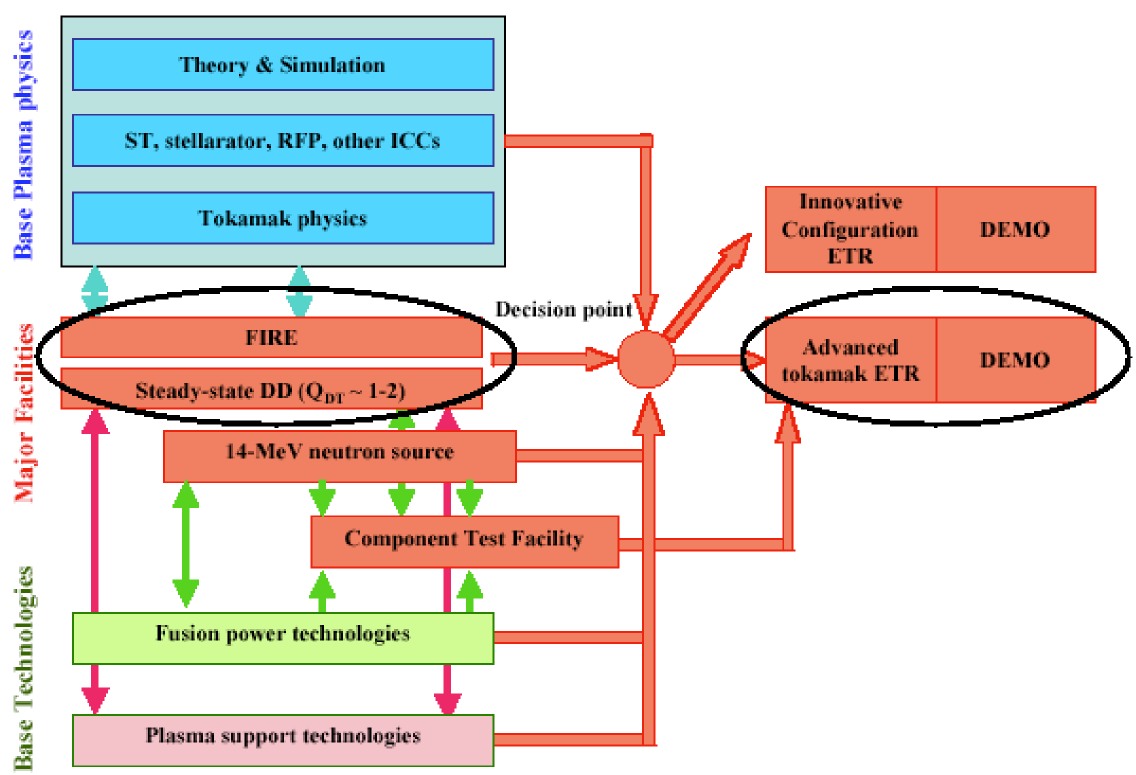


Figure 1.1.2. Schematic of development path based on FIRE-class burning plasma experiment.

1.2 Inertial Fusion Energy

In 1990 the Fusion Policy Advisory Committee recommended that Magnetic Fusion Energy and Inertial Fusion Energy be developed in parallel. This policy was reaffirmed by the Fusion Energy Sciences Advisory Committee in 1999 and the Secretary of Energy Advisory Board in 2000.

As noted earlier, the programmatic issues facing inertial and magnetic fusion are quite different. The burning plasma experiments for inertial fusion, namely the National Ignition Facility (NIF) in the United States and the LMJ in France, are already under construction. Currently plasma ignition on NIF is expected around FY2010, depending on future funding decisions about the pace of funding for diagnostics and cryogenic capabilities. Existing facilities in the United States (e.g., Omega, Z, and Nike) and other facilities worldwide are providing information leading to burning plasma experiments at the NIF and at the LMJ. The domestic facilities have been built, or are being built, under the auspices of the National Nuclear Security Administration (NNSA), primarily for defense purposes.

Although the NIF will provide the needed data on burning IFE plasmas, it does not have the capability to operate at high repetition rates or to manage the fusion power that high repetition rates produce. Moreover the NIF has neither the efficiency nor the durability needed for commercial power production. Substantial scientific and technical issues must be studied and resolved in parallel to enable high repetition rates, good efficiency, and adequate lifetime. The modularity of IFE drivers and the separability of power plant components make it possible to study these issues and issues associated with supporting subsystems in scaled facilities. The IFE community refers to these facilities as “integrated research facilities” or IREs. They are the next major steps in inertial fusion. They are expected to be substantially less expensive than either the magnetic burning plasma experiment or the NIF. While the NIF can demonstrate the creation of fusion energy in single shots, the IREs will provide the foundation of science and technology needed for the subsequent demonstration of net fusion power, and the delivery of net fusion electricity to the grid.

Overview of IFE

An IFE power plant will produce energy by focusing intense beams of light or charged particles, or concentrating intense x rays, onto a small target containing fusion fuel. The fuel will ignite with a burst of fusion reactions releasing much more energy than was invested to cause ignition. The fusion heart of the power plant will have several important systems:

- The fusion targets containing the fuel.
- A factory designed to fabricate millions of targets per year.
- A chamber approximately 6 meters or more in diameter to capture the energy produced by the fusion pulses.
- An injection system to inject or place the targets into the chamber.
- A driver to produce the energy needed for ignition.
- A focusing or concentration system to deliver the driver energy to the target.

There are several types of drivers and focusing systems, many different types of targets, and several types of chambers. To some extent these systems or components are independent so there are many possible combinations. This independence allows modular, cost-effective research on key issues with synergy among the integrated concepts.

There are currently three main kinds of drivers: heavy ion accelerators, lasers, and z pinches driven by pulsed power. The drivers are expected to be the single most expensive part of the power plant. There are substantial research programs in heavy ion accelerators and in krypton fluoride lasers (KrF) and diode pumped solid-state lasers (DPSSLs). The heavy ion fusion program is currently funded through the Office of Fusion Energy Sciences (OFES) and the laser programs are funded through NNSA. There is a smaller, concept exploration program in z pinches that builds on an expanding z-pinch program supported by the NNSA for defense purposes. There are also important IFE programs in target physics, target fabrication (including mass production techniques), target injection, chambers, and focusing systems. These programs are funded through both OFES and NNSA. Despite different funding sources, all the various inertial fusion research programs are very well coordinated.

As noted above, there are many types of targets. In all IFE targets the fusion fuel is compressed before it is ignited. There are two broad methods of compression and two methods of ignition. The fuel is compressed either through an implosion driven directly by the driver beams (direct drive) or by converting the driver energy to x rays that then drive the implosion (indirect drive). The two classes of ignition are hot-spot ignition and fast ignition.

Chambers also fall into a number of general types. The types currently receiving the most attention are dry-wall chambers, wetted-wall chambers, and chambers in which the wall is protected by thick liquid layers. Often the dry wall chambers contain some gas to protect the wall from x-rays, charged particles, and target debris. The wetted walls use thin liquid layers on the wall or sprays of fluid in the chamber to do the same. Thick liquid layers are used to protect the wall from neutrons as well as from x-rays, charged particles, and target debris.

Although there are many possible combinations of drivers, targets, and chambers, resources do not allow the exploration of all combinations. Each integrated approach puts most of its effort into the combination that currently appears to be most compatible. The various combinations of drivers, targets, and chambers must work together and not all combinations are equally compatible or self-consistent. Currently the laser programs emphasize directly driven targets and dry wall chambers. The heavy ion and z pinch programs emphasize indirectly driven targets and thick liquid wall protection.

IFE Plans

Several years ago the IFE community developed a program plan or roadmap (Figure 1.2.1) leading to the integrated research experiments and ultimately to a demonstration power plant.

This plan has three phases preceding a demonstration power plant. The first phase contains research elements at the levels referred to as concept exploration and proof of principle. The IFE community has developed specific milestones that must be met at each level before a concept is ready to advance to the next step.

Phase I, the current phase of the plan, consists of the research preceding the integrated research experiments. In this phase there are research programs in target physics, target fabrication and injection, fusion chambers, and driver physics and technology. The target physics program includes research on the so-called fast ignition approach; an approach that may lead to higher target energy gains at reduced driver energy. In Phase I, two laser driver facilities, Electra and Mercury are under construction and are making excellent progress. Electra is a krypton fluoride (KrF) laser and Mercury is a diode pumped solid-state laser. The needed Phase I facility for heavy ion fusion, the Integrated Beam Experiment (IBX), has not yet been approved. Another driver option, the z-pinch approach, is currently being studied but does not yet have official Department of Energy funding.

In Phase II, those drivers that meet their milestones advance, using the IREs, to the point that the driver information, together with the NIF and advanced research in chambers and target technologies provide the information to determine if IFE is ready to proceed to an Engineering Test Facility (ETF). The ETF will provide a test bed for demonstration of all IFE plant systems at reduced scale, including tritium breeding and recovery and power conversion, as well as accelerated materials and component reliability testing. Information from scaled testing of all IFE plant subsystems will be used in decision making to determine if a full-scale IFE demonstration power plant (Demo) should be built. If the decision is positive, the ETF will also provide the information that is necessary to design and build all plant systems in the Demo.

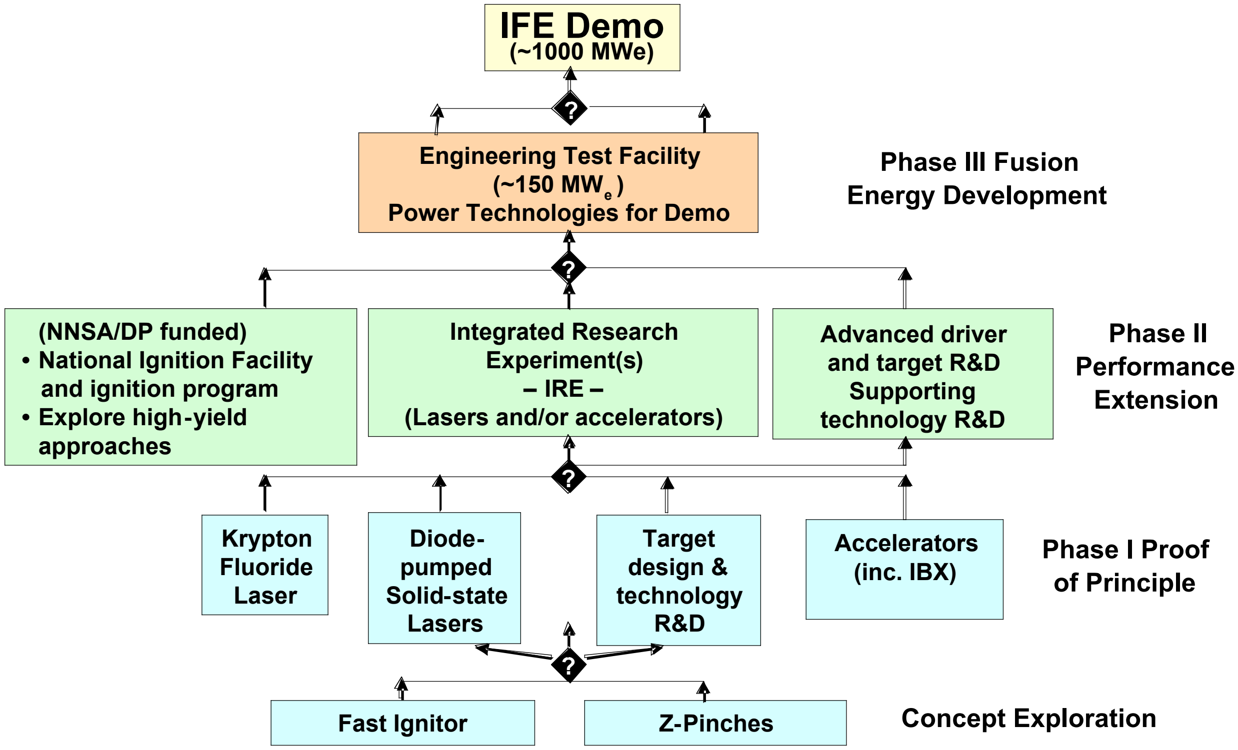


Figure 1.2.1 The Inertial Fusion Energy Roadmap

Assessment of the plans and status at this workshop led to three important conclusions:

1) The various driver programs (lasers, heavy ion accelerators, and Z-pinches) are advancing at different rates because of funding differences and their relative maturity. The most advanced programs are unlikely to be in position to propose an integrated research experiment for several years.

2) The inertial fusion community (both proponents and critics of the individual approaches) believes that the Phase I research plans are sound and that they address the correct technical issues.

3) Phase I funding rates are the programmatic issue. Resolution of this issue will require coordination of the inertial and magnetic programs.

Regarding the second conclusion, it is important to note that there is less agreement about Phase II and some of the quantitative aspects of the milestones needed to advance to Phase II. The various disagreements must be resolved by additional workshops and peer review.

Regarding the third conclusion, there are several important points: The laser approaches have been funded, as a Congressional initiative through NNSA, at approximately the rate recommended by the Fusion Energy Sciences Advisory Committee in 1999. Nevertheless, there is an important issue: The budgets submitted by NNSA do not contain funding for these important laser activities so future funding is uncertain. The heavy ion fusion approach, funded through the Office of Fusion Energy Sciences, has been funded at a little over half the rate recommended by the Fusion Energy Sciences Committee; and, as noted above, the IBX has not yet been approved. The z-pinch approach has not been officially funded and the target physics program (including fast ignition research) is inadequately funded.

Progress and Issues

Despite the significant near-term issues relating to funding, there has been important progress since the last Snowmass Summer Study. We conclude with a summary of this progress and a summary of some of the important remaining issues:

- Laser systems have made impressive progress in efficiency, pulse rate, and lifetime. Efficiency and lifetime remain important issues for KrF lasers. Cost of major components and beam quality are important issues for solid state lasers.
- The heavy ion fusion program has made excellent progress in basic beam science. Several new science experiments have recently begun operations. Fielding integrated experiments (for example the IBX) at moderate beam energy and current and focusing intense beams in the chamber environment remain the important technical issues.
- There has been impressive progress in z-pinch targets and good progress in conceptual power plant designs. Producing economical recyclable transmission lines at low cost remains the most important issue.
- Recent calculations indicate that fluid instabilities in the targets may be controlled by appropriate choice of pulse shape. Both directly driven and indirectly driven targets appear to be feasible.
- Chamber technology and target fabrication and injection are being placed on a sound scientific basis. For example, experiments on dry-wall damage limits are underway. Scaled hydraulics experiments have identified nozzle designs that can create all liquid jet configurations required for thick liquid chambers, and a target injection experiment is under construction. For heavy-ion fusion there is now a chamber design where the final focus magnets and chamber structures have predicted lifetimes exceeding 30 years.
- There is broad international interest in fast ignition. If fast ignition is successful, it will produce higher energy gains than conventional targets. So far the target experiments have been encouraging, particularly the recent Japanese results. Fast ignition power production is at a rudimentary level for all drivers. An integrated research plan is required.

2.0 Introduction

The Fusion Summer Study 2002 was a forum for the critical technical assessment of major next-steps in the fusion energy sciences program. It provided crucial community input to the long-range planning activities undertaken by the DOE and the FESAC. It was an ideal place for a broad community of scientists to examine goals and proposed initiatives in burning plasma science in magnetic fusion energy and integrated research experiments in inertial fusion energy. The meeting was open to every member of the fusion energy science community and included significant international participation.

The scientific and technological views of the participants provided critical fusion community inputs to:

- the decision process of FESAC and DOE in 2002-2003, and
- the review of burning plasma science by the National Academy of Sciences called for by FESAC and Energy Legislation which was passed by the House of Representatives [H. R. 4].

The Snowmass MFE Study feeds into FESAC and NRC reviews by providing an expert consensus view on key issues:

- a clear articulation of the scientific basis for proceeding with a burning plasma experiment,
- identification of principal new physics phenomena and experimental requirements for their study, and
- a uniform technical assessment of approaches to burning plasma research.

The Study was held in Snowmass, Colorado, in July 2002 and benefited from the participation of roughly 280 scientists and engineers, 30 of them from outside the United States. It built on earlier planning activity at the 1999 Snowmass Fusion Summer Study and the scientific assessments at the UFA-sponsored Burning Plasma Science Workshops (Austin, Dec 2000; San Diego, May 2001). The Summer Study was open to every member of the fusion energy science community, both MFE (tokamaks and other concepts) and IFE, and significant international participation was encouraged so that the study could gain the widest range of inputs.

3.0 Magnetic Fusion Energy Next steps

3.1 MFE Approaches to Burning Plasmas

Three approaches to the study of burning plasmas were considered during the Snowmass meeting: IGNITOR, the Fusion Ignition Research Experiment (FIRE), and the International Thermonuclear Experimental Reactor (ITER). Each proposed experiment is described, in the words of its advocates, in the following sections.

3.1.1 IGNITOR

Ignitor is the first experiment that has been proposed and designed to achieve fusion ignition conditions in magnetically confined deuterium-tritium plasmas. Demonstration of ignition, the study of the physics of the ignition process, and the heating and control methods for a burning, magnetically confined plasma are the three most pressing issues in present day research on nuclear fusion and they are specifically addressed by the Ignitor experiment.

The machine is characterized by an optimal combination of high magnetic fields ($B_T \simeq 13$ T), compact dimensions ($R_0 \simeq 1.32$ m), relatively low aspect ratio ($R_0/a \simeq 2.8$) and considerable plasma cross section elongation and triangularity ($\kappa \simeq 1.83$, $\delta \simeq 0.4$). The reference central density for which ignition can be achieved is about 10^{21} m^{-3}. The corresponding line-averaged density is well below the known density limit, that is related to the average plasma current density. The considered plasma current is $I_p \simeq 11$ MA, and the duration exceeds all the intrinsic physical time scales.

Ignition can be achieved by ohmic heating alone shortly after the end of the current ramp phase. The peak temperature, at ignition, is expected to be about $T_{e0} \simeq T_{i0} \simeq 11$ keV for an energy confinement time $\tau_E \simeq 0.6$ sec (see Table 3.1.1.1).

The first wall, lining the plasma chamber with molybdenum tiles, acts as an extended toroidal limiter and the expected peak thermal power loads do not exceed 1.8 MW/m^2. The poloidal field system of Ignitor can also produce magnetic divertor configurations with two up-down symmetric X-points, at 9 MA and $q_{95}(\psi)>3$, to facilitate the access to the so-called H-mode regime. A preliminary analysis of the thermal loads at the strike points with X-points near the first wall indicates that they are acceptable with the first wall as presently designed.

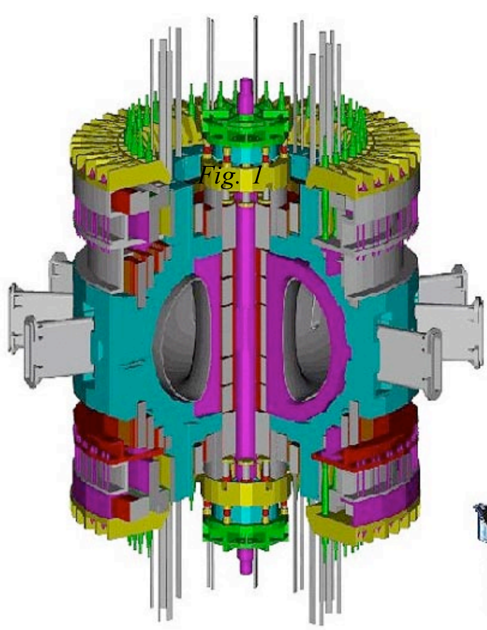


Figure 3.1.1.1

Experiments have shown that attaining high density plasmas is more important for good impurity screening than including a divertor system in the machine design. High density plasmas have higher neutral particle density and lower temperature at the plasma edge. In these regimes, in the absence of transport barriers, the level of impurity contamination has consistently been found to be low by a variety of experiments over the last 25 years. In fact, the standard view of the divertor as the dominant power and particle sink has been challenged by recent experiments, where particle recycling from the main chamber and cross field diffusion in the outer region of the plasma column are observed to play an increasingly important role at higher densities.

A system for the injection of radio frequency power at the ion cyclotron frequency ($\approx 100 - 140$ MHz) is included in the machine design. In order to gain significant control over the evolution of the temperature and current density profiles, and to shorten the time needed to reach ignition, less than 5 MW of absorbed power are sufficient, which can be delivered by antennas using 3 of the 12 equatorial ports. In non-ignited plasmas, it is also possible to operate at higher temperatures and lower densities than those needed for ignition in order to increase substantially the α-particle pressure gradient and enhance the virulence of α–particle driven modes making their analysis easier. For this a higher level of auxiliary heating power up to about 20 MW, using all six antennas, can be employed. The first exploration of fusion burn conditions in tritium-poor plasmas can also be conducted, with significant production of power from D-^{3}He reactions.

Table 3.1.1.1 Example of Plasma Parameters When Ignition is Reached (JETTO CODE)

Toroidal Plasma Current I_p	11 MA
Toroidal Field B_T	13 T
Central Electron Temperature T_{e0}	11.5 keV
Central Ion Temperature T_{i0}	10.5 keV
Central Electron Density n_{e0}	9.5×10^{20} m^{-3}
Central Plasma Pressure p_0	3.3 MPa
Alpha Density Parameter n_α*	1.2×10^{18} m^{-3}
Average Alpha Density $\langle n_\alpha \rangle$	1.1×10^{17} m^{-3}
Fusion Alpha Power P_α	19.2 MW
Plasma Stored Energy W	11.9 MJ
Ohmic Power P_{OH}	11.2 MW
ICRF Power P_{ICRH}	0
Bremsstrahlung Power Loss P_{brem}	3.9 MW
Poloidal Beta $\langle \beta_p \rangle$	0.20
Toroidal Beta $\langle \beta_T \rangle$	1.2 %
Central "safety factor" q_0	$\cong$ 1.1
Edge safety factor $q_\psi = q_\psi(a)$	3.5
Bootstrap Current I_{bs}	0.86 MA
Poloidal Plasma Current	$\cong$ 8.4 MA
Energy Replacement Time τ_E	0.62 sec
Alpha Slowing Down Time $\tau_{\alpha,sd}$	0.05 sec
Average Effective Charge $\langle Z_{eff} \rangle$	1.2

$n_\alpha^* \equiv n_D n_T \langle \sigma v \rangle \tau_{\alpha,sd}$

$\tau_{\alpha,sd} \equiv 0.012 T_{e0}^{3/2}(\text{keV})/n_{e0}\ (10^{20}\ \text{m}^{-3})$

Modest ICRF power levels are also adequate, in combination with the ohmic and fusion alpha heating, to access H-mode regimes, according to the available scalings. While these regimes can exhibit longer energy confinement times, they have the disadvantage, for a burning plasma, of featuring characteristic flat density profiles. This is one of the reasons why a classic divertor with coils inserted inside the toroidal magnet cavity is not adopted in the Ignitor design. The easier accessibility to H-modes that this may allow does not compensate for the degradation of global plasma parameters (e.g., the maximum achievable current I_p) and the complexity that such a system, operating in a high magnetic field environment, would involve.

Given the importance that programming the plasma density rise has in order to attain ignition, a pellet injector, with velocities ~ 4 km/s, is an integral part of the machine design. In particular, this is to be used to produce the peaked density profiles that are optimal for fusion burning, to minimize anomalous ion transport, to promote the formation of internal transport barriers, and for diagnostic purposes.

One of the main criteria around which Ignitor is designed is that it should produce mean poloidal magnetic fields $\bar{\bar{B}}_p = I_p/(5a\sqrt{\kappa})$ around 3.5 T. This ensures the macroscopic stability of the high plasma pressures needed for ignition and maximizes the possibility to reach this regime by ohmic heating alone. Recently $\bar{\bar{B}}_p$ has also been identified as the main parameter of merit to assess the performance of a machine magnet system for the confinement of a toroidal plasma. Therefore, given our present knowledge of the macroscopic stability of well confined plasmas, any larger Ignitor-like device should also maintain this value of $\bar{\bar{B}}_p$.

The machine (Figures 3.1.1.1 and 3.1.1.2) is characterized by a complete structural integration of its major components (toroidal field (TF) system, poloidal field system, central post, C-clamps and plasma chamber). A "split" central solenoid is adopted to provide the flexibility to produce the expected sequence of plasma equilibrium configurations during the plasma current and pressure rise. The structural concept upon which the machine is based involves an optimized combination of "bucking" between the toroidal field coils and the central solenoid with its central post, and "wedging" between the inner legs of the toroidal field magnet coils and between the C-clamps in the outboard region. The core machine systems, consisting of the copper TF coils, the major structural elements (C-clamps, central post, bracing rings) and the plasma chamber, are designed in such a way to withstand the forces produced within it, with the aid of a radial electromagnetic press when necessary. The set of stainless steel C-clamps forms a complete shell, which surrounds the 24 TF coils. These coils are pre-stressed through the C-clamps by means of a permanent mechanical press system (two bracing rings) that creates a vertical pre-load on the inner legs of the TF coils. This permanent press is supplemented by an electromagnetic press that is activated only at the maximum magnet currents, to maintain as closely as possible a hydrostatic stress distribution in order to minimize the von Mises equivalent stresses. The adopted structural solution ensures that the inner legs of the TF coils possess a sufficient degree of mechanical strength to withstand the electrodynamic stresses, while allowing enough deformation to cope with the thermal expansion that occurs during the plasma discharge.

The entire machine core is enclosed by a cryostat. All components, with the exception of the vacuum vessel, are cooled before each plasma pulse by means of He gas, to an optimal temperature of 30 K, where the ratio of the electrical resistivity to the specific heat of copper is minimum.

An important element of the Ignitor experiment is the site where it will operate. The ENEL center of Rondissone, near Turin, has been selected on the basis of its credits. Rondissone is a major node of the European electrical grid and is authorized to accept loads corresponding to the highest plasma currents and fields to be produced in Ignitor. Moreover, Rondissone has the unique advantage of housing the large scale test facilities of the Center for advanced high current technologies and of allowing ready access to the body of relevant expertise of this Center.

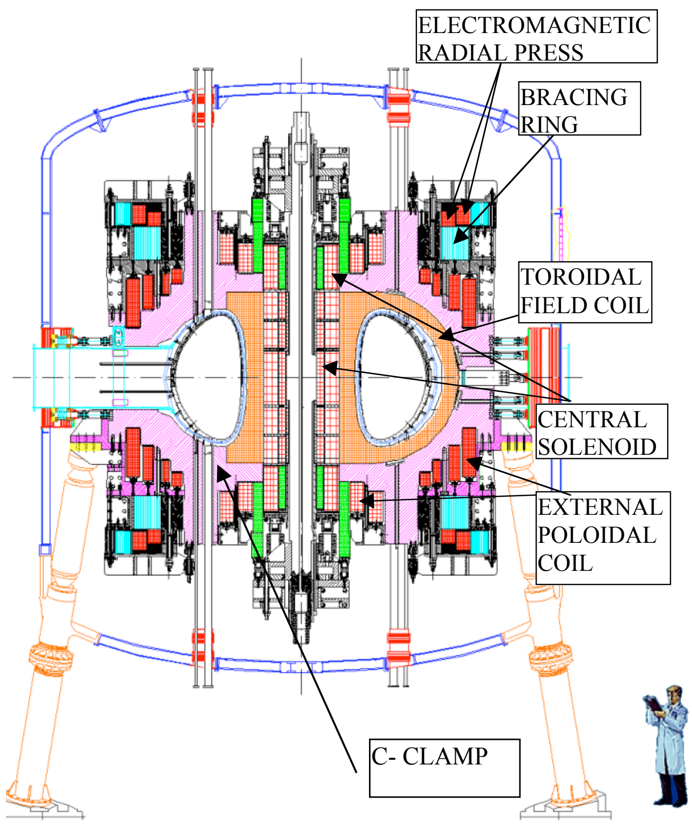


Figure 3.1.1.2

3.1.2 Fusion Ignition Research Experiment (FIRE)

Mission. The FIRE mission is to attain, explore, understand and optimize fusion dominated plasmas to provide knowledge for designing attractive MFE systems. FIRE is envisioned as an extension of the existing advanced tokamak program leading to an attractive magnetic fusion reactor (e.g., ARIES-RS). The FIRE design study of a next step burning plasma experiment has the goal of developing a concept for an experimental facility to explore and understand the strong non-linear coupling among confinement, MHD self-heating, stability, edge physics and wave-particle interactions that is fundamental to fusion plasma behavior. This will require plasmas dominated by alpha heating (Q ≥ 5 with Q ≈ 10 as the target) that are sustained for a duration comparable to characteristic plasma time scales (≥ 10 τ_E, ~ 4τ_{He}, ~2 τ_{skin}). FIRE will have the capability to investigate burning plasma physics issues in both the edge transport barrier (H-Mode) regime, and the advanced tokamak regime with internal transport barriers, high-beta and self-driven currents. FIRE will also contribute significantly to the development of reactor-relevant fusion plasma technologies. The FIRE pre-conceptual design activities, carried out by an U. S. national team, have been undertaken with the objective of finding the minimum size (cost) device to achieve the essential burning plasma science goals.

Machine Description. FIRE activities have focused on the physics and engineering assessment of a compact, high-field tokamak with the capability of achieving Q ≈ 10 in the Elmy H-mode for a duration of ~ 2 plasma current redistribution times (skin times) during an initial burning plasma science phase, and the flexibility to add advanced tokamak hardware (e.g., lower hybrid current drive) later. The configuration chosen for FIRE is similar to that of ARIES-RS, namely a highly shaped plasma, with double-null pumping divertor and aspect ratio ≈ 4. The key "advanced tokamak" features that give FIRE flexibility are: strong plasma shaping, double null poloidal divertors, low toroidal field ripple (< 0.3%), internal control coils and space for wall stabilization capabilities.

The reference design point is R_o = 2.14 m, a = 0.595 m, $B_t(R_o)$ = 10T, I_p = 7.7 MA with a flat top time of 20 s for 150 MW of fusion power with the cross-section shown in Figure 3.1.2.1.

R (m), a (m)	2.14, 0.595
κ_x , κ_a , κ_{95}	2.0, 1.85, 1.77
δ_x , δ_{95}	0.7, ≈ 0.48
q_{95}	> 3
$B_t(R_o)$ (T), I_p (MA)	10, 7.7
$Q = P_{fusion}/(P_{aux} + P_{OH})$	10
H98(y,2)	1.1
β_N	1.81
P_{loss}/P_{LH}	1.3
Z_{eff} (3% Be + He (5 τ_E))	1.4
R[illegible] (%)	3.8

Table 3.1.2.1 FIRE, Q = 10 Parameters

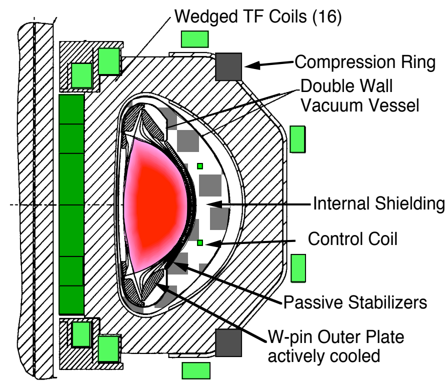


Figure 3.1.2.1 FIRE Configuration

The baseline magnetic fields and pulse lengths can be provided by wedged BeCu/OFHC toroidal field (TF) coils and free-standing OFHC poloidal field (PF) coils that are pre-cooled to 77 °K prior to the pulse and allowed to warm up adiabatically to 373 °K at the end of the pulse. 3-D finite-element stress analyses including electromagnetic, and thermal stress due to ohmic and nuclear heating have shown that this design is robust with a margin of 30% beyond the allowable engineering stress. Large (1.3 m by 0.7 m) mid-plane ports provide access for heating, diagnostics and remote manipulators, while 32 angled ports provide access to the divertor regions for utilities and diagnostics. The initial specifications for FIRE, like the previous BPX design call for 3,000 full field, full power pulses and 30,000 pulses at 2/3 field with a total fusion energy production of 5.5

TJ. The repetition time at full field and full pulse length will be < 3 hr, with much shorter times at reduced field or pulse length R&D and design activities are underway to increase both the number of pulses and to increase the repetition rate. FIRE will provide a comprehensive set of diagnostics that will enable the complete characterization of a single plasma pulse, similar to the capability of TFTR during DT operation.

FIRE will provide reactor-relevant experience for divertor and first-wall power handling since the anticipated thermal power densities on the divertor plates of ~6 MWm^{-2} for detached operation and ~15 MWm^{-2} for semi-attached operation exceed present experiments and approach those anticipated for ARIES-RS. The high plasma triangularity and double null of FIRE are expected to provide access to Type II elms easing the divertor heat load and associated erosion. FIRE would use only reactor relevant metallic materials for plasma facing components, and carbon could not allowed in the vessel due to tritium inventory build-up by co-deposition. The divertor plasma-facing components are tungsten "brush" targets mounted on copper backing plates, similar to a concept developed by the ITER R&D activity. The outer divertor plates and baffle are water-cooled and come into steady-state equilibrium during the pulse. The first wall is comprised of Be plasma-sprayed onto copper tiles. The neutron wall loading in FIRE is ~ 2 MWm^{-2} and produces significant nuclear heating of the first wall and vacuum vessel during the 20s pulse. The inner divertor targets and first wall are cooled by mechanical attachment to water-cooled copper plates inside the vacuum vessel. Remote handling would be provided for the maintenance and replacement of the internal hardware. Sixteen cryo-pumps – closely coupled to the divertor chambers, but behind sufficient neutron shielding – provide pumping (≥100 Pa m^3/s) for D-T and He ash during the pulse. Pellet injection scenarios with high-field-side launch capability will reduce tritium throughput, and enhance fusion performance. The in-device tritium inventory will be determined primarily by the regeneration of the divertor cryo-pumps, and can range from < 2 g for regeneration overnight to ~10 g for weekly regeneration. The tritium usage per shot and inventory is comparable to that of TFTR and therefore will not require a significant step beyond previous US fusion program experience in tritium handling and regulatory approvals.

The construction cost of the tokamak subsystem (magnets, divertor, plasma facing components and mechanical structure) has been estimated to be ≈ $351M (FY02US) including $71 M of contingency. Another ≈ $850 M would be required for auxiliary heating, startup diagnostics, power supplies and buildings to put the project at a new site.

Plasma Performance Projections. The physics issues and physics design guidelines for projecting burning plasma performance in FIRE are similar to those for ITER. The operating regime for FIRE is well matched to the existing H-mode database and can access the density range from $0.3 < n/n_{GW} < 1.0$ through a combination of pellet fueling and divertor pumping. This flexibility is important for investigating the onset of alpha-driven modes at the lower densities and to optimize the edge plasma for confinement studies and optimal divertor operation. The performance of FIRE was projected by selecting JET data with parameters similar to FIRE, namely $\beta_N \geq 1.7$, $Z_{eff} < 2.0$, $\kappa > 1.7$ and $2.7 < q_{95} < 3.5$. The average H(y, 2) and density profile peaking, $n(0)/\langle n \rangle_V$ for these data was found to be 1.1 and 1.2, respectively. This is consistent with the analysis of JET H-mode data presented by Cordey et al. Recent analysis of the JET and ASDEX Upgrade H-Mode data base indicates that H(y, 2) ≈ 1.1 is consistent with the high triangularity (d = 0.7) and modest density (n/n_{GW} = 0.7) anticipated for FIRE operation. A 0-D power balance code was used to calculate the Q-value in FIRE as a function of H-factor as shown in Figure 3.1.2.2. The density profile was assumed to have $n(0)/\langle n \rangle_V$ = 1.2 (x points) or 1.5 (Δ points) with 3% Be and self-consistent alpha ash accumulation. On this basis, FIRE would be expected to achieve Q ≥ 10 for JET-like H-modes thereby attaining the plasma performance needed to carry out the physics mission. Physics based models using marginal stability transport models such as GLF23 also predict a range of Q values from 5 to 15. These models dependent sensitively on the value of the temperature of the H-mode pedestal which is projected to be higher for plasmas with strong shaping (triangularity) and pedestal density low relative to the Greenwald density. A next step experiment, such as FIRE, would provide a strong test of these models and improve their capability for predicting reactor plasma performance. A 1 1/2 -D Tokamak Simulation Code (TSC) simulation of this regime with H(y,2) = 1.1 and n(0)/<n>$_V$ = 1.2 indicates that FIRE can access the H-Mode and sustain alpha-dominated plasmas for > 20 τ_E, > 4 τ_{He} and ~ 2 τ_{skin} as shown in Figure 3.1.2.3. In addition, time is provided for plasma startup and a controlled shutdown to avoid plasma disruptions. The burn phase can study plasma profile evolution, alpha ash accumulation, techniques for burn control and plasma current evolution due to alpha heating.

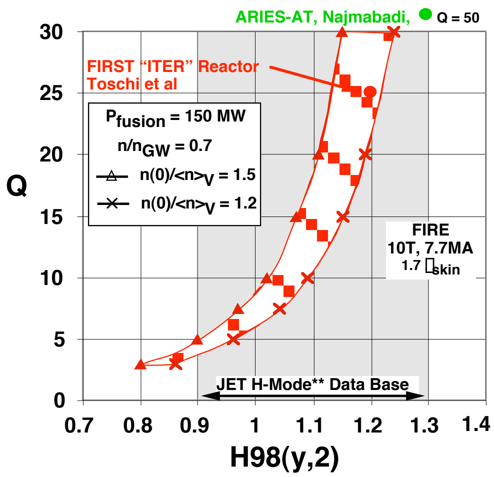


Figure 3.1.2.2 Fusion Gain for FIRE

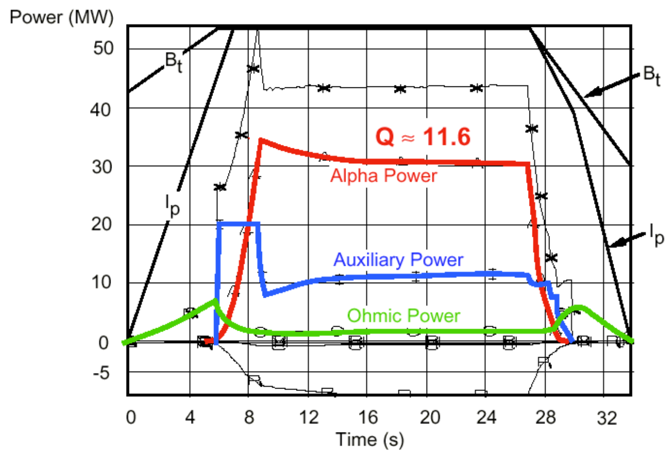


Figure 3.1.2.3 Fusion-dominated Plasma Evolution.

A longer term goal of FIRE is to explore advanced tokamak regimes using pellet injection and current ramps to create reversed shear plasmas (e.g., PEP modes), and then applying lower hybrid current drive to sustain the AT mode at high fusion gain (Q > 5) for a duration of 1 to 3 current redistribution times. Simulations using TSC with self-consistent lower hybrid current drive modeling show that 100% non-inductively driven burning plasmas could be sustained at $b_N \approx 3$, 64% bootstrap current with Q ≈ 7.5, fusion powers of 150 MW if confinement enhancements H(y,2) ≈ 1.6 were attained at B = 8.5T and Ip = 5.5 MA. An important feature of the FIRE cryogenic copper alloy magnets is that the pulse length increases rapidly as the field is reduced with flattops of ~ 40 s at 8 T and ~90 s at 6 T. The primary limitation to exploiting this long pulse capability is the generic problem of handling the plasma exhaust power under reactor relevant conditions.

Assessment. FIRE does not seek to demonstrate that our existing knowledge is correct nor to avoid important physics issues, rather the philosophy of FIRE is to explore the science of burning plasmas as fully as possible within the cost constraints of a $1B class laboratory. FIRE is a natural extension of the existing state of the art tokamaks, and is based on the extensive international H-mode data base for projecting performance to the burning plasma regime. Due to the high magnetic field, the extrapolation required to attain Q ≈ 10 is a relatively modest factor of 3 in terms of the normalized confinement time ($B\tau_E$). The MHD stability characteristics of FIRE, with $q_{95} \approx 3.1$ and $\beta_N \approx 1.8$ for initial burning plasma experiments, are similar to the standard MHD regimes in existing tokamaks and will explore the synergistic effects of energetic alphas and MHD modes such as sawteeth and TAE modes. Operation at $\beta_N \approx 3$ or higher in later phases would begin to explore the important areas of neoclassical tearing modes (NTM) and resistive wall modes (RWM). Lower hybrid current drive and feedback stabilization are being evaluated, and show promise as an experimental tool to investigate the control of NTMs and RWMs. In the lower field advanced tokamak regimes at B ≈ 6.5T, ECCD could also be employed for NTM stabilization. Divertor pumping and pellet fueling will allow FIRE to vary the density, hence the TAE driving terms $R\nabla\beta_\alpha$, by a factor of three providing a good test bed for exploring the instability boundary for TAE modes and determining the transport of energetic alpha particles due to multiple overlapping TAE modes.

The double null divertor configuration produces the strongest plasma shaping which is critical for resolving and exploiting a number of physics effects related to confinement and MHD stability. The double null divertor may also significantly reduce the frequency and intensity of vertical displacement disruptions which is a critical issue for the feasibility of a tokamak based reactor. The high power density in FIRE poses a significant challenge and opportunity for the divertor and first wall designs, but this is a generic issue for magnetic fusion. The success of FIRE in this area would provide yield important benefits for technology development for future fusion devices.

A critical issue for all next step experiments is to supply auxiliary heating power to at high power densities to a fusion plasma. FIRE proposes to use ICRF heating which has been demonstrated on existing experiments but the high power densities and neutron wall loading present in FIRE will require significant plasma technology R&D. This R&D will be needed if ICRF is to be used in a fusion application.

3.1.3 International Thermonuclear Experimental Reactor (ITER)

Motivation and Rationale. During the 1980s and early 1990s, it became clear that tokamaks worldwide exhibited a common physics which pointed to common parameters for a facility that not only could study the new physics introduced by burning plasmas and the physics of true steady state plasmas but also could serve as a test-bed for the development of technologies needed for design, reliable operation, and optimization of a First Fusion Reactor (FFR). Such a facility that is capable of both physics and technological research has come to be termed an integrated research facility. There are compelling arguments that the integrated facility approach is the fastest and cheapest route to the development of fusion energy because: a research program utilizing a succession of less-capable facilities, often called the modular approach, ultimately has higher costs and takes a longer time to complete; and, data from less-capable facilities calls for a larger extrapolation to an FFR. In particular, it is interactions between many diverse physics/technology processes, whose relative importance is scale-dependent, that determine the state of a burning plasma. Since the interactions are scale-dependent, a reactor-scale research facility is ultimately needed in either the integrated or modular approaches to attain the correct balance and minimize the extrapolation of results to an FFR. The International Thermonuclear Experimental Reactor (ITER) is designed to serve as such an integrated, reactor-scale research facility at minimum cost, while avoiding delays waiting for data from less-capable facilities.

In response to the need for an integrated research facility whose parameters were based on common physics, the four ITER Parties (Europe, Japan, Russian Federation, and United States) launched the ITER Engineering Design Activities (ITER-EDA) in 1992 with the programmatic objective to "demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes." The resulting international cost sharing reduced the cost-per-Party to an affordable level. Even though the US withdrew from the ITER project in 1998, the remaining Parties developed a new design originally called ITER-FEAT, but now known simply as ITER. This design has a cost that is roughly 50% of the intermediate 1998 EDA-FDR design. The new ITER design exploits advances in tokamak operations, plasma shaping facilitated by a pancake central solenoid, pellet fuelling, and several plasma heating systems to add flexibility and reduce the plasma size and the facility cost. The Parties determined that attainment of Q=10 (with margin) would fulfill the overall programmatic objective. Moreover, the integrated facility with its steady-state super-conducting magnets, is capable of true steady-state operation, thus permitting investigation of burning plasma behavior for arbitrarily long time scales. If steady-state is chosen to be the operational mode of a FFR, this choice must rest on an unequivocal demonstration on an integrated facility with β>3%.

In addition to exploring the science of burning plasmas, the ITER objectives also include carrying out a program aimed at the development of fusion technologies. From the basic design it is clear that a great deal of fusion technology needs to be integrated into ITER just to enable it to reach its physics objectives. Thus, super-conducting magnets and actively cooled vessel, first wall and divertor target plates are required to achieve fully steady-state plasma conditions. A variety of heating and current drive systems are needed to access various plasma regimes with sufficient flexibility. And fueling and pumping systems are required to inject the fuel, remove the large fraction that is unburned and purify it for reinjection. The construction and operation of ITER must be carried out in a way that will demonstrate the safety advantage that fusion enjoys and will be indicative of fusion's potential to mitigate the problem of long term storage of radioactive waste.

ITER will also serve as a test bed for developing fusion nuclear technology through its blanket testing program. The objective is to "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." The development of blanket modules with tritium breeding ratio (TBR) greater than unity, and which use environmentally attractive materials and coolants, is crucially important to fusion's development as an energy source, ranking on par with the physics goal of achieving high fusion gain. The blanket module development program will begin with non-neutronic thermo-mechanical testing during the earliest phases of ITER's operations. It will evolve in the high gain and high duty cycle phases of ITER's operation to compare measured and calculated TBR's, and to examine issues such as on-line tritium recovery and production and removal of high-grade heat. During the second decade of ITER's operations, it is anticipated that ECCD will suppress neoclassical tearing modes, and that ITER will reach $\beta_N \approx 2.7$ which corresponds to 800 MW fusion power and a neutron flux of 1 MW/m^2. Testing will continue and the net fluence for blanket module testing would be increased from ~0.1 to 3.0 MWa/m^2 -- comparable to that of the 1998 ITER-EDA design. Replacing the shield blanket with a breeding blanket during this period is not precluded.

ITER Design. A summary of the new ITER design is presented in the ITER Final Design Report. Its major features are reported in the archive fusion literature. Details are found in a series of reports described in a system code identified a shallow cost minimum when optimized (at fixed Q=10) over design parameters such as aspect ratio, elongation, maximum toroidal field, poloidal flux consumption during burn, divertor geometry, and vertical stability requirements. The cost minimization rests on our understanding of the physics as embodied in the ITER Physics Basis, results of $800M of manufacturing R&D summarized in a consensus on safety requirements, and a cost target of roughly $5B US (2002). Figure 3.1.31 and Table 3.1.31 present the main parameters of the new ITER design.

Eighteen Nb_3Sn superconducting magnet coils are the source for the toroidal magnetic field which is limited by material properties to be less than 13T at the magnet. Ferritic inserts control field ripple to be less than 1% at the plasma boundary. The poloidal field is determined by the requirement to shape a plasma carrying a full current as expressed by $q_{95} \approx 2.5$ to an elongation $\kappa \approx 2$, while supporting a lower-single-null divertor. A limitation on elongation arises from the power necessary to control a vertical instability of the plasma The six poloidal field coils outside the toroidal field system use NbTi superconducting material, while the central solenoid, which is the source of inductive current drive, has six pancake Nb_3Sn coils operated at up to ±13T to maximize the flux swing. The pancake design means that each coil can carry a separate current and is an important flexibility feature for plasma shaping, especially for increasing triangularity. There are also eighteen saddle coils arranged in three rows of six that can compensate for field errors as well as generate feedback fields to control n=1,2 resistive kink or wall modes. Research in this area is not yet complete, and specifications for the saddle coils should be regarded as provisional.

Parameter	400 MW	500 MW
R/a (m/m)	6.2/2.0	6.2/2.0
Volume (m^3)	831	831
Surface (m^2)	683	683
Sep. length (m)	18.2	18.2
$S_{cross\text{-}sect}$ (m^2)	21.9	21.9
B_T (T)	5.3	5.3
I_p (MA)	15.0	15.0
κ_x/δ_x	1.85/0.48	1.85/0.48
κ_{95}/δ_{95}	1.70/0.33	1.70/0.33
$l_i(3)$	0.84	0.84
V_{loop} (mV)	75	75
q_{95}	3	3
β_N	1.8	2.0
$<n_e>$ (10^{19} m^3)	10.1	11.3
$<n_e>/n_G$	0.85	0.94
$<T_e>$ (keV)	8.8	8.9
$<T_i>$ (keV)	8.0	8.1
β_α (%)	2.5	2.8
β_p	0.65	0.72

Parameter	400 MW	500 MW
$P_{RF}+P_{NB}$ (MW)	7+33	17+33
P_{OH} (MW)	1	1
P_{TOT} (MW)	121	151
P_{BRM} (MW)	21	26
P_{SYN} (MW)	8	8
P_{LINE} (MW)	18	27
P_{RAD} (MW)	47	61
P_{FUS} (MW)	400	500
$P_{LOSS}/P_{L\text{-}H}$	87/48	104/51
Q	10	10
τ_E (s)	3.7	3.4
W_h (MJ)	320	353
W_{fast} (MJ)	32	34
$H_{H98\,(y,2)}$	1.0	1.0
τ_{He}^*/τ_E	5	5
$Z_{eff,ave}$	1.66	1.72
$f_{He,axis\,/\,ave}$ (%)	4.3/3.2	4.4/3.2
$f_{Be,axis}$ (%)	2.0	2.0
$f_{Ar,axis}$ (%)	0.12	0.14

Table 3.1.3.1 ITER Parameters

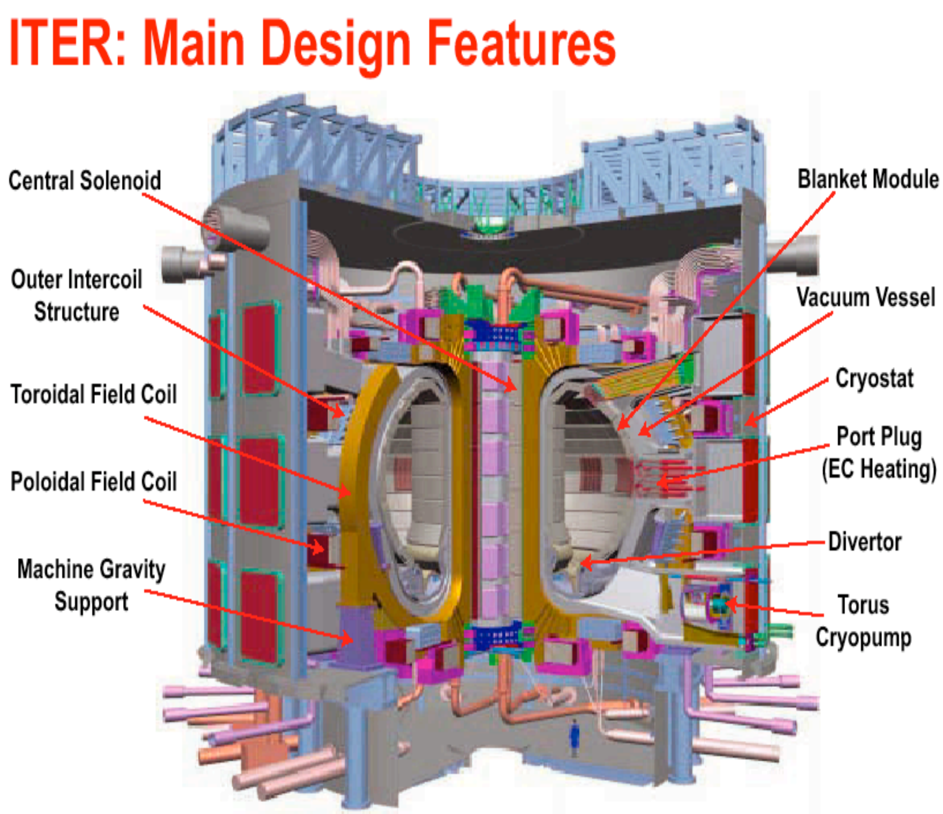


Figure 3.1.3.1 ITER Main Design Features

Mechanically, the double-wall vacuum vessel is attached to thick toroidal field coil cases which rest on gravity supports. The vacuum vessel, in turn, provides support for the blanket modules, port plugs, and divertor cassette assemblies. The first wall and pressurized water cooling system can accommodate neutron heat fluxes up to 1 MW/m^2, which corresponds to 800 MW fusion power. A lower single null (SN) divertor cassette assembly, developed in the EDA fabrication R&D program, has long divertor legs to provide flexibility in divertor operation. All this is summarized in Figure 3.1.3.2.

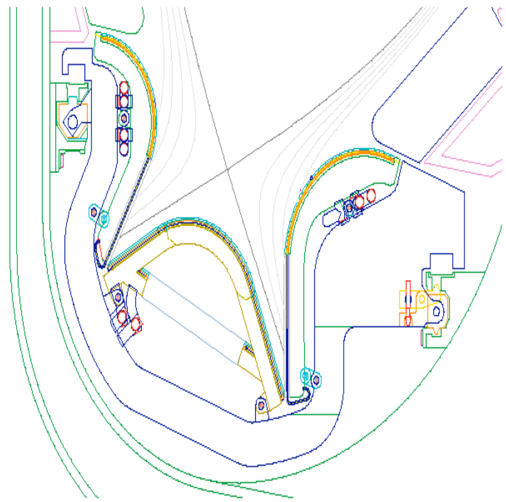


Figure 3.1.3.2 Present ITER Divertor Configuration

Plans for auxiliary heating and fuelling systems call for 33 MW of 1 MeV negative ion neutral beams, 20 MW of 170 GHz gyrotrons for heating and current drive, and 20 MW of 80 MHz ion cyclotron heating. Studies show that 20 MW of gyrotron ECCD power launched through the upper ports will be just sufficient to stabilize neoclassical tearing modes. The initial heating system provides 73 MW of auxiliary power. The design will support an additional 40 MW, the specifics of which will depend on initial experimental results. Gas-puff DT and/or impurity fuelling sources are in the divertor chamber and near top of the plasma. A near-vertical pellet launch capability will be located near the inside of the divertor cassette. Prospects for pellet launchers located in the central-solenoid core are under study.

A major particle control issue concerns recovery of un-reacted tritium which gets trapped in cold carbon-hydrogen co-deposition layers that are shielded from heat flux emanating from the main plasma. Graphite is the preferred divertor strike point material because it doesn't melt and retains its shape during disruptions. But in TFTR and JET, roughly 30% of the tritium introduced into the vacuum chamber remained lodged in carbon-hydrogen co-deposition layers which formed in regions inaccessible to the heat flux from the main plasma. Loss of tritium at the JET/TFTR rate would limit ITER to several hundred full DT burn shots before the administrative limit to the in-vessel tritium inventory would be exceeded. A cold-trap recovery strategy holds promise, but an experimental verification is just in the planning stage. Succinctly, the strategy is to control the location of potential co-deposition by design and then to recover co-deposition material by heating and pumping code position regions periodically (e.g., during transformer recharge periods). Should carbon ultimately be proven to be infeasible as a target material due to co-deposition issues, a tungsten backup solution has been developed for ITER and can be adopted. Since the entire divertor can be replaced in six months, a number of materials and designs for the divertor target plates can be tested during ITER's lifetime. Initial ITER operational plans call for 2-3 years with non-activating proton plasma discharges direct observation of co-deposition regions should be possible.

<u>ITER Physics: Projections, Opportunities, and Steady-State.</u> This section will first discuss how our present knowledge of fusion plasma physics supports the design of ITER through physics-based

projections for plasma performance. Second, we shall argue that ITER will provide exciting opportunities to carry out new experimental physics on plasmas in regimes inaccessible to present experiments and essential for reactor performance. And third, we will discuss how that design features either planned for ITER or available as modifications/upgrades will enable investigations of true steady-state tokamak plasmas with close to 100% bootstrap current, especially their β-limits and confinement scaling.

Initial Performance Projections. There is much physics in common between burning plasmas and discharges in present devices. This suggests the use of nondimensional concepts and parameters to express physics so that the extrapolation principles are clear. The simplest example of a nondimensional parameter is β -- the plasma energy density normalized magnetic pressure. Table 3.1.3.1 shows the nominal ITER Burning plasma discharge with $\beta_N = 1.8$ lies safely below limits $\beta_N \approx 3.0$ computed by ideal MHD stability codes, such as PEST or GATO.

More complicated is energy confinement for H-modes, where two processes are at work: First is the H-mode transport barrier at the plasma edge which determines properties of the pedestal region at the boundary. Experiments have determined that a threshold power flux through the separatrix is required to retain the H-mode barrier. The physics of the pedestal, which is the boundary condition for core transport model, remains unknown. The second process is core fine-scale turbulence which proceeds in a 5-dimensional phase space and governs energy transport in the core. Transport models based on based general properties of this turbulence replicate experimental core temperature profiles quite well, given the experimental edge temperature at the pedestal. Therefore, energy confinement projections are expressed in terms of the pedestal temperature. Databases to empirically determine pedestal temperature are being collected.

In the past, global confinement times have been projected via log-linear regression extrapolation from a confinement time data base. However, a complication in this process has resulted in an unphysical increase of transport losses with increasing β for the commonly used ITER Physics Basis IPB(y,2) scaling relation. Furthermore, this approach is being generalized to distinguish between pedestal and core energy content. Work is in progress.

At present, the best way to project confinement is to extrapolate from discharges in present devices that have the same non-dimensional pressure β_N and collisionality ν^* as the proposed ITER plasma and thus need extrapolation only in the nondimensional gryoradius ρ^*. Scaling in ρ^* can be measured by comparing discharges at different torroidal field but at the same β and ν^*. This approach is termed Demonstration Discharges and is insensitive as to whether it is core physics or pedestal physics which is governing confinement.

Clearly, there is much to be learned regarding transport in tokamaks . Nonetheless, one can argue that Demonstration Discharge projections are good enough to assure that ITER will enter the burning plasma regime., defined by Q>10.

Extrapolation of divertor performance faces similar issues. Acceptable detached or semi-detached SOL plasmas are found experimentally in present devices and are successfully replicated by divertor modeling codes --a major advance since ITER's inception in 1992. But scaling of the cross-field transport diffusivity remains unknown and introduces uncertainty into the projections for ITER. Fortunately, a range of values can be accommodated by the long divertor legs of the design and flexibility in gas and impurity puffing and pumping rates. Erosion of divertor strike points by ELMs will depend pedestal physics and MHD stability. Again, since two or more physics processes are work, the correct balance calls for a reactor-scale facility to produce definitive experimental results directly useful in designing a FFR.

Inaccessible Physics. Burning plasmas introduce new phenomena into fusion plasma physics that are inaccessible to experimental investigations in machines presently available. The *ITER Physics Basis* identifies three classes of new physics: Physics of Energetic α-particles; Physics of self-heating and self-organization of 100% bootstrap current discharges; and Physics attributable to the scale of a facility required so that transport losses approximately balance thermonuclear heating.

A main objective of burning plasma physics is to ascertain the influence of a distribution of α-particles slowing down from their initial 3.5 MeV energy on the stability of Alfven Eigenmodes and on the giant sawtooth phenomena. Nonlinear α-particle loss would diminish heating of the core plasma. Present experiments have already triggered AE instabilities but the energy-distribution and concentrations of energetic ions differ from those of thermonuclear α-particles. These investigations are also of class 3 in that the minimum unstable toroidal mode number n_{min} increases with machine size. from $n_{min} \approx 1$-3 in present experiments to $n_{min} \approx 10$ in ITER. The high ITER value opens the possibility of turbulent AE modes in a reactor-scale device. The relative concentration of α-particles will be highly temperature dependent. One should note that adding extra ECH gyrotron power can maintain or otherwise control the core temperature without introducing an extra, non-thermonuclear source of energetic particles, thereby adding flexibility to AE investigations in burning plasmas.

A fundamental property of burning plasmas is that heating by α-particles deposits most of their energy into electrons leading to plasmas with $T_e \approx T_i$ in contrast with present NBI-heated discharges with $T_i >> T_e$. Since fusion reactivity is temperature dependent, thermal instability is possible, especially if internal transport barriers spontaneously form. Burning plasmas are essential to thermal control investigations, although they can be modeled to some degree with feedback between neutron rates and auxiliary gyrotron heating. Steady-state plasmas with 100% bootstrap current and a thermonuclear energy source (with no current drive) are of particular interest because they are self-organized. With self-heating, they will determine their own current, temperature, and fusion power given the source of DT fuel, ^4He ash removal, a toroidal field and a poloidal field to provide radial force balance. Whether the β-value so determined lies below the MHD stability β-limit is a major unanswered question for Advanced Tokamaks.

In tokamak plasmas, many diverse plasma processes occur concurrently and their mutual interactions depend on the scale of the plasma. Put another way, a burning plasma must have a normalized gyroradius ρ^* about a factor-of-5 smaller than that of contemporary devices. Since many plasma processes, when expressed nondimensionally, depend on ρ^* in different ways, their mutual interaction depends on the scale of the plasma. Examples are: Will the H-mode threshold power exceed transport loss power? Will the core density requirement for a specified fusion power exceed the SOL density needed to accommodate the resulting heat flux through the separatrix? What is confinement scaling with ρ^*? Will energy delivered to the SOL plasma in a disruption thermal quench suffice to vaporize the divertor chamber walls? Investigation of these questions calls for a burning plasma experiment.

<u>Burning plasmas or advanced tokamaks</u>? The integrated approach to fusion plasma physics recognizes that a single facility can, on one hand, assure that fundamental data on burning plasmas are realized while, on the other, have sufficient flexibility to optimize reactor-scale plasmas as well as to robustly demonstrate true steady-state tokamak operations and document its limits. Entry into the burning plasma regime rests on conservative projections of plasma performance, and the data returned will require little extrapolation to a First Fusion Reactor. The ITER facility can accommodate all presently conceivable design features that an Advanced Tokamak with conventional aspect ratio and shaping should have. Table 3.1.3.2 lists the features, modifications, and upgrades that we have been able to conceive of for an aspect ratio $R/a \approx 3$ tokamak. Since the ITER design is intrinsically steady-state, extension to arbitrarily long time scales. Consequently, one does not have to choose between a burning plasma or an advanced tokamak facility. ITER serves both needs. With appropriate modifications, the ITER facility can accommodate them all. as a flexible advanced-tokamak research facility should. Again, steady-state data will need little, if any,

extrapolation. Consequently, one does not have to chose between a burning plasma and an advanced tokamak facility. ITER serves both needs.

Table 3.1.3.2 Design Features and Modifications for Advanced Tokamak Operations on ITER

ITER Improvements and Design Features	Advanced Tokamak ITER Physics	Comments
More gyrotrons (70 MW)	• ECCD Stabilization of NTMs • Burn simulations • Raises core temperature and fusion reactivity • High-bootstrap fraction w/ECCD	Reliable 1 MW, 170 GHz, steady-state gyrotron sources needed
Pancake Solenoid	• Optimizes plasma shape	Elongation of base design
Positive ion NBI	• Angular momentum source exploits plasma rotation	Control of resistive wall instabilities by rotation
Negative ion NBI	• Heats plasma center • Weak source of angular momentum	Counter-rotation NBI physics
Ion cyclotron heating	• Fast wave current drive • Localization of heating • ITB trigger; high bootstrap • TAE simulations	High Bootstrap electron heating 80 MHz minority proton system for burn and α-particle simulations
Lower hybrid	• Efficient current drive • NTM stabilization	Low-power coupling tests required before adaptation
Inside pellet launch	• Efficient control of core density • Innovative launch from inside solenoid	Core density control without affecting SOL plasma
n=1 error-field and feedback coils	• Control of resistive wall mode	Experimental Studies of feedback and rotation.
Low-power interior TAE mode antenna	• Measures TAE mode frequencies and damping decrements; interpretation as disruption precursor	Low-power linear response to swept frequency excitation determines proximity to disruption
Cold trap divertor	• Recovers unburned tritium from walls	Assures greater than 95% tritium burnup
Long Divertor legs	• Flexibility in divertor operation	Creation of detached divertor plasmas
Superconducting Coils	• Supports true steady-state operations of 5000 s or more	TF flatop is marginal for steady-state in present facilities
Heat flux margin for first wall	• Permits fusion power up to 1000MW and β_N=2.7 (vs.400 MW)	Long pulse operation at β_N=2.7

Conclusion. The ITER design fulfills the requirements for an integrated test facility, with constraints consistently and carefully applied to reflect conservative engineering practice. This overview indicates that plasma physics projections find that Q>10 will be achieved with margin. Many phenomena involve interaction between several basic processes and we argue that only reactor-scale experiments will find the correct balances. Indeed, a reactor-scale facility will teach us much regarding basic processes in burning plasmas and is essential to the development of a steady-state capability for tokamaks. Since the steady-state tokamak is a self-organized system, it will be subject to constraints which we do not know now and cannot investigate in existing facilities. But these constraints will be essential limitations for a steady-state reactor. An exciting world of reactor scale experiments awaits us.

3.2. Physics

Each of the three approaches to the study of burning plasmas was evaluated from the physics perspective. Both the projected performances and the abilities of the three approaches to study burning plasma physics were assessed. The physics assessment follows in the sections below.

3.2.1 Wave Particle Interactions

<u>Opportunities to address RF wave-plasma physics issues.</u> The major new element for wave physics in a burning plasma is the presence of a substantial energetic alpha population. We already have some experimental and theoretical experience with RF in the presence of strong beam populations and weak alpha populations (TFTR, JET). However, fusion alphas differ from beam ions in the larger values of $\square_{\square}/T$, $k_{\square}\square$ and the anisotropy of the energetic ion distribution. Alpha particles will modify the wave dispersion and dissipation, and may modify the mode conversion properties as well. Since alphas slow down predominantly on electrons, the electron temperature will be comparable to, or higher than, the ion temperature in a burning plasma. This will result in increased Landau and cyclotron damping and higher current drive efficiencies which should make the plasma response stronger and easier to observe and compare with theory. The dynamic interaction between wave heating, transport and stability in a burning plasma will differ significantly from a non-burning plasma, and we may expect different plasma responses to RF inputs. Ultimately, the nonlinear dynamic interaction between plasma properties and RF/Beam inputs will have to be assessed. We are not proposing to carry out such studies here.

Although not specifically called for in the proponent's operational scenarios, waves have the potential to produce many additional effects, including control of plasma equilibrium, stability and transport. More specific items include: plasma rotation generation, flow-shear and transport barrier control, mode stabilization/destabilization (sawteeth, Alfven eigenmodes, fishbones…), formation of energetic particle populations, wave induced transport, alpha channeling, etc. Although not all of these effects are understood or have been validated in past experiments, they are under investigation in the ongoing base tokamak research program. An extension of one or more of these promising physical processes to a burning plasma device will be of great importance when one considers reactor applications. Consequently, having different types of RF and/or beam systems on a particular device offers a greater opportunity to study wave physics and associated phenomena, including plasma control.

<u>Key Issues.</u> High-power waves can serve a number of functions in heating and control of a burning plasma device. Each of the proposed machines relies to a varying extent on RF systems for its basic operation and for control to reach advanced modes. Therefore, the first task of this working group was to assess whether the proposed RF systems met the requirements of the devices for heating to the required fusion temperatures, and delivered the required current drive for the AT scenarios or for other control processes. These include such basic questions as power and frequency requirements, and an assessment of the power deposition profiles assumed or predicted by the advocates. Further questions included verification of the current drive efficiencies and driven current profiles in advanced mode scenarios.

<u>Methods and Assessment Tools.</u> There is a substantial base of experience with all of the proposed RF techniques. The first line of assessment then is to determine to what extent the proposed scenarios are based on established practice in existing experiments and where are the critical extensions to new regimes. By far, the most significant extensions will be the presence of a substantial population of energetic alpha particles and higher electron temperatures than typically encountered in present-day machines.

In the fusion theory program a collection of modeling codes exists that can adequately treat the main heating and current drive processes. The accuracy and completeness with which these codes describe experiments varies, depending on the frequency regime. For the electron cyclotron range of frequencies the accuracy and agreement with experiment tends to be good. In this case the waves propagate in vacuum so that the launchers can be at a distance from the plasma. The wave propagation is well described by geometrical optics, and is essentially independent of the ions. Absorption is entirely by electrons. Radial drift of the electrons is expected to be unimportant so that evolution of the distribution function can be adequately described by a 2D in velocity space, bounce averaged Fokker-Planck model, local to a flux surface.

For the lower hybrid (LH) and ion cyclotron range, the situation is more complicated. Because these waves do not propagate in vacuum, the launching structure (antenna) must be in close proximity to the plasma edge. The spectrum and coupling efficiency of launched waves depend on geometric details of the launchers as well as edge plasma properties that are difficult to predict for a future experiment, especially in the presence of ELMs. In large, hot devices where the absorption is sufficiently strong that complex, possibly chaotic ray trajectories do not occur, lower hybrid wave propagation is adequately described by geometrical optics. Again, in most cases the evolution of the electron distribution function is adequately described by a 2D Fokker-Planck model local to the flux surface. The experience is that with care in the application of models, reasonably good agreement with experiments can be obtained for bulk current drive efficiency and for approximate localization of the driven current.

Although experiments in the ion cyclotron range of frequencies are more mature than in other frequency regimes, modeling is still in a state of active development. Although ray tracing has been reasonably successful in modeling the main processes required for these scenarios, such as bulk heating or fast wave current drive, the approximations of geometrical optics break down in regions where mode conversion or wave cut-off can occur in the proposed devices. In 1D or 2D full-wave codes, a self-consistent iterative solution with a Fokker-Planck code is still under development. Codes that solve the full-wave equation in 2D are, for the present, restricted to special analytic forms for the distribution function, such as superposition of Maxwellians or slowing down distributions. For the present study we employ a combination of ray tracing and full wave codes to examine wave absorption at single time points during the flat-top phases for each device. We should note here that the TRANSP code has been used by R. Budny to study the time-dependent scenarios proposed for each device by the advocates. We have not assessed this time-dependent evolution and hence can only say that in the flat-top phase the results obtained by Budny are in reasonable agreement with our modeling results. In TRANSP, the RF heating is calculated using a 2D reduced-order full-wave code coupled with a bounce-averaged Fokker-Planck package.

Assessment. The base case heating scenarios for each of the three proposed devices has been examined. These assume ELMy H-mode operation for ITER and FIRE, and L-mode operation for Ignitor, and all require RF for plasma heating. In addition, we considered advanced tokamak (AT) scenarios for ITER and FIRE requiring RF current drive for q profile (reversed shear) control. ECCD was considered for NTM mode stabilization in ITER.

ICRF. All three machines employ ICRF for bulk heating at the 2^{nd} harmonic tritium resonance. This technique has been demonstrated in DT plasmas on both TFTR and JET in the presence of fusion alpha particles. Of course, the energetic alpha density was much smaller than will be the case in burning plasma. Absorption at the second harmonic resonance is proportional to the tritium β so that some type of pre-heating is required before this mechanism becomes efficient. Introducing a small fraction of minority of ^3He allows efficient wave absorption with the same resonance location at low β in DT, or for tritium-free operation in D or H majority. This mechanism is also well demonstrated on many experiments. Ion cyclotron heating on small, high-field, high-density machines relevant to FIRE and IGNITOR has been well demonstrated on Alcator C-Mod, at least for H minority heating. ^3He minority heating will be studied in the near future. For the Snowmass meeting, calculations were done corresponding to the final burning plasma state of the scenarios contained in the design publications for each of the machines. These calculations did not show any problems for the basic ICRF heating in the final state.

Heating scenarios were calculated using the CURRAY ray tracing code and the PICES full-wave ICRF solver for all three machines, using slowing down the model for the alpha distribution. For

ITER, the base case is ELMy H-mode with the 2nd harmonic T and ^{3}He fundamental resonance on axis, f_{RF} = 53 MHz. Both codes show highly peaked heating profiles but there is a discrepancy on the partitioning between plasma species, with the full-wave code PICES giving higher deposition into electrons and less into the ion species than CURRAY. The relative power fractions in CURRAY are (electrons = 45%, T = 8%, ^{3}He = 44%, alphas = 4%). Using PICES, it is found necessary to readjust the frequency to f_{RF} = 58 MHz in order to minimize alpha absorption. This further increases electron absorption giving (electrons = 73%, T = 2%, ^{3}He = 14%, alphas = 11%). Similar results are obtained for the ELMy H-mode scenario in FIRE at f_{RF} = 100 MHz, although less difference seen in power partitioning. The relative power fractions are PICES (electrons = 45%, T = 11%, ^{3}He = 38%, alphas = 5%) compared to CURRAY (electrons = 28%, T = 17%, ^{3}He = 54%, alphas < 1%). We do not, at this point, know if the high fraction of power directly deposited into electrons has significant consequences for the operational scenarios. In none of the calculations is absorption by Beryllium significant. For IGNITOR, the base case is L-mode evaluated during the current flat-top. The calculation was carried out at f_{RF} = 146 MHz to place the minority resonance on axis for the increased magnetic field of 14.5T due to paramagnetism. Here, the electron deposition is much smaller and the code agreement is better. The relative power fractions are PICES (electrons = 21%, T = 7%, ^{3}He = 51%, alphas = 20%) compared to CURRAY (electrons = 18%, T = 11%, ^{3}He = 70%, alphas = 1%).

The METS 1D, full-wave code was used to examine the single pass absorption characteristics for standard scenarios in each of the three devices using parameters taken from the flattop phase of the TRANSP simulations. For each device, the RF heating scenario considered placed the 2T and the fundamental ^{3}He resonances near the magnetic axis. In these simulations, the wave fields and power absorption were computed using Maxwellian distributions for the thermal species and a slowing down distribution for the energetic fusion alphas. With a frequency of 53 MHz and a launched peak wave number, $k_\parallel$, of 10 m^{-1} at the antenna, in an ITER ELMy H-mode, the net single-pass absorption is nearly complete at ~97%. Of this power absorption, the power is partitioned as follows: electrons ~ 42%, tritium ~ 4%, the minority ^{3}He ~ 36%, the energetic alphas ~ 15%, and the carbon and deuterium < 1. For an ELMy H-mode in FIRE at a frequency of 100 MHz and $k_\parallel$ = 9 m^{-1}, the single-pass absorption is ~ 91% (electrons ~30%, the tritium ~ 8%, the ^{3}He ~52%, and < 1% being absorbed on the deuterium and the energetic alphas). For the standard L-mode scenario in IGNITOR at a frequency of 140MHz with $k_{//}$=9 m^{-1}, the single pass absorption is ~ 100% (electrons ~ 22.7%, the tritium ~ 7.7%, the ^{3}He ~69.5%, the alphas and deuterium ~ 0.1%). It is interesting to note that the single pass absorption is > 90% in all three devices. Despite the differences in some of the assumed equilibrium parameters, the results from METS agree reasonably well with those from CURRAY, and somewhat less so with those from PICES. The effects of the shape of the alpha density profile on the power partitioning was explored for the ITER device. The net single pass absorption for the alphas varied between 10% for very narrow density profiles, up to about 23% for much broader profiles.

Fast wave current drive needed to provide on-axis seed current for advanced scenarios has been demonstrated on DIII-D and other machines at high cyclotron harmonics. Although the driven currents were modest (100 kA) due to the relatively low electron temperatures and RF power levels, the measured currents were in reasonable agreement with theory. It is believed that the physics is understood adequately to allow projection to burning devices, although wave absorption by alphas or beam ions may become an issue. Extensive modeling carried out for advanced scenarios on TPX, ARIES and ITER-EDA show that FWCD efficiency can be sensitive to details of antenna characteristics, cyclotron resonance locations, alpha particle and beam energy profiles. In particular, low frequency ($\omega < \omega_D$) current drive, found advantageous to avoid alpha absorption in the EDA design, is apparently not possible for ITER due to the smaller aspect ratio.

For the advanced modes on ITER and FIRE, relatively modest amounts of on-axis fast wave current drive are required as "seed current". The primary extensions are to higher performance plasmas (for example higher T_e with T_e ~ T_i which should result in higher current drive efficiency) and to higher energetic alpha density. These scenarios have been studied using the PICES full-wave ICRF solver and the CURRAY ray tracing codes. For ITER, the codes are in good agreement, showing ~75% of the power going to electrons, 17 to 25% going to tritium and 7% or less going to energetic alphas. The current drive efficiency is ~ 0.09A/W, resulting in a highly-peaked driven current of about I_{FW} ~ 1.75MA for P_{FW} = 20 MW. For the FIRE AT case using the slowing down distribution in PICES we find that considerable power is absorbed by energetic alpha particles (~ 50%), but that adequate on-axis current is obtained nevertheless. With the parameters listed we obtain total driven current of I_{FW} = 0.39 MA. In this case ~43% of the power is absorbed by electrons, ~48% is absorbed by hot alphas and ~8% is absorbed by tritium. These results are sensitive to the antenna spectrum reducing to about I_{FW} = 0.21 MA using the launched spectrum of a single 2-strap antenna. It is also quite sensitive to wave frequency, or equivalently magnetic field due to the proximity of the alpha cyclotron resonances to the plasma edge. The frequency of 95 MHz is carefully adjusted to get both the alpha fundamental cyclotron resonance and the 2nd harmonic resonance out of the plasma. Still the resonances are Doppler-broadened to the point that significant alpha absorption occurs on the inside and outside edges of the plasma. The CURRAY code gives much less power into hot alphas but much more into tritium (28%) so that the net current drive efficiency is in reasonable agreement with PICES. If these estimates are accurate fast wave power of ~18 to 24 MW would be required to obtain the 0.35 MA needed for the scenario.

The working group noted several issues with respect to use of ICRF which deserve mention but which were not deemed high level concerns likely to prevent achievement of machine missions. IGNITOR will require heating during B-field ramp. This implies that the resonance location will move across the plasma during the heat to burn. There is some experimental data indicating that this should be an acceptable scenario, perhaps requiring multiple RF source frequencies. However it was not possible for the working group to evaluate the implications for power deposition profiles or the effect of moving cyclotron resonances on plasma evolution. Similarly for FIRE, the heating is optimized by placing the ^{3}He resonance on axis (f_{RF} = 100MHz), whereas the FWCD is much improved by lowering the frequency to 95MHz thereby moving the minority resonance inside the magnetic axis. The capability of the RF systems to provide the needed heating and current drive during the time dependent phases of plasma build up, heating to burn, or access to AT modes was not assessed for any of the machines, because the information required to run the RF codes was not provided to the working group.

Power handling limits for ICRF antennas are not well understood. This is primarily a technology issues but also overlaps with plasma physics. The consequence of ICRF absorption by alphas or MeV neutral beams near the plasma edge should be assessed, even if only a small amount of power is absorbed, since it may affect the first wall. Also noted is that diagnostic techniques to measure minority concentrations (H, ^{3}He, etc) are badly needed to quantify RF physics scenarios.

Lower-Hybrid Current Drive. ITER and FIRE require lower hybrid current drive in the outer part of the plasma to support advanced tokamak scenarios. The lower-hybrid system is an upgrade for both ITER and FIRE. For IGNITOR it is a possible upgrade, although no modeling results were made available and hence we made no assessment of AT capabilities. Use of lower-hybrid waves for off-axis current drive and profile control has a firm experimental and theoretical base in non-burning plasmas. Lower hybrid has produced the highest current drive efficiency of any of the RF techniques, primarily due to its ability to Landau damp on relatively fast electrons ($v_{phase}/v_{thermal}$ ~ 2.4). In past experiments fully non-inductive discharges have been maintained with lower-hybrid waves on Alcator-C, ASDEX, FT-U, PLT, PBX-M, TORE- SUPRA (up to 2 minutes), TRIUM-1M (2 hrs), JT-60U (up to 3.6 MA), and JET (up to 3 MA). Its use on high-field, high-density machines relevant to FIRE has been demonstrated on Alcator-C, and FTU.

Our modeling with the ACCOME code indicates that the lower-hybrid current drive efficiency (□= 0.3) assumed by ITER is too high, and therefore their goal of steady-state operation at 9.0 MA with 52% bootstrap current, using 30 MW of NBI-CD and 30 MW of LHCD is not likely to be achieved. Our estimate indicates a total current of 7.6MA for the proposed AT scenario, including 59 % bootstrap current. Higher lower hybrid power, or a higher H-factor to maintain the beta, could make up for this deficiency. The projected 30 MW of negative NBI power at 1.0 MeV in ITER is sufficient to drive the required central seed current (2.0 MA) for AT operation (this could be replaced by a comparable amount of FWCD power since the current drive efficiencies are very similar). It should be noted that a desirable AT scenario should achieve at least 70% bootstrap current fraction at a β_n = 3, even without a conducting wall (beta limit) as an interim step toward DEMO-like requirements (80-90% bootstrap current fraction with reasonably low q_ψ to provide satisfactory confinement).

In FIRE, at 8.5 T a desirable AT scenario (β_n = 3 or above, bootstrap fraction at or above 0.7, reversed shear) is obtained with 20 MW of LH power and up to 24 MW of ICRF power for on-axis current drive. LHCD modeling using the new AT reference parameters at 6.5 T was also carried out and owing to the relatively higher value of n/B^2, a LH power of 30 MW is required to achieve the above AT parameters. Values of q_{95} ~ 3.5 are achieved in both cases. Further optimization, using more relevant density profiles, including barriers inside the q_{min} location, may result in reduction of the required LH power.

The present studies indicate that a frequency of order 5 GHz is sufficient to avoid alpha absorption of LH waves in both ITER and FIRE. Also, a possible beneficial effect of the negative portion of the LH power spectrum, included in the present studies, should be considered when assessing MHD stability near the beta limit (partial cancellation of the edge pedestal bootstrap current results in improved stability against the n=1 kink mode, with subsequent enhancement of beta-normal and increased core bootstrap current.

ECH. Only ITER calls for the use of ECH (f_{ECH} = 170 Ghz). The functions are bulk heating, and off axis localized current drive for control of neoclassical tearing modes or other instabilities. The high magnetic fields of FIRE and IGNITOR would necessitate development of power sources at frequencies much higher than presently available. Electron cyclotron heating and current drive has been studied and demonstrated on virtually every type of magnetic confinement device. Since ECH waves do not interact with ions, the presence of energetic alpha particles has essentially no influence. From that standpoint, the only difference between a conventional tokamak and a burning experiment is the high electron temperature resulting in increased Doppler-broadening of the resonance. Studies indicate that NTM stabilization in ITER will be required near r/a ~ 0.8. However the present design location of the above-midplane EC antenna in the ITER design is not optimal and is unlikely to provide robust current density stabilization of the neoclassical tearing mode. For the scenario considered, the local electron density is 10^{20} m^{-3} and the electron temperature is 5.1 keV. By optimizing over launch angle with the launch point above the axis as in the present design, we obtain 190kA of ECCD with 20MW of 170 Ghz injected power. This corresponds to efficiency γ = nIR/P = 0.06 $A10^{20}$ m^{-2}/Watt or a current density of 5.1 A/cm^2 at the q=2 surface (r/a ~ 0.8). This is to be compared to the bootstrap current density that must be replaced, 7 to 10 A/cm^2. Possible improvements include: (i) Moving the above-midplane antenna 1 meter down along the plasma chamber wall towards the equatorial plane. This improves localization of the driven current, increasing the current density by a factor of 2.5, perhaps sufficient for stabilization of a typical NTM mode. (ii) As an alternative, the source frequency could be increased to 200 GHz giving the same current density at r/a = 0.8 as the moved antenna, but current drive at the plasma center could no longer be provided. The trade-offs between launch location and source frequency suggest further optimization studies, possibly developing a two-frequency gyrotron or two separate systems for NTM stabilization and central current drive.

3.2.2 Alpha Physics/ Energetic Particles Assessment

A self-heated (burning) plasma has as its main power source the 3.5 MeV alpha particles released in the d-t fusion reaction. A central issue in burning plasma physics is whether the alphas will transfer their energy to the thermal plasma before escaping from the confinement device. The dominant loss mechanism for energetic alpha particles in a burning plasma occur through the resonant interaction of the alpha particle orbit with toroidal symmetry breaking fields, such as toroidal field ripple or from collective phenomena induced by the alpha particles and/or background plasma. In the following we assess the possible implication of such symmetry breaking fields on the confinement of alpha particles in the three burning plasma options.

Key Issues and Associated Assessment Criteria. The assessment of the three burning plasma options (FIRE, IGNITOR, ITER) is based on two broad criteria: what known phenomena may impact on the operation or otherwise adversely affect the performance of a BPX, and what new physics can be learned from a BPX which is not attainable on present day facilities.

The best understood and most controllable alpha loss mechanism in a tokamak burning plasma arises from the resonant interaction of alpha particles with the ripple in the toroidal magnetic field. The reduction of toroidal field ripple to negligible levels in the ITER by the use of ferritic inserts essentially eliminates alpha particle losses for AT operating modes. The level of ripple induced loss in both FIRE and IGNITOR are at acceptable levels.

The next level of symmetry breaking fields arise from the interaction of alpha particles with MHD modes of the background plasma. The most significant of these modes are the m=1/n=1 sawtooth mode and the Neoclassical Tearing Mode (NTM) in monotonic q-profile plasmas. It is expected that all three BPXs will exhibit sawtooth behavior in their standard operating regimes. The possibility to determine the stability of sawteeth with isotropic alpha particles is an important scientific objective for a burning plasma experiment. The alphas are expected to partially stabilize the m=1/n=1 mode, possibly leading to giant sawteeth and transient fast ion loss. The NTM is only expected in discharges with sufficiently high poloidal beta. These modes may be affected by the presence of alpha particles and may induce alpha particle loss at large amplitude, however the interaction of alpha particles with NTMs is as yet poorly understood.

The next level of symmetry breaking fields arises from the collective behavior of the alpha particles with the background plasma (eg. Toroidal Alfvén eigenmodes - TAEs) or with modes whose existence depends on the presence of the alpha particles (Energetic Particle Modes or EPMs). Of these the modes with the lowest threshold for excitation are the TAEs and related Cascade Modes. The Cascade modes are most relevant for advanced tokamak operating modes. The TAEs occur inside toroidicity induced gaps in the shear Alfvén continuum and are thus common to all toroidal confinement systems. Higher order gaps do exist but these modes tend to be more difficult to excite. Thus, the TAEs and Cascade modes are considered the most unstable collective alpha instabilities in a burning plasma experiment. Other modes do occur near MHD stability limits (such as Kinetic Ballooning Modes - KBMs, and Beta Induced Alfvén Eigenmodes - BAEs) however the nature of these modes and their interaction with energetic particles is not well understood and no quantitative assessment can be given at this stage. At high energetic particle beta the EPMs can be excited, where the mode frequency is determined by a characteristic frequency of the energetic particle distribution. Historically and experimentally the best understood of these is the Fishbone mode at the q=1 surface. Analysis indicates that the fishbone will not be excited in a burning plasma experiment as the expected alpha particle beta is insufficient to drive the mode.

A key point in our evaluation of the three BPXs is whether any of these varieties of collective phenomena can adversely affect the performance or safety of the facility. To address this issue with confidence we would need a robust prediction of the non-linear consequences of the excitation of such modes. Unfortunately such a predictive capability does not yet exist, and the observed losses due to TAE excitation in present devices has not been reproduced with available codes. This is especially true for the case where multiple modes interact with the energetic particle population. Thus, if the modes are predicted to be stable, then there is no consequence, however when a mode is predicted unstable, no reliable extrapolation can be made. The assessment in this case must be based on the linear stability of these modes.

We can also ask what can be learned from a BPX that is not accessible for study in existing facilities. A burning plasma experiment can afford the investigation of new physical phenomena such as avalanche losses produced by multiple overlapping modes. This phenomenon is relevant only at the largest scales required for achieving high fusion gain. In addition, unlike neutral beam or

ICRF heated minority ions in present experiments, the fusion-born alpha particles will have a nearly isotropic velocity distribution. Theory indicates that this feature is important for the mode stability assessment in burning plasma as well as for the nonlinear consequences of the instabilities. The combination of fusion scale and particle isotropy will yield valuable new information on the stability and nonlinear dynamics of alpha driven modes near reactor conditions. The assessment will thus need to take account of the range of toroidal mode numbers which can be excited in a burning plasma experiment in order to assess the possibility of avalanche loss dynamics.

In order to fully exploit the potential of a burning plasma experiment with regards to the alpha particle physics, the devices should possess flexibility in the operational window (plasma temperature, magnetic shear) adequate auxiliary heating capability, and diagnostics for measuring the alpha particle distribution and the unstable modes. The combination of flexibility and relevant diagnostics maximizes the likelihood that a burning plasma experiment will allow the study of new alpha particle phenomena. The assessment of device flexibility (operational window and parameter control) and diagnostic capability must be an important part of the overall assessment of a burning plasma experiment.

<u>Methods for Projecting Plasmas in Future Devices.</u> The prediction of plasma behavior in future devices is based on theoretical understanding and numerical tools that have been developed and applied to existing experimental data. The quality of our current understanding is far from complete and so our predictive capability is relatively primitive. Nonetheless we can place more confidence in estimating linear stability than non-linear dynamics. In addition we can more confidently estimate the linear stability of MHD modes of the background plasma (such as the TAEs) than the stability of EPMs. For this reason much of the quantitative assessment of alpha particle physics is related to the linear stability of gap modes such as TAEs in the standard operating regimes of the three BPX options.

The primary tools for this assessment are the NOVA-K code for predicting linear stability of global perturbative modes nestled within Alfvén gaps, and the HINST code for predicting the local non-perturbative stability of Alfvén modes. These two codes are complimentary, however differences in the output of the two codes can point to some hitherto unresolved physics issues and so caution needs to be shown in their interpretation.

<u>Uniform Assessments of Approaches to Burning Plasmas (FIRE, IGNITOR, and ITER).</u> For collective alpha particle driven phenomena the single most important factor is the copious production of alpha particles. The strong dependence of the alpha particle beta and population density with operating temperature is common to all burning plasma experiments. The main distinction between the three BPX options is in the proposed operating temperature of the devices. For the TAE, analysis bears out that the stability in all three devices can be reduced to a single parametric dependence of toroidal beta vs. operating temperature. Generally, TAEs are expected to be unstable for temperatures exceeding 25 keV in all devices, with lower temperatures possible in IGNITOR because of the lower plasma beta. If the stability phase space in temperature ad beta is to be properly explored in a BPX, then some temperature flexibility is required for the three BPX. Due to the high operating temperature of ITER, in the range where TAEs are expected to be excited, we conclude that ITER is unique among the three BPXs in being able to readily access a regime in temperature and beta where Alfvén eigenmodes may be studied.

Regardless of the operational flexibility of the three proposed experiments, no progress can be made in alpha physics without a detailed diagnostic measurement capability to resolve the spatial redistribution and loss of alpha particles, as well as to obtain evidence for the excitation and magnitude of internal modes in the plasma. These measurements require sophisticated diagnostic tools which have not been fully developed even on present day experiments. Nonetheless, the ITER device has at present the most advanced diagnostic plans, partly owing to its advanced state of design. It should be pointed out however that there are serious physics issues that may limit diagnostic capability in a burning plasma experiment and which may represent very stubborn obstacles to measurement of alpha phenomena. Reflectometry may be of limited use in plasmas exceeding 20 keV owing to wave absorption issues. This may require less well localized interferometry methods for the detection of internal modes, and possible a fan array of interferometer chords on the plasma midplane in order to record the amplitude and spatial localization of alpha driven modes. At present these is no validated alpha particle distribution measurement technique that extrapolates to a burning plasma experiment. The possibility of a future DT experiment on JET should provide the much needed validation platform for burning plasma diagnostic development.

3.2.3 MHD Science in a Burning Plasma

A burning plasma represents a new and unique regime for magnetically confined plasmas, allowing us to investigate scientific questions related to the stability of a complex, self-organized thermonuclear system and the interaction of MHD modes with an isotropic population of fast ions. While not requiring self-heating, a burning plasma-scale experiment will allow us to investigate the dependence of macroscopic stability on plasma size, and the importance of key kinetic effects in plasmas with very low collisionality, small gyroradius, and high Lundquist number.

The production and control of plasma in a self-consistent state with strong self-heating and self-generated current, and the crucial role of MHD stability in determining that state, can only be investigated in a “burning plasma” experiment. In a plasma that is largely self-sustained through alpha heating and bootstrap current drive, the internal profiles of pressure, current density, and rotation will be determined primarily by internal processes that are linked in a complex, nonlinear way. In addition, these profiles can all be modified by MHD instabilities, whose thresholds in turn depend on the profiles. Systems for avoiding or controlling instabilities through profile control or direct feedback stabilization may act differently in such a tightly coupled system. All these interactions are difficult to simulate in present inductively driven, externally heated tokamaks.

The effects of energetic alpha particles on MHD modes, and of MHD modes on confinement of alpha particles, also require a burning plasma for complete study. In conventional tokamak operation, the physics of the m = 1 internal kink mode (sawtooth instability) plays a central role since it can lead to a large-scale redistribution of the plasma pressure and hence of the fusion reaction profile. Theoretical work indicates that MeV alpha particles can provide stabilizing mechanisms for the sawtooth. However, a complete understanding of the alpha particle, q profile evolution and sawtooth instability interaction has not been established experimentally. Only a burning plasma experiment allows a realistic study of the nonlinear interaction between a population of fast ions with an isotropic velocity distribution, the MHD instabilities that it may drive, and the redistribution of fast ions that may result.

Issues related to the scaling of non-ideal MHD stability with plasma size require a high beta plasma with larger radius and/or magnetic field than existing experiments, although not necessarily a burning plasma. For example, present experiments are consistent with an unfavorable scaling of the neoclassical tearing mode (NTM) threshold island size versus normalized ion gyroradius $\rho_i^*=\rho_i/a \sim (aB)^{-1}$. However, uncertainties about the dependence on collisionality and a predicted favorable scaling of the seed island amplitude with magnetic Reynolds number S make the NTM stability threshold in a tokamak reactor difficult to predict with confidence. Edge pedestals with low collisionality and small ρ_i^* are also a regime not easily accessible in present experiments. A burning plasma experiment should bridge the gap between existing tokamaks with a B~1-4 m-T and a reactor prototype which may have a B~16, and provide crucial new understanding of NTM physics.

A burning plasma experiment that lies between existing and reactor-size plasmas, as measured by size scaling parameters such as a B, can address most or all of the macroscopic stability issues that will be present in a reactor-size plasma, and will provide a strong basis for extrapolation in scale of stability limits, alpha-particle effects, and integration issues. Much of the underlying physics should transfer to burning plasma physics in other magnetic configurations such as the ST stellarator, and RFP. Extrapolations in scale and especially in configuration require that the burning plasma experiment be well-diagnosed, in order to provide detailed experimental validation of theoretical and numerical models. MHD science with predictive capability is needed to impact the development of any magnetic configuration, tokamak or other, in a complete and reliable way.

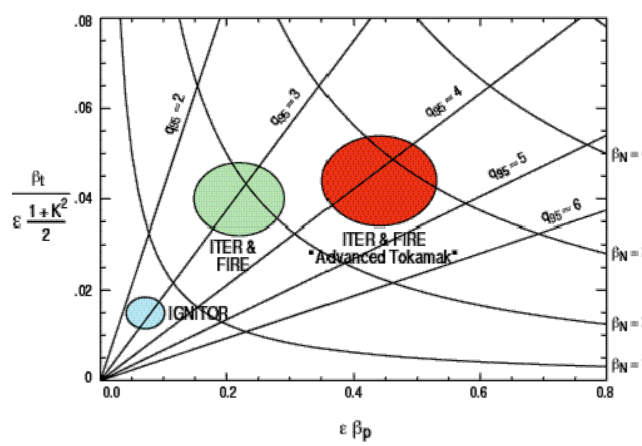
A burning-plasma tokamak experiment also has the potential to make significant contributions to plasma stability science in fields outside of fusion energy, through expanded understanding and validation of non-ideal MHD physics (incorporating effects such as resistivity, FLR, energetic ions, plasma flow, etc.). Validation of the underlying physics in laboratory experiments will increase the confidence in applying these models in settings (extraterrestrial plasmas, for example) where controlled experiments and detailed internal measurements are more difficult.

MHD stability limits are not a fundamental obstacle to the burning-plasma missions of the three proposed machines. The base scenarios are stable to ideal MHD (except the m/n=1/1 mode); IGNITOR in general operates farther from stability limits. Central m/n=1/1 sawtooth instabilities and edge localized modes (in H-mode operation) are anticipated in all three devices, but are not expected to prevent access to the burning plasma regime.

Active control of MHD instabilities will most likely be required in ITER and FIRE. Both have plans for neoclassical tearing mode (NTM) control through localized current drive, although the lower hybrid current drive approach planned for FIRE has less experimental validation. The advanced tokamak scenarios require wall stabilization of the n=1 kink mode, using feedback stabilization alone (FIRE) or feedback plus neutral beam-driven rotation (ITER).

Burning plasmas offer a regime not accessible in existing experiments. IGNITOR, FIRE, and ITER would all yield important new MHD physics in self-heated plasmas with a large population of energetic alpha particles. FIRE and ITER would address additional stability issues in higher beta H-mode plasmas and in advanced tokamak plasmas with a largely self-generated current profile. ITER's long-pulse scenarios will address the stability of plasmas with a fully relaxed current profile. Much of the MHD stability physics learned in a burning tokamak plasma should be applicable to a broad range of confinement concepts.

Introduction. Any magnetic configuration meriting serious consideration for a fusion reactor must satisfy constraints imposed by MHD equilibrium and stability. It is widely appreciated that ideal MHD theory can be used as a foundation for use in the design of next step options. However, non-ideal MHD effects are also important and are not as well understood; the effect of resistivity, neoclassical physics, energetic particles, etc. on next step devices cannot be predicted with as much certainty as ideal MHD. Burning plasma experiments will improve our present understanding of MHD physics by extending the operational space of various non-ideal plasma parameters and by addressing the role of self-consistent interactions of energetic particles, alpha heating, profile evolution and plasma stability.



FIRE and ITER occupy roughly the same regimes of dimensionless parameters relevant to MHD stability: beta, safety factor, etc. The primary distinction between them is size and pulse length. IGNITOR is in a quite different parameter regime due to its lower beta. ITER, FIRE, and IGNITOR all provide the opportunity to study MHD stability in plasmas with relaxed pressure profiles driven by self-heating, with ratios of pulse length to energy confinement time of ~100, 25, and 10, respectively. In the conventional high-Q scenarios, all three devices have pulse lengths that are about 1-2 current relaxation times; ITER's long-pulse advanced tokamak mode would have a fully relaxed current profile with a pulse length of ~10 current relaxation times.

In order to provide a uniform assessment for the three burning plasma options, the MHD physics working group examined the important burning plasma physics issues in a number of topical areas. In each of these areas, we articulate the new physics to be learned from a burning plasma, assess the limitations imposed by MHD physics on the ability of the proposed experiments to achieve their full range of scientific goals, and identify the impact the physics to be learned will have on development of future tokamak and non-tokamak fusion devices.

Equilibria. The MHD equilibria used in the detailed stability calculations are sample snapshot equilibria that were provided by the proponents for each experiment. For each of the devices, conventional tokamak scenarios were studied, as well as "advanced tokamak" (AT) scenarios for FIRE and ITER. In addition to reference cases for each device, equilibria with variations of the pressure and current density profiles were generated in order to perform sensitivity studies.

Ideal MHD. For each of the equilibria generated, we assessed ideal MHD stability with respect to Mercier, ideal ballooning and n=1, 2, 3 internal and external kink modes using the stability codes DCON and PEST. All the equilibria were stable to high-n ballooning modes, unless q(0) was less than unity when the related Mercier criterion was violated. Generally, when a q=1 surface is present, ideal MHD instability to an m=1 internal kink is predicted for sufficiently high plasma β_p. Therefore, beta limits for ideal MHD modes must be assessed by varying q(0) in a range slightly below and slightly above unity. With q(0) greater than unity, the conventional tokamak scenarios for all three machine designs are generally well away from ideal stability limits. With q(0) less than unity, ITER and FIRE are ideally unstable to an m/n=1/1 instability, while IGNITOR by virtue of its lower beta, is stable to somewhat lower q(0). The implications of the unstable 1/1 mode are assessed in the following section. The AT scenarios for ITER and FIRE require wall stabilization for ideal stability.

m=1 Stability. A crucial issue in all three of the burning plasma experiments is the interaction of the alpha particles with m=1 modes, including the effect of an isotropic alpha particle population on the stability properties of the m=1 mode and the redistribution of the alpha population following a sawtooth crash. (Note that the interaction of alpha particles with MHD modes is also discussed in the Energetic Particles section of this report.) Present experiments in regimes relevant to ITER and FIRE [β_p~0.2, q_{95}~3.0] indicate that sawteeth do not have a significant direct effect on stored energy. Previous D-T experiments find that alpha particles are redistributed at a sawtooth crash, but are not lost. More important is the potential impact of a large sawtooth in triggering MHD instabilities such as the neoclassical tearing mode, leading to locked modes or disruptions.

A linear theory of m=1 modes has been developed by Porcelli, et al. that accounts for the ideal MHD stability properties of the internal kink and kinetic effects coming from trapped thermal and alpha particle contributions. Marginal stability of the linear mode can be used to predict the onset of sawteeth. When this calculation is coupled with the TSC transport code, predictions for the sawtooth period and mixing radius can be made with either complete or partial reconnection assumptions. Simulations indicate that for all three devices, large-amplitude, long-period sawteeth will have little direct effect on the total stored energy or fusion power. We note that detailed predictions are complicated by uncertainties in the transport and reconnection models used, and in the precise mathematical structure of the ideal MHD δW used in the sawtooth simulations.

It is important that the experiments have methods for controlling sawteeth through current profile control, either by maintaining q(0) above unity or by stimulating small-amplitude sawteeth. Ion cyclotron heating (ICRH), electron cyclotron current drive (ECCD) and lower hybrid current drive (LHCD) can all be used for this purpose. All three proposed experiments provide good opportunities to investigate the m=1 instability and reconnection physics in the presence of an isotropic population of energetic alphas, a key issue for future reactors.

NTM Stability. One of the crucial issues for any long pulse, high temperature tokamak is the appearance of neoclassical tearing modes (NTMs). Empirical observations indicate that the critical beta for neoclassical tearing mode onset scales with normalized ion gyroradius $\rho^*=\rho_i/a$. Since this scaling is not favorable for larger plasma experiments and there are uncertainties in theoretically predicting the nonlinear island width threshold and seed island mechanisms, NTM physics is one of the key MHD science questions to be addressed in a burning plasma experiment. From analytic estimates, the anticipated saturated island widths produced by NTMs in ITER and FIRE (w/a ~ 0.1-0.2 with poloidal beta ~ 0.5-1.0) are large enough to cause significant reduction of energy confinement and potentially lead to locked modes, loss of H-mode or disruption. The saturated island size is significantly smaller for IGNITOR because of the lower beta, and is not likely to pose a problem. NTMs are expected to be a less severe problem in AT scenarios that have q>2 everywhere and NTM-stabilizing negative magnetic shear in the core.

With the uncertainties in the theory and the anticipation that sawteeth will trigger large NTMs, techniques for controlling NTMs are crucial for ITER and FIRE. ITER plans to use localized ECCD, a technique that has proven successful in ASDEX-Upgrade, DIII-D, and JT-60U. FIRE has proposed using LHCD for NTM control, but this technique is less well validated experimentally. Additionally, methods for controlling the sawtooth amplitude (and hence the seed island mechanism) may be employed to avoid NTM excitation. FIRE and ITER offer opportunities to study NTM stability with ρ^* and magnetic Reynolds number S (important for reconnection physics) intermediate between existing machines and reactor-scale plasmas.

Wall Stabilization. Advanced tokamak scenarios for FIRE and ITER need wall stabilization of the n=1 kink mode, since the anticipated broad current profiles, elevated q(0), and large β_N make these plasmas susceptible to ideal external kinks. In these cases, the plasma can be unstable to resistive wall modes in the presence of a wall with finite conductivity. Resistive wall modes can be stabilized in the presence of sufficient plasma flow and in principle can be controlled using active feedback. MARS modeling, using a sound wave damping model, predicts that stabilization can be achieved with rotation frequencies on the order of 0.5-1.5% of the Alfven frequency at rational surfaces (comparable to typical values of the critical rotation frequency observed in DIII-D), although precise predictions are sensitive to the plasma profiles. The estimated rotation driven by the planned neutral beam power for ITER and FIRE is marginal to sub-marginal, but there is sensitivity to the model used for momentum transport. RF-induced plasma rotation is too poorly understood to make accurate assessments.

The characteristic time constants of the passive stabilizer and conducting structures near feedback control coils differ in FIRE and ITER. The faster response of the anticipated FIRE feedback system indicates that stabilization approaching the ideal wall beta limit can be obtained. With the slower feedback coils of the ITER configuration, only a modest improvement of the beta above the no-wall beta limit can be realized. However, neutral beam-induced rotation would improve the stability further. Both FIRE and ITER will be able to address the resistive wall stability properties of a large burning plasma tokamak experiment in a reactor-relevant regime of little or no external torque.

Error Fields and Critical Rotation. Relatively small non-axisymmetric magnetic fields can slow plasma rotation and cause locked modes or produce seed islands for neoclassical tearing mode growth. Observationally, slowly rotating plasmas with resonant surfaces in the plasma are susceptible to the penetration of field errors with resonant magnetic helicities. This sensitivity rises as β approaches the n=1 ideal kink limits. Analytic estimates, based on field error penetration with rotation at the electron diamagnetic drift frequency, give upper limits on the amplitude of (2,1) resonant error fields to avoid low density locked modes during ohmic startup. These limits should be readily achievable. The error field limits become somewhat smaller for AT plasmas in ITER and FIRE in order to avoid drag from the "error field amplification" effect in AT plasmas above the no-wall stability limit, but should still be achievable with the planned correction coil systems and the possible addition of neutral beam-driven rotation. The three devices would provide an opportunity to investigate the plasma's sensitivity to error fields in a reactor-relevant regime of little or no external torque.

Pedestal Stability. Edge localized modes (ELMs) constitute an important concern for any burning plasma experiment relying on H-mode operation. Chiefly, large ELMs have a deleterious effect on divertor lifetime and can adversely impact high performance operation. In ITER and FIRE, the power loads to the divertor plates from the largest conceivable Type I ELMs are at the respective design limits. However, ELMs also have the beneficial effect of reducing impurity and ash accumulation and allow for steady state density control. Present theoretical efforts toward understanding edge MHD properties focus on intermediate-n ballooning/peeling modes, which may be destabilized by steep edge gradients and the associated edge bootstrap current. The limiting pedestal height predicted by MHD stability is in the range needed for good performance in all three devices, assuming the pedestal width is similar to present experiments (□/a~0.03). It is becoming clear that these instabilities play an important role in Type I ELM onset. Several tools are known for reducing Type I ELM size, creating a transition to smaller Type II ELMs, or eliminating ELMs – discharge shaping, counter-injection of neutral beams, variation of edge plasma collisionality, and shallow pellet injection. There is not sufficient understanding of the crucial physics parameters required to attain alternative, more benign regimes to permit scaling to burning plasma parameters. Nevertheless, it is expected that each of the experiments has sufficient flexibility in varying shape or edge conditions to avoid serious divertor problems.

Coupling to Theory, Modeling, and Other Confinement Concepts. An important scientific outcome for the burning plasma experiment would be the generation of MHD science that could be applied to a broad range of magnetic fusion energy concepts. The physics of magnetic reconnection, wall stabilization, plasma rotation and neoclassical effects are directly relevant not only to future tokamaks but also to other concepts, including STs, stellarators and RFPs. Therefore, it is important that MHD theory and modeling efforts be closely coupled to burning plasma activities. In order to facilitate detailed comparisons between theory and experiment, a burning plasma experiment must be well diagnosed. Critical diagnostics include external magnetic field measurements plus profile measurements of plasma density, temperature, rotation and current density for equilibrium reconstruction; and fluctuation measurements of external magnetic field and internal profiles to determine the toroidal, poloidal, and radial mode structure. Validation of theories and numerical models is the means by which a burning plasma experiment can benefit a broad range of fusion concepts, and perhaps plasma science applications beyond fusion.

3.2.4 Transport

Overarching goals for transport science. Transport research on a BPX should be guided by goals identified by the transport community at the last Snowmass meeting, and endorsed program-wide in various forms since, including in the IPPA document: Pursue the challenging yet realistic goal of developing comprehensive predictive transport models based on physically reasonable assumptions and well-tested against experiments. Theory/experiment comparisons should be done at the turbulence level; and Develop tools and understanding for the control of transport and transport barriers.

Therefore, a burning plasma physics experiment must have the following attributes: It must have sufficient flexibility and diagnostics to stimulate the science required to improve predictive capability for steps beyond itself, including non-tokamaks; and It should enable the development of control tools for managing nonlinear pressure profile dynamics that may exist only in advanced burning plasma regimes.

Developing tools that modify the pressure through turbulence and transport manipulation would have high impact, and requires flexibility, excellent diagnostics, and adequate access.

Assessment summary – prospects for studying generic transport issues: In terms of plasma characteristics, including pulse length, all devices can provide important information on some aspects of pressure profile dynamics. Ignitor's mission places questions of pressure profile control outside of its mission. FIRE and ITER can study dynamics and control, with ITER possessing the most complete set of tools (see below). Scaling of core and edge transport at reactor-relevant dimensionless parameters, □, □*, □*, and n/n_{GW} can be studied on ITER and FIRE. Ignitor cannot match these dimensionless parameters as satisfactorily, owing to the lower □ values. Diagnostic questions exist for all three devices.

Assessment summary – configuration flexibility: A BPX must be able to have its configuration varied about its usual operating point. Needed flexibility includes capability that will increase the chances for access to advanced modes of operation (e.g. transport barriers) and for contributing broadly to plasma science. Desired flexibility elements include:

- Density control (pumping and pellet injection, including inside launch pellets)
- q profile control and capability for long pulse operation.
- Perturbative heating capability of ions and electrons to test transport models
- Neutral beams, where possible, for heating and momentum input for rotation variation and variation of heat and particle deposition
- Shaping and elongation flexibility for internal inductance, pedestal, and ELM control.

All machines offer configuration flexibility that will enable advances in transport science to varying degrees. ITER has the most comprehensive set of tools, followed by FIRE. Each device has limitations compared to present-day advanced tokamaks, however, and so advances will demand a robust base program working to complement the BPX research. For more background, find the table of the Integration Group regarding device flexibility.

ITER's complement of 1 MeV beams (33 MW baseline), together with current drive capability from ECCD, LHCD (upgrade), and on-axis ICRF fast wave will allow for variations of rotational and magnetic shear, increasing the chances for access to advanced operating regimes. Simulations indicate that its current drive tools should enable the q profile to remain elevated and reversed. The various heating schemes will enable separate heating of electrons and ions and perturbative heating for model testing. Pellet injection capability, including HFS launch, is integral to the program, as is divertor pumping. Shaping variability (1.85 < □ < 1.97; .48 < □ < .58) is limited compared to present-day AT's such as DIII-D, owing to the closed divertor geometry. More shaping flexibility, perhaps made possible during a period of low neutron fluence operation and reduced heat management requirements, would inform the science of pedestal physics and ELMs considerably.

FIRE's simulations indicate that its current drive tools (LHCD and on-axis ICRF) should keep q elevated or reversed in a steady-state configuration. Completely separable heating of ions vs. electrons is not possible, but the ratio is variable depending on whether He^3 minority or direct electron heating is employed. 120 keV neutral beam injection is posed as a possible upgrade. At present, such a beam can only be injected nearly perpendicular to the plasma current unless tangential access is enabled by reducing the number of TF coils. Shaping, while aggressive in its target values (2 < □ < 2.1; .65 < □ < .85), is again limited in terms of variability compared to present-day AT's, as it is in ITER, and for similar reasons. More flexibility, perhaps made possible during operations with reduced heat fluxes and relaxed divertor requirements, would enhance the studies of pedestal physics and ELMs. Pellet injection capability is integral to the program, as is divertor pumping.

Ignitor's q-profile control can be achieved through programming of the current rise phase, combined with ICRF heating over a wide range of currents. Simulations indicate that reversed shear cannot be actively maintained with the auxiliary systems planned. Pellet injection is planned and is an integral part of the program. Pumping is not presently in the design, but a pump limiter option is being studied. Studies that demand rotational shear control using deposition of momentum with beams will not be possible in the presently planned Ignitor program. While the range of shapes is not constrained by divertor requirements, it is uncertain how Ignitor will best inform the science of pedestal and ELM physics without a divertor.

Assessment summary: turbulence and profile diagnostics. Concerns exist for all three devices regarding certain profile diagnostics usually regarded as essential for transport studies, and especially turbulence diagnostics. This is in part a consequence of chosen priorities for development up until this time. Up until now, a major priority has been designing diagnostics aimed at protecting the machine and enabling control. ITER has spent considerable resources developing its diagnostic set to this end. FIRE, being in the pre-conceptual design phase, has devoted far fewer resources to this issue. From discussions, it seems that Ignitor's attention to these diagnostic sets has not been a high priority as of yet. Developed turbulence diagnostics proposals are pointedly lacking, and this represents a concern for all three devices especially given the central scientific importance of these measurements to turbulence and its interpretation to burning plasma-related transport issues. The high density and line-integrated densities of these devices make beam-based profile diagnostics a challenge. All three devices propose to use reflectometry for density fluctuation measurements. While likely to succeed in the edge, its utility for the core in the absence of wavefront imaging needs to be assessed. All devices plan to measure temperature fluctuations from ECE emission. Finally, no proposal has been made for short wavelength fluctuation measurements in assessing electron thermal transport, yet this topic remains one of the most vexing in transport science and is critical regarding predicting the performance of any BPX. Discerning the behavior of the electron channel and associated turbulence in the presence of strong electron heating via alphas, and associated questions related to the sustainment of transport barriers, can only be answered in a BPX.

Predicting performance of a BPX. The measure of performance – Q = P_fus/P_ext, the ratio of fusion power produced to external power supplied. Q is important for energy economics. The fraction of alpha self-heating F =Q/(5+Q) □ nT□ the "fusion product" is a less sensitive parameter more relevant to scientific goals. The BPX must have Q greater than 5 which amounts to more than 50% self-heating and preferably Q greater than 10 (66% self-heating) in the D-T phase. The controllability of self-heated devices within MHD stability boundaries is an experimentally open question that must be answered in a burning plasma device. The technology goals for material wall neutronics testing or power handling depend on some required P_fus per surface or circumference, and hence depend on achieving high Q at full design P_ext. Q=10 is the nominal goal of all current designs and the maximum design P_ext is generally set by the threshold power required to obtain good H-mode confinement in a non-burning (D-only) phase.

Uncertainty of extrapolated predictions**.** Q=5F/(1-F) is a very sensitive parameter when F □ nT□ exceed 50% and a variety of empirical and theoretical modeling methods are best used to determine its range. Empirical scaling laws for the energy confinement time □ (e.g. H98y2) typically have an RMSE of 15% so that nT□ (=□^{2} P_ped /V) and F are uncertain by 30%. A specific prediction of Q=5 thus corresponds to 2.7 < Q < 9.3 [or Q=10 to 4.3 < Q < 30]. nT□ is being extrapolated over a considerable distance particularly in the rho_star dimensionless variable. For gyroBohm scaling (typical of H-mode scalings) nT□ □ $B^3 a^{5/3}$ at fixed a/R, q, □, □, and □*. This varies about 15-fold from JET to ITER. Empirical scalings for □ have a shallow minimum in the RMSE. Thus for example a collisionless electrostatic gyroBohm scaling with an RMSE of 16.6% (slightly worse than 14.5% for y2) can obtain Q=30. (F=0.85) compared to Q=5 (F=0.5) for a y2 ITER prediction. This more optimistic scaling does not have the beta-degradation of y2 and is supported by beta-scaling experiments in single machines. Global empirical confinement time statistical scaling rules for □ are typically combined with a 0D power balance dW/dt =P_ext + P_alpha - P_brem –P_ped where P_ped=W/□ is the transport loss power flowing to the H-mode pedestal or cold L-mode edge and P_ext = P_aux + P_oh. Plasma profile peakedness must be supplied or provided by core transport code models.

There has been considerable progress in the development of theory based core transport code models. Two physically comprehensive US models have been extensively benchmarked with the ITER profile database and used in this Snowmass 2002 assessment: MultiMode and GLF23renorm. The latter has been developed by fitting to gyrokinetic stability and nonlinear simulations. Both have comparably good RMSE ~ 10% fits to the total stored energy W_tot given the boundary conditions or the energy stored behind the H-mode pedestal W_ped =3Vn_pedT_ped (or equivalently the L-mode r/a > 90% edge). However both core models have a degree of stiffness and are very dependent on T_ped. A *perfectly stiff* core has T(0) linear in T_ped independent of

P_ped. MultiMode is weakly stiff (e.g. T(0) $\propto$ T_ped$^{0.3}$) and GLF23renorm is very stiff (e.g. T(0) $\propto$ T_ped$^{0.7}$). The net result is that Q scales roughly as Vn_ped beta_pedB^2/P_ext$^{0.25}$ and Vbeta_ped^{2}B^4/P_ext$^{0.9}$ respectively. There is no theory based model for T_ped (or beta_ped). Empirical scaling models for the world database have been developed. A free fit suggests beta_ped $\propto$ P_ped$^{0.3}$ and nearly independent of n_ped but it is generally believed that the P_ped power dependence obtains only at low P_ped and at high P_ped the ELMing H-mode pedestal pressure gradients are limited by MHD stability: beta_ped < (width_ped/R)s/q^2 independent of P_ped. Assuming saturation in P_ped, various P_ped independent statistical fits to beta_ped (or equivalent) have been obtained with RMSE of 30% typical of the best. For these MHD limit rules, the width_ped/R is fit to a combination of dimensionless variables like beta_pol_ped$^{0.5}$ or rho_star_pol_ped (or some fractional power of this). It is very difficult to distinguish these two variables in the pedestal data. In the first case the projected maximum beta_ped will tend to be comparable to present data but in the second case, beta_ped may decrease somewhat and be weakly dependent on n_ped. Thus in an MHD limited pedestal regime, the weakly stiff MultiMode model has Q $\propto$ beta_ped and Q can increase with density and weakly with lower P_ext. In contrast the stiffer GLF23 projections with Q $\propto$ beta_ped^2 are more sensitive to pedestal scaling and Q tends to be independent of operating density but increases almost inversely with decreasing P_ext. Indeed very large Q (essentially ignition or Q= infinity) is not precluded as P_ext is withdrawn, provided beta_ped does not decrease too much (again the issue of pedestal power dependence), and the decreased P_ped does not drop below the H-L transition power. The Q projections for cold edge L_modes are likely to have a similar dependence, but there exists no equivalent of a beta_ped (or beta_edge) database scaling. In any case beta_edge is likely to be strongly dependent on P_ped and not limited by MHD in L-mode.

Assessment Summary of BPX Performance: Applying standard empirical H-mode scaling rules for power access and global confinement time, it is expected that all three devices will achieve their goal of dominant self-heating F > 0.5 (Q=5) and F > 0.66 (Q=10) seems likely. The most widely tested theory based core models combined with a variety of semi-empirical scaling rules for the pedestals support this conclusion. Both ITER and FIRE with standard divertors are designed for full H-mode access and pulse durations of 2-3 current relaxation times with expectations for ITER somewhat more robust. Any added density profile peaking helps performance and the added the rotation from the 1MV NBI gives ITER an added reserve for better performance. IGNITOR is not designed for H-mode but an alternative de-rated current “wall-separatrix” operation may well access a transient H-mode possibly limited by wall power handling. The baseline IGNITOR operation anticipates a cold (L-mode like) edge with significant ohmic heating comparable to the auxiliary L97 global scaling enhancements of about H=1.25-1.4 are needed for Q = 5-10 for peaked density profiles n(0)/<n> = 1.8 Such enhancements (with cold edges) and peaking have been obtained transiently but the database for this is not widely established and steady state demonstration discharges in existing tokamaks are needed. ITER under the same conditions of cold–edge and peaked density require L97 enhancements of 1.4-1.7. The core theoretical models used require hot (H-mode like) edges or pedestals for devices to obtain Q=5-10.

3.2.5 Boundary Modeling of Burning Plasma Experiments

The existence of a Burning Plasma Experiment (BPX) would provide both an opportunity to learn the details of boundary physics in a reactor plasma regime, and a challenge to design plasma facing components which provide sufficient control to permit meaningful operation of the experiment. We define the boundary region as lying between ~95% of the last closed flux surface (LCFS) to a limiting surface outside the LCFS. This covers the region which is typically described as the pedestal for H-mode devices such as ITER and FIRE, and the scrape off layer (SOL). The ability to model this region of a tokamak plasma has advanced tremendously over the last decade with the development of 2D fluid plasma models which contain much of the physics expected to be important. The research effort, driven by the perception of a serious divertor heating problem for the ITER device, led to the development of the “detached divertor” concept in which the power load on the wall designed to receive the exhaust power is reduced by enhanced radiation via impurities. The nature of these detached divertors have been thoroughly summarized by and can be only briefly summarized here. The basic idea is to use impurity radiation to not only spread the exhaust power over a large surface area, but to reduce the electron temperature in the SOL sufficiently to obtain three body recombination sufficient to reduce the ion current to material walls, and the concomitant heating which arises from recombination within the wall. It has been generally found that the exhaust power from a high power burning plasma device would be sufficient to destroy any material surface if care isn't given to designing the exhaust surfaces to operate in some variant of the detached plasma mode.

Adequate design of a system to deal with the exhaust of power and particles from a burning plasma experiment must address many issues. These include: control of exhaust power to prevent destruction of plasma facing surfaces; prevention of impurities used to radiate the power from transporting to the confined plasma and reduce fusion power production; control of primary fuel neutrals recycling from walls; and removal of fusion ash to prevent dilution and reduction of fusion power. The challenge of a BPX for boundary physics is to provide a divertor design sufficient to permit adequate control of the power exhaust to prevent destruction of plasma facing components. The exciting opportunity a BPX offers is to understand how control of the power and particle exhaust can permit optimization of the plasma parameters in the pedestal region, and thus permit optimization of core plasma performance in enhanced confinement modes. Progress made in boundary physics on existing devices permit optimism that the physics of this important region of the plasma is adequately understood to permit design of divertors sufficient for initial operation of the BPX. Two issues which are potentially critical for the initial design of a BPX are control of transient phenomena such as disruptions, and ELMs in H-mode operation, and tritium retention associated with tritium co-deposition with carbon. These are both very active areas of research in the world's fusion program.

Introduction. The results of modeling of the character of the boundary plasma in the three devices being considered for a burning plasma experiment (BPX) FIRE, IGNITOR, and, ITER are reported here (Our simulations for the ITER device are on the upgraded configuration previously referred to as ITER.) We define the boundary as lying between 95% of the last closed (magnetic) flux surface (LCFS) to 105% to 110% outside the LCFS. This covers the region which is typically described as the pedestal for H-mode devices such as FIRE and ITER, and the scrape off layer (SOL). Previous efforts to model the behavior of the boundary plasma have varied widely among these devices, from extensive modeling in ITER, to limited modeling for IGNITOR. We have attempted to apply a uniform model to each device. The ability to model this region of a tokamak plasma has advanced tremendously over the last decade with the development of 2D fluid plasma models which contain much of the physics expected to be important, and which have been validated extensively against experimental data from existing tokamaks. The research effort, driven by the perception of a serious divertor heating problem for the ITER-EDA device, led to the development of the "detached or partially detached divertor" concept in which the power load on the wall designed to receive the exhaust power is reduced by enhanced radiation via impurities. The nature of these detached divertors have been thoroughly summarized and can be only briefly summarized here. The basic idea is to use impurity radiation to spread the exhaust power over a larger surface area, and to reduce the electron temperature in the SOL sufficiently to obtain three body recombination sufficient to reduce the ion current to material walls, and the concomitant heating which arises from recombination within the material. For diverted tokamaks it has been generally found that the exhaust power from a high power burning plasma device would be sufficient to cause unacceptable erosion of any material surface if care isn't given to designing the exhaust surfaces to operate in some variant of the detached plasma mode. It is our purpose to assess the issue of exhaust power handling for each of the three devices.

Adequate design of a system to deal with the exhaust power and particles from a BPX must address many issues. These include: control of exhaust power to prevent destruction of plasma facing components (PFC); prevention of impurities which radiate the power from transporting to the confined plasma and reduce fusion power production; control of primary fuel neutrals recycling from walls; and removal of helium ash to prevent dilution and reduction of fusion power. The assessment presented here is inadequate to address all of these issues in depth. We focus primarily on the power exhaust question. We describe the results obtained by applying the 2D fluid plasma code UEDGE to each of the proposed experimental. We use a combination of a simple model for the effect of impurity radiation, which assumes the impurity ion density is a specified fraction of the primary ion density, and a more complete multi-species model which determines the spatial profile of all ionization states of an impurity. The simple approximation permits estimation of the effect of impurity radiation, but does not address issues associated with impurity transport.

Time dependent phenomena have not been addressed in the simulations discussed here. However, one issue which is recognized as a potential problem for the design of a power exhaust system for a burning plasma system is that of "off normal" events, which generally include disruptions and the effect of Edge Localized Modes (ELMs) associated with operation in H-mode. Space limitations preclude the use of figures to show either the geometry considered for each device, or the details of variations obtained in our modeling.

The boundary physics working group finds that all three devices will likely have acceptable power loads in their base-case designs, but the Ignitor wall limiter is the more difficult to convincingly model because of alignment and potential impurity MARFE issues. While the PFC design of the IGNITOR device appears compatible with the mission of achieving a burning plasma, it appears less relevant for the development of a steady state fusion reactor with simultaneous control of power exhaust, particle exhaust, and impurity flow than the divertor configurations of FIRE and ITER. Estimates for AT scenarios in FIRE and ITER indicate that these modes may be limited by high divertor heat flux. Achievement of the pedestal parameters required for good confinement in these devices will require careful attention to edge fueling and pumping configurations, as well as adequate suppression of pedestal turbulence. Both FIRE and ITER will therefore permit exciting research opportunities for boundary physics studies in reactor relevant regimes.

FIRE. The FIRE device is a high B-field, high density tokamak, with a total fusion power of 150 MW, leading to a SOL heating power of approximately 30 MW; auxiliary heating and core radiation loss make modest adjustments to this power. The anticipated density at the top of the H-mode pedestal is 1.5 to $3.0\times10^{20}\ m^{-3}$. FIRE is planned to operate with tungsten-rod divertors and beryllium first wall, avoiding the use of carbon to avoid the tritium retention problem associated with carbon redeposition. FIRE is envisioned to operate in a double null configuration, in part as a means of minimizing the heat load on the divertors. The base case considered for FIRE has a pedestal density of $3.0\times10^{20}\ m^{-3}$, and transport diffusivities of χ=0.5 m^2/c, D=0.25 m^2/s. The peak heat flux on the outer divertor is calculated to be 16 MW/m^2 for the base case of FIRE with no impurity radiation. This heat flux can be reduced to 6 MW/m^2 with neon injection giving 0.5% impurity level at the core boundary, as planned. Helium pumping is found to be adequate for the base-case. For a DN configuration with a core-edge density of $1.5\times10^{20}\ m^{-3}$, and no impurity radiation, the peak heat flux on the outer divertor plate is 28 MW/m^2 with the base case diffusion coefficients. Both density cases show an almost inverse-linear scaling of the peak heat flux as the thermal diffusivity is varied from 0.25 to 1.0 m^2/s. This shows the importance of understanding the turbulent transport level in the boundary plasma. The peak heat flux to the inner divertor is 3.5 MW/m^2 for the base-case with spatially constant diffusion coefficients; the more realistic case of substantially reduced diffusion on the inboard side can reduce this peak flux below 2 MW/m^2. The engineering design of FIRE anticipates divertors capable of handling up to 25 MW/m^2. Calculations at the highest anticipated FIRE density yield ion densities above $10^{21}\ m^{-3}$ in the divertor region. The neutral density in the divertor region is large enough that re-absorption of the radiation is likely, a process that is not adequately included in the present UEDGE model.

The DN calculations have been done without including the effects of classical cross-field drifts. Calculations and measurements on DIII-D indicate that the up/down asymmetry of the divertor heat loads is sensitive to the magnetic up/down symmetry. This sensitivity arises from the effect of drifts, so these effects must be included in more accurate estimates of the FIRE divertor load. Balancing the up/down heat loads will likely require a feedback system for the optimum magnetic configuration. The most extreme limit of unbalanced DN operation is a single-null (SN) configuration. Calculations of the heat load for a corresponding single-null (SN) configuration indicates about a factor of 2 increase in peak heat load, as expected.

The simulations assume that particle pumping is done through the private flux region where the computational boundary has a neutral albedo of 0.98, i.e. 2% of the neutral flux to the private wall is pumped. Variations of the albedo from 0.96 to 0.99 indicate little sensitivity to this parameter. All other surfaces have unity neutral albedo and ion recycling coefficients of unity. The private-flux boundary albedo of 0.98 gives a total particle throughput rate of 4 kA for FIRE. The FIRE design criteria calls for a maximum particle removal rate of 100 Pa•m^3/s (~5 kA equivalent). Simulations done by the FIRE team indicate this particle exhaust rate will be sufficient to control the plasma density in the divertor region, and thus control the impurity radiation rate.

The DIVIMP impurity code was applied to the UEDGE solution to calculate the probability of leakage of high Z particles released from target and walls to reach the separatrix. Wall sources are likely to be well screened, as also target sources provided flow reversal does not set in close to the target surface.

IGNITOR. The IGNITOR device is designed to operate in a limiter configuration. The LCFS of the plasma core is tangent to the limiter surface at the inboard midplane and the limiter surface is shaped to nearly coincide with the LCFS. Ideally, the particle and heat exhaust would be distributed over the entire surface of the limiter, but the wetted area of the limiter may be considerably smaller due to misalignment between the wall tiles and the magnetic flux surfaces; this effect is not evaluated specifically here. Since there is very limited experience in modeling limited plasmas with UEDGE, two approximations for the limiter have been considered. The first uses a simple wedge-shaped limiter whose vertex is set at the LCFS at the inboard midplane. In addition, the plasma is surrounded by an outer wall that coincides with a magnetic flux surface in the shadow of the wedge limiter. The plasma wetted area of the wedge limiter has been varied by changing the angle of the wedge. As expected, the peak limiter heat load scales inversely with the wetted area of the limiter. The calculated limiter heat load is then extrapolated to the planned limiter surface (wetted area of 3.7 m^2). The second model for the IGNITOR limiter is to assume it is conformal with the flux surface which is tangential to the actual wall at the inner midplane. The flux to the conformal wall is then determined by specifying the poloidally dependent radial scale length of the density at the LCFS.

The innermost flux surface of the simulations is about 4 cm inside the LCFS. The core plasma density at this boundary is 2×10^{20} m^{-3} and the net power into the SOL from the core plasma is 22 MW, split equally between electron and ion channels. (This estimate is obtained by assuming a steady state plasma, and subtracting the expected bremmstrahlung radiation from the sum of the alpha and Ohmic heating powers.) The limiter and wall surfaces surrounding the plasma do not pump particles, so in steady state there is no net particle flux from the core plasma. Radial transport is simulated by assuming L-mode-like diffusivities, using 1.0 m^2/sec for particles and 2.0 m^2/sec for energy. The peak heat load calculated with this wedge limiter, without impurity radiation, extrapolates to about 3MW/m^2 on the inner wall of IGNITOR. Calculations with the conformal wall indicate a peak heat load of about 1 MW/m^2. Because the IGNITOR team states that the wall tiles can accept 10 MW/m^2 for 4 seconds, unless pronounced peaking occurs from misalignment, the heat flux should be acceptable for the base operating modes.

The IGNITOR team anticipates that the limiter peak heat load will be reduced by a radiating mantle in the plasma edge arising from intrinsic low Z impurities. The UEDGE simulation with a simple fixed impurity fraction model with 40% of the input power radiated produced a peak limiter heat load of 1 MW/m^2, extrapolated to the actual limiter geometry. This is a reduction of about a factor of 3 from that calculated without impurity radiation. The exhaust power is approximately evenly split between the limiter and the outer wall. The peak load on the outer wall is approximately

0.25 MW/m^2. A few calculations with the conformal limiter indicate some reduction, although the maximum radiated fraction may be limited because of the formation of a MARFE at the inner midplane, as seen experimentally. More quantitative assessment is needed. The maximum heat loads calculated with the UEDGE model are roughly consistent with those obtained by the IGNITOR team. The electron temperature on the limiter surface is reduced from about 25 eV to 2 eV by the impurity radiation. The electron and ion temperature on the outer SOL surface away from the limiter remains relatively high, around 50 eV. Such high temperatures might lead to significant sputtering of the Mo walls by low Z impurity ions, as is seen in Asdex-U and C-mod. Estimates of the penetration of Mo from the limiter surface, using the DIVIMP code and the UEDGE simulated plasma indicate this source of contamination could be significant.

ITER. The ITER device is designed to operate with a lower single null divertor at moderately high density and triangularity. The high triangularity means there is a second X-point which lies just outside the dominant lower X-point. Although UEDGE is capable of simulating such unbalanced double null configurations, we have not done so for the analysis described here. Rather, the simulations are restricted to a maximum normalized poloidal flux of 1.035 to avoid the second X-point. This radial range is somewhat smaller than usually used for UEDGE analysis, and should be re-visited. The ion density at the 95% flux surface is specified to be 6×10^{19} m^{-3}, which is 60% of the average density. The total power flow across this flux surface is specified to be 100 MW, and is assumed to be equally divided between the electron and ion channels. The gradient scale length of the density and temperatures at the outermost flux surface (1.035) is specified to be 5 cm, and the ion flux to this surface is assumed to recycle as neutrals. Particle throughput is controlled by specifying an albedo for the neutral flux on both private flux and the outermost walls. A neutral albedo of 1.0 is assumed for all surfaces except in the private flux region within 50 cm of the divertor plate, where the albedo is specified to be 0.98, i.e. 2% of the neutral flux to this wall is absorbed. Finally, the dominant impurity species is assumed to be carbon since the divertor plates of ITER are currently specified as carbon. The simple fixed fraction model is used for carbon impurities, and the fraction is varied to obtain a total radiated power similar to that obtained in more detailed modeling by the ITER team.

Using the radial transport rates assumed by the ITER team ($D_\perp$=0.3 m^2/s, χ=0.5 m^2/s) the peak power flux to the divertor is about 10 MW/m^2 when a carbon fraction of 3% is used, giving a total radiated power of 64% of the heating power. This power flux is comparable to that obtained by the more complete modeling of the ITER team, and lends credence to the simplified modeling reported here. (A single simulation using the multi-species impurity model for carbon sputtered from the walls was also done, with peak heat loads consistent with those obtained using the fixed fraction model with 3% carbon.) The carbon impurity is important, in that without impurity radiation, the peak heat flux is 30 MW/m^2 for the base-case. The peak heating power increases to 38 MW/m^2 if the cross field thermal diffusivity is reduced to 0.25 m^2/s (with D/χ=0.5 and a 3% carbon fraction). The divertor plates in ITER are designed for a maximum steady state heat load of 25 MW/m^2. It is anticipated that additional impurities would be introduced in the divertor region if the thermal diffusivity is so low that the peak heat load would exceed the design maximum. The sensitivity of the divertor peak heat flux to transport indicates the importance of better knowledge of the transport rates in the boundary.

The pumping assumptions used in this simplified modeling produces an ion flux across the 95% flux surface of 10 kA (equivalent atom amperes). This is roughly consistent with the pumping capability of 200 Pa m^3/s, but implies a very large fueling source for the core plasma. The large convective power flow associated with this ion flux limits the temperature at the top of the pedestal to only 1.4 keV, significantly lower than required for good H-mode confinement. The simple modeling reported here suggests more complete modeling of the particle flows must be undertaken, and benchmarked against existing experiments, to understand the fueling and pumping behavior of the ITER plasma. Recent simulations by the ITER team indicate that a combination of shallow (gas puffing) and deep (pellets) fueling can be used to achieve adequate pedestal temperatures.

DIVIMP was applied to the UEDGE solution to calculate the probability of leakage of high Z particles released from target and walls to reach the separatrix. Wall sources are likely to be well screened, as also target sources from the regions of significant production.

ELMs. Edge-Localized-Modes (ELMs) represent a serious concern for a next step burning plasma tokamak. The energy released by ELMS into the SOL and divertor can lead to unacceptable divertor erosion if the target surface temperature transiently rises above the ablation threshold at each ELM. For the ITER divertor design this threshold is expected to be an ELM energy density of $0.4 MJ/m^2$ if the deposition time is $300\,\mu sec$ onto a carbon divertor target. For a tungsten target the threshold should rise to about $0.6 MJ/m^2$. (Recent calculations which assume the temporal variation of the ELM energy deposition period is triangular with a total duration of $300\,\mu sec$, indicate the threshold energy density will be $1\,MJ/m^2$ for either carbon or tungsten divertor targets.) For natural H-mode density plasmas, without additional gas puffing, the energy lost at an ELM has been measured to be about 15-20% of the pedestal energy, the pedestal pressure times the plasma volume. More than 50% of this energy is typically observed to fall on the divertor target in an area of 1-2 times the area for the heat flux between ELMs. For ITER these ELM characteristics along with the expected pedestal scaling would lead to ELMs exceeding the ablation threshold by a factor of about 5 (a factor of 2 to 3 for the newest estimate of the ablation threshold). The ELM heat flux is expected to be less severe for a smaller more compact device as the pedestal energy has a stronger size dependence than the divertor wetted area.

A favorable scaling of ELM energy with density has been observed in present tokamaks. For pedestal densities above ~80% of the Greenwald limit ELMs are observed to become much smaller and would be tolerable for ITER at its expected pedestal parameters. However, the scaling of the reduction of ELM energy at high density is still very uncertain. If the reduction in ELM energy requires a high collisionality pedestal to reduce the edge current driving the ELM, then a burning tokamak plasma would likely not benefit because it requires a nearly collisionless pedestal for adequate central confinement. On the other hand it is possible that other processes controlling the ELM energy scale with the normalized Greenwald density. In this case ELMs may be tolerable in a burning plasma if operated near the Greenwald limit.

In general plasma shape does not appear to strongly affect the ELM energy in relation to the pedestal pressure. However, small ELMs have been observed at low collisionality and higher triangularity. Such regimes are currently an active area of research in today's devices. Recent JT–60U experiments suggest this operation regime can be reliably accessed, and projections to ITER are encouraging. Further research will be required to adequately assess the viability of these small ELM regimes to a future burning tokamak plasma.

Transport Simulations. The cross field turbulent transport rates have been examined using the 3D turbulence BOUT code. These initial BOUT simulations use UEDGE generated plasma profiles corresponding to the ITER modeling. The plasma is simulated from the 96% poloidal flux surface to the 103.5% surface. The outermost surface has been selected to avoid the second X-point at the top of plasma. The equilibrium plasma profiles are obtained by using modified hyperbolic tangent fits to the UEDGE simulated plasma density n_e, and electron and ion temperature T_e and T_i at the mid-plane. The mid-plane temperature and density on the separatrix are $T_e=245\,eV$, $T_i=444\,eV$, $n_i=2.71\times10^{19}\ m^{-3}$, respectively, while at the $\psi=96\%$, $T_e=1180\,eV$, $T_i=1024\,eV$, $n_e=5.03\times10^{19}\,m^{-3}$. For these parameters ν_e^* varies from 0.5 to 50 for electrons and ν_i^* from 0.5 to 5 for ions in the boundary region ($\psi > 96\%$). The mean free path for electron-ion collisions is less than the typical connection length ($2\pi * q_{96} * R_0 > 165$ m) outside $\psi=98\%$. The electron-ion collision length varies from 250 m at the boundary between the BOUT simulation volume and the plasma core (~4 cm inside the separatrix at the outboard midplane) to 4.17 m in the far scrape-off layer (~4 cm outside the separatrix at the outboard midplane). This high collisionality justifies the use of Braginskii fluid equations in the SOL, which is the basis of the BOUT simulation code. It is felt that the moderate collisionality found on the closed surfaces permit meaningful simulation of the boundary turbulence using BOUT in this region as well.

Although BOUT contains many sources of free energy which can drive plasma turbulence, our initial attempt for the ITER simulations has found that the dominant instability is the resistive X-point mode (a resistive ballooning mode which is strongly influenced by the magnetic geometry near the X-point). Magnetic curvature is the dominant instability drive for the resistive X-point mode. However, the drift Alfven-wave instability may also play a significant role. Particle transport perpendicular to the magnetic field Γ_r results from correlated fluctuations of the plasma drift velocity δv_E and density n, and can be calculated from $\Gamma_r = <\delta v_E \, \delta n>$. A strong poloidal asymmetry of the turbulent flux of particles is found. The plasma drift velocity $\delta v_E \sim k_\theta \sim 1/B_\theta$ diverges at the X-point null as $B_\theta \rightarrow 0$, and thus a large particle flux is produced near the X-point region. The flux surface averaged particle and heat diffusivity D and χ are about the same order, varying from 0.5 m^2/s in the core-edge region to 10 m^2/s, peaked near the separatrix, then back to 1 m^2/s in the SOL. These transport coefficients are significantly larger than typically observed in current tokamaks. The reason is still under investigation. Possibilities include the lack of consistency of magnetic geometry and plasma profiles used in the boundary region, and too small a radial electric field shear to induce the H-mode.

<u>Advantages of a BPX from the Perspective of the Boundary</u>. The first questions which arises in the mind of someone who focuses on the boundary physics of a fusion device is whether the PFCs will survive the challenging environment, and whether the core plasma performance will be compromised by the interaction of the exhaust plasma with the surrounding walls. Our analysis indicates that all three devices will survive the environment. While there is some concern about the viability of limiters in a BPX, we are optimistic about the compatibility of the boundary plasma with a high performance burning plasma. These affirmative answers permit exploration into what exciting new opportunities a BPX provides for better understanding boundary physics issues relevant to development of fusion power production.

A BPX with an H-mode edge (FIRE and ITER) will permit study of boundary physics with reactor relevant edge collisionality and pedestal densities near the Greenwald limit. This is not possible with today's devices. Size scaling means that small devices can either examine boundary physics at reactor relevant collisionality (low density, high temperature), or with pedestal densities near the Greenwald limit (high density, low temperature), but not both. Since the top of the pedestal region in a fusion reactor is expected to have parallel mean-free paths for electrons and ions much longer than the magnetic connection length and be near the density limit, the next generation BPX will permit the first study of reactor-relevant boundary plasmas.

A BPX will permit detailed study of an issue closely related to that in the previous paragraph, namely the compatibility of techniques used to control power and particle exhaust with the need to maintain high pedestal temperatures for core plasma performance in a reactor relevant plasma. The performance of an H-mode plasma is sensitive to the nature of the plasma at the top of the H-mode pedestal. On the other hand, the pedestal parameters are determined in part by the interaction between the exhaust plasma and the device walls. This core/edge coupling is an important physics study in today's devices, and it will be exciting to extend these studies to reactor relevant plasmas.

A BPX with active pumping (FIRE and ITER) will permit study of the ability to adequately exhaust helium from a burning plasma to prevent ash accumulation. Current experiments can only inject helium from either high energy beams or gas puffing, and examine the ability to pump that injected helium. A BPX will generate the helium internally, as a direct product of the fusion reaction. It will be exciting to demonstrate the efficacy of our present ideas of helium exhaust in a burning plasma, directly relevant to development of a fusion power reactor.

Finally, a BPX will permit examination of the effect of energetic alpha particles on the physics of the boundary plasma. It is currently envisioned that the alpha particles resulting from the fusion reaction will thermalize before being exhausted. If, however, they are lost before complete thermalization, they will affect the ability to adequately exhaust the power in a burning plasma. An understanding of the effect of the alpha particles will be a step forward in the development of fusion power reactors.

3.3 Technology Issues

The MFE Technology Working Group carried out a critical assessment of the technology aspects of major proposed next-step burning plasma devices with an emphasis on ITER, FIRE and IGNITOR. This effort also included an assessment of the research issues and potential benefits to technology development of the various options. The scope of the Working Group included magnets and power supplies, plasma facing components, plasma heating/current drive/fueling systems, vacuum vessel and remote handling systems, safety/tritium/materials topics, and costing. The Technology Working Group actively solicited input and comments from representatives of the three burning plasma design teams.

The Group's overall assessment is that there are no outstanding feasibility issues to prevent the successful design and fabrication of any of the three options, although they are at rather different stages of development. ITER and IGNITOR have well-developed designs. As the most recently initiated activity, the FIRE design has been funded only to the pre-conceptual design level. A comprehensive R&D program has supported ITER. FIRE has benefited from past efforts on CIT/BPX and, more recently, on ITER. Also, IGNITOR has built full-size prototypes for all key components.

The machine designs seem to be adequate to meet the different burning plasma missions of the three options. Cost information has been supplied for ITER and FIRE. The main purpose of the ITER cost information is to estimate the relative value of all ITER tasks to facilitate international negotiations while FIRE is costed as a US construction project. Finally, ITER offers the most benefits in terms of demonstrating reactor-relevant technologies and integrated operating experience.

While there appear to be no major feasibility issues for the construction of the three options and the designs are adequate to the meet the different missions of the three options, there are numerous technical issues and concerns which are described in the following sections. Perhaps the most important set of issues is concerned with plasma facing components: surface erosion due to type I ELMs and tritium retention in carbon-based materials.

3.3.1. Magnet Technology

Expected performance and operating margins: The magnets for all three burning plasma experiments should provide adequate capabilities for meeting all the requirements of the particular burning plasma experiment, but all three magnet systems necessarily have complexity beyond that of current day experiments. The basic structural concept for IGNITOR involves bucking, wedging and an active magnetic press. ITER is somewhat less complex, relying primarily on wedging of the TF coils and shear keys. FIRE has the least complex structural concept. Complexity in structural and magnetic interactions leads to stringent requirements for assembly and the potential for reduced reliability over the long term because of the need for more interactive structures to operate repeatably in a predicted fashion.

To validate the design a substantial design effort was carried out to evaluate fatigue, creep/relaxation, disruption loads, and fault conditions for all three machines. The most intensive studies were performed on the ITER magnet system, but it is fully expected that other designs are, or will be, brought to a similar level of robust system design with adequate margins.

All three designs have comparable design criteria. FIRE is within allowable stress limits with a 30% margin to be used in later design stages, for design development, for improved performance or for a more flexible operational space.

Feasibility of manufacturing: The magnet systems for all three machines do not represent feasibility problems, although the level of detail worked out in the designs are different for these magnet systems: IGNITOR having the highest level of details and demonstration of feasibility and FIRE the lowest. However, FIRE benefits from previous studies such as CIT/BPX, TPX and from involvement with ITER and IGNITOR. ITER not only fabricated the most critical components, but also tested them in operation, simulating operating conditions and also in much more severe conditions, thus testing margins experimentally. IGNITOR and ITER have a well-established on-going collaboration with industry, which addresses and resolves the manufacturing feasibility issues. FIRE also addressed several feasibility issues for the magnet system through an industrial subcontract.

The full scale or representative scale prototypes fabricated for each machine are listed in the table below:

IGNITOR	FIRE	ITER FEAT
Full size CS coil segment	Full scale water-jet cut plate	CS model coil
Full size TF coil fabrication test	Full size BPX BeCu Large Plate	TF model Coil
Full size C-Clamp forging and machining		Case Corner forging - welding, distortion, crack size exercise
Full size Mechanical Jacks		

Readiness for manufacturing, e.g., need for further R&D: The identified and accomplished level of R&D for the machines was quite different. The superconducting magnet system for ITER had the highest effort due to the more complex design and higher risk factors in comparison with resistive magnet systems. The IGNITOR readiness is the highest, having completed the essential R&D, fabricated full-scale prototypes of critical elements, and produced fabrication drawings developed in collaboration with industry.

FIRE is still in an advanced preconceptual design phase, but has identified needed R&D work including: conductor joining, materials characterization and radiation resistant insulation development. FIRE benefits from previous manufacturing development of large BeCu plates

ITER has created the final design drawings, but not fabrication drawings. Most critical elements of the magnet system have gone through cost effectiveness studies in close collaboration with industry. ITER is preparing procurement packages for key elements in collaboration with industry, bringing in industrial capabilities and defining project needs for planning purposes. The ITER R&D work mostly addresses optimization issues, rather than feasibility, but additional R&D will be required to address issues raised by the design change of the CS to a free-standing solenoid. This includes issues related to temperature margin in the superconductor, pancake windings with joints in tension, and fatigue limits due to tension in the CS conduit.

Schedule for construction: The IGNITOR construction schedule is 5 years. The FIRE construction schedule is 6 years after first contracts award, but including R&D and conceptual design work it is 8 years from the contract award, and could be accelerated from the assumed funding limited schedule with additional funding. The magnet system is on the critical path. The ITER construction schedule shows 9 years from the contract award to first plasma. The magnet system is the long lead item and also is the critical path. The production schedule for the most critical element – superconducting strand/conductor - is 6 years due to limited world production capacity.

Risk assessment of cost, schedule, and performance: No systematic data is available for the cost risk assessment on any of the machines. The magnet systems of all machines are on the critical path and therefore represent a high-risk schedule item. ITER has analyzed performance risks in the most detail. One of the ITER fault analyses with overcurrent in the TF coil requires more detailed work, since the consequences were calculated to be unacceptable. ITER requires highly reliable protection system elements – circuit breakers and quench detection units which must have sufficient redundancy and contingency. FIRE identified some faults and has initiated studies. Some IGNITOR fault analyses are complete and others are planned.

Site/Power Issues: The peak power requirements for FIRE and ITER are comparable, but ITER will require the power to be taken from the grid, while FIRE and IGNITOR can use motor-generator storage devices. Several appropriate locations in the United States have been identified for FIRE. IGNITOR has identified an appropriate site in Italy. ITER has received four international site proposals. The site requirements for ITER are significantly more demanding than those for FIRE or IGNITOR. Some of the parts of ITER must be fabricated on site due to size, weight, and transportation issues. Therefore some buildings on site will have a multipurpose use during construction.

Relevance to a DEMO: ITER has the highest overall relevance to a DEMO reactor. FIRE has a higher radiation effects relevance due to much higher nuclear flux and fluence at the magnets, closer to what is expected in a DEMO. ITER is the only device of the three that uses superconducting magnets and offers superconducting magnet system operating experience integrated with other fusion and plasma technologies. DEMO may or may not look like ARIES-AT, but it is certain that superconductor technology will evolve during the long period of ITER construction and operation. As with all technological components the future DEMO will be based on the technology available at the time it is designed and based on previous experience. Thus the superconductor materials and conductor technology will likely be different in the future.

The table below gives a DEMO magnet relevance assessment for the three machine.

	IGNITOR	FIRE	ITER FEAT
Magnet Mechanical Design	Low	Low	Highest relevance assuming DEMO will follow ARIES-AT design which uses large super-conducting TF magnets in a plate structure.
Conductor Type	Low	Low	Medium-High- superconductor development will proceed parallel to ITER construc-tion, and ITER conductor will be obsolete by the time it is built. The technology will evolve – e.g. ARIES-AT design assumes YBCO-coated material in plates topology.
Relevance to DEMO Magnet, radiation effects	Medium	High - >10^{10} rads in TF insulation provides life test for DEMO insulations	Medium - Steady state removal of nuclear heat in a superconducting coil will be demonstrated.

Relevance to other fusion experiments. All three machines have some relevance to planned or on-going experiments. The copper machines disengage superconducting magnet technology and R&D from burning plasma development. The ITER magnet system provides the most direct path for the development of fusion magnet technology.

3.3.2 Plasma Facing Components/Heat Removal

ITER is the most mature of the three designs. The design work for PFC's has been coupled with an extensive R&D program where PFC mock-ups have been tested to prototypic heat flux conditions. Many state-of-the-art computer codes have been used to analyze the predicted response of PFC components under both normal and off-normal conditions. In addition, ITER-specific criteria, SDC-IC, have been developed jointly by the Home Teams for structural design of in-vessel components. In general, the design of the ITER PFC's meets the operating goals, and their design has been well integrated into the entire ITER device. The ITER device is the most DEMO relevant with long burn times and highest neutron fluence of the three devices. Because of the severe conditions expected for PFC components as well as the uncertainties in predicting those conditions, there still are a number of issues and concerns.

- Tritium inventory and co-deposition. Only ITER makes use of carbon in the PFC design. Carbon has been shown to retain high levels of tritium, which can lead to high levels in the machine after a relatively short time.

- High sputtering erosion. For the expected divertor conditions, low-Z materials like carbon are predicted to have high sputtering erosion rates, which could result in reduced lifetimes for the divertor plates.
- Surface losses due to Type I ELM's. Because of the potentially high frequency of ELM's during the plasma burn, the lifetime of PFC surfaces could be unacceptably short. Type I Elm's should be either eliminated or the energy deposition should be below a threshold value where no erosion occurs.
- Long term reliability of bonds at material interfaces. PFC's will experience high thermal stresses from the high heat loads. Often the highest stresses occur at bonds between dissimilar materials. Failure of the bond would result in a hot spot or actual loss of the surface material.
- Scale-up of technology to large-scale components. Fabrication and operation of actively cooled PFC's is relatively new for existing devices, and scale-up to ITER sizes and conditions requires a significant level R&D.
- Long term performance of copper alloys. Copper alloys are sensitive to irradiation and will operate at temperatures where thermal creep is a concern.

FIRE has performed design analyses but does not have a companion R&D program. Much of the work for the FIRE PFC's has been based upon the work done for ITER, and several of the same modeling codes used for ITER have been used for FIRE. Therefore, many of the same conclusions apply to FIRE. Since FIRE does not use carbon, the tritium inventory and sputtering erosion issues are reduced. The level of detailed analysis for FIRE PFC's is not generally as extensive as for the ITER PFC's. Issues and concerns for FIRE PFC's are:

- Erosion and tritium co-deposition performance of mixed beryllium/tungsten divertor surface, such surface resulting from mixing of wall-sputtered Be transported to the divertor. Role of oxygen in T/Be trapping.
- Surface losses due to Type I ELM's. Because of the potentially high frequency of ELM's during the plasma burn, the lifetime of PFC surfaces could be unacceptably short. Type I Elm's should be either eliminated or the energy deposition should be below a threshold value where no erosion is predicted.
- Long-term reliability of bonds at material interfaces. PFC's will experience high thermal stresses from the high heat loads. Often the highest stresses occur at bonds between dissimilar materials. Failure of the bond would result in a hot spot or actual loss of the surface material.
- Scale-up of technology to large-scale components. Fabrication and operation of actively cooled PFC's is relatively new for existing devices, and scale-up to ITER sizes and conditions requires a significant level R&D to assure high reliability.
- Long term performance of copper alloys. Copper alloys are sensitive to irradiation and will operate at temperatures where thermal creep is a concern.

The first wall in IGNITOR is the plasma facing system. It does not have a physical divertor, although it is possible to configure the plasma to a double X-point configuration. In general, the PFC design meets the goals of the machine, but a number of issues have been identified. The current analyses and design effort are focused on the optimization of the tile detailed design. In general, IGNITOR is the least DEMO relevant from the PFC viewpoint due to the absence of a divertor component, and the absence of DEMO relevant plasma facing materials such as tungsten and beryllium. The short pulse times also limit the capability of the device to adequately test actively cooled PFC test modules.

- The ability to sustain a radiating mantle at the edge without hot spots is an issue. This is more of a plasma physics issue, but the results are important for the first wall. In high-density plasmas the low value at the edge temperature and high neutral density reduces sputtering from the wall and the impurities are effectively hindered from entering the core plasma. Any burn through of the mantle or non-uniformity of mantle radiation (MARFES) will result in a hot spot and possibly unacceptable temperatures on the first wall.
- Alignment of first wall tiles. A limiter may be susceptible to hot spots due to misalignment due to the very shallow angle of the magnetic field lines. Most of the power is radiated, which helps alleviate alignment issues. Initial analysis indicates the tiles need to be aligned to a fraction of a

millimeter, and the capability to perform and maintain an alignment below 1 mm is considered questionable. Some experience that confirms the feasibility of the proper alignment relying on remote handling installation has already been acquired on the FTU machine. During the Snowmass Meeting, we learned that the FTU alignment process did not eliminate hot spots and local melting is occurring. It was agreed that R&D was needed to assess the optimum technique for magnetic measurements needed to align the limiter in IGNITOR.

- Response to off-normal conditions. Disruption erosion is a significant issue for ITER and FIRE, and the same is expected for IGNITOR. IGNITOR is not expected to have the same difficulties with ELMs under the proposed operating conditions (operating in L-mode or H-mode with no ELM's).

The overall assessment of the PFC area is given below. The rank in each area is shown in italics, and key issues are identified below the rank.

Criteria	**ITER**	**FIRE**	**IGNITOR**
Meet Performance Requirements	*Issues – being addressed* - Tritium inventory - Carbon sputtering erosion - ELM erosion	*Issues – being addressed* - ELM erosion - Mixed beryllium/tungsten co-deposition	*Issues - being addressed* -R&D needed to confirm alignment tolerances
Margins and Adequate flexibility	*Issues – being addressed* - Peak heat load close to maximum allowable. - In-situ repair is possible	*Issues – being addressed* - Peak heat load close to maximum allowable. - In-situ repair is possible	*Issues - being addressed* -Heat flux peaking factor too low
Feasible fabrication	*Issues – being addressed* - Fabrication technology scale-up - Bond integrity	*Issues – being addressed* - Fabrication technology scale-up - Bond integrity	*Issues - being addressed* - Demonstration of alignment in 1/R gradient field
Issues and R&D needs identified	*Mature*	*Mature*	*Issues - being addressed* - Additional issues raised by assessment team
Credible R&D Plan	*Mature*	*Mature*	*Issues- being addressed* - Revisions suggested
Credible cost estimate	*Mature*	*Mature*	*No cost data provided*
On path to DEMO	*Mature*	*Issues- being addressed* - No actively cooled FW	*Issues –being addressed.* - Mo does not extrapolate to DEMO (activation) - No actively cooled PFCs - Short pulse length
Relevance to other fusion devices	*Mature*	*Mature*	- No divertor
Adequate reliability and maintainability	*Issues – being addressed*	*Issues – being addressed*	*Issues – being addressed* - Failure to achieve tile alignment goals could reduce pulse time.
Interfaces identified	*Mature*	*Issues – being addressed*	*Issues – being addressed*
Maturity of design	*Mature*	*Mature* – divertor technology same as ITER - FW at conceptual design level	*Issues – being addressed*

3.3.3 Heating, Current Drive and Fueling

Summary. The Heating, Current Drive and Fueling Subgroup has assessed Heating/Current Drive/Fueling systems for candidate burning plasma devices to determine their capabilities with regard to meeting the physics objectives, operating performance and flexibility, and ability to test advanced Heating/Current Drive/Fueling systems under reactor relevant conditions. For all three burning plasma experiments, heating and current drive capabilities should be achievable while meeting all safety and reliability goals. Fueling and pumping requirements require pushing beyond the state of the art. The necessary R&D to develop the Heating/Current Drive/Fueling systems has been identified and can be completed in a time frame consistent with construction schedules for the burning plasma devices.

Neutral Bean Injector System. ITER will employ negative-ion NBI at 1-MeV and development programs are underway (now mainly outside the U.S.) to provide steady-state systems to meet these requirements. The primary requirement of a development program is a full-scale independent R&D test stand to establish reliable 1 MeV operation. For reactor applications, highly reliable, steady state systems will be required. NBI is a possible upgrade to FIRE. Access considerations preclude the use of NBI in IGNITOR.

ICRF System. ICRF technology is well developed and should be considered a relatively low risk for implementation on a burning plasma experiment. For ICRF, a combination of development of independent test stands and deployment in burning plasma experiments is needed. By providing multi-MW component testing in a high-heat-flux and neutron environment, IGNITOR, FIRE and ITER would all contribute very significantly to the development and qualification of ICRF concepts for reactor application. For ITER, heating and current drive are both planned (40 MW, 40-55 MHz). Heating is planned for FIRE (20 MW, 80-120 MHz) and Ignitor (5 MW, 100-140 MHz).

LHCD System. LHCD is very attractive for current drive and could be critical to advanced tokamak operation scenarios. To develop LHH for burning plasma applications, a combination of independent test stands and deployment in burning plasma experiments would be needed. Since the frequency for LHCD is at least 5 GHz, where 1 MW klystrons are not available, klystron development would be necessary, but this is considered low risk. Although none of the three machines has selected LHH as a necessary component of its heating and current drive system, any of them could be used to test the basic heating concept in a reactor-level plasma. Adequate port space must be made available to keep launcher power densities at reasonable levels.

ECH/ECCD System. ECH / ECCD systems are very attractive for burning plasma experiments because they take up a small space and can be implemented with a very small impact on the nuclear environment. ECH / ECCD (20 MW, 170 GHz) is in the baseline design for ITER and development efforts worldwide are very promising to guarantee availability. The high frequency gyrotrons needed for FIRE or Ignitor would require development and are not included in the baseline designs.

Fueling and Pumping System. The ITER fueling and pumping system designs are very well advanced. The fueling system design, in particular the pellet injection system is beyond the current state of the art and will require a significant technology development effort. The steady-state extrusion requirements are well beyond what has been demonstrated to date. The requirements on the pellet guide tube system are questionable. For instance the inner wall guide tube diameter is rather large at 5cm diameter, while the largest pellet size envisioned is 1cm diameter. A study of injection through this size curved guide tube is needed to ascertain if this is a reasonable choice. The FIRE fueling system requirements are well developed and seem reasonable to meet the performance objectives of the machine. The Ignitor fueling and pumping system are not as well defined at those for ITER and FIRE. The anticipated two-stage light gas gun pellet injector will need some development effort to bring that technology to a reliable state. The lack of any defined pumping system for Ignitor makes it difficult to determine whether the machine will be able to control density and pump helium resulting from DT fusion reactions.

Fueling and Pumping System Requirements	**ITER**	**FIRE**	**Ignitor**
Fuel Isotope	Pellet (90%)	Gas (1-5%T) Pellet (40-99%T)	Unknown
Gas Fueling Rate (Torr l/s)	600	200 for 20s	Unknown
Pellet Fueling Rate	600	200 for 20s	Unknown
Pumping volume (m^3)	~1000	35	~15
He pumping speed (l/s)	60,000	3200	Unknown
Torus pumping rate (Torr l/s)	1500	200	Unknown

Summary Assessment	**ITER**	**FIRE**	**IGNITOR**
Meet Performance Requirements	*Mature* -NBI system needs further development - Need LH power increase from 20 to 30 MW?	*Mature* -LH power needs to increase from 20 to 30 MW?	*Issues –being addressed* -Pumping system needs definition
Margins and Adequate flexibility	*Mature* - Possible system upgrades defined.	*Mature* - Possible system upgrades defined.	*Mature* - System upgrades may be needed.
Feasible fabrication	*Mature*	*Mature*	*Mature*
Issues and R&D needs identified	*Issues* -Need dedicated R&D test stand for heating system. -Need pellet fueling development to provide high fueling rate	*Issues* - Dedicated R&D test stands needed for heating systems.	*Issues - being addressed* - Pellet injector needs development. - ICRF power level needs be determined
Credible R&D Plan	*Mature*	*Mature*	*Issues – being addressed*
Credible cost estimate	*Issues – being addressed* - ICRF system cost is low.	*Mature*	*Issues – being addressed* - Need more detailed cost information.
On path to DEMO	*Mature*	*Mature*	*Mature*
Relevance to other fusion devices	*Mature*	*Mature*	*Mature*
Adequate reliability and maintainability	*Mature*	*Mature*	*Issues – being addressed* - Need more design info.
Interfaces identified	*Mature*	*Mature*	*Mature*
Maturity of design	*Mature*	*Mature*	*Issues – being addressed*

3.3.4 Vacuum Vessel and Remote Handling

The ITER, FIRE and Ignitor designs represent different levels of performance and cost as well as design maturity. Although the vacuum vessel configurations are somewhat different, all provide the necessary plasma vacuum and safety barrier functions. The ITER vessel is the largest, by far, of the three vessels, measuring almost 20m in diameter at the outer extent of the torus and weighing almost 5600 tonnes. The FIRE and Ignitor vessels are smaller, weighing just 165 and 16 tonnes respectively.

The ITER vessel could be considered prototypical of a DEMO device, while the FIRE and Ignitor designs have much less direct relevance. The vacuum vessel for any burning plasma experiment must take a significant step towards reactor relevance due to the high activation levels and tritium inventories connected with such an experiment. The vessel is the primary safety barrier, and so must be designed and built in strict accordance with a recognized design code, such as the ASME Boiler and Pressure Vessel Code. All of the vessels must withstand the severe loads associated with a plasma disruption, in combination with the normal operating loads.

Remote Handling. In-vessel components of ITER, Ignitor and FIRE must be remotely maintained due to neutron activation. The remote maintenance approach of these machines is similar although the details of the handling equipment, machine components and the level of design maturity varies considerably. Fusion remote handling requirements and applications are very specialized and all machines require significant R&D to develop and demonstrate feasible design solutions.

ITER is by far the most advanced in terms of design and has completed some major remote maintenance R&D projects (e.g., L6, L7 projects). FIRE has followed closely the approach of ITER, leveraging its experiences and developments. Ignitor remote handling costs are not reported, but the design approach appears to follow that of other machines. ITER and FIRE design and R&D should be achievable within their respective time and cost projections.

ITER is the most representative of the remote handling requirements of DEMO, with activation levels, and component designs and sizes more typical of a power reactor, and would substantially advance the technology. FIRE and Ignitor would also provide invaluable experience in fusion specific remote handling and may present less technical risks given their lower activation levels and less demanding requirements. Ignitor must perform assembly operations remotely as well due to the small port size

The overall assessment of the vacuum vessels for each machine is given the Table below.

Vacuum Vessel Assessment	**ITER**	**FIRE**	**Ignitor**
Key issues and R&D identified	yes	yes	yes
Maturity of design	Preliminary to detailed	Advanced pre-conceptual	Preliminary to detailed
Expected Performance	Meets by analysis and R&D	Meets by analysis to date, needs R&D	Meets by analysis to date and R&D
Operating margins and flexibility	Margin for all load categories	More analysis required, new size	More analysis required for revised loads
Feasibility of manufacturing	Full scale prototype completed	Bonding of Cu plates to shell	Full scale prototype completed
Need for further R&D	Very little	Cu bonding, remote welding, prototype	Remote welding
Credibility of capital and operating costs	Credible (1)	Preliminary (2)	Not available
Relevance to demo	Very high relevance (size, integral shielding, config.)	Modest relevance (integral shielding)	Modest relevance
Relevance to other fusion experiments and applications	Addresses all design and regulatory issues for fusion safety boundary	Addresses limited set of issues	Addresses issues with limited tritium inventory and activation levels
Reliability, off-normal conditions considered	Yes, all analyzed in detail	Identified, but not all are analyzed	Identified, disruption analysis in detail
Interfaces identified and addressed	Yes, in detail	Yes	Yes, in detail

Notes:

1) ITER cost estimate may not include design or R&D costs
2) FIRE cost estimate must be updated for modified dimensions

Remote Handling Assessment	ITER	FIRE	Ignitor
Key issues and R&D identified	Yes, high level of definition	Yes, with limited assessment	Yes, some R&D identified
Maturity of design	Preliminary to detailed	Advanced pre-conceptual	Pre-conceptual to detailed
Expected Performance	Meets by analysis and R&D to date	Meets by limited analysis to date, needs R&D	Meets by limited analysis to date, needs R&D
Operating margins and flexibility	Meets by analysis and R&D to date	Meets, limited analysis to date, needs R&D	Needs further definition
Feasibility of manufacturing	Several prototypes built, significant manufacturing issues not expected	No prototypes to date, no significant manufact-uring issues expected	Reported as no manufacturing issues expected
Need for further R&D	Some major R&D completed, more required	No R&D completed to date, some ITER R&D relevant	Some welding experience - R&D to be performed in addition to ITER R&D
Credibility of capital and operating costs	Credible (1)	Credible for level of design	Undefined
Relevance to demo	Highest relevance (blanket and divertor modules)	Good relevance (divertor modules, FW tiles)	Moderate relevance (FW tiles only)
Relevance to other fusion experiments & applications	High relevance to other activated, DT machines	High relevance to other activated, DT machines	Reported high relevance to other activated, DT machines
Reliability / off-normal conditions considered	Yes, in good detail	Identified with limited assessment	Needs further definition
Interfaces identified and addressed	Yes, in detail	Yes, many details need further development	In-Vessel only, details need further development

Note: (1) ITER cost estimate may not include all elements of cost, e.g., design or R&D costs to date.
(2) Ignitor in-vessel system based on existing FTU articulated boom system.

3.3.5. Safety/Tritium/Materials

Safety and Environment. Several safety and environmental (S&E) issues have been considered, including the safety approach, safety design guidelines adopted by each design, the ability to obtain a generalized regulatory approval, and demonstration of the S&E potential of fusion.

Each of the three designs has documented that public, worker, and environmental safety is a recognized part of the design process. Each design has implemented a safety approach that minimizes the environmental impact during normal operation. Each design has a goal to preclude the need for a public evacuation plan in the event of an accident, and each design has also addressed the issue of radioactive waste generation and minimization. All three of the stated safety approaches would be found generally acceptable to regulators.

The safety assessments for each of the three designs are generally commensurate with the level of detail in the design and each has shown that it can meet the regulatory requirements by using safety design guidelines and safety provisions. These provisions are based on a graded approach of adding additional passive barriers as inventories increase, reducing inventories where possible, and using engineered safety systems for active mitigation of radiological or toxicological releases. Each machine has established safety design goals consistent with the regulatory body's requirements, so a generalized regulatory approval could be obtained for any of the three designs.

Regarding demonstration of the S&E potential of fusion, FIRE is judged to be medium, since the FIRE design has few inventories relevant to a power plant and the machine design (HHF divertor, etc.) presents only a few safety challenges. IGNITOR is judged to be low because the design does not pose any significant safety challenges and inventories are not relevant to a fusion power reactor. ITER is ranked high because of the power plant relevant hazards present and the implementation of safety limits to mitigate those hazards, the high degree of safety integration in the design at the system and subsystem level, the depth and rigor of the safety analysis, and the detailed safety R&D that has answered key safety questions. Of the three candidate designs, ITER poses many, if not most, of the safety concerns associated with a fusion power plant. Thus, the regulatory approval of the safety approach used by ITER, concurrence and validation of the safety analysis, and safe operation of the facility would in large part demonstrate the S&E potential of fusion power plants.

Tritium Systems. The three burning plasma options have very different tritium plant requirements.

IGNITOR and FIRE are within the present knowledge base as exemplified by TSTA. However, ITER is a substantial extension beyond the present knowledge base. Compared to the other burning plasma options, the tritium processing for ITER will have to handle a larger amount of tritium, will have to process it more rapidly, and will have to perform with higher reliability. Comparing ITER to TSTA, the ITER tritium systems processing rate is an extension of about an order of magnitude, the tritium inventory is an extension of about an order of magnitude, and the tritium cycle time (the time available for processing) is about an order of magnitude shorter. Additionally ITER will require a high throughput tritiated water processing system, which has yet to be demonstrated on an integrated fusion fuel processing system.

IGNITOR and FIRE do not consume large quantities of tritium. However, ITER will consume large quantities of tritium, and it will not breed tritium. It is estimated that ITER may consume approximately half of the world tritium available for fusion research. Especially, if coupled with other experiments which consume significant quantities of tritium such as a volumetric neutron source, attention must be given to conserving available tritium. This issues is all the more important given that sufficient tritium must remain for DEMO startup.

For the tritium systems the following conclusions can be drawn:

- The IGNITOR and FIRE tritium systems requires are within the bounds of present experience set by experiments such as TSTA, while ITER is beyond present experience.
  - There is high confidence in successful operation of the IGNITOR and FIRE tritium systems.
  - There is greater technical risk associated with operating the ITER tritium systems.
- The tritium systems for ITER and DEMO will be similar, while the systems for IGNITOR and FIRE will be much less demanding.
  - The greatest value can be drawn from the ITER tritium systems
  - While, the IGNITOR and FIRE tritium systems are not at DEMO levels, there have been only two tritium systems attached to fusion machines, so significant value can be drawn from these systems.
- Tritiated water processing at ITER/DEMO conditions has not been tested.
- The design team is addressing the R&D for ITER systems.
- For ITER care must be given to wise use of the world tritium available to fusion research.

Materials. For all three of the burning plasma devices, it is essential that the chosen structural materials have performance windows that are sufficiently wide to allow each machine to meet its operating performance goals and also have the capacity to withstand off-normal events. In broad terms, materials performance is determined by the effects of temperature and neutron flux on both the mechanical properties (strength, ductility, fracture toughness, fatigue resistance) and on physical properties (moduli, thermal and electrical conductivity). There is an excellent data base for many of the materials selected for all three machines; in some cases the database includes information on full-scale commercial heats (e.g., for stainless steels). However, commercial–scale fabrication methods coupled with welding and joining methods which result in the development of the required physical and mechanical properties in the full-sized component, have not been established in all cases.

The output of the ITER materials R&D program is summarized in the Design Description Report, the Materials Assessment Report, the Materials Properties Handbook, and the Interim Structural Design Criteria Document. Together this body of data represents the most complete set of information available on the behavior of structural materials for near-term devices and is sufficient to provide a basis for detailed engineering design and for obtaining regulatory approval. This database provides a mature benchmark for the evaluation of materials selections for FIRE and IGNITOR.

For ITER, the fluxes of high–energy neutrons and the projected component lifetimes are of sufficient magnitude that radiation damage significantly affects the mechanical behavior of all the structural materials. Neutron doses in the first wall and divertor range from 0.1 up to 3.0 dpa. Because of the shorter pulse lengths and smaller number of full power pulses, the accumulated neutron doses are more than an order of magnitude lower in FIRE than in ITER and for IGNITOR neutron exposures are lower by a further order-of–magnitude. Consequently, although radiation effects are very significant for ITER, there are relatively few issues related to neutron damage for the other machines.

Full-scale component fabrication methods introduce thermal histories that differ significantly from the thermal-mechanical cycles used to prepare materials at the developmental stage. For both ITER and FIRE more data are required on the properties of materials, particularly the copper heat sink material CuCrZr, that have undergone full scale thermal –mechanical treatments. For ITER, the radiation response of such materials needs to be documented. This is not an issue for IGNITOR. A second area requiring further R&D for both ITER and FIRE are the methods for bonding the plasma facing materials to the copper heat sink and the joining of the heat sink to the stainless steel structure. Additionally there is relatively little known about the integrity of multi-layered bonded structures subjected to repeated thermal cycles coupled with radiation damage. Again, this situation does not apply to IGNITOR. Further data are required for ITER on the effects of low levels of neutron exposure on the fatigue life and fracture behavior of both Be and W and on the thermal conductivity of CFCs. The FIRE magnet utilizes a high strength CuNiBe alloy (C17510) which was only partially covered under ITER and recommendations for further R&D include fracture toughness, fatigue and fatigue crack growth measurements and investigating the possible impact of radiation hardening at below room temperature. For IGNITOR the main materials issue is the selection of Inconel 625 as the vacuum vessel material since it has been shown that the stresses developed during current quench will significantly exceed the yield strength.

3.3.6 Cost

Summary. A cost assessment has been accomplished for the burning plasma devices based upon the cost information provided by ITER and FIRE. IGNITOR chose not to publicly provide privileged cost information. The data has been normalized to current US dollars and a common cost breakdown structure. The cost data for ITER and FIRE are compared in the common format. Cost concerns were discussed with the project representatives and remaining issues are highlighted. It has not been possible to accomplish an in-depth cost analysis at this time due to available resources and differing project purposes for their costing efforts as discussed below.

The cost to construct ITER is approximately \$5B (2002\$), including direct capital cost and indirect capital cost, including R&D during construction. Site and site improvements are not included. ITER developed its cost information for the purpose of providing accurate industrial estimates from all parties to determine the relative value of 80+ procurement packages. This cost data is necessary to facilitate international negotiations on task sharing. The cost information is based upon a large engineering effort (~ 1000 PPY) and a large R&D effort (~ \$900M) to reduce the project technical and cost risk. The provided ITER cost information does not have explicit contingencies. In the future, actual cost estimates may be developed by each party using its own procedures with appropriate contingencies. The US will need to carefully estimate the cost of any potential contributions to the ITER project and additional R&D, if needed. These estimates should include an adequate cost contingency to mitigate potential cost increases.

The FIRE cost estimate was defined using ground rules consistent with U.S. construction. It is based on an advanced pre-conceptual design using in-house and vendor estimates. The estimate also draws upon construction estimates from previous similar fusion experimental devices (CIT and BPX). Due to its pre-conceptual design status, additional R&D and design definition effort is required to reduce the project risk. The total capital cost estimate to design and construct FIRE is approximately $1.2B (2002$), which includes a 25% contingency.

Confidence of the cost estimates is a great concern to consider before embarking on a large construction project. This concern arises from prior experiences with cost escalation during construction of other projects. Confidence of the cost estimate can be enhanced if there are firm design requirements, detailed design definition, substantial supporting R&D to reduce risk, and significant involvement of industrial partners who will be involved in the fabrication and construction. Further, project costs during construction can be controlled with requirements and design change control, effective project construction management and cost control, and continuity of project management from design through construction.

<u>Cost Basis</u>. The ITER costs are based upon a process of using detailed design bases submitted to the international home teams for subsystem quotations. These quotation packages were compared to each other and to an estimate prepared by the Central Team to define a uniform and substantiated cost basis (e.g., "value") for all the ITER systems and facilities. The ITER estimate was escalated to current 2002 U.S. dollars by using GDP Price Level Deflator indices. ITER provided a conversion factor to project its estimates from the kIUA (thousands of ITER Units of Account in 1989 currency) basis to 2000$. ITER used US labor rates.

FIRE compiled a detailed project cost estimate on June 19, 2002 based upon detailed vendor and in-house quotes. The FIRE Team updated its cost estimate on July 1, 2002 to reflect the 2.14-m major radius machine in 2002$ cost basis.

A common Cost Breakdown Structure (CBS) was defined to compare both devices. Escalated project estimates were translated into the CBS to compare systems, subsystems, facilities, and utilities on a uniform basis. No site or site improvements were considered.

<u>Detailed Cost Assessment</u>. Cost comparisons below level 2 may be misleading as the two projects have accounted the costs using different approaches and placed costs in divergent categories. Some conclusions are noted for the following subsystems:

- In-Vessel Components – The ITER cost is $348M and the FIRE cost is $82M, both of which seem reasonable. The FIRE first wall estimate is $20.68M, reflecting simple, passively-cooled, Be-coated copper tiles. Costs for ITER shielding port modules seem to be missing.
- Vacuum Vessel and In-Vessel Structures – At the top level, the costs in this account look appropriate. ITER is $330M and FIRE is $53.6 M. There is some question if the ITER estimate includes shielding, heating and cooling, support structures, and local I&C.
- Toroidal Magnets and Structures – The ITER practice of collecting all the superconducting conductors into a single account complicated the direct comparison of TF and PF magnet systems. The 1995 ITER Interim Design Review provided a subdivision of the conductors into TF, PF, and CS systems, which was used to allocate the present conductor and feeder estimates. The total costs for the ITER Nb_3Sn superconducting and the normally conducting FIRE TF systems look appropriate. The ITER cases, supports, and coil assembly are higher than FIRE, however the ITER estimate contains the machine support structure.
- Poloidal Field Magnets and Structures - The same scheme for allocating the conductors and feeders apply to this system. The overall PF costs have a ratio of 9.3/1 for ITER to FIRE. This compares to a stored energy ratio of around 9/1, so the costs are probably in the right proportion.
- Cryostat - The ITER cryostat shell and structure is $108.8M compared to FIRE at $2.14M.

FIRE justifies their estimate, as the volume ratio is 10:1. The remaining cost difference is due to ITER being a helium-cooled cryostat with an atmospheric pressure differential while FIRE is nitrogen-cooled with a small pressure-differential. ITER uses a thermal shield ($41.4M); FIRE does not require one.

- Machine Support Structure – ITER does not explicitly identify the cost of the machine support structure, which is in the magnet costs. FIRE includes about $11M for support structure.
- Machine Assembly – ITER has about $72.2M for assembly operations, while FIRE has allocated $6.68M for Torus Assembly. FIRE allocated $1.95M for Ancillary Systems Assembly and Installation, but ITER has not identified any cost specifically for this effort. The FIRE tooling is much less because the overall weight is a factor of 1/10, smaller power core components, and self-aligning first wall components ($60.9M for ITER vs. $2.39M for FIRE).
- Fueling System – These cost accounts are clearly identified in both ITER and FIRE with well-defined costs with functional differences.
- Vacuum Pumping System – These system cost accounts are clearly identified in both ITER and FIRE with well-defined costs. They seem to be in the right proportion (2.3/1).
- Fuel Processing and Handling System – The ITER costs seem somewhat high ($103M), but they are clearly defined. FIRE seems low at $4.6M, however FIRE has much less fuel processing demands than ITER. Also FIRE plans to use the TFTR tritium processing systems. FIRE also did not presently include atmospheric or water detritiation costs.
- ICRF Heating and Current Drive – The cost of these systems is highly dependent on the power delivered, frequency range, duty cycle, and required shielding. The RF heating and current drive system costs have been examined a good deal in this assessment and the adjusted costs now seem reasonable.
- ECRF Heating and Current Drive – The ITER cost of $111.4M was assessed as being reasonable at $5.57/W. FIRE is planning ECRF as a future upgrade.
- Neutral Beam Heating and Current Drive – ITER's NB system is estimated to cost $137.9M for 16.5 MW of power, which is $8.37/W. FIRE does not plan to employ neutral beams.
- Device Instrumentation and Diagnostics - ITER well defines its instrumentation and diagnostics subsystems, which are estimated to cost $214.3M, including deferred costs. ITER assumes the national parties will supply the exo-power core diagnostics that are not included. FIRE's estimate is $25.4M, which includes only Phase I diagnostics.
- Central Instrumentation and Control – The ratio of the system costs seem appropriate, but ITER only provided a rough estimate at the highest level. FIRE provided a detailed breakdown.
- Power Supplies (Magnets Only) – Both ITER and FIRE provided a separate major cost account for the magnet power supplies. The RF systems and all other systems contained their own power supplies. ITER accounted all the accounts by type of hardware, whereas FIRE designated the power supplies by the system served. With the new 2.14 m FIRE estimate, the system-level costs seem much more in line with the power handling capabilities.
- Remote Handling – Remote operations on very large, heavy, and radioactive components represent a significant advancement in the state of the art. The FIRE project recently separated remote handling R&D costs from the remote handling capital costs that now look reasonable compared to ITER. The ITER costs for port remote handling and transporters may be missing.
- Buildings and Structures – The ITER safety-related (concrete) building costs are substantially higher than FIRE ($447M vs. $72M), because ITER uses two leak-tight concrete vaults, 2-m thick shielding roof, and the safety-rated buildings are earthquake rated. The costs for non-safety buildings for both projects appear appropriate.
- Support and Facility Systems – The FIRE water cooling system seems low at $7.26M compared to ITER at $198M, but the FIRE rejected power is 53 times lower. The ITER waste handling and treatment system is low at $11.8M for the larger volume of building and water volumes (the FIRE estimate is $10.55M.) ITER had no provision for facility utility systems. The site electrical power costs are reasonable, except ITER did not identify an emergency power supply and uninterruptible power supply. Plant safety systems (fire, chemical, and radiological) were not addressed by either device, except ITER did provide $5.7M for radiological protection.
- Project Support and Oversight – This is the overhead or indirect cost account that manages the

project and construction, provides systems engineering, supplies environmental and safety oversight, and provides scientific support during the construction period. ITER estimated this by projecting professional and support staffing, both for the international and global teams. On the other hand, FIRE estimated the support need by the function. Both estimates yielded reasonably credible estimates, with ITER being $685M and FIRE $144M. R&D support is also provided in the indirect costs.
- Commissioning (or Preparation for Operations) – ITER set-aside the first year of operations as a period to commission the device. Thus the cost is equivalent to an operational year plus funding for the initial spares ($468M). The period of time for commissioning for FIRE is unknown, but the project did set aside $24.5M for this purpose. This amount would be only sufficient for a short commissioning duration.
- Annual Operating Costs – ITER estimated an annual budget of $272M/yr for approximately 20 years. FIRE has not yet estimated this item.
- Decommissioning – ITER has provided a period of 11 years to decommission ITER with a $481M budget. FIRE has not yet determined the decommissioning activities, duration, or cost.

3.4 Experimental Approaches and Objectives

The Experimental Approaches and Objectives group included four subgroups, each concerned with an important element of the three proposed burning plasma experiments (BPX) and its place in a development path leading to the availability of fusion as an energy source. The Diagnostics subgroup evaluated the diagnostic sets on each BPX, in particular the suitability of the diagnostic set to carry out the mission defined by the proponents. The Integrated Scenarios subgroup performed extensive calculations to evaluate the projected fusion performance of each BPX and evaluated the flexibility of each device to address advanced scenarios. The Physics Operations subgroup evaluated the experimental operations plans of each BPX, including their compatibility with the physics and engineering features of each device. The Development Path subgroup identified how each device might contribute toward the ultimate goal of commercially available fusion energy.

3.4.1 Diagnostics

The success of any BPX will be dependent on the capability to make reliable plasma measurements with sufficient resolution and accuracy in a very hostile environment. These measurements will enable the development of understanding of the plasma behavior, provide the data necessary for achieving different operational modes and, significantly, be used as input for real-time control of the plasma behavior. All the proposed BPX devices will carry out physics studies needing very high quality information from diagnostics, similar to those in present-day operation. Many different diagnostic techniques will have to be integrated to provide the necessary operational guidance.

However, there will be a major change in performance as alpha-particle heating becomes significant and a relatively untested set of techniques for following the behavior of these particles is introduced. Diagnosis of the alpha particles with sufficient detail represents a severe challenge in any BPX, and will require substantial development. Also the study of the transport of the bulk plasma and of the alpha particles require detailed measurements of fluctuations in the core of the plasma.

The design experience of CIT and ITER revealed the serious impact of the burning plasma environment on measurement capability. The need to maintain overall shielding integrity and the sensitivity of diagnostic components to radiation effects leads to the location of components and their integration with shielding material. This shielding requirement tends to limit the field of view and solid angle available for diagnostics. Reflecting optics must be used in the first shielding module, and the plasma-facing mirror is vulnerable to radiation effects as well as sputtering and deposition. Remote maintenance of the shield modules in the radial ports and of the divertor modules (if present) leads to careful, and costly, design at the tokamak boundary. Sensitive components need to be housed in a distant shielded location. Stable calibration of many diagnostics will be a challenge, considering the complexity of remote handling, shielding and access conditions.

A very significant effort (~ 100 professional person years (ppy)) has been expended in designing the diagnostic systems for ITER. A carefully developed and evolving set of measurement requirements has been created for meeting the ultimate mission of a long controlled burn. The diagnostics to meet these requirements have been defined and many have undergone relatively detailed design studies. The integration of these systems into the available port space and shielding is well advanced, making use of the knowledge of the radiation shielding requirements that have evolved through the R&D studies of radiation effects. High priority has been given to measurements needed for control and machine protection. Detailed evaluations have been made of the ability of the various diagnostic techniques to provide the necessary temporal and spatial resolution, coverage, and accuracy. At this time, most of the measurements needed for control meet requirements, but a few, such as the measurement of current profile, need further study.

Only a fraction of a person year has been spent on preparing for FIRE diagnostics, though very much of the ITER experience can be applied to assessing the measurement capability. The mission of understanding and optimizing fusion-dominated plasmas in a smaller, higher density, RF-heated plasma leads to more extreme environmental conditions for some diagnostic components. Most proposed techniques appear plausible, but more detailed design work is necessary to demonstrate that measurement requirements can be satisfied. Active spectroscopy techniques, such as charge exchange spectroscopy and motional Stark effect polarimetry (the technique most commonly used to measure the current density profile), are dependent on the penetration of a neutral beam for full spatial coverage. Adequate penetration of a conventional (~100 keV) beam is problematic on FIRE due to its high density, and beam development appears essential.

Preparation for the diagnostics for IGNITOR is at a conceptual level, and measurement requirements have not been defined, limiting our ability to assess its diagnostic capability. The mission of achieving burn in short pulse plasmas without a divertor, and operating well away from stability limits for MHD modes, requires a less extensive set of measurements. The goal of investigating plasma heating, transport processes and stability of fusion-generated alpha-particles may be difficult, due to the lack of detailed profile and confined alpha diagnostics.

The diagnostic technology developed for tokamak BPXs is expected to be important for burning Innovative Confinement Concepts.

Assessing the planned set of plasma measurements for the different devices requires a careful analysis of what actually drives requirements for a specific measurement. Different objectives set different requirements, and thus an assessment should be made on the limits and constraints imposed by the device design on diagnostics as a generic group. Evaluation of individual diagnostic systems is beyond the scope of this review, although it should be noted that such designs exist only for ITER.

Access. First and foremost is the fundamental issue of diagnostic access. Most plasma measurements require a direct access to the discharge, and consequently the number of ports, their size, shape, location and orientation directly affect the measurement capability. In addition, in the presence of a divertor, additional measurements are required in an area, which is very restricted. Finally, many measurements benefit from a tangential view (looking in the toroidal direction), which, when absent, cannot always be compensated by the use of mirrors. Note that the ports must contain significant quantities of shielding to limit the neutron streaming, and diagnostic sightlines typically must follow labyrinthine paths through this shielding for it to be effective. The result of this integration with the shielding is generally a restriction in the achieved field of view and solid angle available for diagnostic viewing.

Radiation Environment. While one criterion of success is ultimately the realization of a large alpha particle production, the consequence is also a much larger neutron and gamma flux and fluence on diagnostic components than encountered in present-day experiments. Without mitigation, these are expected to produce effects such as radiation induced conductivity (RIC) in insulators, radiation-induced EMF (RIEMF) in mineral-insulated cables and nuclear heating. Because of radiation-induced absorption and fluorescence in optical components, reflective optical components have to be used close to the plasma.

Erosion and Deposition. The large neutral particle fluxes found in a tokamak at or near the first wall mean that the erosion and redeposition of plasma facing materials occurring in existing devices would be present as well in a BPX. These phenomena could affect the stability of the calibration of optical diagnostics, perhaps even during a single long pulse. Ultimately, this could be a survivability issue for optical systems.

Beam-based Diagnostics. Many measurements in present day tokamaks rely on techniques which use a neutral beam to diagnose the plasma, often with a dedicated diagnostic beam not used for auxiliary heating. Many of these measurements do not have equivalent techniques and consequently a serious gap may exist if beam-based diagnostics cannot be fielded effectively. However, in general those techniques do not easily extrapolate to a BPX because of beam penetration issues and access, and consequently they require additional development as well.

Cost. Stringent measurement requirements, combined with the access limitations and harsh environment expected on a BPX, drive the implementation schedule and overall costs of diagnostic systems. In this context, it should be recognized that the implementation of diagnostics cannot simply be an "after-thought", as it has been traditionally. The funding profile for the diagnostic set should be accelerated compared to the historical record. This will allow adequate integration of the diagnostics into the machine design as well as permitting advanced development of currently unavailable or uncertain measurement capabilities.

Assessment. As mentioned above, diagnostic implementations have traditionally proceeded only after the design of the tokamak is largely completed. That is the case for FIRE and IGNITOR, though the FIRE device is in an early design phase. However, since ITER went through a significant redesign (from ITER-98 to ITER), the diagnostic features are better integrated into the device at the design inception. Overall, the diagnostic set proposed by the ITER team appears credible, with a good probability of success. The unresolved issues are identified and subject to active R&D. The FIRE diagnostic set is plausible, but many issues remain, some of which may be resolved during detailed engineering design. For IGNITOR, it has not been possible to fully evaluate the diagnostics set.

For all three devices, the impact of access will be very important. ITER has certainly the largest port access, and consequently can accommodate a large set of diagnostics, although some limit may ultimately exist for the complement needed for physics evaluation. FIRE has more ports assigned to diagnostics, even though they are smaller. The integration of multiple diagnostics into these ports has not yet been done, so the adequacy of diagnostic access is more uncertain. FIRE retains basically the same measurement requirements as ITER, but adds the complexity of diagnosing a second divertor, a very demanding task. In the case of IGNITOR, the restriction in access is severe, but similar to that encountered on Alcator C-Mod, for example. Presently, with the available information, concerns remain that some basic measurements cannot be done effectively in IGNITOR for physics evaluation, particularly for detailed spatial profiles. However, many details are not yet formalized in the design, and it should be noted that the lack of a divertor simplifies the requirements. In all three cases the access is also limited by the necessary shielding, which is required to limit the radiation dose to the diagnostic components, the field coils, and the rest of the facility.

Of particular concern for studying burning plasma physics is the lack of convincing alpha particle diagnostic techniques. Particular attention is required to properly measure the confined and escaping alphas, and appropriate diagnostic testing must be done prior to their integration into a BPX. In parallel, attention should be given to refine the measurement requirements for alphas in the context of difficulty in implementing the techniques. The study of turbulence-driven transport in the alpha-dominated regime also requires a good set of fluctuation diagnostics for the core of the discharge. These systems should be part of the integration design studies as early as possible.

The diagnostics will experience a harsh nuclear environment in all three options. Neutron and gamma fluxes will be about 20 – 50 times larger for magnetic sensors and their cables for FIRE and IGNITOR than in ITER, and material selection and careful design will be more crucial. In all options, optical systems will require reflective optics close to the plasma to avoid absorption and fluorescence effects. Neutron streaming must be limited, leading to the integration of the diagnostic components with shielding and a resulting restriction of diagnostic views. While the fluence is highest in ITER with its long pulses, it is not likely to be a serious issue for diagnostics when integrated with appropriate mitigation. Remote handling is a necessity for all three devices, and will feature strongly in design integration. All diagnostic systems should be installed and tested prior to any tritium operation.

The impact of erosion and deposition will be the most severe for ITER, with its long pulse length, although the lack of easy access would mean that all three options have to consider these effects on the long-term reliability and availability of diagnostic systems, including their effects on calibration. The integration of diagnostic shutters to protect viewing optics will present an engineering challenge.

Neutral-beam-based diagnostics are presently planned for ITER and FIRE. For ITER, the feasibility of these diagnostics is under study, and the prospects are encouraging, although the short beam pulse length may still be an issue. The lack of beam penetration in FIRE is recognized to require the development of a new specialized beam. IGNITOR does not plan on a neutral beam, and a few important parameters may not be measurable. Adequate measurements of the current profile and the toroidal rotation may be problematic on IGNITOR using other techniques.

In all cases, an aggressive and dedicated R&D program is required for full implementation of the necessary measurements in the three options, building on the extensive ITER R&D effort. Research in radiation effects in materials must be pursued in a timely fashion, for the first components (such as magnetic loops) to be manufactured and tested. Many new diagnostic techniques require testing in existing experiments prior to their fielding in a BPX. A predictive understanding of erosion and redeposition on first mirrors and other exposed components require development as well. Additional, specific details of the assessment can be found in the appendix.

Contribution to ICC. Plasma diagnostic development has always been a discipline where the transferability between device concepts has been productive and direct. Diagnostic-specific issues encountered in tokamak BPXs will likely be relevant to the development of a burning ICC. Examples include measurement of alpha particle population, neutron flux and current profile. Issues of operation and survivability will be common to all BPX devices, including ICCs, although specific access solutions usually are not. Diagnostic integration experience gained on a tokamak-based BPX will also produce many benefits. For one, all the R&D required for mitigating radiation effects will apply to any burning ICC. New materials or improved techniques for the first mirrors will also be important. In addition, the development of reliable, simpler diagnostic techniques will be beneficial to any future reactor, be it a tokamak or not.

Benefits. Going ahead with a US participation in a BPX program will provide a strong motivation for developing new diagnostics, either new techniques or significant upgrades of current systems. This might reduce the recent decline in the U.S. effort in plasma diagnostic development. Many of these new diagnostics will be developed at universities, providing excellent opportunities for training young scientists. Many of the components will be developed by U.S. instrumentation companies, often small businesses.

Plasma instrumentation is required to provide the highest quality of physics information, while operating at new standards of stability and reliability, and will have to draw on U.S. expertise outside our field. On the technological side, the study and implementation of radiation-mitigation techniques will have spinoffs to areas of nuclear handling. All diagnostic developments, and their successful implementation on these BPX devices, will go a long way to defining the plasma measurements necessary for the control of a fusion power plant, with ITER obviously providing a greater degree of definition.

3.4.2. Integrated Scenarios

Introduction. The Integrated Scenarios group examined operational scenarios for each of the three burning plasma experiments (BPX): ITER, FIRE and Ignitor. Due to the sensitivity of fusion performance on interacting physical processes, a BPX will provide an unprecedented test of and challenge for integrated modeling. Properly diagnosed BPX experiments will provide a valuable set of data to improve our understanding of fusion science as we continue to study these interactions.

The operational scenarios for each of the devices were evaluated using both 0D and 1.5D calculations. In general, the results from both approaches were consistent, and indicate that each device is capable of meeting its expectations based on their advocates' assumptions.

The 0D calculations are done for equilibrium conditions using the IPB98(y,2) database scaling. This scaling relation is based on a database containing over 2000 discharges from many tokamak experiments. Although single tokamak databases show dependences on triangularity, proximity to the Greenwald density limit and density peaking, these are not reflected in the IPB98(y,2) scaling. In addition, this scaling relation produces some trends that are not observed in single tokamak databases, such as the well-known beta degradation. These dependencies are being actively researched, and may have significant impact on the projections. Scans were performed over density and temperature profiles, effective helium confinement time, density relative to Greenwald density, and impurities. The operating space was defined as $Q > 5$, $\beta_N < 2$, $n/n_{Gr} < 1$, $P_{loss}/P_{thr} > 1.0$, and $P_{aux} < P_{aux}$(max). This operating space was displayed in POPCON diagrams to determine the boundaries of operation.

Plasma energy transport is of central importance to the more sophisticated 1.5D calculations. These simulations rely on models for the core energy transport, and the pedestal in the case of H-mode. The MMM95 and the "stiffer" GLF23 models are used to calculate core transport for the ELMy H-mode in these simulations. The Coppi-Tang model was also used to simulate L–mode conditions. Several US dynamic transport codes were used in the assessments including Baldur, Corsica, GTWHIST, ONETWO, TRANSP, TSC, WHIST and XPTOR. The H–mode pedestal is handled in two ways: First, scans of an assumed pedestal value are used to determine the required pedestal temperature for a given Q value. Second, a theory-based model fit to experimental data is used to calculate the pedestal values directly. Although this model reproduces experimental trends, there remains uncertainty (32%) in predicting pedestal temperatures using this model.

Most results are based on the advocates' impurity assumptions, although these assumptions were scanned in the 0D assessment. A database was set up for the ITER design for the Z_{eff}, and this is typically used to project impurity content.

Accessibility to advanced tokamak or advanced performance plasma was assessed by considering the following features that are required to take advantage of this operation regime: 1) plasma shaping, 2) current profile control, 3) MHD control (NTM and RWM), 4) flattop durations $\tau_{dur}/\tau_{cr} \geq 2-3$, 5) profile diagnostics, and 6) pressure profile control. The device's proposed tools for AT operation, 1.5D simulations, and 0D operating space were used to determine the device's capability to examine burning AT plasmas.

ITER. The burning plasma operating space for the ELMy H-mode in ITER is large and robust to uncertainties in profiles, helium concentrations, and impurities for $H_{98(y,2)} = 1.0$, as determined by 0D analysis. For this operating space Q values of 10 or more are accessed. More sophisticated 1.5D simulations support the 0D operating space projections. Any level of density profile peaking allows larger operating space.

The 1.5D analysis of the ELMy H-mode in ITER indicates that the projected plasma performance can be met. The pedestal temperature range $T_{PED} = 2.7-5.0$ keV provides a range of fusion gain values $Q = 5-16$, depending on injected auxiliary power. Model predictions of T_{PED} are within this range, and the Q values predicted by these calculations is at the midrange.

The ITER device as proposed has a combination of baseline and upgrade heating/CD/MHD control sources that include NNBI, LHCD, ICRF and ECH/ECCD. Resistive wall mode control is anticipated to be possible with the error field correction coils, which are outside of the vessel. There exists access to Advanced Tokamak (AT) regimes, here defined as stationary 100% non-inductive current plasmas, with the goal of achieving progressively greater beta and bootstrap fraction. In addition, they are pursuing few thousand second flattop times, which correspond to greater than 5 current redistribution times. The long pulses obtainable in ITER are considered a significant benefit to studying stationary plasmas. Targeted plasmas obtain $\beta_N = 2.9$, $f_{bs} = 50\%$, $q_{95} = 5.3$, $H_{98(y,2)} = 1.6$, with RWM stabilization of the $n = 1$ mode. Achieving higher betas in combination with injection of CD power increases the total power lost from the plasma and the nuclear heating, and can challenge the power handling capability of the systems designed for nominal operation at 500 MW of fusion power for 400 s. The achievable flattop time is bounded by the blanket module cooling at higher fusion powers. Access to 3000 s pulse lengths at a maximum of 700 MW requires less significant facility upgrades of the heat rejection system. Access to higher fusion powers would require more substantial upgrades. Scrape off power, blanket module nuclear heating, divertor nuclear heating, first wall heat flux, and divertor particle heat flux limits provide the boundary to ITER's AT operating space.

FIRE. The burning plasma operating space for the ELMy H-mode in FIRE is large and robust to uncertainties in profiles, helium concentrations, and impurities for $H_{98(y,2)} = 1.1$, as determined by 0D analysis. For this operating space Q values of 10 or more are accessed. More sophisticated 1.5D simulations support the 0D operating space projections. The requirement of $H_{98(y,2)} = 1.0$ is considered reasonable based on observations that discharges selected from the confinement database for high triangularity and lower n/n_{Gr} values do average about $H_{98(y,2)} = 1.1$. Any level of density peaking further enlarges the operating space.

The 1.5D analysis of the ELMy H-mode in FIRE indicates that the projected plasma performance can be met. The pedestal temperature range $T_{PED} = 2.3-5.5$ provides a range of fusion gain values $Q = 4-15$, depending on injected auxiliary power. Model predictions of T_{PED} are at the low end of this range for both temperature and fusion gain.

The FIRE device as proposed has a combination of baseline and upgrade heating/CD/MHD control sources that include ICRF/FW, LHCD, and possibly ECH/ECCD (at lower fields for AT, not examined in detail). High beta values are expected to be facilitated by FIRE's strong shaping and internal coils for resistive wall mode control. There exists access to AT plasmas, here defined as stationary 100% non-inductive current plasmas, with the goal of achieving greater beta and bootstrap fraction. Targeted plasmas obtain $\beta_N = 3.7$, $f_{bs} = 70\%$, $q_{95} = 3.5$, $H_{98(y,2)} = 1.6$, with RWM stabilization of the $n = 1$ mode. The internal RWM coils for $n = 1$ mode feedback are considered an advantage for obtaining beta values approaching the wall stabilized limit. The device is capable of flattop times ranging from 1 to 3 current redistribution times. The flattop time restrictions may limit the range and depth of AT study. Achieving higher betas in combination with injection of CD power increases the total power lost from the plasma and the nuclear heating, and can challenge the power handling capability of the systems designed for nominal operation. The flattop time achievable is determined by the nuclear heating of the vacuum vessel, or heating of the TF coil. The radiated powers from the core plasma and in the divertor, and the particle heat load to the divertor provide the limits to FIRE's AT operating space.

IGNITOR. The Ignitor operational scenarios do not achieve a stationary burning plasma with respect to energy confinement time, effective particle confinement time, or current diffusion times scales. Since the 0D analysis inherently depends on the plasma reaching equilibrium, it was not done for Ignitor and our conclusions are based on 1.5D calculations.

The reference operation is limited on the first wall and is assumed by the advocates to be in an improved L–mode confinement regime. It is possible to access $Q = 10$ with varying combinations of density profile peaking and confinement enhancement over ITER97L L–mode global scaling. This can be achieved both with either strictly Ohmic or auxiliary assisted heating. The level of confinement enhancement ranges from $H_{97L} = 1.2$ for peaked density ($n(0)/\langle n\rangle = 2.3$) profiles, $H_{97L} = 1.4$ for broad density profiles ($n(0)/\langle n\rangle = 1.4$), and $H_{97L} = 1.7$ with both broad temperature and density profiles. Although peaked density profiles may be possible on a transient basis, a scenario for sustaining such peaking has not been established. Since the confinement enhancements over L-mode are associated with density peaking, the sustainment issue provides the primary uncertainty in projecting fusion performance.

Although H-mode operation in Ignitor is not the primary operation mode, it was discussed as a possibility. Although the device is not optimally designed to produce a divertor configuration, either single- or double-null divertor configurations are possible at slightly reduced current. H–mode operation should be available in such configurations. A smaller set of 0D and 1.5D simulations have been done, indicating that $Q > 5$ should be accessible in Ignitor H–mode discharges.

The Ignitor device as proposed has a large amount of ICRF (24 MW), but lacks the current drive and MHD control, and particle removal tools normally associated with AT operation. This is largely due to the choice of the advocates to not include AT operation as part of Ignitor's mission. It may be possible to operate the ICRF system to produce fast waves for current drive purposes, but this capability would only be useful to drive current on the magnetic axis and does not provide for the off-axis current drive needed for most AT scenarios.

Several scenarios for advanced performance have been proposed for Ignitor. These focus on transiently obtaining reversed magnetic shear and internal transport barriers using combinations of fast current ramps and auxiliary heating. This technique is widely used in present-day devices, and there is little doubt that such regimes could be produced in Ignitor and would increase performance. Sustainment of these configurations is not possible due to current diffusion.

3.4.3 Physics Operations

Introduction. Operational issues impact most aspects of a machine design. In addition to specific operation plan and schedule issues, the Physics Operations working group has selected a set of issues to evaluate which strongly impact the ability to operate a burning plasma device in a manner consistent with the stated mission. Issues addressed include the experimental operations plan, operational issues which impact the divertor, first wall, and structure, in-vessel tritium inventory, equilibrium operating flexibility and achievability, and controllability of the target equilibria. Assessments have relied on analysis performed by the working group as well as reports from BPX design teams.

The principal difference among the three devices under assessment is their different level of operational design maturity. ITER's operational plan is consistent with its mature level of engineering design, and is capable of supporting its scientific and technical objectives of long pulse burning plasma study integrated with reactor-relevant technology. The ITER project includes a credible and well-defined R&D plan to address outstanding operational issues. The FIRE operational plan is consistent with its pre-conceptual level of design maturity, and is likely to be capable of supporting its scientific and technical objectives of burning plasma study with AT control on timescales of 1-3 current relaxation times, reduced reactor-relevant technology integration goals relative to ITER, at reduced cost relative to ITER. The R&D plan and operations schedule provide a credible plan for addressing outstanding issues. Ignitor is at an early stage of operational planning, although at an advanced stage of technology development in some areas (e.g. magnets). Some key elements of the operational plan have yet to be addressed with significant analysis. However, the proposed plan is consistent with this level of design, and consistent with the scientific and technical objectives of burning plasma study over approximately one current redistribution time. Methods and prospects for resolving many outstanding issues have been proposed, and several are currently being incorporated in the design.

Experimental Operations. The experimental plan must provide sufficient pulse lengths to allow study of relevant phenomena. The present devices are strongly separated in this characteristic. ITER provides 12000 full-power equivalent 400 second pulses out of a total of 18000 discharges (67% of the total). FIRE provides 3000 full-power equivalent 20 second pulses out of a total of 30000 discharges (10%). Ignitor provides 600 full-power equivalent 4 second pulses out of a nominal plan of ~25000 discharges (3%).

A relevant figure of merit for pulse lengths is how many time constants corresponding to various physics and technology phenomena are contained in a single pulse. As a common basis for comparison, we consider only the high-performance full-field ("Full Power Equivalent", FPE) discharges in each device. The current relaxation time, τ_{CR}, is a measure of the relevance to advanced tokamak control scenarios. As calculated in TSC simulations, Ignitor provides 0.92τ_{CR} in a single pulse of 4-second burn time, FIRE provides 2.0τ_{CR} in its 20-second pulse, and ITER provides 2.6τ_{CR} in its 400-second pulse. The same analysis applied to energy confinement times provides a measure of confinement physics productivity potential. Ignitor, FIRE, and ITER pulses contain ~7, ~20, and ~120 τ_E respectively.

Divertor/First Wall and Structure Operational Impacts. Heat loads on divertors and first walls pose a particular challenge to all BPX devices, with typical steady state divertor fluxes (in the absence of radiative mitigation) in the range of 5-15 MW/m^2 and disruption thermal loads of 30-100 MJ/m^2 in less than a few milliseconds. Type I ELMs in particular are a divertor lifetime-limiting issue in both FIRE and ITER, whose baseline operating regime is presently the ELMy H-mode. Both devices are expected to produce similar heat loads to their divertors in the range of 1-5 MJ/m^2/ELM. A single type I ELM will produce a melted layer in the W divertor plate in both FIRE and ITER (if a W divertor is installed in the later phases of operation), of approximately 10-100 μm. The operations consequences of limited melting of plasma facing surfaces are not well-understood, but available machine experience indicates that operating with previously melted surfaces can degrade plasma purity, increase disruptivity, and increase the fraction of discharges that fail to reach performance goals. Assuming complete loss of the melt layer, the divertor lifetime is limited to approximately 100 discharges in each device (similar for tungsten or carbon divertors). Solutions to this severely limiting divertor erosion include developing target equilibria which produce type II instead of type I ELMs, operating in a stationary ELM-free mode (e.g. EDA, QH), or mitigating ELM effects with impurity injection. The baseline high-triangularity, DN configuration of FIRE is favorable for achieving type II ELMing regimes. Ignitor envisions a near-DN operating scenario which could produce ELMs, but the details of this scenario are presently not sufficiently well-defined to allow assessment of the wall impact.

Disruption heat loads produce a similar lifetime limitation in the BPX devices. ITER and FIRE experience comparable divertor heat loads of 1000-4000 MJ/m^2/s$^{1/2}$ in unmitigated full-performance disruptions, which produces a tungsten melt layer of ~100 μm in each device (for the ITER W divertor option). A similar thickness of carbon is ablated from the ITER C divertor, which is planned in the early phases of operation. A molybdenum melt layer of ~100 μm is likely produced on the first wall in each Ignitor disruption. In each device the lifetime of the divertors (FIRE, ITER) and first wall (Ignitor) is approximately 100 disruptions (~1000 discharges assuming 10% disruptivity) before replacement is necessary. Injection of large quantities of noble gas produces a pre-emptive radiative collapse which distributes the thermal and magnetic energy isotropically to the first wall. Typically less than 1-2% of the energy is conducted to the divertor in this process. Calculations show that following injection of neon the ITER and FIRE first walls would experience thermal loads of 15-20 MJ/m^2/s$^{1/2}$, below but near the melt limit of the Be first wall material they share (~20-25 MJ/m^2/s$^{1/2}$). Employing this method of mitigation in Ignitor serves to increase the area over which the energy (> 43 MJ/m^2/s$^{1/2}$) is deposited relative to the unmitigated case, but still somewhat exceeds the melt limit of Mo (~40 MJ/m^2/s$^{1/2}$). The resulting melt layer thickness is estimated to be 10-100 μm. The wall lifetime, even if all disruptions are mitigated, is therefore similar to the unmitigated lifetime of ITER and FIRE divertors and the Ignitor first wall (1000 discharges, assuming complete erosion loss of the facing material and a disruption rate of 10%).

An important and often dominant source of local electromagnetic (EM) stress in tokamaks is the poloidal halo current which is driven during disruptions when the plasma becomes limited. Calculations of halo currents driven during VDE's show that Ignitor, with the largest vertical growth rate (~80 rad/sec) and toroidal field (13 T) among the three devices, produces the largest pressure on the vessel (> 6 Mpa) during a VDE of all the BPX designs. This pressure produces several cm of displacement of the 2.6 cm-thick Ignitor vacuum vessel during the dynamic evolution of the disruption, and exceeds the plastic yield limit. Ignitor is presently undergoing a redesign of vessel support structures in order to withstand these forces, but in-vessel misalignment of PFC's is still a concern. While the EM loads in FIRE VDE scenarios are below allowables (calculated to be ~4 Mpa in the nominal VDE scenario), the addition of neutron heating-induced thermal stresses in the vessel to the disruption EM loads produces a total stress near cyclic allowables in the present design. Redesign of the vessel, support structure, FW tiles, and structure heaters is underway to reduce the total stress. ITER loads are well within allowables in all disruption scenarios (calculated to be ~0.5 Mpa in the nominal VDE scenario). Impurity injection mitigation would reduce the halo pressure in all devices.

Calculations show that at least 50% of the plasma current is likely to be converted to runaway current carried by high-energy (typically ~10 MeV) electrons in disruption current quenches. The limiting level of damage to plasma facing components (PFC) before replacement or repair is necessary is difficult to assess, but given the total energy of the runaway channel (~240 MJ in ITER for I_{RE}~$0.5I_P$), it is likely to be essential that runaways be suppressed in all three BPX's. Gas mitigation has been demonstrated to suppress runaway production completely with sufficient injected density, and analysis of BPX scenarios shows that a relatively modest amount of impurity (producing a neutral density ~10^{22} m^{-3}, 10^{-3} bar) at easily achievable gas jet pressures (~2-10 bar, limits corresponding to ITER and Ignitor, respectively) will provide this complete suppression.

RE suppression specifies a maximum allowable current quench time. However, disruption thermal load mitigation by gas injection requires a sufficiently *long* current quench time in order to produce a sufficiently low radiated power. In FIRE and ITER there exists a common scenario for simultaneous mitigation of RE and reduction of thermal loads below the Be melt limit. Because of its uniquely large magnetic energy, the majority of the energy in a mitigated Ignitor disruption is released during the current quench. Simulations of Ignitor disruption mitigation suggest that there is no single current quench time (and therefore no single-species injection scenario) which allows simultaneous RE suppression and reduction of wall thermal loads to below the Mo melt limit. Multi-species gas mitigation offers an opportunity to satisfy both timescales, but is still an early, unproven concept. For all BPX's, disruption mitigation presumes a disruption prediction/detection system that is sufficiently fast and reliable. Because such systems presently exist only in rudimentary form, insufficient for a BPX, significant development will be required to satisfy this need.

The ITER baseline plan includes operation with carbon divertor plates in the initial phases of operation (including initial DT operation). Modeling and simple extrapolation from experimental data suggest that a 1.5% retention rate is likely in the ITER C divertor. The ITER fueling level of 250 g of T results in ~3.8 g per discharge, or ~100 shots before reaching the ITER in-vessel limit of 350 g-T, requiring shutdown and in situ cleaning or replacement of the divertor. The ITER baseline plan includes replacement of the C targets with W before entering full power DT operations, while R&D efforts plan to alleviate T retention by use of such methods as dedicated C/T cold traps. The FIRE baseline divertor design includes W targets during the entire life of the machine, with T retention below 1% allowing more than 5000 discharges before replacement. Ignitor's Mo first wall baseline design allows execution of its full operational plan without requiring replacement due to tritium loading.

Equilibrium Operational Issues. The shaping flexibility and scenarios of ITER are adequate for its stated mission. Extensive design and analysis has demonstrated that the power supplies and PF coil capabilities can achieve the range of shapes required with reasonable variation in profile parameters. All planned scenarios are controllable with demonstrated dynamic performance consistent with both physics and machine operational requirements. Near-DN equilibria can be achieved in the reference scenario (□~1.85, □~0.55). Long pulse AT capability is provided, with demonstrated capability to achieve □>2.1, □>0.6 with pedestal parameters consistent with DIII-D AT scenarios. FIRE has chosen strong DN shaping parameters for nominal values, consistent with and adequate for its mission. Satisfactory shape control performance has been demonstrated consistent with the pre-conceptual level of the FIRE design in TSC simulations. The elongation can be varied very little about 2.0 at the X-point, ± 0.05 at full minor radius. Triangularity can be varied from 0.6 to 0.8 at full minor radius. Vertical stability analysis and some degree of dynamic shape control analysis has been done.

Nominal achievement of Ignitor limited and near-DN equilibria within power supply and PF coil constraints has been demonstrated. However, only limited dynamic control studies have been done. Vertical stability control has been assessed in the limiter configuration and found to be consistent with power supply and PF coil constraints. Detailed dynamic control characteristics and requirements of the near-DN configuration are expected to be demanding and deserve careful analysis. In the absence of strong pumping, the rapid ~3.5 MA/s rampdown in the standard Ignitor scenario is likely to produce an extremely high internal inductance and a density limit disruption. Pre-emptive gas mitigation just prior to the disruption may provide a solution, since much of the magnetic energy will have been dissipated by that time. The baseline rampup scenario for Ignitor requires very large voltages and ramp rates compared with present machines. Consistency with achieving stable, monotonic target equilibrium current profiles at 11 keV by the end of rampup has yet to be demonstrated.

3.4.4 MFE Development Paths

The development path to realize fusion as a practical energy source must include four additional essential elements: Fundamental understanding of the underlying science and technology; Plasma physics research in a burning plasma experiment; Configuration optimization such as high performance, steady-state operation; and Development of low-activation materials and fusion technologies

Burning plasma physics and configuration optimization: A diversified and integrated portfolio consisting of burning plasma experiment(s), steady-state DD tokamak experiments, ICCs, and theory/simulation is needed to develop the necessary predictive capability in burning plasma physics and high-performance state operation and concept operation. The BPX should be flexible and well diagnosed in order to provide fundamental understanding and physics and technology data for the entire toroidal concept portfolio.

Plasma Support Technologies: A strong program in plasma support technologies (fueling, magnets, heating, PFC) including experiments on test stands is necessary to develop advanced technologies necessary for power plants. Experience on present and future high performance and steady state device as well as the BPX will provide a wealth of data on individual technologies. Among the proposed BPX experiments, ITER will provide valuable data on integration of power-plant relevant plasma support technologies.

Low-activation material and fusion power technologies: All scenarios considered require development of low activation material and fusion power technologies for integration at a subsequent device to BPX. Fusion power technologies are in their infancy and are probably a pace setting element of fusion development. Development of fusion power technologies require:

- A strong technology research program including testing of components in non-nuclear environment as well as fission reactors.
- A materials program including an intense neutron source to develop and qualify low-activation material. International Fusion Material Irradiation Facility (IFMIF) is an example of such a material test facility and has been included in fusion development plan worldwide.
- A Component Test Facility (CTF) which is sometimes referred to as a volume neutron source (VNS) for integration and test of power technologies in a fusion environment with a high duty

factor. Such a device should test and integrate fusion power technologies under proto-typical power and neutron flux and fluence conditions and should address reliability of components in a power-plant environment.

Fusion development scenario based on ITER-class burning plasma experiment

Burning plasma physics and configuration optimization: It is highly unlikely that an ITER-class experiment would be the only large tokamak experiment in the world. National or regional programs will include performance-extension tokamak devices. These devices are needed to ensure continuation and growth of national expertise and capabilities. More importantly, physics investigations on these performance-extension devices will allow optimum utilization of ITER-class experiment. Smaller devices would allow thorough investigation of individual physics phenomena and act as a test bed for ideas, which can be tested in an integrated manner in ITER. As such, an international tokamak research program centered around ITER and including these national performance-extension devices have the highest chance of success in thorough examination of burning plasma physics in advanced tokamak modes.

Non-tokamak facilities to extend physics understanding, and to develop and test the innovations for improving toroidal magnetic configurations are an essential part of the magnetic fusion program. Diversified facilities at various stages of scientific exploration are needed to carry this fusion program forward, and thus to provide assurance that an adequate magnetic configuration is available at the time of the DEMO decision point.

Plasma Support Technologies: Because of its size, its relatively high duty factor, and its neutron flux and fluence, ITER will provide valuable data on integration of power-plant relevant plasma support technologies.

Low-Activation Material and Fusion Power Technologies: A unique aspect of an ITER-class burning plasma is the capability for limited testing of fusion power technologies. However, because of the low base-line fluence of 0.3 MW.yr/m^2 and relatively low neutron flux, there would be a high risk to proceed to an electricity producing device solely based on ITER testing program. As described above, a strong base program, an intense neutron source facility and a CTF/VNS is necessary before proceeding with the DEMO. ITER capability in testing fusion power technologies as well as the ITER experience on integration and operation of a variety of fusion technologies are valuable to CTF/VNS operation.

Decision Point: An ITER-class BPX allows leapfrog in fusion development path by combination three areas of burning plasma physics, advanced tokamak modes, and plasma support technologies. Successful completion of ITER experimental program (demonstration of high-performance AT burning plasma) will allow tokamak concept to move to fusion power demonstration (DEMO) leading to the shortest development time for fusion. Here DEMO is defined as a device which incorporates all physics and technologies necessary for an attractive commercial power plant. Alternatively, the tokamak configuration may be replaced by an alternative configuration. The technology integration within the ITER program will allow significant acceleration of alternative configurations at this stage.

Fusion development scenario based on FIRE-class burning plasma experiment

Burning plasma physics and configuration optimization: The major next step plasma physics facilities in the International Portfolio Approach are:

- Advanced tokamak physics facilities to address the high-□, high-bootstrap and non-burning plasma physics issues needed for attractive power plants. The programs planned for KSTAR, now under construction in South Korea, and JT-60SC under design in would be sufficient to address these issues in a non-burning plasma. The larger of these facilities would have advanced tokamak performance capability sufficient to achieve equivalent $Q_{DT} \sim 1 – 2$ while operating in deuterium. Very limited DT experiments might also be carried out. These facilities would also address the integration of plasma technologies in DD plasmas.

- Burning plasma facility(s) to address the burning plasma physics issues expected in power plants. The most expeditious way to do this is to incorporate the results from the advanced tokamak facilities into the later phases of the burning plasma experiment. The FIRE experiment, being designed in the US with a construction cost of ≈ $1.2B, has adopted strong plasma shaping, geometry and other advanced features identified by ARIES power plant studies.
- Fusion Plasma Simulator to contain comprehensive coupled self-consistent models of all important plasma phenomena that would be used to guide experiments and be updated with ongoing experimental results.
- Non-tokamak facilities to extend physics understanding, and to develop and test the innovations to improve the toroidal magnetic configuration are an essential part of the magnetic fusion program. Diversified facilities at various stages of scientific exploration are needed to carry this fusion program forward, and thus to provide assurance that an adequate magnetic configuration is available at the time of the DEMO decision point.

Plasma Support Technologies: Experience on present and future high performance and steady state device as well as FIRE will provide a wealth data on individual technologies. Complete integration with burning plasmas is deferred to the follow-up step.

Low-Activation Material and Fusion Power Technologies: As described above, a strong base program, an intense neutron source facility and a CTF/VNS is necessary before proceeding with the DEMO.

Decision Point: Integration of Program Elements is needed to provide the technical basis for the decision on an Advanced Engineering Test Reactor (ETR). FIRE in combination with non-burning KSTAR and JT-60 SC and a strong burning plasma simulation program would provide the integrated physics basis (advanced confinement, high power plasma exhaust and burning plasma) needed for the Decision on proceeding with a tokamak based Advanced ETR. The integration of technology from the CTF/VNS with the superconducting long-pulse advanced tokamak and the advanced burning plasma tokamak would provide the technology basis for the decision on a tokamak Advanced ETR. During the initial operating phase of the advanced ETR the integration of the physics and technologies would be validated, and the facility would evolve into the DEMO. Alternatively, the tokamak configuration may be replaced by an alternative configuration which has been developed within the configuration optimization program.

Fusion development scenario based on IGNITOR burning plasma experiment

The major advantage of IGNITOR is demonstration of fusion burn, a major milestone for fusion energy development, at earliest date and at the lowest cost. Because of its short pulse length, IGNITOR cannot thoroughly investigate burn control and/or advanced tokamak modes.

As an element of a national base program, IGNITOR would support ITER-based or FIRE-based development scenarios.

Relationship between the MFE innovative confinement concepts (ICCs) and tokamak burning plasmas (science and technology)

The Innovative Confinement Concepts are a core part of the U.S. base Fusion Energy Sciences Program, along with the Advanced Tokamak (AT) program and the theory and computational modeling program. The ICC program responses to Goal 2 of the Integrated Program Planning Activity:

> ***Resolve outstanding scientific issues and establish reduced-cost paths to more attractive fusion energy systems by investigating a broad range of innovative magnetic confinement configurations.***

The ICC experiments address several programmatic and fusion energy science objectives by:

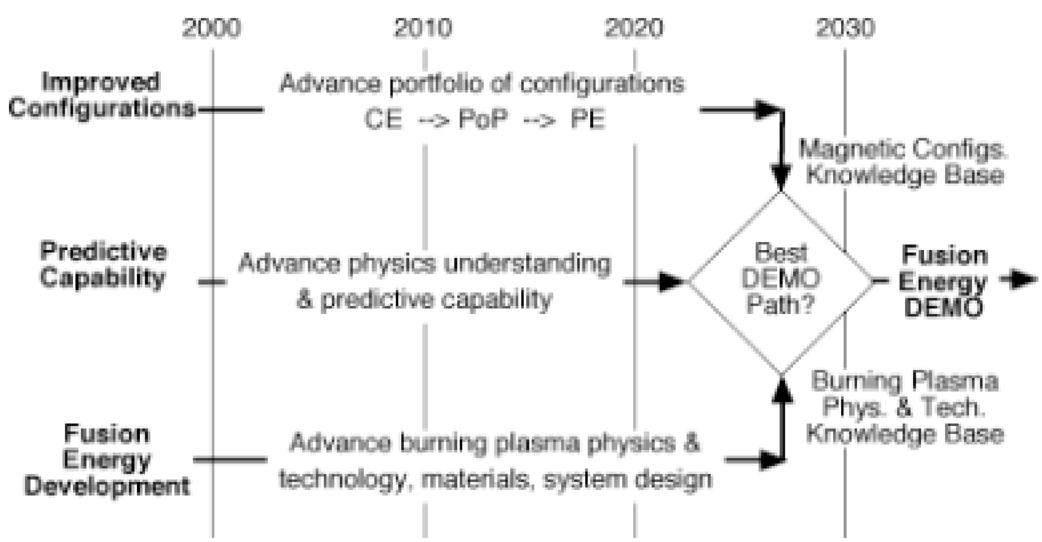
- Working within a broad range of plasma and fusion energy sciences, including cross fertilization with other fields of plasma science;

- Seeking concepts and innovations that work better or change the paradigm for fusion energy;
- Broadening the physics of toroidal magnetic confinement by operating in parameter regimes inaccessible by the tokamak;
- Strengthening university plasma science and technology programs, engaging faculty by providing opportunities to contribute to plasma and fusion science with small-to-medium size experiments; and
- Attracting bright, young talent with the vision of unlimited energy for mankind while providing the opportunity to participate in experiments they can "get their hands around."

The contribution to development of fusion energy by the PoP ICCs is discussed above in terms of the U.S. multiple path strategy. Generally, we envision a progression through the PE phase in parallel with the tokamak, coupled through the predictive science knowledge base to the BPX physics. Assuming progress on one or more of the ICCs, at the appropriate time a decision will be made as to the best DEMO. A reactor based on these concepts is anticipated to look much like one based on the tokamak, although obviously with differences arising from operating in a different part of toroidal-physics parameter space. A reactor based on the toroidal CEs may follow a similar path, although these configurations are expected to pass through the PoP stage before becoming a PE.

There are, in addition, possible reactor scenarios for several of the CEs which would result in a very different reactor implementation. Preliminary concepts have been explored for a spheromak reactor including a conventional (tokamak-like) device using coaxial helicity injection; a steady state, flowing liquid wall reactor; a pulsed, liquid wall reactor; and a reactor driven by multiple, merging spheromaks. For any of these to become viable will require a better understanding of the physics than we have today. The FRC is considering rotational magnetic field current drive and may take advantage of the ability to move the plasma from one vacuum chamber into another in order to separate the formation and burn phases. Magnetized target fusion is examining reaching pulsed burn by the compression of a ST, FRC, spheromak, z-pinch, or other plasma. At the extreme, concepts like Inertial Electrostatic Confinement will generate small, net power in each of many cells, perhaps using no magnetic field; the assembly into a reactor looks much more like a fission assembly than a tokamak. Such ideas move the fusion power options far from the tokamak burning plasma and in some cases have as much or more in common with the IFE ideas than MFE.

Integrating burning plasma physics with an advanced portfolio of configurations. The decision on the best DEMO path will include consideration of the need for a burning plasma experiment for a candidate advanced configuration in the context of the predictive science base developed from the BPX and the toroidal ICCs.



To couple effectively the BPX and the ICCs, it will be essential to develop and focus on the science. To see this, it is useful to consider some examples from the 3 Proof-of-Principle (PoP) and almost 30 Concept Exploration (CE) experiments. Consider first two toroidal concepts:

Spherical Torus (ST): The ST, a PoP-level experiment, is a tokamak squeezed to the lower limit of aspect ratio, resulting in strong field line curvature, a low outboard toroidal magnetic field, high toroidal beta with a central value ~ 1, and a strong diamagnetic well (~ 30%). There is a strong overlap with tokamak burning physics, allowing application of the lessons learned therein to STs ranging from the PoP level to a burning plasma. However, several differences will broaden contributions to and from the ST. As in the BPX, alpha-generated pressure may impact operation at high beta, MHD stability, and control of the discharge; however, the strong diamagnetic effects and deep magnetic well will modify the response of the ST plasma in ways that will be clear only through experimentation and deep scientific understanding. Energy confinement may retain substantial features of electrostatic turbulence, but the relatively large ion gyroradii man result in strong magnetic effects on the turbulence. Stability and waves will be affected by the large Alfvén Mach number (~ 4) particles and the large dielectric constant (~□00). Plasma-boundary interactions will require a divertor or other means of handling high power, but the large mirror-ratio may modify plasma boundary physics.

Spheromak: The spheromak, a toroidal CE-level experiment, operates in a very different regime from the tokamak, with $q \lesssim 1$, transport driven by magnetic turbulence (at least in present experiments), and a singly connected volume with no central column to generate a vacuum magnetic field or inductive current drive. BP physics will, however, carry over by enhancing understanding of MHD equilibria and stability, the effects of self-consistent pressure profiles from fusion products, alpha particle drive for Alfvén and other modes, and high-power plasma-wall interactions. The spheromak may also be considered as the limit in which the ST vacuum toroidal magnetic field reaches zero, thus benefiting from applications to the ST but extending toroidal physics beyond the ST regime.

Toroidal physics in these and other ICCs thus occupies a broader parameter space than in the tokamak alone. The reversed-field pinch (RFP) operates with a weak magnetic field and thus $q < 1$ and large shear; the plasma current consequently plays a more dominant role in it than in the tokamak, and the reversal of the safety factor, q, opens parameter space not accessible to the spheromak. Compact stellarators will test the physics of quasi-symmetric magnetic fields, extending the axisymmetric physics of the tokamak without requiring plasma toroidal current. The quasi-axisymmetric stellarator, which does make modest use of bootstrap currents, will clarify the trade-offs between poloidal fields generated by external coils and by internal currents, and offers a fusion opportunity that combines features of the tokamak with those of other stellarator configurations. The compact stellarators will also test neoclassical transport levels, plasma stability, and accessibility of enhanced confinement regimes. The Electric Tokamak (ET) has a large aspect ratio and low magnetic field and the goal of reaching unit beta; the study of this part of toroidal parameter space will significantly extend the tokamak operating space and may lead to interesting energy options. The Field-Reversed Configuration (FRC) and levitated dipole experiment (LDX) extend the toroidal operating space to zero toroidal magnetic field. Both are anticipated to operate at high beta but with the plasma current ranging from dominant and perpendicular to the magnetic field in the FRC to negligible in the LDX.

Thus, the synergistic combination of the tokamak burning plasma, the advanced tokamak, and the toroidal ICCs will generate a much broader scientific understanding of toroidal physics than would be obtained by operating tokamaks, leading to a better optimization of the toroidal confinement concept. This understanding will support the ICCs in their scientific and energy missions and are likely to contribute to broadening the scientific options available to the BPX, plasma-based test facilities, and DEMO.

Successful coupling through a predictive science base will require outstanding diagnostics both in the BPX and the ICCs. However, measurements will be very difficult in the intense radiation environment of the BPX, and the resources available to the CEs require them to focus on a narrow set of issues. Most AT experiments are very well diagnosed, so they and the PoP ICCs can provide a bridge between the BPX and the CE ICCs, helping to unify the program by providing an opportunity to explore physics to a detailed and integrated manner. The ATs can also test new physics generated in the ICCs and from science experiments focused on narrow physics issues such as magnetic reconnection. In doing this, it can help guide experiments on the BPX and strengthen the coupling of results to theory and simulation.

Much of the knowledge determined from a tokamak BPX will be encapsulated in computational models. Because of the cost of a BPX, successful ICCs will need to take advantage of this knowledge to minimize the cost of their bp step by making it more focused, taking a more aggressive step, possibly leapfrogging the BPX step. In the process of doing this, they will contribute to this vertical integration of modeling across the concepts, complementing the horizontal physics goals of existing integrated simulation codes. This will strengthen our modeling and simulation capabilities, providing a tool to develop and enhance the toroidal physics discussed above.

Technology transfer between the BPX and the ICCs will be as important as physics transfer. The BPX will develop operating experience and improve availability in a real fusion environment for key magnetic fusion technologies, most of which are applicable to one or more ICCs. Technologies developed by the ICCs may be applicable to the BPX as well:

- Magnetic coils – Superconducting coils will be needed for most steady-state devices, although resistive coils may be acceptable if the plasma beta is high.
- Heating and current drive technologies – These technologies are applicable to many ICCs; also, new current drive techniques such as helicity injection and rotating magnetic fields are being developed in the ICCs and may find application to a BPX.
- Fueling technologies – Gas puffing or pellet injection are sufficiently flexible to be adapted to most toroidal ICCs. Compact toroid injection is an advanced option which arises from research on the spheromak and FRC and may be used in BPX.
- Plasma facing components – Toroidal ICCs have similar or greater wall power and particle handling issues as the tokamak, so this may be one of the most important spin-offs from the BPX to the ICCs.
- Remote handling – Development of and experience with remote handling will be of major utility to eventual burning plasma ICCs.
- Fusion Power Technologies – Breeder materials tests and blanket designs will be applicable to many ICC concepts.

There are thus significant couplings between the BPX and the ICCs which promise benefits to both. To achieve these couplings, the chosen burning plasma experiment must be capable of exploring a broad range of physics parameters, have good access for diagnostics and a well thought-out diagnostic plan, and be supported by a strong theoretical and computation modeling and simulation effort coupled especially to the toroidal ICC physics experiments.

4.0 Inertial Fusion Energy Next Steps

In 1990 the Fusion Policy Advisory Committee recommended that inertial fusion energy (IFE) and magnetic fusion energy (MFE) be developed in parallel. The Fusion Energy Sciences Advisory Committee, following the 1999 Snowmass Workshop, reaffirmed this policy and recommended funding levels for both approaches. During the past three years, both approaches have made excellent technical progress. The needed next steps for the two programs are, however, quite different. While both require burning plasma experiments, IFE can obtain much of the needed scientific data from the National Ignition Facility (NIF), a facility currently being built at Lawrence Livermore National Laboratory under the auspices of the National Nuclear Security Administration (NNSA). Inertial fusion energy does not require the construction of an additional burning plasma experiment. Also, IFE is currently obtaining essential plasma physics data from existing NNSA facilities such as Z, Omega, Nike, and a number of foreign facilities.

Although the NIF will provide the needed data on burning IFE plasmas, it does not have the capability to operate at high repetition rates or to manage the fusion power that high repetition rates produce. Moreover the NIF has neither the efficiency nor the durability needed for commercial power production. Substantial scientific and technical issues must be studied and resolved in parallel to enable high repetition rates, good efficiency, and adequate lifetime. The modularity of IFE drivers and the separability of power plant components make it possible to study these issues and issues associated with supporting subsystems in scaled facilities. The IFE community refers to these facilities as "integrated research facilities" or IREs. They are the next major steps in inertial fusion. They are expected to be substantially less expensive than either the magnetic burning plasma experiment or the NIF. While the NIF can demonstrate the creation of fusion energy in single shots, the IREs will provide the foundation of science and technology needed for the subsequent demonstration of net fusion power, and the delivery of net fusion electricity to the grid.

Finally, it is noteworthy that this Snowmass Workshop did not have the charter to coordinate the MFE and IFE programs and their important areas of synergistic overlap. This coordination will be accomplished at a later date.

4.1. Overview of Inertial Fusion Energy

An IFE power plant will produce energy by focusing intense beams of light or charged particles, or concentrating intense x rays, onto a small target containing fusion fuel. The fuel will ignite with a burst of fusion reactions releasing much more energy than was invested to cause ignition. The fusion heart of the power plant will have several important systems:

- The fusion targets containing the fuel.
- A factory designed to fabricate millions of targets per year.
- A chamber approximately 6 meters or more in diameter to capture the energy produced by the fusion pulses.
- An injection system to inject or place the targets into the chamber.
- A driver to produce the energy needed for ignition.
- A focusing or concentration system to deliver the driver energy to the target.

There are many different types of targets, several types of chambers, and several types of drivers and focusing systems. To some extent these systems or components are independent so there are many possible combinations. This independence allows modular, cost-effective research on key issues with synergy among the integrated concepts. One of the reasons for independence among the various components is the fact that the many of the components are spatially separated. For example the driver, in most concepts, may be many meters away from the chamber. Additionally, the phenomena that control target energy gain, chamber repetition rate, and power plant reliability separate into groups that occur on vastly different time scales, and can be studied independently, as explained in Section 4.3.

At the present time IFE research focuses on three main kinds of drivers: heavy ion accelerators, lasers, and z pinches driven by pulsed power. The drivers are expected to be the single most expensive part of the power plant. There are substantial research programs in heavy ion accelerators and in krypton fluoride lasers (KrF) and diode pumped solid state lasers (DPSSLs). The heavy ion fusion program is currently funded through the Office of Fusion Energy Sciences (OFES) and the laser programs are funded through NNSA. There is a smaller, concept exploration program in z pinches which builds on an expanding z-pinch program supported by the NNSA for defense purposes. There are also important IFE programs in target physics, target fabrication (including mass production techniques), target injection, chambers, and focusing systems. These programs are funded through both OFES and NNSA. Despite two different funding sources and despite the relative independence of the various systems, the research programs for the drivers are tightly integrated with the programs in target physics, chambers, and target fabrication and injection.

As noted above, there are many types of targets. In all IFE targets the fusion fuel is compressed before it is ignited. There are two broad methods of compression and two methods of ignition. The fuel is compressed either through an implosion driven directly by the driver beams (direct drive) or by converting the driver energy to x rays that then drive the implosion (indirect drive). The two classes of ignition are hot-spot ignition and fast ignition. In hot-spot ignition the implosion itself compresses and heats a hot spot in the center of the fuel. This spot ignites and subsequently ignites the rest of the fuel. In fast ignition, a separate short-pulse driver heats the fuel to ignition. There is broad international interest in fast ignition but there are also important technical uncertainties.

Chambers also fall into a number of general types. The types currently receiving the most attention are dry-wall chambers, wetted-wall chambers, and chambers in which the wall is protected by thick liquid layers. Often the dry wall chambers contain some gas to protect the wall from x-rays, charged particles, and target debris. The wetted walls use thin liquid layers on the wall or sprays of fluid in the chamber to do the same. Thick liquid layers are used to protect the wall from neutrons as well as from x-rays, charged particles, and target debris.

Although there are many possible combinations of drivers, targets, and chambers, resources do not allow the exploration of all combinations. Each integrated approach puts most of its effort into the combination that currently appears to be most compatible. The various combinations of drivers, targets, and chambers must work together and not all combinations are equally compatible or self-consistent. Currently the laser programs emphasize directly driven targets and dry wall chambers. The heavy ion and z pinch programs emphasize indirectly driven targets and thick liquid wall protection.

4.1.1. Program Roadmap

Several years ago the inertial fusion community developed a program plan or roadmap leading to a demonstration power plant. This plan is shown in Figure 4.1.1.1.

This plan has three phases preceding a demonstration power plant. The first phase contains research elements at the levels referred to as concept exploration and proof of principle.

The IFE community has developed specific milestones that must be met at each level before a concept is ready to advance to the next step.

In Phase II, those drivers that meet their milestones advance, using the IREs, to the point that the driver information, together with the NIF and advanced research in chambers and target technologies provide the information to determine if IFE is ready to proceed to an Engineering Test Facility (ETF). Because it is possible to ignite inertial targets at small scales and yields, the ETF will be designed as a scaled facility to operate at a fusion power level of 200 to 1000 MW, corresponding to a net electrical power output of between 100 and 300 MW. The ETF will provide a test bed for demonstration of all IFE plant systems at reduced scale, including tritium breeding and recovery and power conversion, as well as accelerated materials and component reliability testing. Information from scaled testing of all IFE plant subsystems will be used in decision making to determine if a full-scale IFE demonstration power plant (Demo) should be built. If the decision is positive, the ETF will also provide the information that is necessary to design and build all plant systems in the Demo.

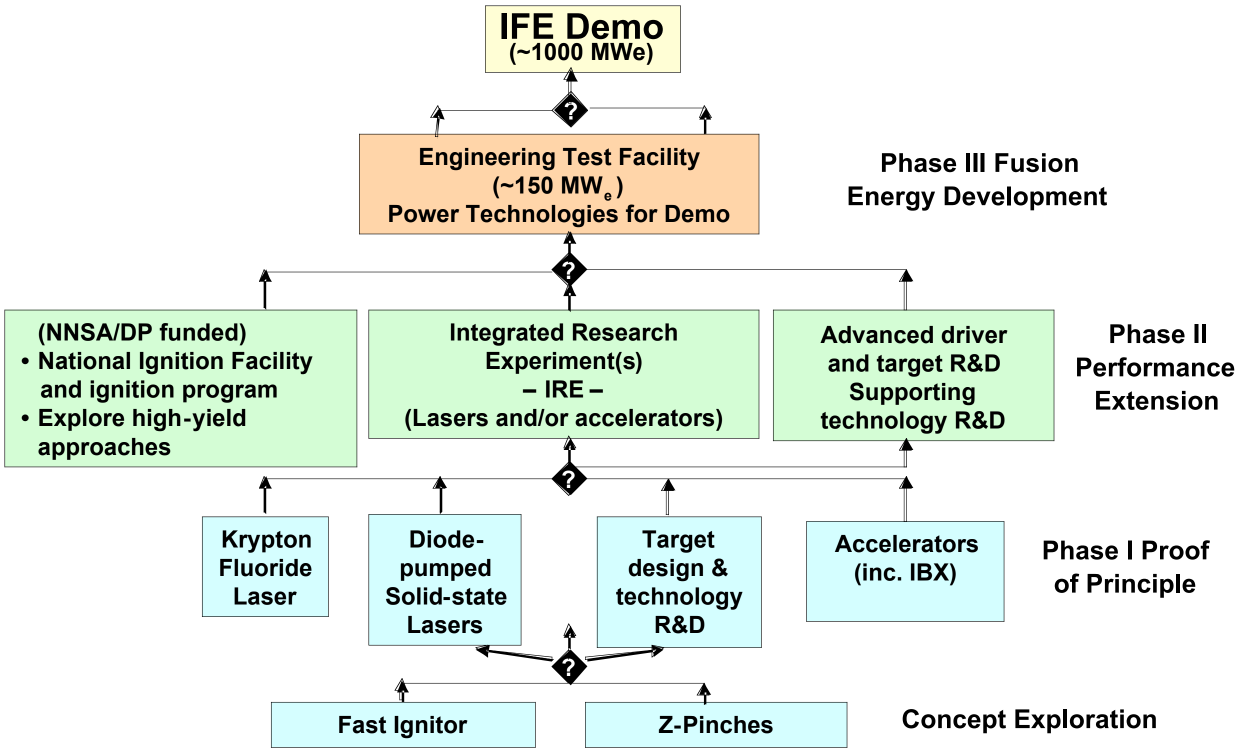


Figure 4.1.1.1 The Inertial Fusion energy Roadmap

One important conclusion of this Workshop is the fact that the entire IFE community (including the proponents and critics of the various IFE approaches) believes that the research programs that have been developed for Phase I are sound and that they address the correct technical issues. There is less agreement about Phase II and some of the quantitative aspects of the milestones needed to advance to Phase II. The various driver programs are advancing at different rates because of funding differences and their relative maturity. The most advanced programs are unlikely to be in position to propose an integrated research experiment for several years. The various disagreements must be resolved by additional workshops and peer review.

Because of the NIF's importance as the main inertial fusion burning plasma experiment, we conclude this section with the expected NIF schedule (Figure 4.1.1.2). As shown, the experimental program will begin soon after first light with the first 4 beams (a quad), currently expected toward the end of FY03. As more beams are added, increasingly complex target experiments will be possible. By the end of FY07, hohlraum symmetry will be adequate to begin high quality symmetry experiments in support of ignition. The full capability of NIF for ignition experiments will be available at the end of FY08. NIF will be capable of testing a variety of ignition target approaches including both direct drive and indirect drive, by moving some of the beams to different chamber ports. NIF will initially be configured for indirect drive. Technology is currently being developed which would allow NIF to deliver as much as 100-200 kJ in a short pulse to demonstrate fast ignition.

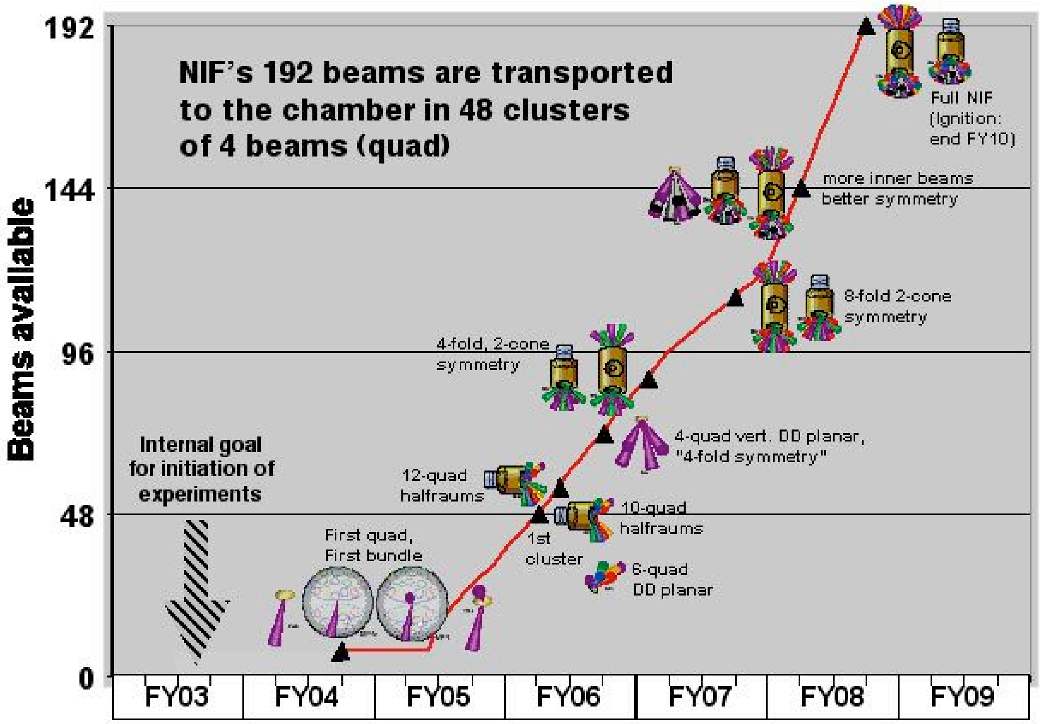


Figure 4.1.1.2 The NIF experimental program is expected to start soon after first light in FY03 with the first four beams (a "quad").

4.1.2 Program Highlights

IFE requires two key ingredients: the capability to ignite fusion fuel and release more energy than was invested, and the capability to reliably repeat this process at high repetition rates, and extract useful energy. The 3 years since the 1999 Snowmass Workshop saw major technical progress in both areas.

Both NNSA and international research made major progress toward the goal of achieving fusion ignition. In the United States, the Advanced Strategic Computing Initiative has improved the computational tools, and increased the computing power, used to design and predict the performance of inertial fusion targets. Target designs for the National Ignition Facility have improved in both robustness and predicted yield, with new design features that have allowed an order-of-magnitude relaxation of capsule surface finishes, and have increased predicted target yields from the base-line 15 MJ to over 100 MJ in some designs. The Omega laser at the University of Rochester has fielded and imploded cryogenic capsules. Internationally, experiments with the Gekko laser in Japan have proved that a fast-ignition laser pulse can be delivered to compressed fusion fuel using the recently developed "cone focus" method. More recently, the Z facility at Sandia National Laboratories has demonstrated z-pinch driven fuel capsule implosions with capsule compression ratios of 14 to 20. Target designs for a z-pinch driver have predicted yields of 0.5 GJ to more than 3 GJ. All three of the major ICF implosion facilities in the United States—NIF, Omega, and Z—are now studying approaches to upgrade beam lines in their laser systems to provide the multi-kilojoule, multi-petawatt performance required to pursue fast ignition experiments.

While the broad-based national and international inertial fusion research programs have advanced the understanding and robustness of target performance, the National Ignition Facility is expected to be the central facility for inertial fusion ignition experiments.

Additional highlights are described below:

The funding for the laser IFE approaches has been approximately equal to the FESAC recommendations and therefore an integrated program has been fielded that deals with laser, target fabrication and chamber issues. Both the KrF lasers and DPPSLs have made impressive progress in durability, efficiency, and technology. KrF lasers have made important progress in the electron beam diode and in simulation tools, the DPSSLs have made important progress in manufacturing the needed crystal laser media. Directly driven laser targets have also made substantial progress. Three years ago there were serious doubts about the hydrodynamic stability of these targets, but recent calculations indicate that the instability may be substantially reduced by shock heating of the ablator using an appropriate pulse shape. In addition experiments on Nike have shown that a thin high-Z layer can reduce laser imprint that seeds instability.

There has been rapid progress in understanding the basic science of dry-wall chambers, target injections systems, and final optics protection for the laser option. Nevertheless there are remaining issues for target survival during injection, first wall materials selection, and final optics durability. Although promising paths have been identified to achieve the IFE requirements, KrF lasers still need to demonstrate substantial improvements in efficiency and durability; and solid state lasers require substantial reductions in the cost of key components. Beam quality is also an issue for solid state lasers because of limited bandwidth with their laser media, but this issue may be mitigated by advances that are yielding more stable targets.

In contrast to lasers, the heavy ion driver program has received substantially less funding than recommended three years ago. The program has dropped almost all accelerator technology development and is concentrating on basic beam science in injectors, transport, and focusing. Three new, modest-scale science experiments have recently begun operations. These three experiments study, separately, the most important issues for accelerator beam dynamics. They include the new Injector Test Stand, designed to study innovative ion sources and multiple-beamlet merging; the High Current Experiment, which addresses issues of high current beam transport; and the Neutralized Transport Experiment, which reproduces the phenomena that control beam focusing in HIF chambers. Fully integrated target designs are now available with the required gain for energy production, and self consistency with current predictions for driver focusing performance.

Separate efforts are addressing HIF chamber and target technologies, and integrated system modeling. Scaled hydraulics experiments have identified nozzle designs that can create all of the liquid jets configurations required for thick liquid chambers, including innovative vortex nozzles that can coat the insides of driver beam tubes with thick liquid layers. For heavy-ion fusion there is now a chamber/target/driver point design where the final focus magnets and chamber structures all have predicted life times exceeding 30 years.

The next step in heavy ion drivers is called the Integrated Beam Experiment (IBX). The IBX is part of Phase I of the development and precedes the Integrated Research Experiment. The IBX would integrate key physics issues at moderate beam current and beam energy in an integrated facility, including injection, acceleration, drift compression, and final focus to a small spot in a target chamber. Focusing intense beams in the fusion environment remains the most important issue.

There has been impressive computational and experimental progress in z-pinch targets and good progress in conceptual power plant designs. Despite recommendations of the Fusion Energy Sciences Advisory Committee, the Department of Energy does not fund this program directly. Producing economical, recyclable transmission lines remains an important issue.

There is broad international interest in fast ignition and increased interest by NNSA. So far the target experiments have been encouraging but the laser-target interaction is more complex (and less understood) than the other approaches and an integrated model (through burn) of the physics is needed. Moreover, a self-consistent approach to power production is still needed.

4.2. Integrated IFE Concepts and Development Plans

IFE has considerable flexibility in the combination of target, driver and chamber types to configure a power plant. The fuel in IFE targets must be compressed and ignited. The two basic types of compression are direct drive, where the beams directly illuminate a fuel capsule, and indirect drive, where the fuel capsule is contained in a hohlraum that coverts the drive energy to x-rays that compress the target. The fuel can be ignited in two ways. In the first, the compression process itself provides the heat to ignite the fuel. In the second, called fast ignition, short pulse (~20 ps) beams ignite a spot on the edge of compressed fuel core. Fast ignition can be used with either method of compression.

At the Snowmass Fusion Study, we considered four categories of driver: lasers, including krypton fluoride (KrF) and diode pumped solid state lasers (DPSSL); heavy ion accelerators (HI); pulsed-power driven z-pinches (Z-pinch); and drivers for fast ignition (FI), which include a short pulse ignitor laser in addition to one of the previous drivers used for fuel compression.

The basic tapes of fusion chamber are dry wall, wetted wall, and thick liquid wall. Dry-wall chambers typically use low-pressure (~10's of mtorr) gas to prevent vaporization of the first wall material (e.g., W coated SiC). Wetted-wall chambers use a thin liquid film or spray (e.g., Pb, PbLi, or flibe) to absorb the target x-rays and debris. Thick-liquid-wall chambers use a neutronically thick flowing blanket (e.g., flibe) to absorb x-rays and debris and also to reduce neutron damage and activation in the chamber structures with the goal of a 30-year life for the chamber.

The US laser, HI and Z-pinch programs have each selected a baseline combination of target and chamber to focus their integrated power plant R&D. These are indicated in Table 4.2.1 along with nominal operating parameters. Note that other driver/target/chamber combinations are possible and several have been the subjects of previous conceptual design studies (e.g. laser and HI with wetted walls). Concepts have been proposed, but not yet developed, for lasers with direct drive targets and thick-liquid-wall chambers. Work is only beginning on power plant consideration for FI, but in principle, FI could be used with the baseline configurations with some added constraints necessary to accommodate the delivery of the ignitor beams (e.g., limits on chamber pressure) and protection of the ignitor laser final optics, which may have to be closer to the target than the typical compression laser final optics. Proposed Development plans for the various approaches to IFE are given in Section 4.2.2-4.2.5.

Driver type	Laser	HI	Z-Pinch
Target type	Direct drive	Indirect drive	Indirect drive
Chamber type	Dry wall	Thick liquid wall	Thick liquid wall
Driver energy	2-4 MJ	6-7 MJ	40-45
Target yield	300-400 MJ	~400 MJ	~3000 MJ
Rep-rate per chamber	5-10 Hz	~6 Hz	~ 0.1 Hz
Chambers/plant	1	1	10
Net power	1 GWe	1 GWe	1 GWe

Table 4.2.1 Nominal Baseline Power Plant Concepts.

4.2.1 Objectives and Metrics for Development Phases

The IFE roadmap (Figure 4.1.1.1) progresses through several phases: Concept Exploration, Phase-1 (characterized by basic concept R&D), Phase-2 (includes the IRE), Phase 3 (includes the engineering test facility (ETF)), and a demonstration power plant. Each of the drivers is at a different stage of development and will progress through theses phases at a different time and rate dependent on funding and success in meeting R&D objectives. Some common, top-level metrics needed to proceed to the next development phase (adapted from the 1999 FESAC-Knoxville Panel on OFES Program Balance and Priorities meeting) are summarized in Table 4.2.1.1. In addition to these generic milestones, each driver type has specific issues to address and objectives to meet at each phase. These are discussed in the following driver-specific development plans.

Table 4.2.1.1 Generic Requirements to Proceed to IRE and ETF Phases of Development.

Requirements to qualify an IRE	Requirements to qualify an ETF
• Resolve key proof-of-principle driver issues (efficiency, reliability, focusability, cost) that are specific to each approach. • Adequate gain IFE target designs with 2-D hydrostability for plausible beam non-uniformities. • Plausible pathways for target fabrication and injection. • A chamber design concept that is self-consistent with target illumination geometry, final focus and beam propagation, chamber clearing, and adequate lifetime.	• Adequate target physics data from NIF and other ICF facilities on implosion symmetry and capsule/fuel layer smoothness • IRE-level confirmation of scaling for ETF driver cost, lifetime at rep-rate, and adequate efficiency • Test injection of prototype ETF targets in simulated fusion chamber environment (could be separate facility) • Final focus lifetime data for sufficient number of shots under ETF prototypical conditions • Experiments that resolve issues of beams driving ETF targets (e.g., beam balance, symmetry, imprint, filamentation, chamber plasma/gas interaction) • Experimentally validated chamber models that show acceptable clearing times; evidence of acceptable chamber materials reliability, coolant chemistry control, and tritium control and recovery.

4.2.2 Proposed Laser IFE Development Plan

This section primarily represents the views of the Laser IFE Program advocates. Some of comments they received at Snowmass 2002 are highlighted in Section 4.2.6.

Laser drivers for IFE are being developed under the NNSA funded High Average Power Laser (HAPL) Program. Two very different approaches to providing the beam energy at the required pulse repetition rate are being pursued: Krypton Fluoride (KrF) gas lasers at the Naval Research Laboratory (NRL) and Diode-Pumped Solid State Lasers (DPSSL) at Lawrence Livermore National Laboratory (LLNL). Both have the potential to meet the fusion energy requirements for rep-rate, efficiency, durability and cost. Both lasers expect to have “first light” during CY 02. The R&D for laser driven power plants also includes many participants in chamber and target R&D. The progress and proposed R&D plan are summarized here.

Introduction. The HAPL program is carrying out an R&D program to develop fusion energy with laser drivers and direct drive targets. This is an integrated research program that develops the main components simultaneously. This ensures that the key interface issues are properly addressed. In this approach, an array of high-energy laser beams symmetrically and directly illuminates a cryogenic target that has been injected into a chamber. The deuterium-tritium fuel in the target undergoes thermonuclear burn and the energy is used to generate electricity. This is illustrated in Figure 4.2.2.1.

The attractiveness of this concept lies in its inherent simplicity, its separable architecture, and the modular nature of the laser driver. The targets are spherical shells, which in principal can be fabricated in a single droplet generator. Thus, they naturally lend themselves to automated, low-cost production. Moreover, none of the target components need to be recycled. The first wall is a passive structure that does not have to hold vacuum. Not having to worry about vacuum integrity allows more choices for the first wall material, such as advanced composites or two-component structures. It also allows the wall to be made in individual sectors that can be replaced during the plant lifetime. The separable nature of the power plant allows the principal components to be developed separately before being integrated into the system. Just as importantly, it allows economical upgrades as new technologies are developed. The laser is modular, and would consist of a number (about 60) of identical beam lines. Hence it is only necessary to develop one of these lines to develop the entire system. All of these factors significantly reduce the development costs for this approach.

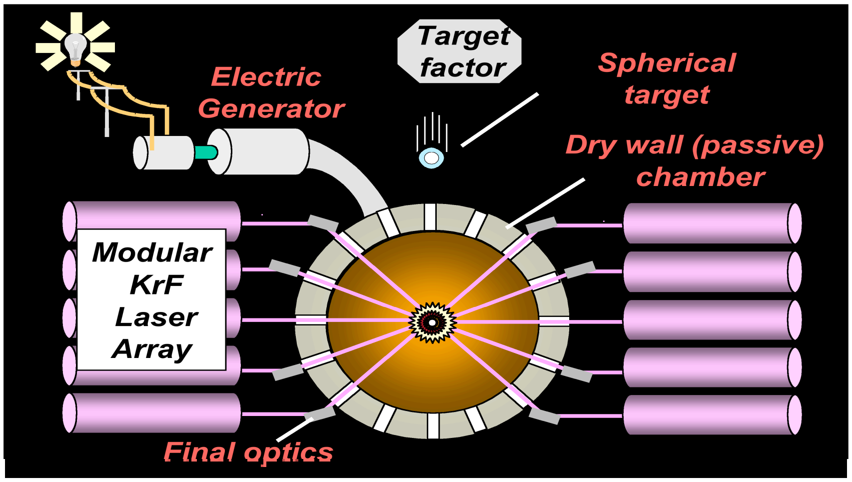


Figure 4.2.2.1 Conceptual Drawing of a laser IFE Power Plant

This program leverages off the target design, laser development, and high energy density physics research carried out in DOE/NNSA Defense Programs, as well as the materials and component research carried out in the DOE/Office of Science fusion program. Thus it capitalizes on two main research thrusts in DOE to provide a solution to our nation's long-term energy needs.

The Laser IFE Program follows three key overarching principles: 1) It is a coordinated, integrated effort; all the components of Laser IFE are developed in concert with one another, 2) the program addresses issues that are unique to Laser IFE, and leaves generic issues (e.g. blankets, some materials, breeders, safety, etc.) to the much larger fusion program and to future research, and 3) the program stresses experimental validation and predictive capability.

Progress. While there are still science challenges that must be met to realize this concept, there have also been sufficient advances in target design, target experiments, lasers, and associated technologies to make this a front-runner for fusion energy. Progress in the various areas is summarized here.

Target Designs. The Laser IFE Program has developed target designs, based on codes that are being benchmarked with experiments, which show gains of 120-180. This is sufficient for a fusion power plant. These are 1-D calculations. Integrated 2-D designs are under development and look promising. The ablator of current high-performance targets is a layer of low-density foam with DT wicked into it. The foam can significantly increase the laser absorption. These designs also make use of the experimental observation that a thin high Z coating outside the target significantly reduces the laser imprint, and hence mitigates the growth of hydrodynamic instabilities.

KrF Laser. The Electra KrF laser uses double-sided electron beam pumping of the laser gas. NRL has commissioned a "first generation" pulsed power facility that is being used to develop the laser components. This facility produces two 500 kV, 100 kA, 100 ns electron beams. The pulsed power system can run continuously at 5 Hz for five hours. This is unprecedented for a system this size. We have used our experimentally validated beam propagation/deposition codes to develop a hibachi concept that can meet the efficiency requirements. This is achieved by contouring the electron beam to "miss" the foil support structure ribs. A cooled foil structure to achieve the durability requirements is under construction. NRL has developed an advanced KrF kinetics model that predicts the results of existing experiments and is now being used to design future systems. They also demonstrated an advanced laser triggered solid-state switch that will be the basis for a durable, efficient pulsed power system.

DPSSL. The "Mercury" laser system employs four technological advances: efficient and reliable diodes, high quality Yb-doped laser crystals in the large sizes crystals needed for this system, active cooling with near-sonic helium gas flow for rep-rated operation, and an angular multiplexed, relay imaged architecture. All of these have been demonstrated. The dual-ended, longitudinal pumping design allows for more uniform pumping and thermal loading on the crystals than traditional side pumping schemes. The power amplifiers are four-passed to extract the stored energy into the laser beam. The first laser head assembly has been fabricated and installed. This includes four 80 kW diode arrays, relay telescopes, low distortion gas cooling of the amplifier head, and the reverser hardware which allows the beam to be re-injected for multi-pass operation. "First light" has been achieved on Mercury, using four of seven planned crystals in the amplifier head. The system has demonstrated 11.8 J at a rep-rate of 0.1 Hz.

Target Fabrication. In target fabrication, program participants have investigated the properties of several high-Z coatings. Measurements show an Au-Pd alloy meets the requirements for DT permeation times, and has high IR reflectivity to help the target survive as it traverses the hot chamber. An advanced divinyl-benzene foam that can meet the requirements for low oxygen content and straightforward over coating has been developed and shells have been made from this material. Economic analysis shows the targets can be made for less than $0.16 each, which is below the economic requirement from power plant studies.

Target Injection. For target injection, researchers are performing experiments to determine the thermal response and mechanical properties of solid DT. They have demonstrated the concept of a separable sabot to protect the target during acceleration, and have started construction of a system to study injection and tracking. This injector is designed to accelerate any IFE target (both indirect and direct drive) and thus is important for the entire inertial fusion program.

Final Optics. In final optics, experiments have established that, at least in small laser spot sizes, a grazing incidence aluminum mirror is both highly reflective (>98%) and can significantly exceed the required laser damage threshold (> 50 J/cm^2 vs. the required 5-8 J/cm^2).

Chambers. In fusion chamber designs, models for how the chamber conditions evolve between shots are being developed. We have established an operating window for target yield, chamber radius, and chamber gas pressure that will avoid first wall vaporization, allow target injection without compromising the frozen DT fuel, and operate at a reasonable efficiency. Long-term material behavior under alpha bombardment is an open issue. Consequently, experiments have been started to exposed candidate first wall materials to ions and x-rays at fusion relevant fluences, spectra, and temperature. Past experiments used the Z (x-rays) and RHEPP (ions) facilities at Sandia. These will be augmented with a new repetitive x-ray source at LLNL, as well as the triple ion beam facility at ORNL.

Power Plants. Power plant studies have shown that this approach can be economically attractive.

Proposed Development Plan. The Laser IFE Program has formulated a three-phase program to develop laser-fusion energy. (This structure has been agreed upon by the fusion community.) Specific milestones must be met to go to the next phase:

Phase I (1999 to about 2005): The present "Proof of Principle" program.

Lasers. Develop technologies that can meet fusion energy requirements for efficiency (> 6%), repetition rate (5-10 Hz) and durability (> 100,000,000 shots continuous). The lasers will demonstrate the beam quality and pulse shaping needed for fusion. The laser technologies employed must scale to reactor size laser modules and have attractive costs for commercial fusion energy.

Final Optics. Achieve laser induced damage threshold (LIDT) requirements of more than 5 Joules/cm^2. Demonstrate this in large area optics. Develop a credible final optics design resistant to degradation from the neutrons, x-rays, gamma rays, and energetic ions.

Chambers. Develop a viable first wall concept that is economically feasible for a fusion power plant. Demonstrate through modeling chamber gas cooling and equilibration that is sufficiently rapid.

Produce a chamber point design for an IFE system that is self consistent with final-optics reliability and target injection and tracking.

Target Fabrication. Develop methods to mass produce cryogenic DT targets that meet the requirements from the Target Design codes. Combine these methods with established mass production costing models to show targets cost will be less than $0.25.

Target Injection/Tracking. Build an injector that accelerates targets the equivalent distance of the chamber (6.5 m) in less than 60 milliseconds. Demonstrate target tracking with sufficient accuracy for a power plant (+/- 20 microns).

Target Design. Develop credible target designs, using 2D and 3D modeling, that have sufficient gain (> 100) and stability for fusion energy. Benchmark underlying codes with experiments on Nike & Omega. Integrate design into needs of target fabrication, injection and reactor chamber.

Phase II (approx 2006-2012): The Integrated Research Experiments (IRE). These bring together the key components. There will be two facilities, a laser facility and a target facility, experiments and modeling in target design, a full-scale power plant study, plus material evaluation.

The Laser Facility component of the IRE would include the following R&D objectives:

Lasers. Build a full-scale (power plant sized) laser beam line using the best laser choice to emerge from Phase I. The beam line will demonstrate all the driver fusion energy requirements, including efficiency, rep-rate, durability and cost basis. In the case of the KrF the beam line would have energy of 50-100 kJ. This would be the size of a power plant beam line. In the case of the DPPSL, the beam line would have energy of 4-12 kJ. This would be the size for a module for a power plant beam line.

Final Optics/Target Injection. Demonstrate the full-scale beam line can be steered to hit a target that is repetitively injected into a chamber, with the required precision and optics LIDT durability.

Chamber Dynamics. Evaluate chamber-clearing models using a “Mini Chamber” in the Laser IFE facility Chamber Materials. Study candidate wall and/or optics materials.

The objectives of the Target Facility component of the IRE are:

Target Fabrication. Demonstrate mass production of fusion class targets.

Target Injection. Demonstrate injection into a fusion chamber environment.

Other Phase II R&D and objectives include:

Power Plant Design. Produce a credible design for a laser fusion power plant that meets the technical and economic requirements for commercial power.

Chamber and Final Optics. Validate candidate materials/structures in a non-fusion environment using available exposure facilities (ions, x-rays, debris). We acknowledge that much of this work must be left to the ETF.

Target Design. Perform integrated, high resolution, 3D target modeling, and validate design codes with target physics experiments at fusion scale laser energies on the NIF, and at rep-rate, on the Laser Facility.

Phase III (2012 to about 2020): The Engineering Test Facility (ETF). This full-scale laser facility (1.5-2.5 MJ) would demonstrate repetitive high fusion yield. The ETF would also evaluate components and demonstrate fusion power.

Major Participants are:

Government Labs: Naval Research Laboratory, Lawrence Livermore National Laboratory, Los Alamos National Laboratory, Oak Ridge National Laboratory, Sandia National Laboratory, Princeton Plasma Physics Laboratory, Idaho National Engineering and Environmental Laboratory, Argonne National Laboratory.

Industry: General Atomics, Titan-Pulse Sciences Division, Schafer Corp, Science Applications International Corp, Northrop-Grumman Corp, Coherent, Inc, Commonwealth Technology, Inc, Onyx Corp, Crystal Systems.

University: University of California San Diego, University of Wisconsin, University of California Los Angeles, University of California Berkeley, University of California Santa Barbara, University of Rochester Laboratory for Laser Energetics, Georgia Institute of Technology.

4.2.3 Proposed Heavy Ion IFE Development Plan

Introduction

This section primarily represents the views of the Heavy Ion IFE advocates. Some of comments they received at Snowmass 2002 are highlighted in Section 4.2.6.

The Snowmass driver, chamber and target working groups reviewed the requirements, issues, progress, and next steps in research and associated budget needs for inertial fusion energy driven by heavy-ion accelerators with suitable targets and chambers, an integrated national program called heavy-ion fusion (HIF). For the past two decades the U.S. Department of Energy and several peer reviews have recommended heavy-ion fusion as an attractive candidate approach to inertial fusion energy (IFE). Motivations for HIF include (a) high energy particle accelerators of MJ-beam energy scale have separately exhibited intrinsic efficiencies, pulse-rates, power levels, and durability required for IFE (noting that HIF requires lower range and higher peak power); (b) HIF targets driven indirectly by x-rays within hohlraums benefit in many ways from the large target physics research effort led by the NNSA program in the U.S.; (c) thick-liquid protected chambers with potential 30 year lifetimes are compatible with the indirect-drive target illumination geometry; (d) cryogenic-DT fuel capsules injected into hot chambers are protected by the surrounding hohlraum when injected into hot chambers; (e) magnets used to focus intense heavy-ion beams to the target could avoid direct line-of-sight damage from target debris, neutron and gamma radiation, and (f) several heavy-ion power plant studies over the past 10 years (Osiris, Prometheus-H and HYLIFE-II) showed that heavy-ion power plants would have attractive economics and environmental characteristics. Over the past decade, U.S HIF research has been carried out by LBNL, LLNL, PPPL, GA, NRL, SNL, and several universities at funding levels varying between $6-12 M/year nationally. There are some important but smaller contributions to HIF beam science from nuclear-physics research facilities in Europe, Japan, and Russia.

There are significant issues to the achievement of heavy ion fusion energy:

- Beam quality- emittance growth, electron/gas interactions at high current, evolution of beam distributions over long path lengths. Maximum-average beam current densities achievable in injection, transport and acceleration experiments for future multi-beam arraysLongitudinal bunch compression with velocity-tilt after accelerationBeam perveance (beam space charge potential over ion kinetic energy) and neutralization limits to final focus and chamber transportMultiple-beam interactions -electrostatic and magnetic beam-beam coupling in the accelerator, and when converging together in the chamberImproved accelerator technology for the IRE and drivers: compact multiple-beam injectors and superconducting magnet arrays, solid state pulsers, low cost/loss ferromagnetic core materials, high gradient/low cost insulators.Heavy-ion targets that can be mass produced at low cost and injectedHeavy-ion liquid chamber at power-plant scale allowing >5 Hz pulse rates.Assuming increased funding for

fusion in the U.S. can be obtained in the near future, the heavy-ion fusion program is technically ready for a much larger effort for the following reasons:

- Published HIF target designs exist with more than adequate gain and 2-D hydrostability for IFE.
- The HIF Program has successfully completed a number of scaled beam experiments at low current but with relevant dimensionless perveance (beam space-charge potential over ion kinetic energy) that show adequate brightness for IFE in present experiments on injection, transport, and final focus with neutralizing electrons.

- The HIF Program has an HIF power plant design based on an induction linac that meets the pulse rate, cost, and efficiency required for IFE that is consistent with requirements for HIF target beam energy, peak power, pulse shape, ion range, and focal spot size, and which is consistent with a thick-liquid protected chamber, target illumination geometry, and propagation through hot chamber vapor, and which is also consistent with detailed particle-in-cell simulations for final focus and plasma interaction in the chamber (Appendix 1-HIF driver/final-focus/baseline physics and systems analysis).
- Experiments with water jets scaled to be hydrodynamically-equivalent to a power plant liquid chamber have shown both the required smooth jets and the principle of oscillation for rapid clearing after shock disruptions.
- We have identified feasible parameter ranges and available technology to commence conceptual design of a ~$50 M Proof-of-Principle Integrated Beam Experiment (IBX), which could benchmark integrated beam models for a heavy-ion beam that is injected, accelerated to ~10 MeV, bunch compressed, and focused into a test chamber with preformed plasma to neutralize the beam, at sufficient currents (>0.3 A) to study electron/wall effects.

Phase-I research for HIF accelerators, targets and chambers includes several specialized high current experiments described below, plus IBX, and supporting theory, small-scale experiments, accelerator technology, target design, fabrication and injection R&D. Phase-II would include a follow-on accelerator/ focus/chamber facility called an Integrated Research Experiment (IRE) costing approximately $150 to 300 M. The IRE, together with target physics data from the NNSA program and other target and chamber R&D specifically for IFE, would provide the basis for a decision to upgrade the IRE to an Engineering Test Facility (ETF) (phase III), capable of testing chambers at high average fusion power. Assuming a suitable chamber, target system and driver interface is developed in the ETF-II, the chamber and target systems could be again upgraded to a net tritium and net power DEMO in a final phase IV.

The major elements of the HIF development strategy include:

(a) Complete initial experiments with high current beams (0.1 to 1 amp) in transport, focusing and injection through FY04. These experiments will extend successful results with earlier scaled experiments at low current to higher current (0.1 to 1 A). A 2 MV injector experiment has already produced adequate injection current at 800 mA, but current work seeks to improve the beam uniformity and brightness.

(b) Build and operate IBX, an extended Proof-of-Principle facility (~ $50 M), to validate integrated beam models for injection, acceleration, drift compression, final focus and chamber transport. This will improve our predictive capability for final focus for the next IRE step.

(c) Build and test a multiple-beam IRE facility (~$150-300 M) to validate prototype driver technology and cost scaling, and also to demonstrate multiple-ion-beam aiming, pulse shaping and coupling into scaled hohlraums experiments.

(d) Assuming IRE tests with multiple ion beam driven targets are successful, and assuming the IRE were built on a site capable of handling modest levels of fusion power, the IRE beam energy (both in number of beams and ion kinetic energy), would be upgraded to drive an ETF. After testing several candidate chambers, the ETF chamber and pulse rate could be upgraded to a net power DEMO. This IRE/staged ETF on a qualified site would greatly reduce the overall development time for IFE based on HIF.

Integrated Beam Experiment (IBX) (~$50 M construction cost, beginning FY04-05)

Figure 4.2.3.1 shows a conceptual design of the Integrated Beam Experiment (IBX), and Table 4.2.3.1 lists the parameter range considered for IBX from a national IBX workshop held October 9-10, 2001. The narrow range given in Table 4.2.3.1 reflects the consensus of the IBX workshop, and a further narrowing of parameters will be done before preconceptual and conceptual designs, which require additional funding in FY03. The IBX mission is to validate integrated beam models for at least a single beam that goes through all important steps in a driver: injection, acceleration, bending, drift compression, final focus and propagation in chamber plasma to the focus spot. The scale of IBX, 5 to 20 MeV, is set by requirements to have sufficient initial beam current, line-charge density and beam potential (~ >300 mA for beam potentials ~> 1 kV) to include electron cloud effects, and then to longitudinally compress such a high current beam against its space-charge forces by a factor of 10 or more. This will put IBX at a significant point between present experiments and the IRE in the space of transport length versus line-charge density (Figure 4.2.3.2). The integrated beam models benchmarked with data from IBX would then allow higher confidence in predicting the corresponding transport limits and focal spot size capability for the IRE, and would lead to a more optimum and capable IRE design.

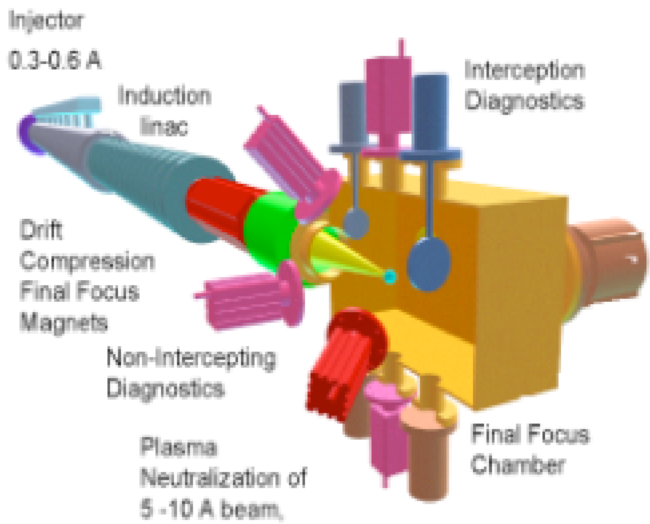


Figure 4.2.3.1 The Integrated Beam Experiment (IBX) will compare measurements and simulations of a high line-charge density beam undergoing all the beam manipulations in a driver: injection, acceleration, drift compression, final focusing and chamber propagation.

The IBX mission is thus primarily a beam physics mission, and to expedite that important mission, this plan proposes to use existing accelerator technology, which is available to build IBX "now", while in parallel support the improved technology R&D over several years that is required for a cost-effective IRE. Some IRE-relevant upgrades of selected IBX components are planned and will be discussed later on. The HIF Program is confident that the IBX construction cost goal of ~ $50 M can be met because of past detailed design studies of machines of similar scale, because it would use existing technology (e.g., use of spark gap switches such as used in DARHT for limited number of pulses), and because the IBX mission can be met with much shorter pulse widths (a few hundred ns) compared to DARHT (2 ms).

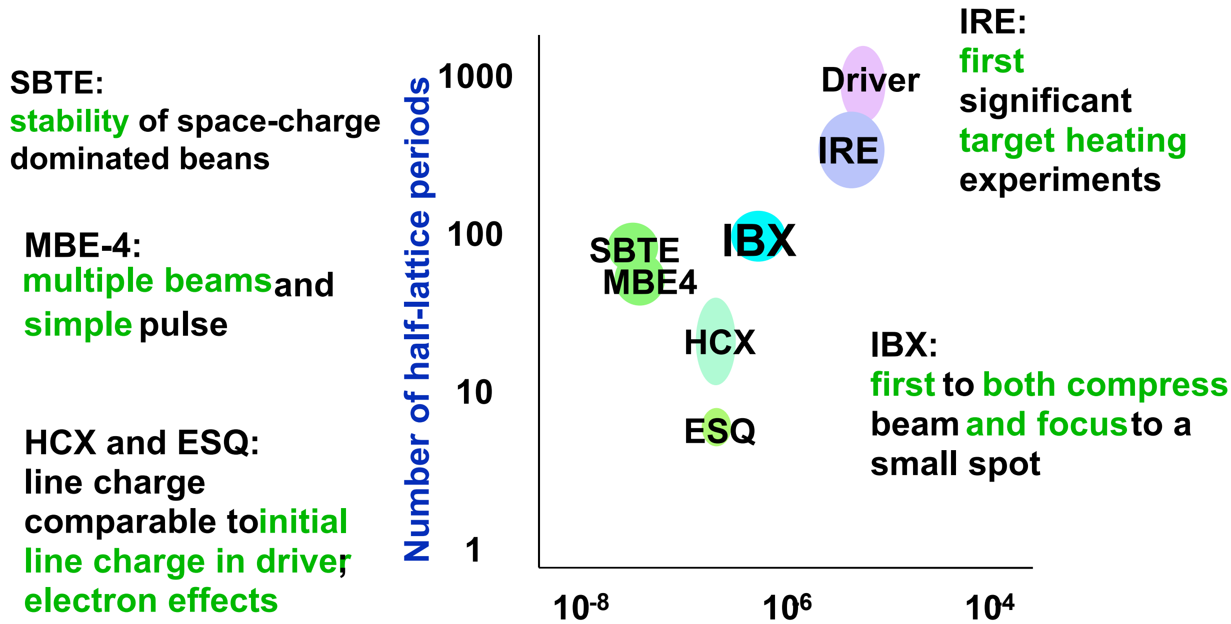


Figure 4.2.3.2 IBX would significantly advance HIF between existing experiments and the IRE/driver in terms of transport length and bean line charge density.

IBX would support a variety of intense ion beam dynamics experiments, including:

1. Integrated physics
   a) Inject, accelerate, compress, and focus a heavy ion beam at high line charge density.
   b) Simulate a 3D beam from source-to-target, predicting final spot radius and profile.

2. Longitudinal physics
   a) Physics of drift compression and space charge stagnation: Measure residual velocity tilt and spread after compression by factor of ~10.
   b) Physics of longitudinal heating during acceleration and compression.

3. Transverse/longitudinal coupling physics
   a) Matching and beam control with velocity tilt and acceleration.
   b) Time dependent (upstream) final focus correction physics.
   c) Bending physics.
   d) Transverse/longitudinal temperature anisotropy instability.
   e) Beam "end" physics.

Metrics

To manage risk for the higher costs of successive steps, each step must satisfy a set of quantifiable performance metrics to qualify the following step. In addition to the metrics listed in Section 4.2.1, HIF has specific metrics. Table 4.2.3.2 presents additional metrics for the extended HIF proof-of-principle phase of R&D, which includes the IBX, to qualify an IRE.

	Team 1	Team 2	Team 3	Consensus
ION	K^+ Ar^+ Rb^+ Kr^+	K or Ar	K	K^+ A ~ 40
N	1 But need 4 for neutraliz.; ≥ 9 for accel. issues	1 But design to allow 4-9 retrofit	1 Upgrade to 4	1 Upgrade to 4 or more
E_{final}	≥ 10MeV	≥ 10MeV	10MeV or more	10MeV-20 MeV
□$_{initial}$	1-2□s Compress x2 in Accel. x10 Drift	0.5 - 1□s flattop	Adjust to fit injector with suitable ion source	0.2□s - 1□s flattop (Injector)
□$_{final}$	□$_i$ ~ 0.2 □ C/m □$_f$ ~ 2 □ C/m K ~ 10^{-3}	□$_i$ ~ 0.2 □ C/m		□$_f$ ~ 1 □ 2 □ C/m K_f | | 10^{-3} compress ~ x10
Focus Technology	SC + , - Pulsed + , -	Magnetic. Decide SC vs pulsed based on HCX, NTX	No real cost difference in SC or pulsed, but pulsed easier	Magnets

- Cost goal < ~40-50 M$ may be achievable with short pulses-more work needed
- Goal is integrated beam science, not fixed MeV, current, or Joules
- Use existing technologies- injector, induction cores, sc transport magnets

Table 4.2.3.1 A consensus for IBX parameters emerged from the October 20001 IBX Workshop

To address a wide range of issues cost-effectively, and to keep an integrated approach so that accelerators, chambers and targets will work together to achieve successful heavy-ion fusion energy, a balanced portfolio approach is used, so that the plan includes:

- Continued operation of smaller experiments during construction of the major facilities IBX and IRE to improve major facility operation, diagnostics, and staffing.
- Early target physics data using existing laser facilities and intense micro-heavy-ion beams generated by short-pulse lasers (included in "HIF Target Experiments")
- Innovative high-gradient accelerator R&D to further improve IRE capability/cost and competitiveness of HIF power plants.
- Sufficient HIF target physics, fabrication, injection and chamber R&D to support sophisticated/integrated IRE target experiments sufficient to qualify an upgrade to ETF.

In constructing this plan, several basic guidelines and principles were adhered to in order to maximize likelihood of success of heavy-ion fusion energy in the long run:

- To maximize attractiveness x probability of success, this plan allocates the majority of effort (~90%) on critical issues for a selected baseline HIF scenario based on induction linacs, indirect-drive distributed radiator targets, ballistic neutralized final focusing, and thick-liquid protected chambers. The remaining 10% is reserved to explore advanced accelerator, target and chamber options that can potentially make major reductions in IRE/driver cost and complexity.
- Smaller experiments (at slightly reduced funding levels) will be continued during constructing of the major facilities (IBX and IRE), to provide data to better utilize and optimize operation of the major facilities, to develop diagnostics and modeling that would be needed, and to train young scientists and engineers to participate in the major facilities. Thus the annual construction funding of a major facility in any given year would be less than 50% of the total HIF program budget.
- Sufficient funding growth for HIF target and chamber science and technology is required to support integrated driver-chamber and target experiments in the IRE, and so that the target and

chamber R&D programs can take major responsibility for target and chamber experiments using the IRE beams. This is to insure that a heavy-ion IRE will address all of the key issues required to qualify for an upgrade to an accelerator-driven ETF.

Table 4.2.3.2 Metrics for the HIF extended proof-of-principle phase to qualify an HIF-IRE

- Transport with low emittance growth for aperture fill factors > 0.5 and for > 40 - 80 lattice periods
- Average acceleration gradient sufficient for an IRE
- Compact multi-beam injectors with normalized emittance < 1 p mm-mr, and overall average current density > 30 A/m^2 adequate for a multi-beam IRE
- Final focus to near-emittance-size spots after > 5 x longitudinal bunch compression, beam perveance > 2 x 10^{-4}, and >90% plasma neutralization.
- End to end simulation of a full scale driver
- Affordable technology for an IRE: low loss cores ($5/kG), high gradient insulators (0.01 $/Volt), solid state pulsers (< 10^{-5} $/watt), SC quad arrays @ (<10$ kA-m).

4.2.4 Proposed Z-Pinch IFE Development Plan

Introduction

This section primarily represents the views of the Z-Pinch IFE advocates. Some of comments they received at Snowmass 2002 are highlighted in Section 4.2.6.

Z, the world’s most powerful x-ray source for z-pinch driven targets, produces 1.6 MJ of x-rays at a power level of 230 TW in a pulse of about 6 ns. Recent capsule compression experiments on Z have produced capsule convergence ratios up to 14-20, and symmetry experiments at the few % level are in progress. The electrical conversion efficiency on Z from wall-plug to x-rays is very large (□15%), and this can probably be optimized to 25%. These impressive results mean that z-pinches are particularly attractive for inertial fusion energy, provided that a suitable method for rep-rated standoff (separation of target and driver) can be devised. Although several concepts for repetitively replacing the final magnetically insulated transmission line that connects the driver to the target have been proposed (e.g., liquid metal electrodes, inverse diodes, etc.), the simplest and most robust is the Recyclable Transmission Line (RTL) concept. In this concept, an RTL is made from a solid coolant (e.g., FLIBE), or a material that is easily separable from the coolant. The RTL/target assembly is inserted through a single opening at the top of the power plant chamber. The shot is fired, portions of the RTL are vaporized and end up mixed with the coolant to be recycled, the upper remnant of the RTL is removed, and the cycle is repeated. Since pulsed power energy has the lowest cost in $/Joule of all IFE drivers, and since some time is needed to change out the RTL for each shot, the present strategy for Z-Pinch IFE is to use high yield (□3 GJ/shot) and low repetition rate (□0.1 Hz). This paper describes a plan for development of z-pinches for energy.

Progress

For Z-Pinch IFE, recent developments in the areas of (1) IFE target design, (2) capsule compression experiments on Z/ZR, (3) rep-rated pulsed power, (4) RTL optimization, (5) manufacturing/cost analysis, and (6) optimization of the overall Z-Pinch IFE power plant concept are summarized.

Target Design. Three target concepts are being developed for potential use in Z-Pinch IFE. The z-pinch driven hohlraum (ZPDH) target uses two z-pinches to drive a hohlraum between them (Figure 4.2.4.1). This approach is the most conservative, and easiest to diagnose. The dynamic hohlraum (DH) target uses a single wire array to collapse on a foam that contains the capsule. A strong radiating shock forms that heats the capsule while the imploding wire array acts as a converging hohlraum wall. Also being investigated is the fast ignition (FI) target concept, in which the z-pinch drives a cone compression of the target fuel, and a FI laser ignites a hot spot.

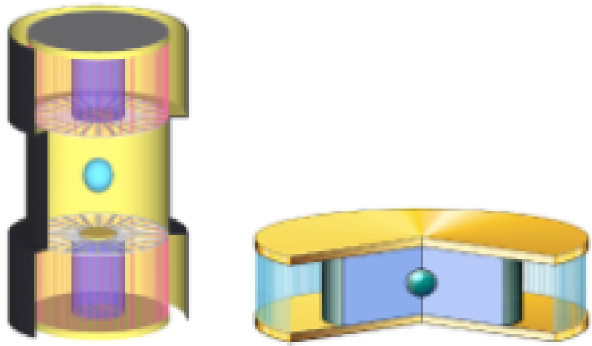


Figure 4.2.4.1 Z-Pinch targets: z-pinch driven hohlraum (left) and dynamic hohlraum (right)

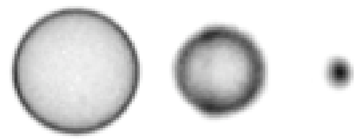


Figure 4.2.4.2 Capsule compression experiments on Z.

Target Experiments. Capsule compression experiments on Z are progressing rapidly for all three targets (ZPDH, DH, FI). ZBL backlighter pictures of ZPDH target implosions on Z show capsule compression ratios of 14-20 (Figure 4.2.4.2). DH experiments on Z show an axial view of a compressed core with a compression ratio of ~10. Cone compression experiments on Z have shown a radial compression of the hemispherical capsule by a factor of 2.

Pulsed Power. Rep-rated pulsed power is needed for Z-Pinch IFE. Rep-rated pulsed power for IFE can be achieved by using (i) fast cycle Marx/water line technology, (ii) magnetic pulse compression technology as in RHEEP II (Repetitive High Energy Pulsed Power) at SNL, or (iii) the new LTD (Linear Transformer Driver) technology. The LTD technology uses direct charging of a ring of low inductance capacitors to drive an inductive voltage adder cavity. The technology eliminates oil storage, water storage, and the usual vacuum insulator stack. A recent study for replacing Saturn (which operates at 10 MA) with LTD technology shows that the accelerator would occupy about _ the area of Saturn. This LTD would be the IRE scale LTD as shown in Figure 4.2.4.3.

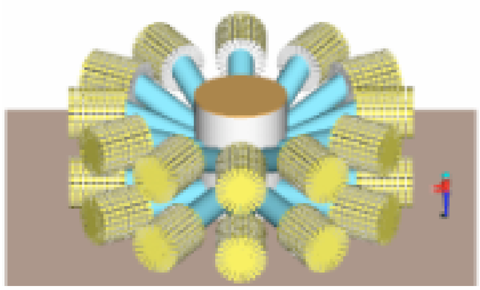


Figure 4.2.4.3 IRE scale Linear Transformer Driver (LTD).

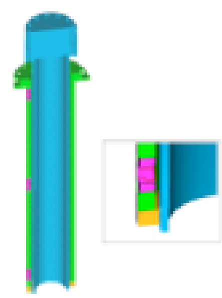


Figure 4.2.4.4 Recyclable Transmission Line (RTL) on Saturn

Recyclable Transmission Line (RTL). The RTL must be optimized for low inductance, proper electrical turn-on, adequate electrical conductivity, low mass, structural integrity, proper vacuum and electrical connections, and other properties. From a pulsed power viewpoint, the most worrisome issues are electrical turn-on and electrical conductivity. The turn-on issue was addressed in experiments on Saturn at the 10 MA level in 2000. An RTL (Figure 4.2.4.4) was designed to have power-plant-level current density and electrical stresses (30 cm height, 8 cm diameter), and the uniformity of electrical turn-on was measured with a series of B-dots. No current loss was observed for any materials tested (tin, aluminum, stainless-steel indicating excellent electrical turn-on and uniformity. A second set of experiments was designed to study the low-mass limit and electrical conductivity of thin electrodes of candidate RTL materials. These experiments on Saturn were completed in 2001 for a series of materials (25 μ mylar, and 50μ, 100μ, 250 μ carbon steel).

Conductivities deduced from the experimental results scale to show that an RTL with mass 80 kg would have small resistive losses (□10 %).

Manufacturing/Cost Analysis. Costs for the RTL/target assembly are an important economic consideration for Z-Pinch IFE. For electricity at 5¢/kWh, about 25¢/target is normally allowed for IFE systems with yields of □0.4 GJ. For yields of □3 GJ, this scales to allow about $1.90 per RTL/z-pinch/target for Z-Pinch IFE. Initial estimates from the SNL Advanced Manufacturing Group for forming the RTL and making the wire array (based on $3x10^7$/year, 0.1 Hz per chamber with 10 chamber per power plant), and assuming the capsule costs 25¢, gives a total cost of $1.80, which is already in the acceptable range. It may also be possible to replace the wire array with a foil for IFE targets, which should simplify the process and lower the cost further.

Z-Pinch Power Plant. The first Z-Pinch Power Plant study has produced a plausible scenario for RTL operation, including the vacuum and electrical connections. The chamber has a single opening at the top for the insertion of the RTL/target assembly, and uses thick liquid walls. As shown in Figure 4.2.4.5, the RTL/target is inserted, the shot is fired, the lower portion of the RTL is sheared off, and the process is repeated.

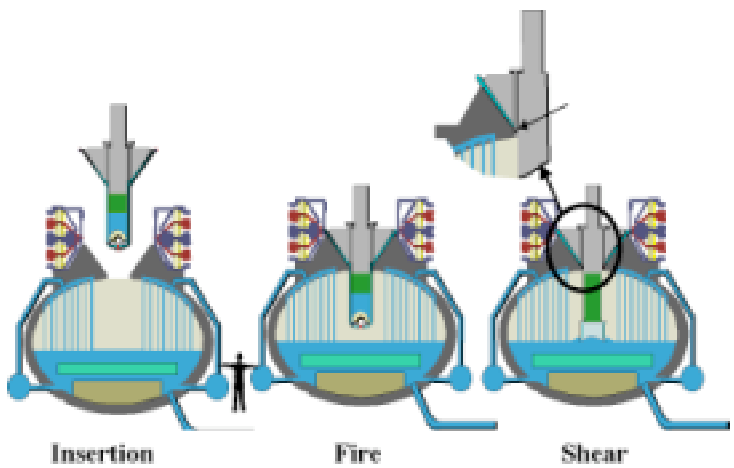


Figure 4.2.4.5 Concept for repetitive replacement of RTL for Z IFE.

Development Plan. Z-Pinch IFE is being developed as an integrated system, including targets, rep-rated driver, standoff (RTL), and power plant concept. A series of integrated, rep-rated, scaled RTL cycle experiments at 0.1 Hz is planned, as noted in Table 4.2.4.1. For the past three years, Z-Pinch IFE has been supported through SNL internal funds at the CE level to explore RTL and power plant concepts. For the next phase (PoP), rep-rated driver development and scaled integrated RTL cycle experiments will be initiated. Figure 4.2.4.6 shows the Z-Pinch IFE Road Map and Figure 4.2.4.7 shows the Z-Pinch IFE Development Path.

Single Shot Facilities. The Z-Pinch IFE Road Map combines the single-shot development of high yield (funded by DOE NNSA DP) with the rep-rated development of high yield for IFE (which merits funding by DOE OFES). The single-shot development (shown in yellow) includes both NIF and Z-Pinches. NIF will be starting to come on line soon, and is scheduled to demonstrate ignition (laser indirect drive) by 2013. These results will be valuable for all approaches to IFE. Simultaneously, z-pinch driven targets are being developed on Z, and soon on ZR. In 2008-2010, DOE will decide on the next step facility. For z-pinches, this will be a facility that demonstrates ignition and high yield (0.5 GJ) with z-pinches.

Rep-Rated Facilities: PoP, IRE and ETF. Parallel to this development, the basis for rep-rated Z-Pinch IFE (shown in blue) will be developed in stages. The first stage (CE) has been supported by internal funds from SNL. Z-Pinch IFE is ready for the next stage (PoP), and it will last 3-5 years.

This stage (PoP) will include development areas in (1) RTL optimization, (2) rep-rated pulsed power, (3) containment/shock mitigation, (4) scaled RTL cycle experiments at 0.1 Hz, (5) Z-Pinch IFE power plant development, (6) Z-Pinch IFE target design, and (7) Z-Pinch IFE target fabrication and power plant technology development. Each of areas (1)-(4) will include an experiment – (1) RTL demonstration on Z, replacing the current MITL with an RTL, (2) rep-rated LTD accelerator construction [1 MA, 1 MV, 100 ns, 0.1 Hz], (3) shock mitigation/containment experiments in a separate facility using explosives and water flow to model the coolant, and (4) integrated, scaled, rep-rated LTD, RTL, z-pinch experiments [5 kJ x-rays at 0.1 Hz for N shots]. The following stage (IRE) would be at 10 MA, 2 MV, 100 ns, 0.1 Hz, 0.5 MJ of x-rays. The next stage (ETF) would combine all results in a rep-rated high yield facility to demonstrate fusion and production of electricity (□60 MA, 5 MV, 100 ns, 0.1 Hz, and fusion yields of 0.5 GJ)

Table 4.2.4.1 Z-Pinch IFE Development at the CE, PoP, and IRE Stages

	CE	**PoP**	**IRE**
Targets (capsule compression, symmetry experiments, etc. on Z, ZR, etc.) (DP funded) Z-Pinch Driven Target Dynamic Hohlraum Target Fast Ignition, Advanced Targets	(X)	(X)	(X)
Driver (Rep-Rated) Fast cycling Marx RHEPP technology LTD technology		X	X
Standoff (Recyclable Transmission Line - RTL) RTL material choice, optimum shape, low inductance, etc. Electrical turn-on, conductivity, low mass, mechanical strength, etc. RTL/Chamber interface issues, etc.	X	X	X
Chamber and Power Plant Thick liquid walls, solid/voids, containment, etc. Power Plant design Manufacturing, target fabrication, costing, etc.	X		X
Integrated, Rep-Rated, Scaled RTL Cycle Experiment @ 0.1 Hz Model RTL cycle Couple with z-pinch load Automated, rep-rated, demonstration experiments		X	X

The Z-Pinch IFE development path elaborates on this plan by showing a plot of x-ray energy per shot vs. repetition rate (Figure 4.2.4.7). The single-shot development line includes Zebra (1 MA), Saturn (10 MA), Z (18 MA), ZR (26 MA), and HY (□60 MA). The IFE development line includes PoP (1 MA), IRE (10 MA), and ETF (□60 MA). Also shown is RHEPP II, which has operated at an average power level of 300 kW. The two steps (PoP and IRE) are a reasonable minimum number of steps to develop and understand rep-rated LTD, RTL-cycle, z-pinch/target operation. The results from the IRE combined with the results from NIF and/or a Z-Pinch High Yield facility would qualify moving to the ETF phase of development of Z-Pinch IFE.

Given the recent progress in z-pinch target design, capsule implosion experiments on Z, rep-rated pulsed power concepts, the RTL concept, and initial z-pinch power plant studies, the development of Z-Pinch IFE appears to be an attractive approach to energy.

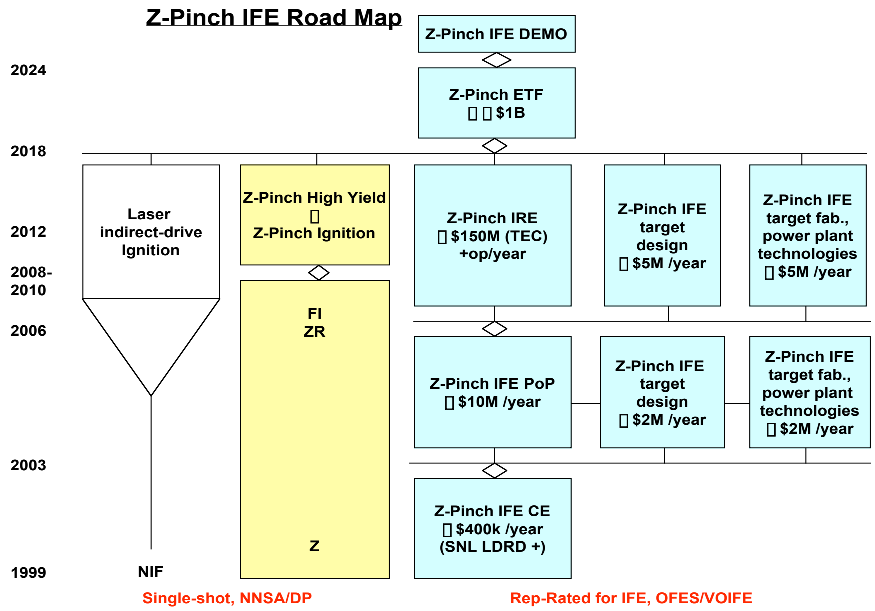


Figure 4.2.4.6 Z-Pinch Road Map.

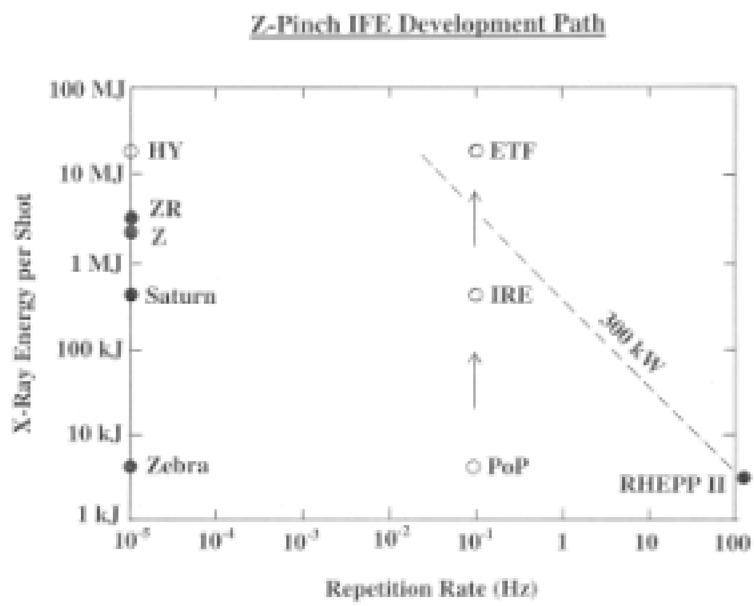


Figure 4.2.4.7 Z-Pinch IFE Development Path.

4.2.5 Proposed Fast Ignition IFE Development Program

Introduction

This section primarily represents the views of the Fast Ignition IFE advocates. Some of comments they received at Snowmass 2002 are highlighted in Section 4.2.6.

Fast Ignition (FI) is a proposed variant approach to Inertial Fusion Energy (IFE) that exploits the advantages of igniting dense fuel in a near isochoric state (uniform density target). This is achieved by separating the ignition step from the compression of the fuel in an anisotropic target geometry. Figure

4.2.5.1 shows a target geometry that has been successfully tested in recent experiments. An attractive feature of FI is that the ignition threshold could be crossed at lower drive energy, with potentially higher gains. There are, however, significant challenges to understanding the physics and achieving fast ignition. The ignition step (delivery of many kJ of energy into the high density, compressed core) involves laser-plasma interaction and electron propagation in a relativistic regime that is only recently accessible by experiment, and remains as yet poorly understood. As a result, Fast Ignition, while potentially attractive for IFE, is still somewhat speculative and requires significant physics validation.

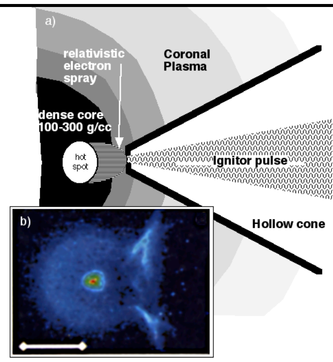


Figure 4.2.5.1: a) Ignition process for a Fast Ignition target with a short-pulse petawatt beam. The laser pulse creates a spray of MeV electrons that transport the ignition energy into the core of the target. The target was previously compressed by longer pulse driver beams. b) Self-emission image of a compressed core heated by a ps "ignitor" pulse, increasing neutron yield 10X. The white bar is 250 μm long.

From the time FI was proposed in 1994, FI research groups have been making rapid progress – assembling compact masses from anisotropic targets, demonstrating the directed propagation of electron energy through dense plasmas, and providing an early demonstration that those electrons can efficiently heat a compressed core (laser-core coupling efficiency ~30%). More work is needed—particularly development of models benchmarked to experiments—before proceeding to the Proof of Principle (PoP) stage; and more powerful capabilities are critical—particularly a multi-PW facility conjoined with symmetric compression—for the experiments needed at that stage.

The present status and proposed path of a Fast Ignition development program is described here. The plan includes the current concept exploration phase, the deliverables required for a transition to PoP, the elements and goals of PoP studies, and the demonstration phase. This program is integrated with other IFE program components and anticipated facility developments (Figure 4.2.5.2). It leverages substantial international efforts that are already underway, and also those efforts proposed for petawatt laser development within the US.

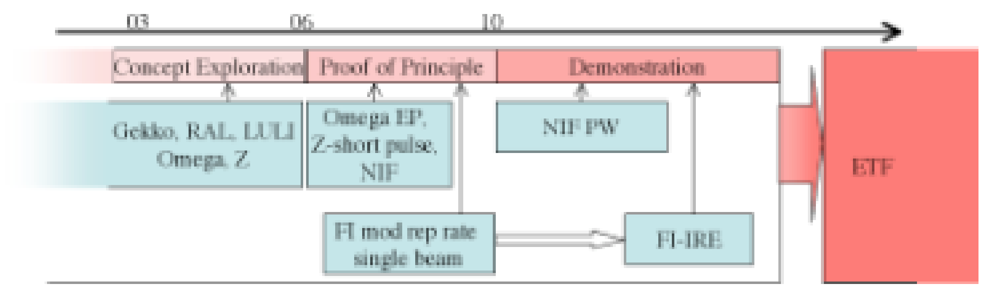


Figure 4.2.5.2 Proposed timeline of FI development program showing the facilities that would be used at each stage and the test facilities specifically needed for FI.

Concept Description

Target Physics. In conventional central hot spot (CHS) ignition DT fuel is compressed with a low density central hot spot, in isobaric equilibrium with the dense, main fuel mass. In contrast, a FI target is assembled with uniform density (isochoric), and the ignition energy is deposited separately in a pulse short compared to the stagnation time (Figure 4.2.5.3). The resulting FI ignition hot spot is much denser than that in a CHS target, so can contain much less mass, leading to reduced ignition threshold. The fuel density is lower than in CHS ignition with less energy invested in compression leading to the possibility of higher gain.

Figure 4.2.5.3 temperature and density profiles of conventional and fast ignitor targets at stagnation

For ignition, the energy E required to be deposited by a fast-ignition beam is $E = 140(100/\square)^{1.8}$ kJ, deposited in a time less than the disassembly time of the hot spot and into a spot size that heats sufficient areal density of fuel (0.5 g/cm^2) to start a propagating burn. For a typical density of 300 g/cm^3 the ignition energy is ~20 kJ disassembly time ~20 ps, and the spot size ~30 μm in diameter. The ignition pulse will likely be supplied by a short-pulse laser whose energy is converted to relativistic electrons at the critical surface of the target plasma. The intensity should be ~10^{20} W/cm^2 to create electrons with the appropriate energy. The mean electron energy should be about 1MeV, giving a range about that of a fusion alpha. Typical FI target design concepts include a reentrant cone in the side of the shell to provide clear access of the ignitor beam to the target, a controlled surface for creation of the electrons, and possibly to concentrate the beam energy.

Ignition Laser. Of the High Average Power Laser (HAPL) drivers currently under consideration for IFE {Diode Pumped Solid State Lasers (DPSSL) and KrF}, the DPSSL can readily be adapted to be used as a FI driver (Table 4.2.5.1), with the addition of a short pulse oscillator and a pulse stretcher at the front end, and a compressor on the output. In addition, spectral sculpting (like that used to support beam smoothing in direct drive IFE) is needed to support a pulse duration of ~30 ps. The KrF laser is also a potential ignitor but it is necessary to deal with its short wavelength and low saturation energy (giving a lower laser induced damage threshold (LIDT) for the final optics and a reduced effectiveness in coupling to electrons, and a limited extraction fluence). In either case, the high intensity output requires the final optical train to be all reflective. To limit the solid angle used by the laser entrance holes in the reactor chamber, the laser induced damage threshold (LIDT) of the final mirror should be greater than 2 J/cm^2. The laser must deliver the ignition energy to the target with an uncertainty < ±50 μm. This is better accuracy than required for other Laser-IFE drivers, and would put more stringent requirements on laser pointing and target delivery; the reentrant cones featured in most FI target designs might relax this requirement substantially.

FI Impact on Compression Driver. Any of the proposed IFE drivers (KrF, DPSSL, HI, Z) being considered for CHS IFE are also applicable in the FI concept. Technology development requirements are comparable to drivers for CHS IFE, but some systems requirements may be relaxed as indicated in Table 4.2.5.2. This paper focuses on the short-pulse ignition driver, assumed to be a laser. Details on the status, issues and R&D plans for the compression drivers are discussed in other sections of this Snowmass report.

FI Impact on Chamber. It appears that the impact of FI on chambers proposed for CHS is relatively minor. There are three issues: final optic, ignitor pulse chamber propagation, and debris.

- Laser IFE chambers are already adapted to control damage to laser final focus optics, making the extension to supporting FI final optics relatively straightforward. However, FI laser targets are likely to generate significantly larger quantities of debris than hot-spot direct-drive targets, so management of this additional debris, and the potential requirement for debris recovery from the chamber, must be considered in FI laser chamber design. Heavy-ion and Z-pinch drivers have no line-of sight access to the compressor final transport elements, so these chambers are not currently designed to control debris and x-ray damage to final focus elements with line-of-

sight access to targets. Significant investigation is therefore required to adapt current HI and Z chambers to fast ignition.
- The high intensity ignition laser pulse puts a limitation on chamber pressure (~10 mTorr) for any of the drivers; laser-induced ionization can potentially steer or defocus the ignition pulse for higher pressures. Low pressures are readily obtained in liquid-protected chambers (e.g. current HI and Z baseline chambers, but this might be an issue for the protection of dry wall chambers.
- The relatively massive reentrant cone feature of many FI target designs might generate highly directed debris jets, compared to other targets.

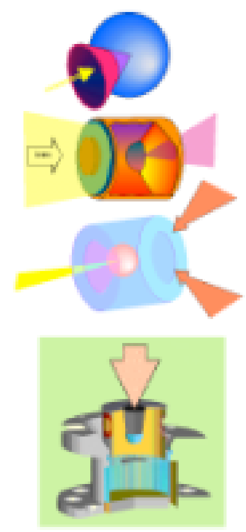


Figure 4.2.5.4 FI target concepts for laser direct drive, heavy ion beam, laser indirect drive, and Z-pinch

Development Plan: Concept Exploration Program. Validation of FI-IFE, and its promotion to serious consideration in proof-of-principle experiments can come once a convincing extrapolation from experimental results and modeling to an ignition prediction can be made. And that in turn requires delivery of:
- A better understanding of the ignition physics – particularly laser matter interaction and collective electron transport in dense plasmas, supported by benchmarked models.
- An ignition target design that can be built.
- A favorable reactor concept including ignitor and compression drivers.
- Suitable facilities for proof-of-principle experiments.

Status

Ignition Physics. The ignition physics for FI is promising. Electron propagation experiments using buried fluorescence layers and rear surface Planckian and OTR emission have shown that electrons propagate as compact beams for over 100 μm in aluminum. Very recent integrated experiments at ILE on the GEKKO XIII facility indicate that these electrons can be coupled to a compressed core with an efficiency >20% (preliminary analysis as of June 2002).

Models exist to describe the fast ignition process, but are not integrated and have varying levels of completeness. Hydro codes to describe the implosion of a shell are most advanced, and are beginning to be tested on Fast Ignition concept targets. Calculations (for laser direct and indirect drive, and Z-pinch) show assembly of a compact, ignitable mass and agree well with preliminary experimental results.

PIC and hybrid codes to describe the laser-matter interaction that generates the ignitor electron beams, and their subsequent propagation to the core of the target are inadequate. These models are lacking in either their volume or duration, or their physics. PIC codes show extremely strong electron beam attenuation that does not seem to be realized in practice, and the hybrid models do not predict some experimental phenomena, such as divergent transport in Al.

Ignition Target Design. Initial target concepts exist for each driver (Laser direct and indirect, HIB and Z-pinch – Figure 4.2.5.4), and details of their behavior are being examined.

Chamber Issues. Preliminary studies have indicated that chamber designs like Hylife II or Koyo could be adapted to the Fast Ignition approach, but at this point the tradeoffs are not well enough evaluated for serious design efforts.

Ignitor Driver. Diode Pumped Solid State Lasers (DPSSL) can be adapted to be used as a FI driver with only modest changes. KrF lasers can also produce the short pulses necessary, but adaptation of a FI system to their use presents more difficulties.

Compression. Drivers currently under development (lasers, HI, and Z-pinch) are suitable for FI compression.

Milestones to be Reached Before Transition to a ‘Proof of Principle’ program

The following milestones must be reached to transition to the PoP phase of development:

1) Sufficient physics agreement between modeling and experiment to have confidence in design capability -
   - Sub scale 3D PIC modeling of absorption, electron production and transport (Weibel etc). This is well underway. There are some issues still to resolve – notably PIC study of electron flux limit /minimum transport area questions.
   - Hybrid 3D PIC modeling at experimental scale for electron transport. This is just beginning. It is expected that the LSP code (developed by MRC/SNL) will be useful for these calculations.
   - Transport experiments with conductivity and scattering close to FI conditions. Previous measurements have been in relatively high Z cold solids; development of lower Z, hot plasma targets is starting this year.
   - Evaluation of the use of laser-generated ions for delivering the ignition energy.
   - Cone transport modeling and experiments to assess effectiveness of energy concentration. Currently only the first modeling ideas are being explored; they need quantitative substantiation.
   - Fuel assembly experiments and hydro models.

2) Proof of principle target designs. Use hydro and transport models to estimate fusion yield. Existing designs will be refined in step with the physics program. The proof of principal experiment design will be fairly detailed but not use an integrated model.

3) Evaluation of the components for FI reactor concepts. Find no ‘show stopping’ problems and provide a basis for more detailed work.
   - Final optics - start work to understand and to increase laser induced damage threshold (LIDT).
   - Target fab/injection.
   - Reactor designs with emphasis on FI final optics protection.

4) Design studies to set the requirements for a reactor relevant FI driver.

5) Conceptual design of a FI ignitor laser.
   - DPSSL: Extrapolation of single pulse HEPW developments to high rep rate.
   - KrF: Conceptual solutions to the problems of low saturation fluence, large multiplicity of beams and reduced electron coupling (low $I\lambda^2$) and reduced LIDT.

6) Availability of facilities appropriate for the proof-of-principle experiments.
   - Proposed NNSA funded multi-PW laser/compression facilities (NIF, Omega-EP, or Z).

Development Plan: Proof of Principle Program

The goal of this program is to show an integrated understanding of the requirements for ignition and gain, and of components required for IFE reactor designs, to give confidence in attractive paths to fusion energy.

FI Target & Chamber Development

This requires a multi-pronged, well-coordinated research effort with four components.

- <u>An experimental program</u> focused on integrated experiments to validate energy coupling to compressed plasmas. This program requires the resources to develop new diagnostics and test new targets as well as access to NNSA funded next-step laser facilities (with multi-kJ, multi-petawatt short pulse capability along with 10’s to 100’s of kJ of symmetric compression for assembling the fuel or producing relevant plasma conditions).
- <u>An integrated modeling program</u> - to develop the capability to couple hydro, transport, interaction and propagation codes so that a target design can be evaluated in an integrated manner. Access to modest dedicated computer clusters is required to test codes and to perform

demo calculations. Further access to substantial computing power will be required for serious analyses, as the processes in these designs cover wide ranges in time and space scale lengths, and are inherently 3D.

- A selection of optimal target design concepts and an estimation of their ignition parameter space. Direct electron ignition and ion ignition will be considered, as well as direct and indirect compression, and allowance for the benefits of RT unstable target aspect ratios and irregular fuel distributions that would be impossible in central hot spot designs. DT ice layers in such asymmetric designs might be very different than used in CHS designs, and require testing of their feasibility.
- An evaluation of FI-IFE reactor designs. Much of the effort here will be spent on testing FI specific reactor and driver issues, chiefly ignitor pulse generation (including studies and demonstrations of CPA adaptations of HAPL facilities). Transport to the target (final optic and chamber transport). These results, in addition to theoretical studies of FI-IFE target fabrication and injection concepts, will be used to evaluate modified CHS designs like Koyo and Hylife II.

Milestones to Transition to a ‘Demonstration’ Program

1. Integrated understanding of the FI gain curve.
   - Integrated full-scale models coupling hydro, optical propagation, hybrid PIC transport, and burn.
   - Integrated proof of principle experiments on proposed NNSA facilities.
2. High gain, buildable targets consistent with mass production fabrication.
   - Ignition target design based on benchmarked integrated model and use of NNSA facilities (NIF or Z).
3. Detailed Power Plant Design.
   - Final optics R&D.
   - Target fabrication R&D.
   - Target injection R&D.
   - Power plant design studies.
4. Rep-rated compression and ignition driver concepts and PoP demonstrations.

Development Plan: Ignition and Demonstration Program

This program should demonstrate ignition and high gain with single shots, and test all driver components at full scale with rep-rated drivers in an FI-IRE. Full power-plant design including target factory and injection system at the smallest feasible scale should be developed. Either NIF or possibly Z could be adapted for ignition and high gain

FI-IRE development activities. The ignitor IRE should demonstrate an ignitor beam line of short pulse (including final optics) at power plant scale and rep rate and lifetime - providing a cost effective engineering solution for full scale implementation.

A down select between DPPSSL and KrF should have been made in the PoP phase.

Adaptation of the target injection IRE for FI targets should be used to validate injection tracking and irradiation capabilities needed for the power plant.

ETF Design. It is clear that FI offers the opportunity to make a uniquely small demonstration plant by trading off compression and ignition energy.

The most cost effective approach to an FI ETF would be a ~100 kJ FI laser coupled to a ~200 kJ compression laser. The present gain scaling curves indicate that a G □ 100 could be obtained (Figure 4.2.5.5), which translates into a yield of □30MJ. A target chamber to contain these yields is much smaller and cheaper than a high yield >> 100 MJ target chamber for a conventional CHS ETF.

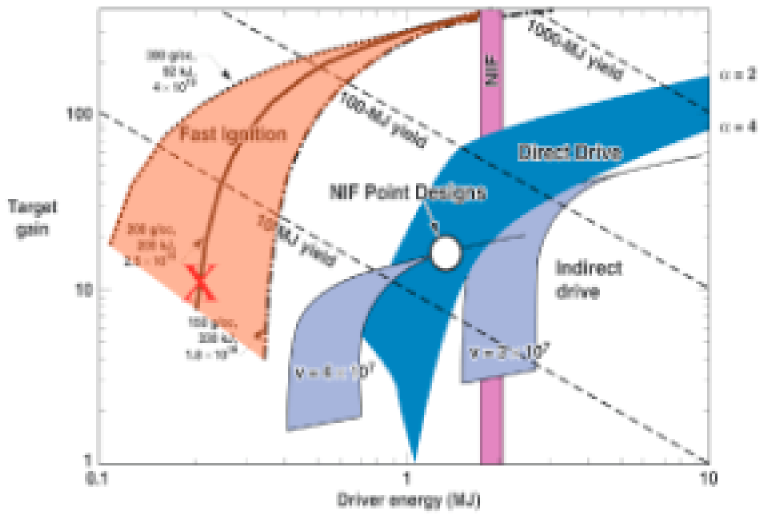


Figure 4.2.5.5 Gain curve vs. driver energy for different laser IFE techniques. A FI ETF would operate with only 200KJ drive, shown with the red X.

Ignition Driver. DPSSL or KrF TBD. The ignition and compression drivers together need deliver only ~300kJ (see Figure 4.2.5.5) – about 10X less than required for a CHS demonstration.

Milestones for Successful FI Ignition and Demonstration phase

- High gain ignition.
- Ignitor laser and target IRE's.
- Chamber and power plant engineering design.
- Cryo targets with path to mass production.
- Pellet factory engineering design.

Table 4.2.5.1 Nominal requirements for the Fast Ignition laser pulse.

Parameter	**Requirement**
Beam energy on target	100-200 kJ depending on coupling to fuel (assuming here 10-20%).
Spot size	≤30 μm diameter. The reentrant cone might relax this requirement.
Pointing accuracy	~±50 μm. The reentrant cone might relax this requirement.
Pulse duration	~20 ps
Wavelength	~ 1 μm – shorter wavelengths have poorer coupling to electrons and lower damage thresholds.
Bandwidth	~ 0.1 THz (both DPSSL and KrF lasers have sufficient bandwidth for this purpose)
Laser efficiency	Could be less than compression driver for compression dominated implosions.
Cost	Some cost premium over compression driver cost (on $/J basis) is acceptable. System analysis is needed to quantify the amount.

Table 4.2.5.2 IFE Compression driver requirement changes for FI targets.

Parameter	**HS Driver**	**FI Compression Driver**
Total energy	Laser: 1.5 - 4 MJ HI: 5-7 MJ Z ?	Could be ~2-3X less than for CHS. Demonstration plant could be as much as 10X less.
Illumination geom.	Laser: base = direct-drive, uniform illumination HI, Z: base = indirect drive with two-sided illumination	Laser – could be asymmetric direct-drive or indirect drive, possibly one-sided compression drive. Compression beam uniformity requirements could be relaxed. HI, possibly one-sided compression drive.
Efficiency	5 - 50%	Could be ~2-3x lower for same recirculating power
Rep-rate	5-10 Hz	Similar for same net power
Laser Wavelength	0.35 or 0.25 μm	Could be relaxed to 0.5 or even 1 μm

4.2.6 Comments on Proposed R&D Plans

As previously indicated, Sections 4.2.2-4.2.6 largely represent the views and proposed plans of the respective driver advocates. In this section, some of the key comments resulting from the Snowmass presentations and written plans are noted. Some of the issues that were raised cannot be resolved at this time. They must be resolved by appropriate workshops and reviews before major new IFE facilities are built.

Laser IFE

With regard to the proposed Laser IFE plan, the following key points were raised:

- While most concurred with the basic R&D steps for Laser IFE, some expressed the opinion that of the proposed time scale to proceed to IRE and ETF was too short. Advocates argue that with adequate funding and national determination, the proposed schedule is reasonable.
- Questions were raised about what product of driver efficiency (η) and target gain (G) was needed for IFE and concern was expressed that the laser systems do not have much margin for possible reduction in either term. It was argued that ηG is not a good metric since it does not account for differences in driver cost; the cost of electricity is the real bottom-line metric.
- There was some debate about the proposed scope of the laser IRE, with some suggesting that a smaller laser might be adequate and others questioning the value of the proposed chamber simulation experiments compared to experiments performed in existing facilities. Proponents felt that the proposed scale was required to make the following step to an ETF. They also argued that the IRE mini-chamber experiments would help improve chamber dynamics codes even if experiments did not simulate prototypical environments in all aspects.
- An opinion was expressed that key laser issues of durability, efficiency and cost could and should be addressed prior to the engineering integration of an IRE, which should be straightforward. Advocates have enough confidence that these issues will be resolved and that integration in an IRE as soon as possible is the appropriate next step.

Heavy Ion Fusion

With respect to the proposed HI plan, the following key points were raised:

- There were questions about the feasibility of upgrading IRE and ETF drivers to allow a demonstration power plant by 2027. Advocates claim this is a potential advantage for IFE and will continue to pursue the driver-upgrade strategy. They agreed however, to look at the proposed time sequence of steps again in more detail.
- During detailed review of the proposed IBX, concern was raised that the current designs may be too expensive. Advocates agreed to improve the designs or reduce the scope of the project to achieve a projected cost less than ~$50M.

- With respect to the final-focus-magnet/chamber interface, there were questions about whether the levels of magnet activation were acceptable from a waste disposal standpoint. The shielding design and optimization is continuing to try to ameliorate this issue.

Z-Pinch IFE

The following points were made about the proposed Z-Pinch plan:

- There were questions whether the time scale to get to a Z-Pinch DEMO was too fast. Advocates respond that on the IFE Road Map, the combination of a Z-Pinch High Yield facility and a Z-Pinch ETF would be from 2010 to 2024, a timescale of 14 years, which is not unreasonable.
- Concerns were raised about the repetitive mechanical connection of the RTLs and the practicality of circulating and refabricating large masses of radioactive material from recyclable transmission lines (RTLs). Advocates noted that these issues are being studied and stated that they are also looking at methods to reduce the RTL mass.
- Concerns were raised about the ability of the chamber to deal with shocks and high velocity debris and the amount of explosion energy (magnetic, kinetic, or radiant) that will be transported up the RTL to the permanent driver structure. Advocates stated that designs are still preliminary and more work on these issues is already planned.

Fast Ignition

Key points raised with respect to the proposed fast ignition plan were:

- Survivability of the FI final optics is key issue that needs more prominence in the plan; a conceptual solution is needed to show that FI is a viable option for IFE. Advocates acknowledge this issue, but note that there is preliminary evidence that cone-target design may allow larger spot sizes, which would allow the optics to be farther from the target. The plan does address the important target physics issues.
- Concerns were raised about ability to meet the spatial and temporal accuracy requirements for FI beams to hit an injected FI target. Advocates claim the ignitor beams could be timed off the firing of the compression driver. Pointing requirements will depend on the ability of the cone to channel energy. Stationary z-pinch targets would be easier to hit, but would produce more debris that must be prevented from hitting the FI laser optics.

4.3 IFE Chamber and Target Technology Plans

The DOE OFES and the High Average Power Laser (HAPL) program have applied a systematic approach to study chamber and target technology issues. Each of the driver programs has adopted a phased approach to studying chamber and target fabrication systems capable of operating reliably and safely at high repetition rates, using phenomenological models and experiments of systematically increasing scale. To ensure that the modeling and experimental efforts are comprehensive, a structured approach (where the integrated system operation is decomposed into time phases and spatial regions, Section 4.3.1.2) has been used to systematically identify phenomena that affect the primary system functions: driver energy propagation and target ignition and gain (nanosecond phenomena), high repetition rates (millisecond phenomena), and safety and reliability (quasi-steady phenomena).

Many of the key issues for IFE target chambers can be studied in *separate effects experiments*, which have artificially imposed initial and boundary conditions, and which may be scaled in geometry and energy, and may use simulant working materials. Often separate effects experiments can provide great insights into understanding modeling specific phenomena in complex systems.

Inertial fusion energy systems have specific scaling features which make it easier to design separate effects experiments that have small distortions in their initial and boundary conditions. Due to the extremely wide range of phenomena time scales in pulsed IFE systems—from the nanoseconds for initial target phenomena to the milliseconds for chamber clearing—fast, short length-scale phenomena appear as initial conditions slower phenomena. For example, microsecond x-ray ablation processes are relatively insensitive to the detailed time history of the x-ray deposition, which occurs over a few tens of nanoseconds, and instead respond only to the integrated effect of the short pulse. This allows existing z-pinch and other x-ray sources to be used in materials response experiments to study IFE target effects, with very small distortion in the ablation response.

Due to the scalability and modularity of IFE subsystems, IFE researchers believe that the major issues governing reliable, high-repetition-rate system operation can be addressed at much smaller energies than are required for actual ignition of targets. The next-step driver experiments, which will occur in "integrated research facilities," or IREs, are scaled in energy by a factor of around 100 from the driver energy required for ignition. Because the IREs will not ignite targets, the energy available in the chamber is further reduced, to be around a factor of 10^4 lower than in a prototypical plant.

The different phenomena that control chamber clearing and reliability tend to scale differently as energy and chamber length scales are reduced, some following roughly volumetric scaling while others follow roughly surface area scaling. Therefore the integrated clearing and reliability phenomena in a scaled IRE chamber do not reproduce those in a prototypical chamber, and instead integrated chamber models must be used to predict and optimize the performance of prototypical chamber systems. Validation of these models can then rely on well designed separate effects experiments, as well as experiments that integrate subsets of the chamber phenomena together. The development of detailed, integrated chamber response models is the principal strategy for chamber R&D during the IRE phase.

Target fabrication and injection technologies can be studied at full scale with prototypical materials. The IRE phases of all of the drivers include such experimental studies, including injection experiments where the thermal and chamber gas conditions are reproduced at full scale.

The combination of integrated chamber response models, and full-scale demonstration of target fabrication and injection methods, will provide the necessary tools to perform engineering design of reliable repetition-capable chamber and target systems. With the parallel development of driver technology and target physics, this provides the ingredients required to build the first average-power IFE facility, typically referred to as the Engineering Test Facility (ETF).

Contrasting to magnetically confined plasmas, inertially confined plasmas can be ignited at quite small scale. The NIF, for example, will demonstrate target yields in the range of 20 MJ, around 5% of that called for in prototypical HI and laser power plants. By operating with target yields around 10% to 50% of the full-scale prototypical yields, the ETF can use substantially smaller chamber and balance of plant systems. Properly scaled in geometric size and repetition rate, these systems can reproduce all of the major phenomena that govern chamber repetition rate, and can accelerate processes that affect reliability so that full-life component testing can occur in time periods of around a year. The ETF provides the technology basis for the subsequent, commercial scale DEMO plant.

While the generic plans for IFE R&D are the same for all of the drivers, the specific issues each face have important differences. The key issues for each driver, that will be addressed under the phased R&D approach, can be summarized as:

Issues for Laser Concepts The base-line design for laser drivers uses a dry-wall chamber with direct-drive targets. Recent research has specified the output expected from direct-drive targets more precisely, so that research can now focus on the key issues for chamber design:

- How target energy output is modified (shifting from debris ion energy to x-rays) by chamber gas,
- First wall response to target output, to the modified target output and resulting lifetime, considering the effects of the wall temperature,
- Final optics response to target output, to the modified target output, and resulting lifetime,
- Chamber gas radiation hydrodynamics and equilibration,
- Target injection and survival,
- Transport of target and first-wall debris in chamber and contamination of final optics,

- Mechanical design of chamber systems to accommodate mechanical and thermal stress loading while maintaining final-focus alignment, and
- Further system integration, through a design study to identify a point design that can serve as a useful reference for judging progress in addressing issues, and in assessing the potential system economic performance.

Dry wall chambers share similar blanket design issues with tokamaks. While some of these blanket design issues are challenging, such as component lifetime under neutron irradiation, the IT working groups concurred that these blanket design issues are best addressed under the larger MFE blanket materials development program. The working groups noted that fabrication for direct-drive targets appears to be well in hand, so that the most important issues focus on identifying windows of operating parameters where target injection can work, and protecting final optics and first wall surfaces from target output to provide sufficient reliability.

Issues for Heavy-Ion Concepts. The base-line design for heavy-ion drivers uses a thick-liquid chamber with indirect-drive targets:

- How do the two-dimensional characteristics of indirect-drive targets affect the output of x-rays and debris,
- How do ablation and venting processes affect the impulse loads to target-facing surfaces, the total quantity of coolant that is vaporized, and the transport of debris, particularly up beam lines,
- How do liquid pockets respond to disruption, and particular how are droplets generated and where do they go,
- How do pockets regenerate, and what are the detailed fluid mechanics that control the geometries of various jet types,
- What are optimal gas densities along beam lines, and how can neutralizing plasmas be introduced,
- What materials should be selected for use in targets, and how can they be fabricated into targets and recovered from chambers,
- What configurations of liquid and solid shielding materials can work to protect the final focus magnets and other structures, and
- What mechanical designs of the final-focus hardware and coolant nozzle systems are needed to allow maintenance and control mechanical and thermal stresses while maintaining alignment?

The working groups noted that target injection appears to be straight-forward for heavy ion chambers, but that target materials selection, fabrication, and recovery are key issues. The fluid mechanics of single jets are now well understood and are known to be controllable, but further work is required to characterize multiple jet interactions, droplet generation, and pocket clearing processes. Close interaction between driver, target, and chamber researchers is required in identifying approaches that permit reliable beam focusing, high target gain, and rapid chamber clearing.

Issues for Z-Pinch Concepts. The base-line design for z-pinch drivers uses a thick liquid chamber, with indirect-drive targets driven electrically by a recyclable transmission line (RTL). Because the major progress in z-pinch target physics and design has occurred quite recently, the current chamber and target designs are still quite conceptual. The most important issues involve the blast effects implied by the large fusion yields (few gigajoule) and large target masses (due to electrical drive), which contribute to significantly increased impulse loading to blanket materials compared to heavy-ion and laser targets. The working groups identified several key issues:

- What are the mass flows and mass inventories in the system, consistent with radiation levels that can be accommodated by remote fabrication and insertion equipment for RTL’s,
- What is the minimum possible mass for an RTL, consistent with mechanical strength required to accommodate insertion loads and maintain wire array alignment, and sufficiently low inductance to allow target drive,
- How can fast plasma debris be prevented from damaging the permanent electrode hardware, particularly insulator materials

- During the largely two-dimensional target disassembly process, how can the radiative cooling of the target debris be controlled to minimize x-ray ablation of blanket materials that protect the permanent electrode hardware, and what are the total impulse loads delivered to the blanket,
- How can the blanket be designed to mitigate damage to the electrode hardware, and how can the momentum of the blanket be transferred to structures with reasonable stress levels, and
- How can the interface between the permanent and recyclable electrodes be made reliably and automatically?

These were the highest level issues that were identified during the working groups' review. The groups felt that most of these issues could be investigated analytically and with existing computational capabilities, and that this effort is warranted given the low cost of pulsed power relative to other driver options. Further investigation is needed as well to design pulsed power systems capable of reliable rep-rated operation, but these issues were judged to be less fundamental to the viability of z-pinch IFE.

Issues for Fast Ignition Concepts. Fast ignition has the potential to increase target gain, and decrease driver cost, for all of the major IFE driver concepts. Thus fast ignition shares issues with each of the driver types it will potentially be deployed with. Recognizing this, in their review the working groups focused on those unique issues which are raised by fast-ignition systems:

- How can survivable and reliable fast-ignition final optics be designed for each driver and chamber combination,
- How closely do the target tracking and beam pointing need to be controlled to achieve the pointing accuracy needed for fast ignition,
- How is timing achieved for the fast ignition pulse, particularly for heavy-ion or z-pinch drivers, and
- What changes must be made in the target design, and how are these implemented?

The major issues for fast ignition center around protection of the final optics, which consist of several optical elements for pulse compression that must be kept inside the chamber vacuum boundary. In addition to protection of the final optic that views the target, the other elements must also be resistant to the high radiation environment, and accommodate the effects of debris that may be transported into that region. Target design, and optimization of the relative energy of the compression and fast ignition drivers, is the second major issue requiring resolution.

4.4. Target Physics

Inertial fusion targets can be categorized by the ignition scheme, the implosion mechanism and the driver technology used to supply the compression and the ignition energy. We will briefly review each of these elements. There are two ignition methods currently being considered. The first, called *hotspot ignition*, heats a central core of the compressed fuel to ignition temperatures. The assembly of a sufficiently large hotspot is accomplished by stagnation of a convergent flow. The assembled configuration of the hotspot, and surrounding compressed, low temperature fuel, will be approximately isobaric. The second ignition technique, called *fast ignition*, heats cold compressed fuel to ignition temperatures directly with an external source of heat. This technique has become feasible due to the advent of short-pulse, high-intensity lasers using chirped-pulse-amplification (CPA), that can compress laser pulses to extremely high power. If focused appropriately, these fast-ignition laser beams can provide the same power densities as result from the hydrodynamic flow stagnation of the first technique.

Inertial fusion fuel can be compressed by two techniques, referred to as *direct* and *indirect* drive. Directly driven capsules directly absorb energy delivered by the external compression driver and use it to implode the fusion fuel. Indirectly driven targets absorb the external energy in material away from the capsule, which converts it into x-rays. The x rays are contained in a hohlraum fabricated from high atomic weight material, that symmetrizes the x rays. The capsule then absorbs these x-rays to compress the fuel.

There are four compression drivers being developed for inertial fusion energy: *heavy ion accelerators* (with h = 25%-45% efficiency); *diode-pumped solid state lasers* (DPSSL) (with approximately 10% efficiency); *KrF lasers* (with approximately 7-8.5% efficiency), and *Z-pinches* (with 15-20% efficiency). Target designs differ in detail from driver to driver to accommodate the form of energy delivery while driving the capsules with adequate symmetry and producing adequate gain (G). The gain requirement is set so that less than about 25% (depending on the capital cost of the driver) of output electrical power is required to run the plant. This figure of merit corresponds to a driver-efficiency gain product (hG) that is greater than 10.

In principle, we can choose a target concept by selecting one of sixteen combinations composed by matching an implosion technique, an ignition technique and a driver. The designs that have been studied most extensively use hotspot ignition. Most effort on target design has concentrated on the following three approaches: targets directly driven by lasers; targets indirectly driven by ion beams; and targets indirectly driven by Z-pinches. Figure 4.1.1 shows some of the possible configurations.

These hotspot ignited capsule designs are judged by three fundamental criteria: symmetry, stability and gain. The target designer must determine that the fuel will be imploded symmetrically; that it is sufficiently stable to prevent either shell break up in flight or excessive mixing of the hot ignition region with the cold main fuel; and that the design produces adequate gain. These target requirements set performance criteria for the rest of the system: driver scale and pulse shape to control fuel entropy in order to meet the gain requirement; illumination geometry to meet symmetry requirements; beam quality and smoothness to minimize seeds for hydrodynamic and plasma instabilities; beam brightness to control convergence ratio; ablator/fuel roughness to control final perturbation levels after amplification by the Rayleigh-Taylor instability; and hohlraum design and materials for symmetry control and coupling efficiency.

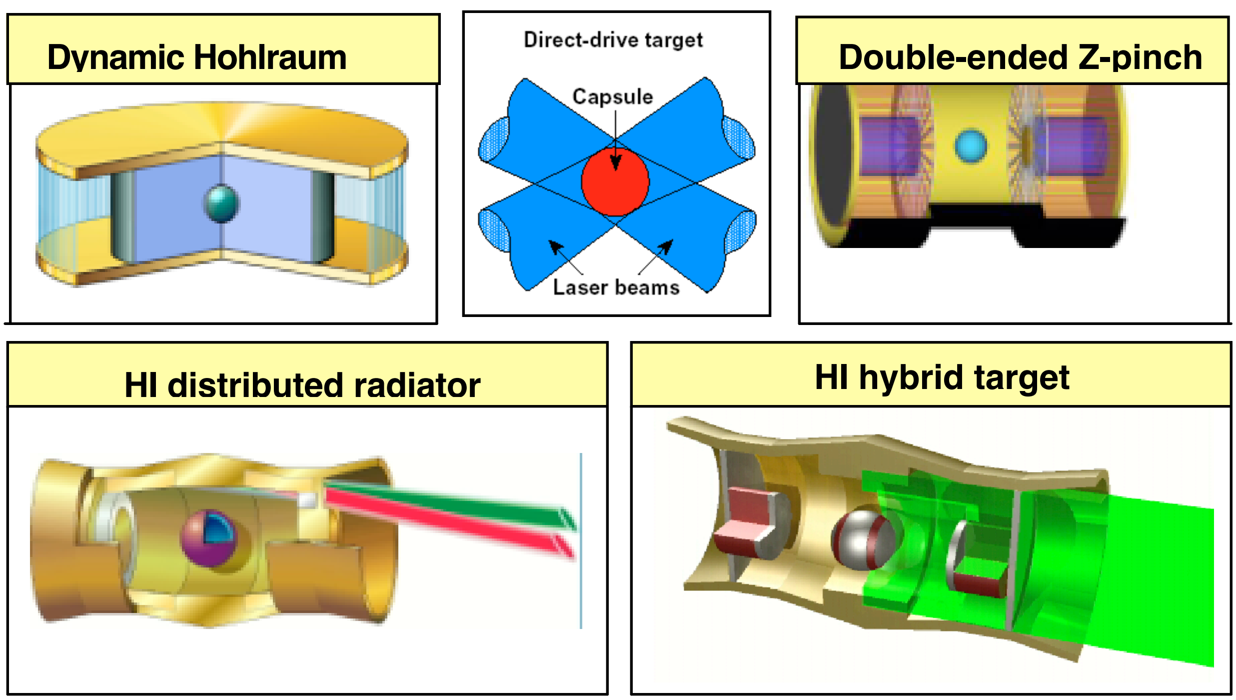


Figure 4.4.1 Shown are two Z-pinch designs (dynamic hohlraum and double ended), two heavy ion designs (distributed radiator and hybrid target) and a schematic of a direct drive laser design.

We can understand how the various requirements arise by beginning with ignition and working back to the implosion criteria. The stagnated fuel assembly is approximately isobaric. The energy invested in this assembly is the sum of the energy in the main fuel, given by a multiple of the Fermi-degenerate energy, and the hotspot. Minimizing this energy with respect to the pressure of the assembly leads to a minimum ignition energy . Detailed simulations of a wide range of implosions have found a scaling law: $E_{ign}\ \mu\ \alpha^{2.7} v_{impl}^{-7.2}$ where α is the ratio of the main fuel pressure to the Fermi-degenerate pressure at the fuel density and v_{impl} is the implosion velocity. Hence, for a given energy invested in the fuel, higher implosion velocities lead to more "excess" energy above the minimum. This is interpreted as more 1-D (without transverse spatial perturbations) robustness.

How do the non-ideal effects manifest themselves as v_{impl} increases? Typically, peak drive pressure is limited by beam focusing capabilities or plasma instabilities. The acceleration distance is approximately half the original radius, R_I. Driving to higher velocities therefore requires larger initial radii and longer acceleration durations. As a rough estimate, we require that the compressed fuel have distortions less than half the compressed radius. This leads to a symmetry requirement that $dI/I < 1/2\ R_F/R_I$, where R_F is the converged radius and I is the incident intensity. Therefore, lower I and larger v_{impl} lead to more stringent symmetry requirements. These are met for direct drive systems by supplying enough beams (typically 60-90) arranged symmetrically. For indirect drive systems, the hohlraum and beam geometry are used to provide a symmetric flux on the implosion capsule. Three mechanisms symmetrize the drive. First, X rays are absorbed and re-emitted from the hohlraum walls multiple times, leading to a large number of effective scatters (which reduces the asymmetry). Second, the capsule samples much of the wall, wall-to-capsule transport smoothes spatial asymmetries with Legendre mode, $l > 5$ by at least two orders of magnitude. Third, proper placement of beam spots can eliminate the low l modes. This simple analysis has been validated by 3D viewfactor calculations, integrated 2D/3D radiation transport/hydrodynamic calculations and in an extensive series of laser-driven experiments on the Nova and OMEGA lasers. Symmetry control at R_F/R_I exceeding IFE requirements will be demonstrated on the National Ignition Facility (NIF). This symmetry methodology generalizes from indirectly driven laser targets (where most of the database exists) to all of the indirectly driven concepts as recently confirmed by experiments on the Z facility at Sandia.

The growth rate of the Rayleigh-Taylor instability during the ablatively driven acceleration phase is of the form:

$$g = a(Agk)^{1/2} - bkv_A,$$

where a and b are of order 1, A is the Atwood number, k is the wave number and v_A is the ablation velocity, given by the mass ablation rate divided by the peak density. There are two ways to reduce the number of e-folds: shorten the acceleration distance (by reducing the implosion velocity and increasing the fuel mass) or change v_A by changing the adiabat of the implosion(and consequently the density). The first of these techniques improves capsule stability while reducing 1-D robustness and increasing gain. An optimization over implosion velocity for fixed drive pressure and coupled energy has been performed using multimode 2D direct numerical simulations with imposed perturbations on the outer ablator surface and the inner ice surface. Power-plant scale indirectly driven capsules with plastic ablators, when driven with a radiation temperature of 265 eV, gave full yield with 10-20 times the NIF roughness specification. This optimization can be tested at the NIF scale. Similar calculations have been performed for capsules tested at the Omega laser. The calculations and experiments agreed within a factor of 2.

Increasing the implosion adiabat (α) stabilizesthe implosion. However, it reduces the 1-D robustness and the gain. Two dimensional multimode calculation of the directly driven a=3 NIF point design produces a gain of 30 at 1.5 MJ incident energy. This gain is insufficient for energy applications. However, by shaping the adiabat profile the fuel can be well compressed (on a low adiabat) to produce good gain, while the ablator can be placed on a high adiabat for adequate stability. This can be accomplished by sending an early time pressure spike through the shell. The decaying shock, thus produced maintains a low a for the fuel while raising it for the ablator. Recent single mode calculations suggest that this approach can produce gains above 100 for lasers of 2-4

MJ, while maintaining adequate stability. Multimode calculations and Omega experiments are planned to verify this effect. The technique can also be tested on the NIF.

Fluctuations in the incident laser beam can imprint perturbations on direct-drive capsules. These perturbations act as seeds for subsequent Rayleigh-Taylor growth. Recent experiments on the Nike laser at NRL have shown that this imprint can be significantly reduced by over coating the capsule with several tens of nanometers of high-Z metal.

The coupling of the beam to the plasma is an efficiency issue as well as affecting symmetry and implosion adiabat (through preheat). These issues have been studied extensively for laser-plasma coupling as part of the NNSA ignition program. Favorable coupling is essential for success of the ignition mission on NIF. For indirect drive, the major concerns are stimulated Brillioun scattering (inefficiency because light is scattered out of the hohlraum), stimulated Raman scattering (inefficiency and preheat from high energy electrons generated in the coupling process) and filamentation (local hotspots driving other instabilities and beam deflections affecting symmetry). Currently, beam smoothing via SSD, and polarization smoothing, together with control of damping mechanisms by varying hohlraum gas compositions, seem to provide adequate control of these instabilities with blue light. In fact, recent experiments with green light have shown instability levels at the few percent level. The two-plasmon decay instability is the principal concern for directly driven targets where this instability acts as a preheat source. Simple scaling of current experiments indicates that this source of preheat may be problematic at a power-plant scale. A remedy may be a reduction in peak laser intensity. In addition there is some evidence that the instability saturates at a tolerable level.

Heavy ion deposition is thought to be classical. However, there are theoretical uncertainties in particle ranges in dense plasmas. These are due mainly to uncertainties in the effective charge of the projectile. Experiments under relevant conditions (material density in the range 0.01-1.0 g/cc and temperature 100-300 eV) are just beginning or are in the planning stages. Target designs can be adjusted to accommodate factors of 2-3 uncertainties in particle ranges and deposition profiles varying from no-Bragg peak to a very sharp one. Possible sources of preheat come from gamma rays and projectile fragments produced in nuclear interactions as well as x rays produced during the ionization–recombination cycle of the projectile passing through the background plasma. Early estimates suggest that these preheating effects are tolerable. New calculations using current beam and target parameters should be performed.

Fast Ignition is a relatively new approach in which an external heating source drives the fusion hotspot to ignition temperatures. Its optimal target configuration for given absorbed energy corresponds to lower peak density than does the conventional implosion hotspot design. Hence, Fast Ignited targets converge less than do the conventional designs, leading to relaxed symmetry requirements and fabrication requirements. Finally, Fast Ignition leads to gain curves 2-3 times above conventional designs, leading to adequate gain for smaller drivers. Success of this approach requires three ingredients: assembly of an optimal fuel configuration; efficient coupling from the external heating source (nominally a short pulse laser although accelerator produced ion beams have also been suggested) to the plasma to generate relativistic electrons; and the coupling of the relativistic electrons to the ignition region (either directly or via ion intermediates).

Experiments over the past decade have shown coupling efficiencies from laser light to relativistic electrons in the range 15-50% with the coupling efficiency positively correlated with laser intensity. Proton beams have been formed with efficiency 1-25%. The leading design to assemble the compressed fuel is the cone focus target in which a dense annular cone is inserted into the side of a capsule. The fuel is compressed along the outside of the cone, and the interior of the cone gives the ignition laser beam access to the imploded core. Recent experiments at GEKKO XIII at ILE, Osaka using this design showed a hundred-fold increase in neutron yield when the short-pulse ignitor beam was timed with peak compression. The inferred energy coupling of the ignitor laser to the compressed fuel was 20-30%. A detailed understanding of electron transport is required to see if these results scale to power-plant conditions. Optimized implosion designs are also necessary.

Recent proposals for NNSA-funded multi-kJ class short pulse lasers co-located with compression drivers may lead to an ignition/high gain demonstration using Fast Ignition.

The following summarizes the outstanding target physics issues for the various concepts. A target physics program plan is sketched to address the issues identified for each concept.

Direct drive laser:

Symmetry: Adequate with quasi-uniform illumination with 60-90 appropriately shaped beams. Designs using asymmetric illumination that might be required by chamber considerations are in progress.

Stability: Designs using recently adopted pulse shapes have much better stability properties than previous designs and lead to adequate gain in the 2-3 MJ laser drive regime. Experiments and design improvements are still possible. Multi-mode calculations are required.

Coupling: The two-plasmon decay instability is a preheat mechanism that requires further study. This process could limit peak laser intensity or require other modifications in the target design to compensate for the preheat.

Laser direct-drive target physics plans:

The NNSA target physics program focuses on underlying physics and design ideas in preparation for ultimate validation in ignition experiments on the NIF. The experimental program will be carried out at the Nike and Omega lasers and planar experiments are planned for the NIF:

- Adiabat shaping experiments (with intensity spike)
- Planar experiments with wetted foam to measure EOS, laser imprint, and RT growth
- Warm foam spherical implosions
- Combine foam shells with cryogenic D_2
- Wetted foam implosions
- Wetted foam implosions using palladium overcoat
- Calculation of multimode implosions in 2D/3D
- Quantitative understanding of saturation phenomena in convergent geometry
- Modeling of effect of laser beam refraction through perturbed coronal plasma
- Production of a benchmarked model of the two plasmon decay instability with relevant scaled predictions

Indirect drive ion beam:

Major progress has been made in the development of heavy-ion targets, based on the concept of distributed radiators, where low-density converter material to stop ions is located close to the capsule P4 locations (at 20° and 60° from the capsule equator). The use of very low density stopping material and high-Z hohlraum wall material, and low-Z pressure-balancing material in the other regions, greatly reduces the hydrodynamic motion of the converter and wall material. The target physics working group summarized the status of the physics understanding in various categories:

Gain: Calculations show adequate gain for several illumination assumptions.

Stability: Capsules with plastic ablators are predicted to be adequately stable in multimode stability calculations.

Coupling: More data is necessary concerning the stopping power of ion beams in dense plasmas. In addition theoretical predictions that there are no significant collective beam plasma effects need to be verified experimentally. On the other hand, the target designs can be tuned to accommodate large uncertainties. Some design features need experimental verification beyond current data: pressure balanced hohlraums and capsule shims.

Heavy ion indirect-drive target design plans:

- Design and field experiments to test the concept of pressure balance. These experiments might be done using radiation from a laser or z-pinch to supersonically heat the low and high Z foams. It may also be possible to design experiments to heat these foams using heavy ions generated by a short-pulse laser.
- Design and field experiments to create the appropriate physical conditions in the low density, high Z materials in-situ using a collection of separated wires or thin sheets which are then heated and expand to fill the voids. The collection of wires or thin sheets can be heated either by radiation or by a lower power ion beam.
- Design and field experiments to measure the wall loss for alloys or mixtures of hohlraum materials that are suitable for IFE.
- Design and field experiments with short-pulse laser-produced heavy-ion beams to study beam target interaction in candidate HIF target materials at reactor level-parameters
- Continue theoretical and computational work on understanding capsule performance including Rayleigh-Taylor instability growth, and low order asymmetries with varying drive temperature. Include 3-d effects such as pointing errors, power balance, and finite number of beams.
- Improve the interface between the target and the rest of the power plant. Simplify the hohlraum and specify tolerances for target fabrication. Calculate target output to allow more accurate chamber response calculations. Set beam requirements on pulse shape, power balance, beam distribution
- Study fast ignition using ion beams to compress fuel and a short pulse laser or accelerator to ignite it.

Indirect drive Z-pinch

The double-pinch hohlraum and the dynamic hohlraum approaches are both being pursued.

Gain: Numerical simulations indicate that high yield and adequate gain can be obtained with either approach.

Need to demonstrate reproducible radiation pulse shaping techniques for both schemes, compatible with radiation symmetry control requirements.

Symmetry: P2 symmetry control has been shown in the double pinch between +/-3%. Total even mode radiation symmetry must be reduced by a factor of 2 to show scaling to high yield requirements.

P2 symmetry remains to be optimized. P4 symmetry control experiments must still be done. Dynamic hohlraum experiments show a reproducible, azimuthally uniform shock (the heating source for the internal capsule), which implies that radiation symmetry control should be possible. However, shocks are stable objects, so the dynamic hohlraum walls may be more perturbed than the observed shock. Need to show adequate control of pole to equator radiation symmetry for the dynamic hohlraum.

Z-pinch target plans

Double-Pinch

- P2 optimization experiments FY03
- P4 control experiments FY04 Control and understanding of even mode radiation flux asymmetry FY05
- Demonstrate reproducible pulse shaping control in capsule hohlraum (e.g. factor of two step 35 eV to 70 eV with Z) FY05
- Develop 2D and 3D numerical pinch models consistent with pulse shape control experiments and measured pinch mass distribution FY07
- Demonstrate scaled pinch power and scaled coupling efficiency with hohlraums that have adequate scaled pulse shape, and scaled symmetry FY09

Dynamic-Hohlraum

- Demonstrate ability to diagnose polar radiation symmetry internal to a dynamic hohlraum

- Demonstrate understanding and control of polar radiation symmetry
- Demonstrate reproducible pulse shape control
- Develop 2- and 3-D numerical pinch models consistent with pulse shaping and radiation symmetry control experiments
- Design and demonstrate implosion system for Fast Ignition FY08

Indirect drive laser:

Gain: Need design with adequate gain. Best current design has gain=45 and 2.4 MJ of blue light.

Coupling: SBS, SRS, filamentation appear to be adequately controlled with beam smoothing control, but continuing work is necessary.

Symmetry: A two-sided design is needed if there is to be geometry advantage relative to direct drive.

Indirect drive laser plans:
- Design improved coupling target with adequate symmetry
- Design laser distributed radiator target with adequate coupling
- Assess utility of longer wavelength lasers in hohlraum designs
- Assess laser-plasma coupling in the small hohlraums required for good coupling efficiency

Direct drive with ion beams:

Gain: Offers improved gain compared to other ion beam options.

Stability: Early calculations showed poor stability properties. Recent calculations suggest that the earlier work was pessimistic.

Symmetry: No existing direct drive ion beam designs are consistent with two-sided illumination.

Coupling: This approach is more sensitive to details of beam-plasma interaction than indirect drive ion beams.

Direct drive with ion beam design plans:
- Study stability of direct drive designs
- If stability looks acceptable, study direct drive in a geometry with the beams entering in a small number (~ 2) of cones for compatibility with thick liquid chambers such as HYLIFE-II.

Fast ignition:

Gain: Offers marked improvement of gain curve and relaxed stability and symmetry requirements for any driver. Techniques are needed that efficiently convert kinetic energy to compressional energy without forming a low-density central region.

Stability and Symmetry: Need to quantify how much stability and symmetry are relaxed.

Coupling: Cone focus design significantly reduces distance between critical surface and ignition region. Electron transport requires detailed modeling and continuing experimental program.

Fast-ignition target physics plans

Current experimental fast ignition research is being carried out on facilities around the world with the aim of understanding the basic laser-plasma interaction, the transport of electrons in dense plasmas and the generation and focusing properties of ions. These facilities vary in scale from a few joules to 0.5 kJ. Larger facilities are under consideration by the NNSA. Integrated experiments would be performed on these larger facilities. Specific activities in the concept exploration phase(next three years) are sketched below:
- Design and demonstrate in scaled experiments atc Omega and Z implosion systems that efficiently convert implosion energy to fuel compression energy without central voids and with small distances between the FI laser critical surface and the compressed fuel

- Successfully model existing and continuing electron transport experiments in materials at normal density in slab targets as well as those with attached cones using 2D/3D PIC/collisional PIC/hybrid PIC codes
- Develop and demonstrate electron transport control techniques
- Conduct and model a set of transport experiments in significantly compressed materials. Using appropriately models thus appropriately benchmarked, predict coupling efficiency for ignition targets.
- Model and demonstrate focusing of FI ion beams in open geometry (from segments of a spherical surface) as well as in geometries where the spherical surface is attached to a cone.
- Transition to proof of principle campaign using HEPW facilities provided by NNSA
- Develop integrated design tools(PIC/hybrid PIC/ hydrodynamics/burn)
- Design and conduct integrated experiments at available NNSA laser facilities(possibly at Omega, Z or NIF)
- Benchmark integrated codes and design full scale ignition experiment
- Transition to ignition campaign
- Demonstrate integrated design and ignition/gain at 100 kJ short pulse laser scale