Snowmass Assessment of FIRE

Executive Summary followed by references to FIRE in the body of the report

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).

<u>Common benefits from all three candidate burning plasma devices include the following:</u>

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.

Key benefits from FIRE are the following:

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).

Key benefits from ITER are the following:

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 $Q = 5 – 15$ 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.

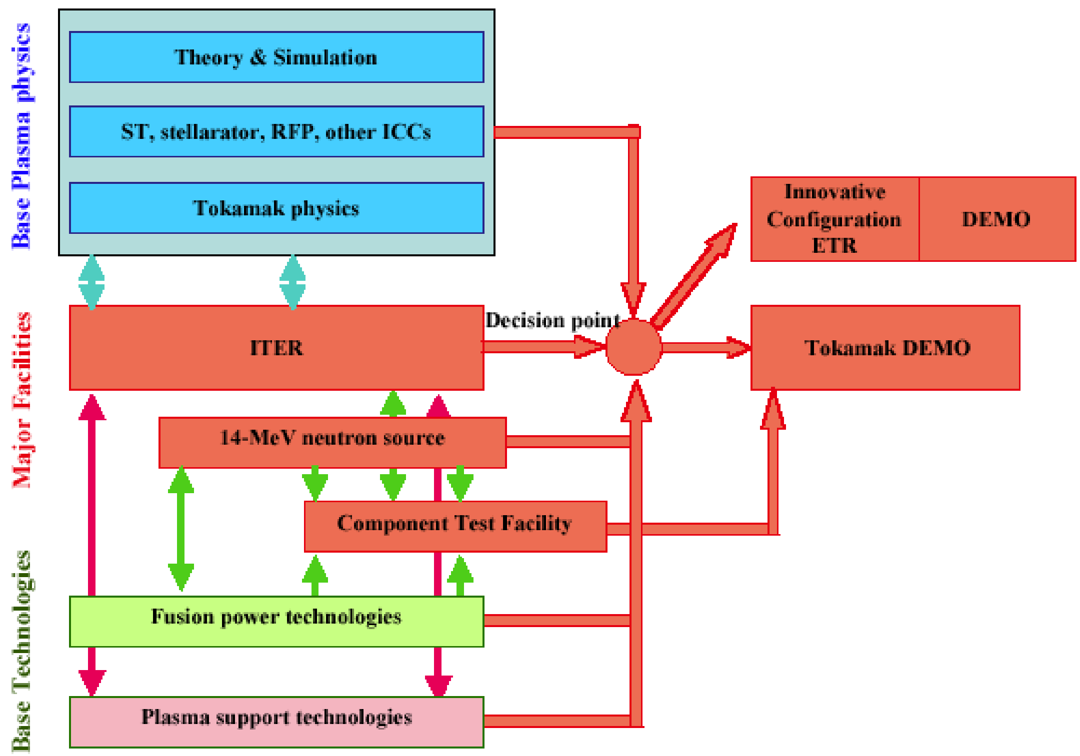

Figure 1.1.1. Schematic of development path based on ITER-class burning plasma experiment.

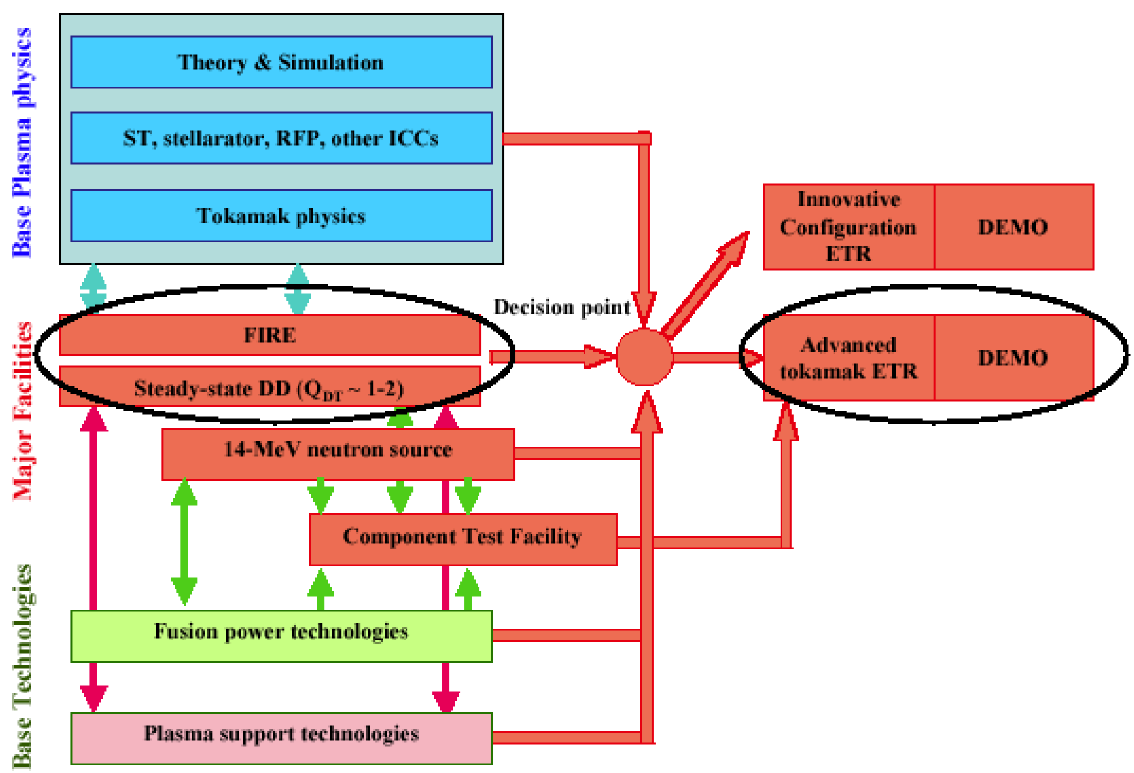

Figure 1.1.2. Schematic of development path based on FIRE-class burning plasma experiment.

Other Comments related to FIRE

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 ($Q \geq 5$ with $Q \approx 10$ as the target) that are sustained for a duration comparable to characteristic plasma time scales ($\geq 10\ \tau_E$, $\sim 4\tau_{He}$, $\sim 2\ \tau_{skin}$). 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 $Q \approx 10$ 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, $B_t(R_o) = 10$T, $I_p = 7.7$ 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
$R\nabla\beta_\alpha$ (%)	3.8

Table 3.1.2.1 FIRE, Q = 10 Parameters

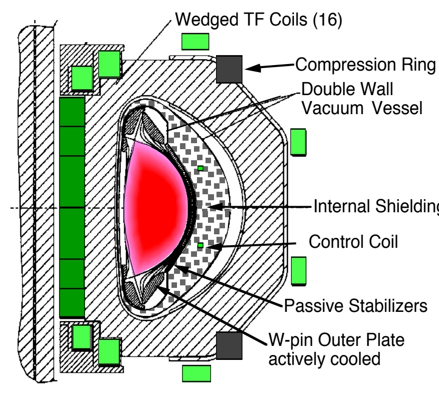

Figure 3.1.2.1 FIRE Configuration

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 (δ_x = 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 $n(0)/\langle n\rangle_V$ = 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.

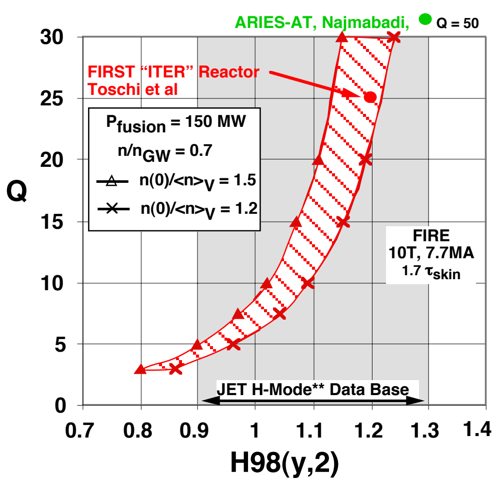

Figure 3.1.2.2 Fusion Gain for FIRE

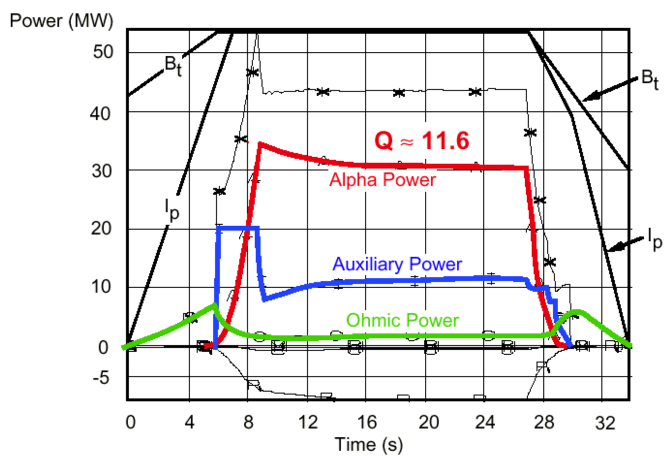

Figure 3.1.2.3 Fusion-dominated Plasma Evolution.

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 $_N$ ≈ 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 $Q \approx 10$ 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 $B \approx 6.5$T, 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.

3.2. 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>Assessment.</u> 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.

<u>ICRF</u> -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.

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.

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.

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 advanced modes on ITER and FIRE, relatively modest amounts of on-axis fast wave current drive are required as "seed current".

These scenarios have been studied using the PICES full-wave ICRF solver and the CURRAY ray tracing codes.

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.

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.

Lower-Hybrid Current Drive. ITER and FIRE require lower hybrid current drive in the outer part of the plasma to support advanced tokamak scenarios.

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.

ECRH- developed for FIRE after Snowmass

3.2.2 Alpha Physics/ Energetic Particles Assessment

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.

Uniform Assessments of Approaches to Burning Plasmas (FIRE, IGNITOR, and ITER). 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 and 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.

(reference Gorelenkov paper on TAEs)

3.2.3 MHD Science in a Burning Plasma

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.

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.

Present experiments in regimes relevant to ITER and FIRE [$\varepsilon\beta_p$~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.

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.

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 ($\Delta/a \sim 0.03$). 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.

3.2.4 Transport

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, β, ρ^*, ν^*, and n/n_{GW} can be studied on ITER and FIRE. Ignitor cannot match these dimensionless parameters as satisfactorily, owing to the lower β values. Diagnostic questions exist for all three devices.

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.

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 ($2 < \kappa < 2.1$; $.65 < \delta < .85$), 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.

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.
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.

3.2.5 Boundary Modeling of Burning Plasma Experiments

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 3.0×10^{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 3.0×10^{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 1.5×10^{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.

<u>ELMs.</u> 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.

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.

Advantages of a BPX from the Perspective of the Boundary.

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.

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.

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.

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.

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:

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.

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:

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:

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

Schedule for construction: 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.

3.3.2 Plasma Facing Components/Heat Removal

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 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 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*

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

3.3.4 Vacuum Vessel and Remote Handling

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

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

3.3.5. Safety/Tritium/Materials

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.

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.

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.

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.

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.

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.

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.

3.4 Experimental Approaches and Objectives

3.4.1 Diagnostics

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.

Assessment. The FIRE diagnostic set is plausible, but many issues remain, some of which may be resolved during detailed engineering design. For all three devices, the impact of access will be very important. 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.

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.

Neutral-beam-based diagnostics are presently planned for ITER and FIRE. The lack of beam penetration in FIRE is recognized to require the development of a new specialized beam.

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.

3.4.2. Integrated Scenarios

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.

3.4.3 Physics Operations

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.

Experimental Operations. The experimental plan must provide sufficient pulse lengths to allow study of relevant phenomena. FIRE provides 3000 full-power equivalent 20 second pulses out of a total of 30000 discharges (10%).

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.

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. 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.

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. 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. 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.

Equilibrium Operational Issues.

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.

3.4.4 MFE Development Paths

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-β, 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.