3.0 Modular Program Pathway

3.1 Pathway Overview

The major issues in fusion R&D can be described as: (1) the achievement and understanding of self -heated plasmas with high energy gain that have characteristics similar to those expected in a fusion energy source, (2) the achievement and understanding of sustained self-heated plasmas with characteristics (steady-state or high duty factor pulsed systems) similar to those expected in a competitive fusion system and (3) the development of the nuclear technologies needed for fusion energy sources. These general categories can be used to describe both the magnetic and inertial fusion R&D programs which have historically pursued a modular approach with the individual modules focused on the technical issues described above. The 1995 PCAST review of Magnetic Fusion recommended that the modular strategy be continued with programs and facilities specialized to address the ignition, steady-state and technology issues. This modular pathway, with burning plasma physics as the highest priority element, was the central recommendation of the Grunder FESAC Panel (January 1998) and was the option that was preferred by many of the fusion community researchers at a workshop on approaches to burning plasma physics held in Madison, Wisconsin (April 1998). The continuation of the modular approach for the next major steps in magnetic fusion enhances the likelihood of successfully realizing a viable fusion power source.

The proposed Modular Program Pathway to Magnetic Fusion (Fig. 2.1) would have four major initiatives aimed at: (1) developing innovations in steady-state advanced magnetic confinement configurations, (2) exploration, optimization and understanding of strongly burning plasmas, (3) development of technologies and materials needed to make magnetic fusion an economically and environmentally attractive energy source, and (4) a Strategic Simulation Initiative to facilitate the fundamental science understanding in each of the first three initiatives and to then serve as a mechanism to intellectually integrate the science of these initiatives.

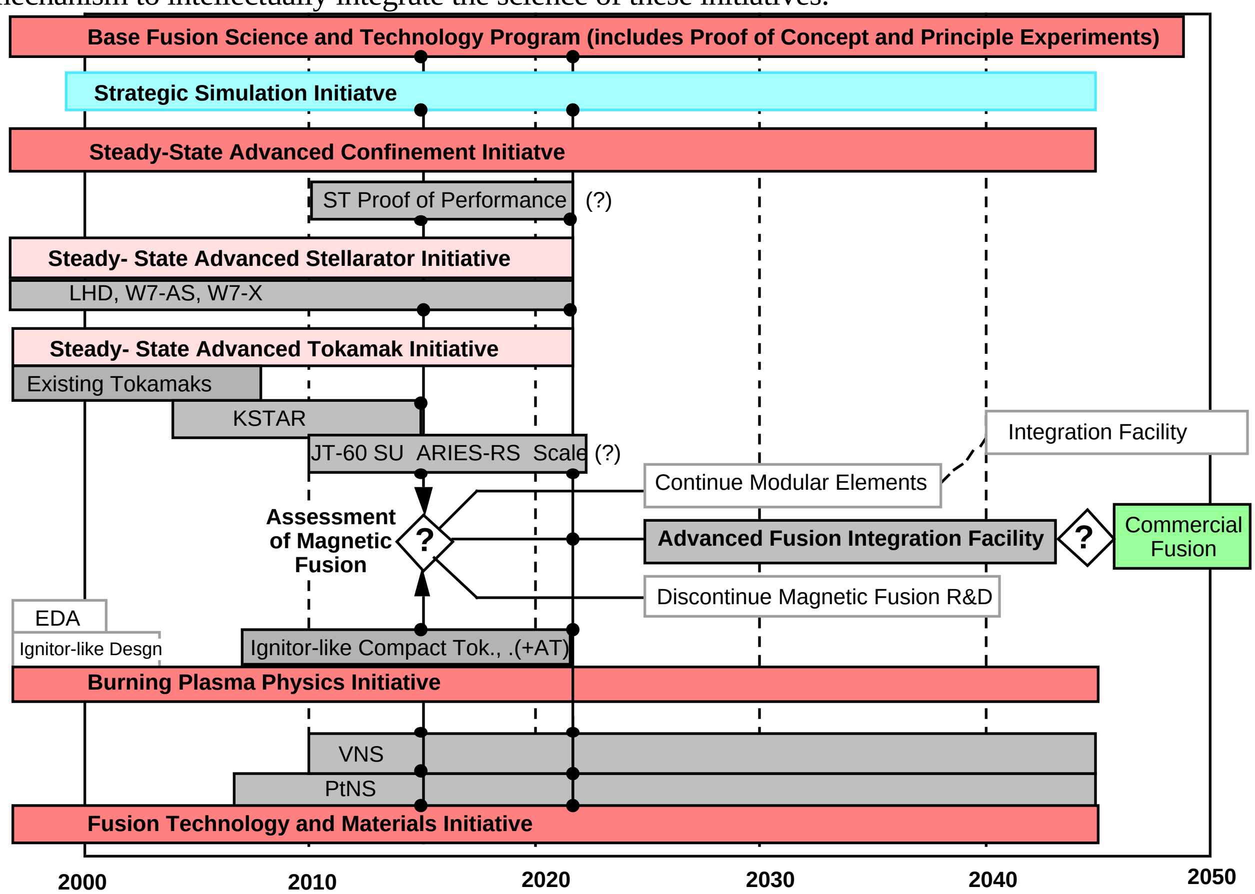

Fig. 3.1 Modular Strategy for the Magnetic Fusion Program (the grey bars correspond to operation)

The Steady-State Advanced Confinement Initiative would be addressed by extensions of ongoing research with advanced tokamaks (DIII-D, C-Mod, JET and JT-60 U), very long pulse superconducting tokamaks (Triam and Tore Supra), new superconducting tokamaks under construction (KSTAR, SST-1), long pulse stellarators (W-7AS), two new $1B class superconducting stellarators (LHD, W-7X) and new facilities in the spherical torus and advanced stellarator configurations. A possible major new facility in this Initiative is the JT-60 SU, which if constructed would be capable of addressing fully the physics of steady-state advanced tokamak physics at the reactor plasma scale of ARIES-RS. Many of these advanced toroidal configurations are able to take advantage of fundamental toroidal plasma physics that was first developed and understood using the pulsed tokamak as a research tool to access and understand fusion plasma conditions. It is probable that the Steady-State portion of the modular pathway which carries forward the superconducting tokamak development path will be implemented by the machines listed above that will be built outside the United States.

There are currently no facilities in the world magnetic fusion program capable of the study of high-energy-gain, burning plasma issues. TFTR and JET carried out successful initial experiments with weakly burning D-T plasmas that were limited in plasma duration in 1993-97. JET is scheduled to carry out another series of weakly burning D-T experiments near the end of 2002. The TFTR and JET experiments have not only produced D-T fusion plasmas with Lawson parameters ($n_i\tau_E T_i$) within a factor of 10 of that required for ignition but most importantly confirmed that D-T experiments could be carried out safely in the laboratory. The magnetic fusion program is technically ready today to begin construction of a $1B scale Ignitor-like compact ignition tokamak. The major thrust of the proposed Modular Pathway is to build a burning plasma facility at the earliest possible time as recommended by the Grunder FESAC Panel. The objective for the burning plasma initiative is to achieve, explore, understand and optimize strongly burning plasmas in a toroidal magnetic configuration. An analysis using the present tokamak data base indicates that a compact tokamak configuration would achieve the desired burning D-T plasma performance ($Q \geq 10$) for pulse duration (>> energy confinement time and ~ plasma current redistribution time) needed to satisfy the burning plasma physics objectives. An important characteristic of the compact tokamak is that ignition can be achieved in a physical size much smaller than the final power plant such as ARIES-RS. Therefore, the incremental construction cost of this facility might be minimized to ~ ≤$1B with construction taking ~7 - 8 years. The generic toroidal burning plasma physics information from this initiative would provide a foundation for understanding burning plasmas in the advanced tokamak, advanced stellarator and spherical torus configurations.

The Strategic Simulation Initiative (SSI) is a key element of the Modular Strategy. First, the SSI will be a powerful capability in developing the fundamental physics understanding of the Steady-State Advanced Confinement Initiative (Advanced Tokamaks, Advanced Stellarators and Spherical Tori) and in the Burning Plasma Initiative which uses the pulsed tokamak to cost-effectively access burning plasma conditions. The major advantage of the SSI will be to intellectually integrate the fundamental burning plasma physics understanding from the Burning Plasma Initiative and the fundamental physics understanding from the Steady-State Advanced Magnetic Confinement Initiative that will allow the development of an optimized step forward in magnetic fusion, the Advanced Fusion Integration Facility. The SSI is expected, in fact, to play a key role in all three pathways discussed in this report.

The Fusion Technology and Materials Initiative would focus on the critical task of developing and testing advanced materials that would lead to an attractive fusion power plant. An essential capability needed in this area is an intense neutron source capable of irradiating candidate materials to power plant scale fluences. A conceptual design for the Point Neutron Source (PtNS) has been developed through an IAEA collaboration and is estimated to cost ~$0.8B. A volume neutron source would test larger size (~10m^2) sub-components to prior to reactor scale integration and is expected to have a construction cost in the range of $1-2B. Such facilities will be needed in the other two pathways described in this report.

Because the costs of the various facilities all exceed the amount that would be available in the US fusion budget under the present constraints, international collaboration would be required to implement this modular strategy.

The Three Major Fusion Initiatives would be carried forward to a Magnetic Fusion Assessment Check point in ~2015 which would review the status of magnetic fusion and decide whether to (1) proceed forward to an Advanced Fusion Integration Facility, (2) extend the Modular phase or (3) move to another innovative confinement concept.

3.2 Rationale for the Modular Program Strategy

3.2.1 Hardware Integration Strategy

The fusion R&D program has used the modular approach for the first decades of research and has understood that these program modules would be integrated near the final stages of fusion development. However, fusion is still in the research phase at this time. Significant progress has been made in producing reactor plasma conditions for short durations in the laboratory that gives encouragement that a solution is possible, but the knowledge base does not exist at the present time to build an attractive fusion power system.

The most efficient approach to pursue fusion R&D objectives at this time is to focus on critical issues in each sub-area, and to develop the knowledge in each sub-area to near that needed for integration at the energy production scale. The advantages of this approach are:

- allows the flexibility needed in a research program,
- reduces cost and time for individual steps, and
- allows innovation to be incorporated earlier.

Systems studies which have been used to evaluate linkages between sub-areas and the Strategic Simulation Initiative will be utilized to provide a “virtual” integration of the research modules. Physical hardware integration of the three main research modules should be done only when needed to address issues in those sub-areas, or when the status is near reactor levels and integration is the main objective.

Fusion has a particular challenge at this time to not only demonstrate the scientific and technological feasibility of magnetic fusion, but to also develop economically and environmentally attractive fusion power systems. Keys to this are advanced magnetic confinement systems with high fusion gain, high power density and high duty cycle preferably steady-state plasmas, the corresponding enabling technology and the necessary nuclear technologies with attractive environmental characteristics such as low activation and the ability to withstand the neutron fluence. The Modular Program Pathway has focused program elements or initiatives that are targeted on addressing these issues.

3.2.2 Toroidal magnetic confinement systems have generic physics and technology issues, and the pulsed tokamak is an effective tool for developing the generic physics and technology.

Now that fusion plasmas have been produced in the laboratory using the pulsed tokamak configuration, the emphasis is broadening to develop features for improving the characteristics of toroidal magnetic confinement as an economic and environmentally attractive energy source. The advanced tokamak, the spherical torus and the advanced stellarator all emphasize features which are based on the fundamental science of toroidal plasmas developed by the conventional tokamak such as shaping of plasma and magnetic profiles for increased β, utilization of the self-produced bootstrap current to optimize the magnetic configuration, and sheared plasma flows to reduce losses due to plasma turbulence. The initiatives underway and proposed for spherical tori and stellarators emphasize the common features and generic nature of the physics and technology of toroidal magnetic systems. The stellarator will explore configurations with low recirculating power that are expected to avoid plasma disruptions. The spherical torus may offer a cheaper next step in the development path since it allows smaller burning plasma devices than can be built using superconducting coil technology. While the spherical torus and the advanced stellarator are presently not as well developed as the advanced tokamak, they are expected to benefit greatly from the tokamak knowledge and infrastructure base.

Two major issues for toroidal magnetic confinement are: (1) the scaling of confinement in alpha heated plasmas and (2) the effect of dominant alpha heating on the magnetic configuration, plasma energy confinement and potential alpha driven instabilities. The basic physics of these processes has been studied using neutral beams or radio-frequency waves to simulate the effects of alpha heating. Information on the scaling of confinement during strong alpha heating and the magnetic configuration parameters required for ignition is central to developing magnetic fusion. In addition, alpha heating depends on the local plasma parameters, which in turn depend on local plasma confinement and alpha heating. Understanding and controlling this complicated non-linear feedback loop is a critical issue for all advanced toroidal magnetic systems - advanced tokamak, spherical torus and advanced stellarator, and experiments with high gain plasmas are needed. The basic strategy for the Burning Plasma Physics Initiative is to continue to use the pulsed tokamak as a research tool to cost effectively access strongly burning plasmas and to address these fundamental burning plasma issues for all toroidal configurations.

Plasma heating, current drive, fueling, particle and power exhaust are also generic and common plasma technology issues for the advanced tokamak, spherical torus and advanced stellarator. Tritium retention and handling, remote maintenance and blanket technology are closely related nuclear technologies for all toroidal systems as well. Detailed systems studies of potential power plants based on the advanced tokamak, spherical torus and modular stellarator confirms that these toroidal systems are almost identical in their capital cost and cost of electricity (COE), and are very similar in other characteristics such as plasma volume and magnet energy as shown in Table I.

Table I. System studies of advanced tokamak, spherical torus and stellarator power plants. Advanced Reactor Innovation Evaluation Study (ARIES)

	Spherical Torus (A = 1.6)	Modular Stellarator	Advanced Tokamak
Power (Thermal), GW	3.8	2.3	2.6
Power (Net Elec), GW	1.0	1.0	1.0
Capital Cost, $B(1992$)	4.3	4.3	4.2
COE, mil/kWh (1992$)	84	75	76
Plasma Volume (m^3)	581	735	349
Magnetic Energy, GJ	76	80	85
Plasma Current, MA	29	~0	11.3

3.3 Technical Contributions of the Modular Plan

3.3.1 Burning Plasma Physics Initiative

The fusion program needs the capability to extend the frontiers of fusion plasma physics that will enable discoveries in previously unexplored parameter space that have the possibility to lead to more attractive fusion regimes. The coupling of advanced toroidal physics with strongly alpha-heated plasmas is a key issue for the development of attractive toroidal magnetic reactors whether they are classified as advanced tokamaks, spherical tori advanced stellarators or reversed field pinches. The achievement of an ignited ($Q \geq 10$) plasma will allow these scientific objectives to be achieved.

Objectives for the Burning Plasma Physics Initiative

- Determination of the conditions required to achieve high Q energy producing plasmas.

- Control of high Q plasmas through modification of plasma profiles and external sources.
- Determination of the effects of fast alpha particles on plasma stability.
- Sustainment of high Q plasma - high power density exhaust of plasma particles and energy and alpha ash exhaust, some evaluation of alpha heating on bootstrap current profiles.
- Exploration of high Q burning plasma physics in some advanced configurations/operating modes that have the potential to lead to attractive fusion applications.

These objectives would be pursued with a phased operating program very similar to that proposed for the Reduced Cost ITER program.

Phase 0 : Evaluate plasma regimes and confirm hardware capability using deuterium plasmas.

Phase I : Demonstrate, control and optimize strongly burning D-T plasmas ($Q \geq 10$) for an extended duration. Assume base line ITER performance [$H_H \sim$ 0.85*ITER 93-H(Elm-free), $\beta_N \leq 2.5$] in line with the present conventional tokamak data base.

Phase II: Demonstrate, control and optimize enhanced performance (e.g. gyroBohm) or advanced physics modes (e.g.,TPX/ARIES-RS) in strongly burning plasmas for an extended duration.

Phase III: Demonstrate controlled ignition ($Q \geq 10$) and extended burn.

Coupled technology issues (e.g., plasma exhaust/plasma facing components, tritium handling and remote handling) will also be addressed at conditions approaching those anticipated in a fusion system with the exception of some steady-state requirements.

<u>Physics Requirements for an Advanced Burning Plasma Experiment</u>

The physics of a burning plasma can be explored if the parameters listed below are attained.

$Q \geq 10$, $P_\alpha/P_{Heat} > 66\%$,	- alpha heating dominant but still easily controlled
Burn time $\geq 10\ \tau_{\alpha s}$	- alpha heating, fast alpha effects (e.g., TAE)
$\geq 10\ \tau_E$	- pressure profile evolution due to alpha heating
$\geq 3\ \tau_{He}$	- helium ash accumulation
$\geq 3\ \tau_{cr}(\tau_{current\ redistribution})$	- evolution of bootstrap current

The initial D-T experiments on TFTR and JET confirmed the single particle confinement requirements for alpha particles and were able to detect weak alpha heating in agreement with expectations (TFTR-1995, JET-1997). At $Q > 10$, alpha heating will dominate the plasma heating and the effect on energy confinement and pressure profile can be determined. The alpha slowing down time is in the range of 0.1 to 0.5 s for the Burning Plasma experiments to be discussed and is sufficiently short so that the alpha distribution is in equilibrium. The energy confinement time ranges from 0.6 to 3 s for experiments to be discussed and is short compared to burn times anticipated. The alpha ash confinement time is expected to range from 4 to 10 τ_E or from 2.4 to 30s. The devices with shorter pulse lengths operate at higher densities which means the τ_E is shorter for the same $n\tau_E$. The normal conductor devices under consideration are expected to have burn times of several helium ash transport times. The current redistribution due to alpha heating modifications of the bootstrap current profile is a key issue for advanced burning plasma experiments. This requirement is more difficult to satisfy and must be determined for each device and specific operating mode. The current redistribution time, τ_{cr}, is $\approx 22\ \kappa a^2 T_e^{3/2}$ s where κ is the elongation, a is the minor radius in meters,

T_e is the average electron temperature in 10 keV and τ_{cr} is in seconds. The larger devices tend to have longer burn times but τ_{cr} increases at roughly the same rate. A lower temperature obtained by operating at higher density (bounded by the Greenwald limit) allows τ_{cr} to be reduced. For baseline physics assumptions and full magnetic fields the typical pulse lengths correspond to ~ 1 τ_{cr}. Fortunately, as advanced tokamak performance is attained the plasma current and magnetic field can be reduced allowing a very substantial increase in pulse length for normal conductor devices so that pulse lengths of several τ_{cr} can be attained.

Possible Facilities for the Burning Plasma Initiative

A normal conductor burning plasma device has the advantage of providing high magnetic fields and plasma currents at a reduced size and cost relative to a superconducting system since a neutron shield is not require to protect the toroidal and poloidal coils thereby allowing the major radius to be reduced. A significant cost savings is also realized by using copper alloys and inertially cooled cryogenic technologies. The copper coil systems could also allow for stronger and more flexible plasma shaping which is desirable for advanced burning plasma experiments.

A number of copper coil burning plasma devices have been studied including Ignitor (1978-98), CIT (1986-89), BPX(1990-91), BPX-AT(1991-1998) which are precooled to cryogenic temperatures prior to the pulse, HLT(1990) which was actively cooled with LN and PCAST(1996) which was precooled to liquid nitogen. The general physical parameters of these devices are summarized and compared with ARIES-RS and ITER-EDA in the following table. Projections of D-T performance for these facilities using the same methodology as ITER RC is given in a later section.

Table II. Parameters of Burning Plasma Facilities

	R (m)	a (m)	κ	A	B (T)	W_{mag} (GJ)	Ip (MA)	FlatTop (s)	Pfus (MW)	Cost ($B)
TFTR	2.5	0.9	1.0	2.8	5.6	1.5	2.7	2	11	~1
JET	2.85	0.85	1.6		3.8	1.5	4.2	5	16	~1
Ignitor	1.32	0.47	1.85	2.8	13	4	12	5	150	0.5
Spherical Torus	1.05	0.75	3.0	1.4	2.6	0.5	16	steady	800	0.8
BPX-AT	2.00	0.50	2.0	4.0	10	4	6.25	12	150	0.70
CIT	2.14	0.65	2.0	3.3	10	4	11	10	150	0.73
BPX	2.59	0.79	2.0	3.3	8.1	9	10.6	10	400	1.63
Gyro-Bohm JET	3.0	1.0	2.0	3	6		10		360	~3
HLT	3.40	1.20	2.0	2.8	8	15	20	40	500	NA
PCAST	5.00	1.50	1.75	3.3	7	35	15.3	120	400	5.3
ARIES-RS	5.52	1.38	1.7	4.0	7.98	85	11.3	steady	2170	4.6
ITER-EDA	8.14	2.80	1.6	2.9	5.68	120	21	1000	1500	10.7

The cost estimates are in FY1995$ and are for the construction project not including site costs.

The construction costs of fusion confinement facilities are a strong function of physical size, plasma current and energy stored in the magnets as suggested in Table II. An empirical fit to the cost estimates for several devices yields cost ~ $B^{0.87}$ $R^{1.85}$ (J.Schmidt, 1995 PCAST Machine Study) and Aamodt (Madison Forum) has found a similar relation with cost ≈ $25BR^2$. The costs for CIT and BPX are the cost estimates at the end of their Preliminary Design Phase escalated to FY1995$. The BPX-AT cost estimate is based on a Conceptual Study during the 1991 New Initiatives Task Force activity and was a detailed cost estimate based on scaling down from BPX. The Ignitor cost estimate of $0.5B is on the high end of various cost “estimates”. Ignitor, CIT, BPX-AT and BPX were costed on the basis of siting at an existing large tokamak or equivalent site. The PCAST and ITER cost estimates do not include site costs outside the fusion facility and its direct facilities costs. The ARIES cost is the total direct cost for the 10th of a kind. Consideration should be given to special studies that look for new design features or manufacturing techniques to reduce the unit construction costs for the tokamak core. The general conclusion is that compact tokamak facilities (Ignitor, BPX-

AT and CIT) with major radii ≤ 2m have construction costs in the ≤$1B class, while larger superconducting devices necessarily have major radii ≥ 5m with costs in the $5B range.

Estimated Fusion Performance of Normal Conductor Burning Plasma Experiments

The performance of PCAST, Ignitor, CIT, BPX and BPX-AT was estimated using a zero-D model assuming ITER-93H (ELM-free) confinement scaling with alpha heating and fuel depletion due to alpha ash accumulation calculated self-consistently. The plasma profiles were taken to be the same with a flat density profile ($\alpha_n = 0.1$) and modestly peaked temperature profile ($\alpha_T = 1$). The impurity levels were taken to be 3% Be and the alpha ash was assumed to have a confinement time $\tau_{He} = 5\ \tau_E$ resulting in $Z_{eff} \approx 1.5$. These assumptions are the same as the modeling assumptions made for the Reduced Cost ITER with the exception that ITER-RC assumes $\tau_{He} = 10\ \tau_E$ for the baseline performance mode and $\tau_{He} = 5\ \tau_E$ for advanced performance mode. For BPX-AT at Q = 10 increasing τ_{He} from 5 to 10 τ_E increases the required H93-Elmfree H factor by 10%. Some of these calculations are summarized in Fig. 3.2 below.

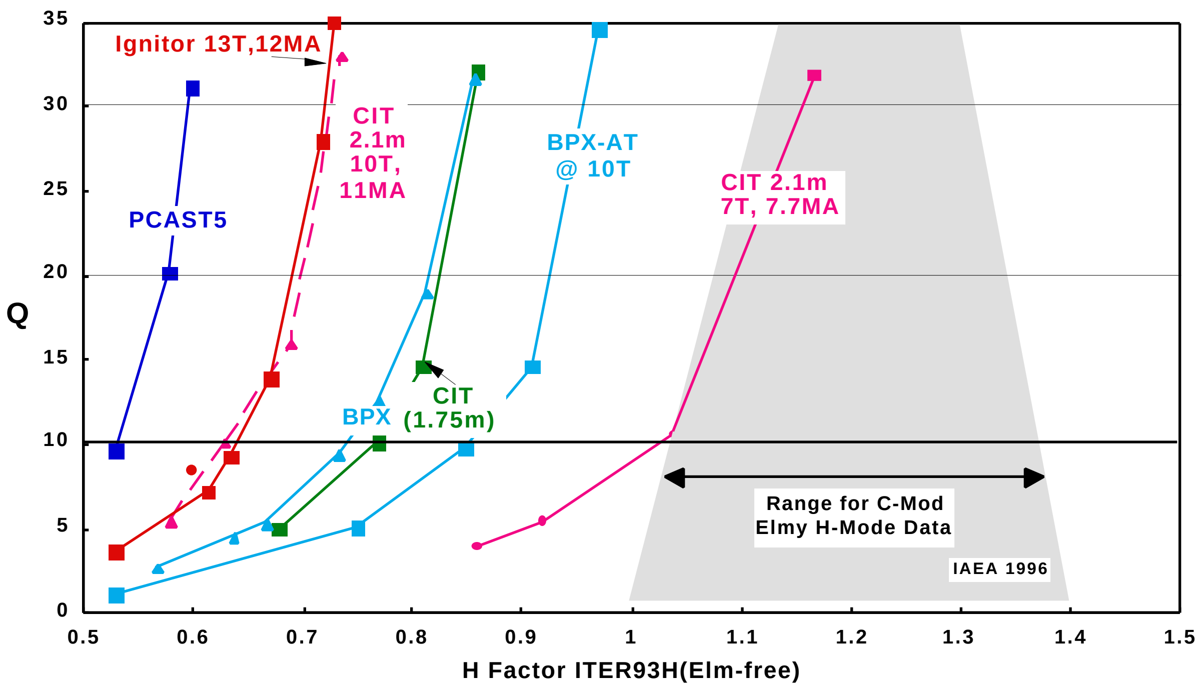

Fig. 3.2. Fusion power gain estimates for several burning plasma experiments.

The existing confinement data base from all tokamaks is centered about an ITER93(Elm-free) H factor of ≈ 0.85 while the confinement data from Alcator C-mod, a prototype for the compact ignition tokamaks, has ITER93(Elm-free) H factors of ≈ 1.2. Ignitor and CIT are projected to ignite for ITER93(Elm-free) H factors of ≈ 0.85 which is the same requirement as for ignition in the ITER-EDA. Other Ignitor-like devices with somewhat lower field and plasma current such as BPX-AT are projected to achieve Q ≥ 10 ITER93(Elm-free) H factors of ≈ 0.85 and would ignite at ITER93(Elm-free) H factors less than those achieved in Alcator C-mod. Note that the full field performance of Ignitor and CIT(2.1m) is nearly the same suggesting that Ignitor would ignite at ~70% of full field and current if the enhanced Alcator C-Mod performance could be attained in D-T. At this reduced field, the flattop of Ignitor would be extended from 5 seconds to 15s.

Compact Ignition Tokamaks

The compact tokamaks (e.g., Ignitor and BPX-AT) first advocated by Coppi with higher magnetic field and higher plasma densities have additional advantages with respect to beta limits, operating density limits, impurities and fast alpha particle limits and are well suited for studying burning plasma physics during the time scales of interest. Recent results from tokamak confinement experiments, in particular Alcator C-Mod, confirm the high field compact ignition tokamak design assumptions with regard to confinement, ICRF heating, power handling and impurity control. Alcator C-Mod with ELMing H-modes tends to operate with 20-30% higher confinement relative to ITER93H scaling than larger lower field lower density tokamaks and provides confidence that Ignitor-like compact tokamaks would achieve the performance required to achieve the burning plasma objectives.

Detailed engineering designs have been carried out for copper alloy coils cooled to cryogenic temperatures and significant operating experience has been obtained. Ignitor is designed to have copper alloy coils that are precooled to 30 °K with liquid hydrogen while BPX-AT, CIT, BPX designs have copper alloy coils that are precooled by LN to 77 °K. The pulse length is determined by the adiabatic temperature rise of the conductor/structure thermal mass during the pulse. A small reduction in the peak coil current allows the pulse length to be increased dramatically for the coil/cooling configuration. For example, reduction of the magnetic field in BPX-AT from 10 T to 7 T allows the magnetic field flat top to be extended from 12 s to 56 s. This feature of Ignitor-like compact tokamaks can be used to advantage for studying advanced tokamak regimes where improved confinement and β allow the plasma current and magnetic field to be reduced as shown in the example below for BPX-AT (Fig. 3.3). In order to exploit this capability the initial design would need to incorporate or allow upgrades for active cooling of internal divertor components (as in BPX-AT) and techniques to pump helium ash during the pulse.

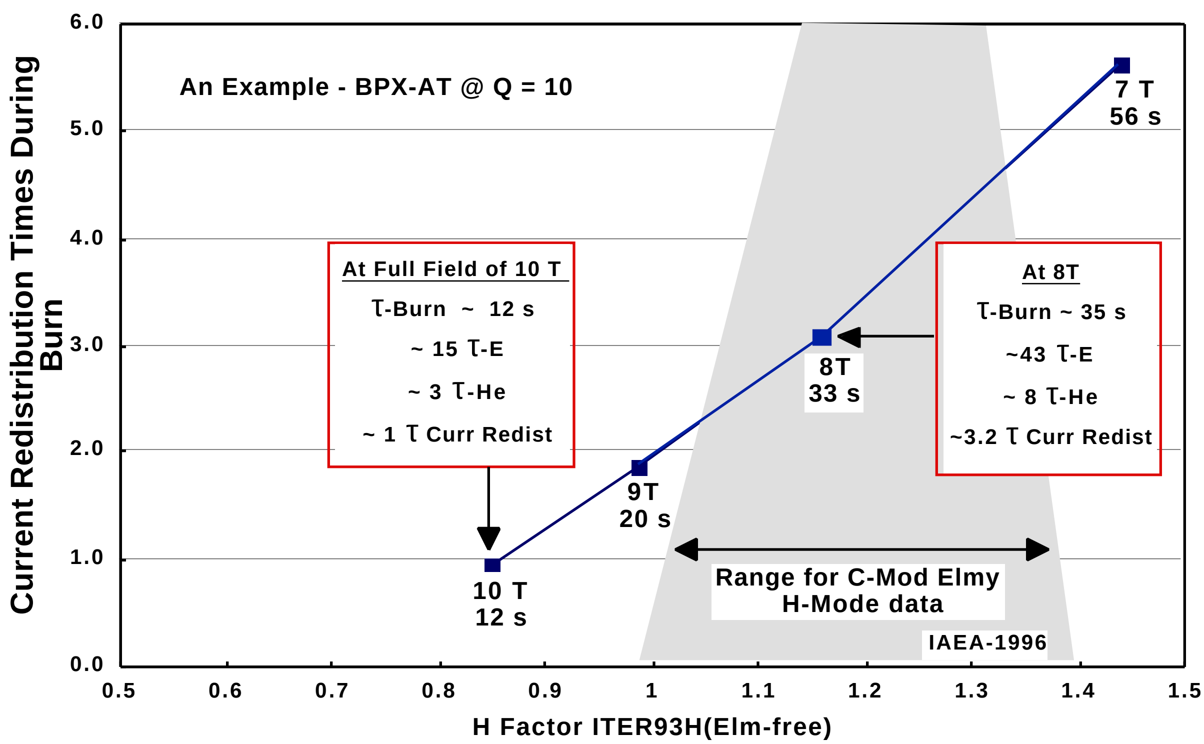

Fig. 5 Capability of an inertially cooled Ignitor-like tokamak to produce pulses with long burn (magnetic field flat top) times that would allow several plasma current redistribution times for studying advanced modes. Compact helium ash pumping systems would have to be provided in the divertor for the long pulses available.

An important point is that the Ignitor-like compact tokamaks can explore a broad range of experimental operating space by density variations and by reduction in the magnetic field/plasma current since they have large margins with regard to density, MHD beta and TAE limits.

After alpha heating, the primary alpha physics issue concerns alpha-driven toroidal Alfvén eigen (TAE) mode instabilities which could cause the loss of energetic alpha particles before effective alpha heating had occurred. Ideally, the burning plasma experiment should be able to avoid these instabilities while achieving high Q and to then controllably approach the stability boundary to determine the physics constraints for future devices. The TAE instability occurs when the alpha particle speed, V_α, is ≥ the Alfvén speed, $V_{Alfvén}$, and (R/a) β_α exceeds a threshold that depends on details of the plasma and magnetic profile (e.g., β and shear). It can be shown that β_α depends on $\beta T_e^{5/2}$ and the temperature can be varied by adjusting the plasma density. The curves in Fig. 3.4 are density scans for 0-D calculations where Q was held constant (by adjusting H) while maintaining constant helium ash and impurity fractions. The Ignitor-like compact tokamaks can scan the same general range of TAE instability parameters space as the ITER-EDA and PCAST devices.

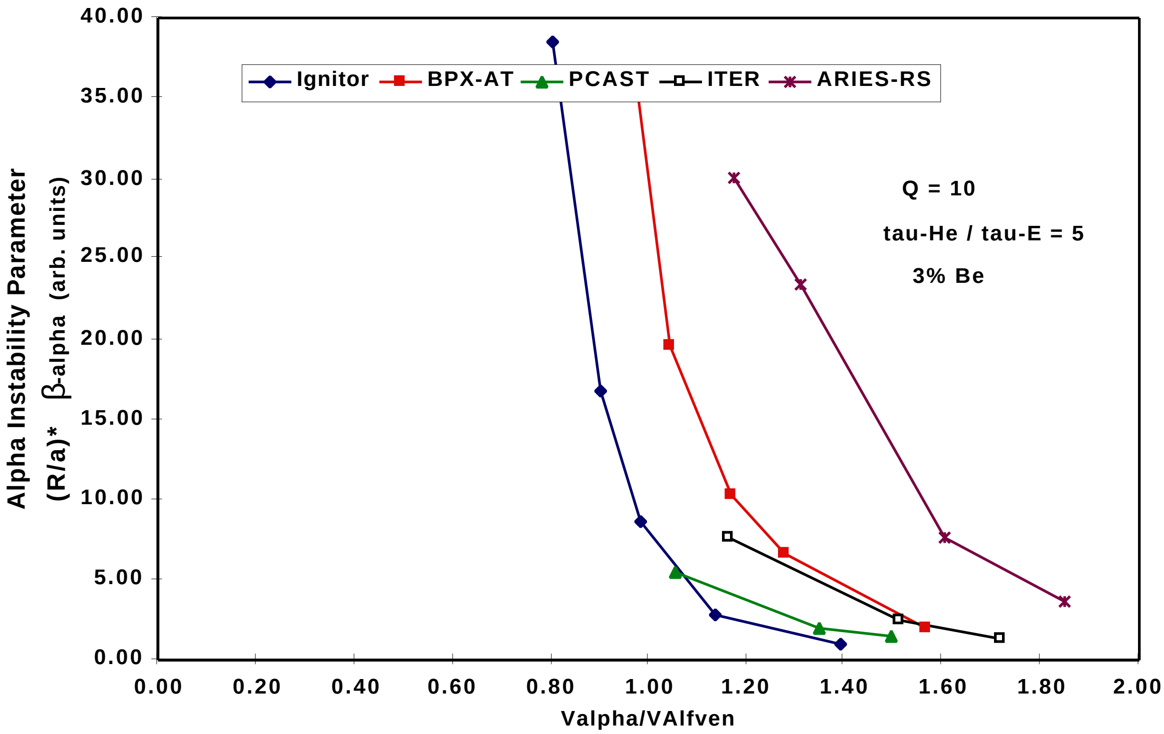

Fig. 3.4. Comparison of parameters that determine the instability boundary for alpha-driven toroidal Alfvén eigen (TAE) modes.

Intermediate Sized Ignition Experiment based on Gyro Bohm Scaling

A moderate size normal conductor tokamak (R = 3m, B ~ 6 T, I_p ~ 10 MA) experiment based on extrapolation of DIII-D/JET gyro-Bohm scaling experiments has been proposed. This physics mode is not considered to be an advanced tokamak mode, such as the reversed shear mode, and therefore enhanced performance is achieved without the need for strong active plasma profile control. The advantage is a more robust (i.e., reliable) plasma configuration with potentially fewer plasma disruptions. A concept study for a similar size tokamak has been carried out for a high-performance long-pulse tokamak (HLT) with parameters as shown in Table II. The HLT design, which is also inertially cooled, increases its adiabatic pulse length by using the thermal mass of an external liquid nitrogen reservoir which is initially subcooled to 63.5 °K prior to each pulse. The toroidal field coil and central solenoid are actively cooled during the pulse by circulating the LN through the cooling channels. This design allows the pulse length to be arbitrarily extended by enlarging the liquid nitrogen reservoir, but reduces the maximum attainable magnetic field to accommodate the cooling channels. Active LN cooling of this type is being evaluated as a possibility for extending the pulse length on compact tokamaks as well as moderate scale tokamaks. Further work is needed in a multiple machine program to test the gyro-Bohm scaling and to investigate pedestal scaling especially at higher β_N.

Findings on the Burning Plasma Physics Initiative

1. The compact high field tokamak utilizing cryogenic normal conductors is a potential pathway to access $Q \geq 10$ conditions and address burning plasma physics with a facility costing ≤$1B. In addition, evaluation of burning plasma physics in an advanced configuration for up to several skin times is possible.
2. Technological issues associated with the longer pulses (e.g. helium pumping and other internal components) in a compact high field tokamak need more detailed analysis and should be updated to include recent results.
3. Intermediate size (JET-scale) normal conductor tokamaks offer interesting possibilities for burning plasma physics research with costs in the ~$1.5B range.
4. Larger burning plasma tokamaks utilizing superconducting coils while somewhat more capable have costs in the several $B range.

3.3.2 Steady-state Advanced Confinement Physics Program Initiative

Objectives of the Steady-state Advanced Confinement Physics Initiative

The development, exploration and detailed understanding of high confinement, high fusion power density and high duty-cycle (steady-state) plasmas is needed for the development of economically and environmentally attractive applications of fusion power. This initiative includes subprograms on steady-state advanced tokamaks, steady-state advanced stellarators and other Proof of Performance experiments that might emerge from the ongoing Base Fusion Science and Technology Program such as the Spherical Torus.

Steady-state Advanced Tokamak Program Initiative

Outstanding progress in exploring and understanding advanced plasma regimes for short pulse magnetically confined plasmas has been made by a number of specialized medium-sized and several large size pulsed tokamaks. The next frontier in the advanced tokamak configuration is to extend high performance advanced tokamak regimes to near steady-state conditions in fusion-relevant plasmas so the plasma physics (with the exception of actual alpha physics) of a magnetic fusion plasma can be understood and optimized. The strategy is to use H and D plasmas rather than D-T plasmas to increase experimental flexibility and to reduce costs associated with tritium handling and neutron activation that are present at burning plasma facilities.

The steady-state advanced tokamak program will be dedicated to the development of the scientific basis for a compact and continuously operating tokamak fusion reactor. It will explore techniques for optimizing steady-state plasma performance through active control of the current profile, the pressure profile, the radial electric field profile, transport barrier formation, of plasma-wall interactions, and by advanced plasma shaping. Key areas to be optimized are the averaged plasma pressure (or β) through wall stabilization, the plasma confinement through transport barriers, and the current drive efficiency. This will involve making efficient use of the self-driven bootstrap current to provide a substantial fraction of the total plasma current. The integration of optimized plasma performance and efficient continuous operation will be a key issue, as will be control of major plasma disruption. Advanced tokamak operation will be at reduced current levels compared to conventional tokamaks and this will reduce the impact of disruptions.

Draft
7/17/98

Requirements for the Next Step Steady-state Advanced Tokamak Initiative

The plasmas needed to resolve the steady-state advanced confinement issues should have physics phenomena similar to that projected for a fusion plasma. This requirement would be satisfied if the dimensionless plasma parameters ρ , ν and β are comparable to those in a strongly burning (e.g., Q > 10) D-T plasma. Kadomtsev (Nuclear Fusion 1975) has shown that plasmas can have the same ρ , ν and β if they have the same similarity parameter, $Ba^{5/4}$. In addition, the pulse length must be sufficient to allow a thorough study and controlled modification of plasma current evolution in advanced tokamak modes (e.g., pulse length $\geq 10\ \tau_{cr}$), and to allow the plasma wall/divertor interaction to come into equilibrium. Some aspects of alpha particle physics (e.g., TAE modes in reversed shear magnetic configurations) can be simulated using the injection of high power beams and RF power. The physics issues and criteria have been previously discussed in detail in the Tokamak Physics Experiment (TPX) conceptual design.(e.g., IAEA paper 1994)

In addition to having dimensionless physics parameters similar to those in a strongly burning plasma and having long pulses for current control and wall equilibration, there are several other physics issues for this initiative. An advanced flexible divertor configuration is required to study the interplay between edge conditions needed for enhanced confinement modes and the requirements for heat dispersal and limited recycling back to the main plasma. The poloidal field system and divertor configuration must allow a wide range of plasma shapes and plasma current profiles to be explored. In addition, detailed steady-state current profile control is needed, necessitating the requirement for multiple current drive systems. Plasma diagnostics are particularly important in this experiment, both for fundamental understanding and for active control.

Possible Facilities for the Steady-State Advanced Tokamak Initiative

During the next 5 years the existing tokamak facilities will be exploited to address the steady-state advanced tokamak issue over the parameter space available to those devices. A number of facilities could be upgraded to extend their capabilities in this program element. However, significant new facilities are needed in the world program to fully address this issue for an advanced tokamak reactor as exemplified by ARIES-RS. The possibilities that have been considered are listed in Table III. The parameters shown are the nominal full field parameters with the exception of Alcator C-Mod.

Table III. Parameters of Steady-State (long-pulse) Advanced Confinement Tokamaks*

	R (m)	a (m)	κ	A	B (T)	$\frac{a/\rho}{\text{ARIES-RS}}$	Ip (MA)	τ_{cr} (s)	FlatTop (s)	Cost ($B)
Alcator C-Mod (Cu)	0.67	0.21	1.8	3.2	5	0.15	1	0.5	5	
DIII-D (Cu)	1.67	0.67	1.8	2.5	2.2	0.23	2	3.2	5-10	
JET (Cu)	2.85	0.85	1.6	2.8	3.8	0.41	4	10	10	~1
JT-60U (Cu)	3.1	0.9	1.7	3.5	4.1	0.45	4	14	10	~1
Triam (S/C)	0.8	0.12	1.5	6.7	8	0.13	0.03	0.2	7200	
Tore Supra (S/C)	2.3	0.80	1.0	4.0	4.2	0.42	2	5.6	120	0.5
SST-1 (S/C)	1.05	0.20	1.8	5.3	3	0.10	0.22	0.3	60	0.2
KSTAR (S/C)	1.8	0.50	2.0	4.0	3.5	0.25	2	3.2	300	~0.4
TPX (S/C)	2.25	0.56	2.0	4.0	3.35	0.27	2	4.3	1000	0.8
JT-60SU (S/C)	5.00	1.40	1.7	3.5	6.25	0.86	10(5)	64	1000(ss)	~3
ARIES-RS (S/C)	5.52	1.38	1.7	4.0	7.98	1.00	11.3	83	steady	4.6
ITER-EDA (S/C)	8.14	2.80	1.6	2.9	5.68	1.44	21	280	1000	~10

The cost estimates are in FY1995$ and are for the construction project not including site costs.

The ability to address advanced tokamak issues depends on the range of dimensionless parameters that can be accessed. Table III assumes that β/ν was constant for all devices when calculating the number of gyro-radii, a/ρ , and when calculating the plasma temperature to determine the current redistribution time, τ_{cr}. KSTAR and TPX would extend the plasma duration of JET/JT-60U plasma regimes with modest ρ , ν and β ($a/\rho \approx 0.2$ to 0.4) from ~1 current redistribution time to >> 10 current redistribution times allowing current profiles to approach equilibrium. JT-60SU would have the capability to address the steady-state advanced tokamak issue fully with a plasma duration of 14 τ_{cr} at nearly ARIES-RS values for ρ , ν and β .

An important point to note in this regard is that JT-60 SU has almost the same dimensional as well as dimensionless parameters as ARIES-RS, and hence one would expect the costs (exclusive of D-T and nuclear technologies) to be comparable. If the requirement is to study advanced tokamak physics at ARIES-RS dimensionless parameters then the test plasma must have $BR^{5/4} \approx$ to ARIES, and since the rough cost scaling for both superconducting and copper machines is cost $\sim B^{0.87}R^{1.85}$ (J. Schmidt), then the cost of the test plasma device will scale roughly as cost $\sim (BR^{5/4})^{1.48}/B^{0.61}$. A possible cost reduction could be realized by increasing the magnetic field and decreasing the major radius.

The non-burning steady-state advanced confinement experiments (e.g., JT-60 SU) would need extensive plasma heating power(~60 MW) to bring the reactor size plasmas to high temperature and flexible high power current drive systems. The high-performance long-pulse tokamaks would initially operate with hydrogen, and later transition to deuterium operation to minimize neutron activation of the structure near the plasma. KSTAR, TPX and JT-60 SU are designed with modest shielding between the plasma and the superconducting coils, and have incorporated low activation materials into the mechanical structure near the plasma.

Integration of the Burning Plasma and Steady-state Advanced Confinement Initiatives

The programs of the Advanced Burning Plasma experiment and the Steady-State Advanced Confinement initiatives would be coordinated and integrated programmatically. A comparison of plasma relevant parameters for possible devices in the two programs are shown in Table IV.

Table IV. Comparison of Potential Elements for Advanced Burning Plasma and Steady-state advanced confinement for the Next Step Modular Program.

	R (m)	a (m)	B (T)	$Ba^{5/4}$	Ip (MA)	τ_{cr} (s)	Flattop (s)	Q_{DT}	P_{Fusion} (MW)	Cost ($B)
Ignitor (full B)	1.32	0.47	13	5.1	12	8.4	5	ign	150	~0.4
BPX-AT (full B)	2	0.5	10	4.2	6.25	9.7	12	≥10	150	
BPX-AT (80%B)	2	0.5	8	3.4	5	7.7	33	≥10*	150	0.7
PCAST	5	1.5	7	11.6	15.3	85	120	ign	400	5.0
ARIES-RS	5.5	1.4	7.98	11.9	11.3	83	steady	ign	2150	4.6
JT-60 SU	5	1.4	6.25	9.5	10	64	1000	(5)*	(250)	~3
TPX	2.25	0.56	3.35	1.6	2	4.3	1000			0.8
KSTAR	1.8	0.5	3.5	1.5	2	3.2	300			~0.4

* JT-60SU and BPX-AT with $H_{ITER-89P} \approx \beta_N \approx 3$. In this table, the temperature was varied to keep β/ν constant for all devices. The plasma temperature for ignited plasmas in Ignitor can be reduced below this value by increasing the plasma density resulting in shorter current redistribution times at lower β/ν .

The burning plasma experiments can extend somewhat into advanced regimes while the advanced confinement experiments would extend into some limited burning plasma studies. JT-60 SU has considered a D-T option with additional shielding which could achieve Q ~ 5 (P_{fusion} ~ 250 MW) if $H_{ITER-89P}$ ~ 3 and β_N = 3 could be attained. This direct integration of burning plasma physics and advanced tokamak physics is an important feature that gives additional flexibility and breadth to the modular approach.

3.3.3 Strategic Simulation Initiative

The Fusion Energy Sciences leg of the Strategic Simulation Initiative will provide additional capability to integrate these two initiatives, and to transfer the generic information from the tokamak initiatives to the advanced stellarator and spherical torus confinement initiatives. This Strategic Simulation Initiative will develop and employ advanced computational methods and multi-teraflop computing resources to greatly accelerate the development of a comprehensive simulation capability for magnetically confined fusion plasmas. The elements of the Simulation Initiative include both microscopic modeling of plasma turbulence that leads to anomalous energy transport and macroscopic modeling of global plasma instabilities that lead to plasma disruptions. By calibrating this simulation capability against both the burning plasma experiment and the steady-state advanced confinement experiment, we will gain the confidence needed to use this as a design integration capability for use in defining the optimum advanced magnetic configuration and in designing the next step integration experiment. To the extent that these devices allow additional direct integration of burning plasma physics and advanced tokamak physics, it will be possible to perform additional calibrations with this simulation capability to gain additional confidence in their projections to the next step.

3.3.4 Fusion Materials and Technology

Fusion Technologies

Technology and materials will be the eventual keys to the realization of fusion's potential as an energy source. Achieving acceptable performance for a fusion power system in the areas of economics, safety and environmental acceptability, is critically dependent on performance of the blanket and divertor systems which are the primary heat recovery, plasma purification, and tritium breeding systems. A fundamental understanding of key materials and technology issues is required to develop a compatible combination of materials (breeder, coolant, and structure) for the blanket system. Key issues include chemical compatibility, neutronics, thermohydraulics and stress conditions in addition to the irradiation effects and tritium interactions . The development of blanket systems require integrated prototype module testing in a 14 MeV environment. Such testing could be conducted in the proposed Advanced Fusion Integration Facility, or in a smaller dedicated D-T test facility such as the Volume Neutron Source (VNS).

Materials

Design and performance of these key components rests in turn upon the properties and characteristics of the structural materials. Temperature limits imposed by the properties of materials are the major limitation in the quest for high thermal efficiency. The major in-vessel systems will have a finite lifetime and will require remote maintenance and replacement. Reliability and lifetime, which are primarily determined by the performance of materials, will have a significant impact on plant availability, a second major factor in the cost of energy equation. Mechanical properties of structural materials under conditions associated with off-normal events, radioactive isotope inventory, and release paths are key considerations in designing for safety. The initial levels of radioactivity of materials on removal from service and the rate of decay of the various radioactive isotopes, dictate acceptable storage and disposal methods and the possibility of recycle of materials, both being major considerations in the environmental acceptability of fusion. The development of low activation materials is a long-term endeavor and a critical issue on the path to fusion energy. Fission reactors are currently used for irradiation studies. However, a 14 MeV point neutron source (PtNS) will ultimately be needed to fully qualify materials for fusion reactor applications. Conceptual design of such a materials test facility is being pursued as part of an international collaboration.

Enabling Technologies

Technology also plays a vital role in the continued progress in fusion science. Exploitation of existing and planned experiments will depend on further advances in certain technologies, particularly those related to manipulation and control of the plasma, extension to longer pulses, and operation in a radiation environment. Within the context of the fusion program's goals to develop a low-cost, next-step device and the knowledge base for a more attractive fusion power source, the likely reduction in the size and complexity envisioned to accomplish these objectives will require improved technologies to handle higher heat loads, produce lower cost magnet designs, develop safe and efficient tritium processing systems and develop more efficient and flexible heating, current drive, and fueling systems.

Synergy Between the Inertial Fusion Energy (IFE) and Magnetic Fusion Energy (MFE) Programs

The ICF community is currently developing an integrated plan for the development of all aspects of IFE including drivers, targets, target fabrication and injection, chambers, and final optical systems that will survive in a fusion environment. There are three driver options: heavy ion accelerators, diode-pumped solid state lasers and krypton fluoride lasers. The proposed plan calls for the construction of an Integrated Research Experiment based on one of these drivers beginning in FY 2003. The potential exists for synergy between magnetic and inertial fusion energy (IFE) technology development. Several of the technologies under development by the MFE program are applicable to IFE such as:

- Pellet/target technology and delivery systems
- PFC liquid metal surfaces / IFE chamber technology
- Tritium systems
- IFE driver technology
- Materials (radiation effects)

A natural overlap between MFE and IFE is that of liquid metal surfaces for the IFE chamber wall protection which is also applicable to divertor plate protection in MFE devices. The benefits of coordination in technology development are twofold; namely, (1) both programs can leverage technology advances made in each other's field of research and (2) a reduction in duplication of effort will lead to substantial cost savings in future development programs.

Major Technology Facilities

Fusion materials must operate in a very demanding environment which includes various combinations of high temperatures, chemical interactions, time dependent thermal and mechanical loads, and intense 14 MeV neutron fluxes. Neutron irradiation is a particularly important issue, due both to its effects on physical and mechanical properties as well as the production of radioactive materials, and is the most difficult to investigate with currently available facilities.
At present fission reactors are the primary means to investigate the effects of irradiation on fusion materials. In addition there are specialized studies which can be done in ion beams (REF). However the response of materials to these various radiation fields can be quite different from that due to a fusion neutron spectrum. Various techniques have been used to more nearly reproduce the fusion environment, but a 14MeV neutron source will ultimately be needed to develop and qualify fusion materials. The source must reproduce the fusion spectrum, particularly in terms of the ratio of transmutation products to atomic displacements, and must have flux and fluence capabilities sufficient to allow accelerated testing to fluences up to end-of-lifetime. The international community has proposed as a Point Neutron Source an accelerator facility based on the D-Li interaction (IFMIF). The characteristics of the IFMIF are summarized in Table V. The Final Report of the Conceptual Design Activity estimates the cost of the facility to be ~0.8B$. Development and qualification of structural materials in such a device would be required before embarking on a high-fluence fusion-based system.

Table V. Point Neutron Source Parameters

Neutron Flux Volume	> 1 MW /m^2 >6.0 liter
	> 5 MW /m^2 0.4 liter
	> 20 MW /m^2 0.1 liter
Particle	D+
Beam Current	2@125mA
Beam Energy	32, 36 or 40 MeV
Beam spot	5 cm x 20 cm
Duty Factor	100 % (CW)
Plant Factor	70 %

The technologies needed to build a source like IFMIF are essentially in hand, and share much in common with high-power spallation sources, and the APT program. The primary uncertainties relate to the high reliability and availability requirements.

Engineering testing of nuclear components

The overall goal of the fusion technologies area is to develop a blanket system that meets the performance requirements of tritium self-sufficiency and efficient energy recovery, and at the same time achieves the desired level of safety and environmental attractiveness. While much work can be done without a high flux neutron environment, development and testing of fusion blankets and other nuclear components will eventually require a large volume plasma-based neutron source. The required technical features, summarized in Table VI, are taken from the IEA Study Summary which describes requirements for testing in fusion facilities, with emphasis on testing needs to construct a DEMO blanket.

Table VI. Fusion Nuclear Technologies Testing Requirements

Neutron Wall Load, MW/m^2	1 - 2
Plasma Mode of Operation	Steady-State
Minimum Continuous Operating Time, Weeks	1 - 2
Neutron Fluence (MW•y/m^2) at Test Module	
Stage I: Initial Fusion "Break-in"	0.3
Stage II: Concept Performance Verification	1 - 3
Stage III: Component Engineering Development and Reliability Growth	4 - 6
Total Neutron Fluence for Test Device, MW•y /m^2	> 6
Total Test Area, m^2	> 10

Fusion nuclear systems could be tested in any sufficiently sized facility, such as the Advanced Fusion Integration Facility, to demonstrate performance and reliability and activation features. However, the risk and cost of such tests are dependent on the cost and scale of the facility, leading to the idea of smaller options tailored to the testing task. A number of design studies have been performed that address the concept of VNS-like facilities, which are typically highly driven devices with Q ~ 1-2, and water cooled magnets. Examples include the Steady Burn Experiment work at MIT, the Spherical Torus concept at ORNL, the Small Fusion Development Plant at ORNL and two Small Business Innovation Research Projects focusing on Spherical and Conventional Tokamak confinement. A related but differently scoped facility, called a Pilot Plant, was pursued in 1988/89. These device concepts generally are small (e.g., R~2 meters or less), low power (e.g., 100 MW), normally conducting, jointed magnet tokamaks where the designs emphasize maximum access to the fusion core. Costs have been estimated in the 1-2B$ range. While the plasma performance required is less than that for a high-gain burning plasma facility, significant technology and materials development would be needed before embarking on this step.

3.4. Pathway Implications for the Modular Plan Option

The elements of a Road Map for magnetic fusion based on the modular approach are shown in Fig. 3.1. The burning plasma (BP), steady-state (SS) and nuclear technology mission elements of ITER are separated and addressed by specialized focused modular initiatives. The modular initiatives have greater flexibility to explore and exploit innovations to improve the attractiveness of magnetic fusion. For example, the burning plasma physics element can begin studies of more conventional regimes and then incorporate advanced tokamak physics into the later phases at lower cost. In addition, the steady-state non-burning plasma element is relieved of nuclear technology constraints and has much more flexibility to pursue experiments dedicated to developing innovative advanced tokamak concepts. This pathway also allows the alternate magnetic configurations to be evaluated at the Proof of Performance level before the specifications of the Advanced Fusion Integration Facility are determined.

While both the steady-state advanced tokamak and the burning plasma physics elements will develop significant enabling technologies, this plan explicitly calls for facilities dedicated to the development of fusion technologies and materials. The need for a VNS has been identified by several studies (Finesse, IEA) which called for the capability to test components with ~$10m^2$ exposed to neutron fluxes of ~2 MWm^{-2} and fluences up to 6 $MWam^{-2}$. In the modular plan, the focus is toward integrating advanced plasma physics, nuclear technology and advanced materials in an Advanced Fusion Integration Facility thereby allowing time for materials to be tested in the PtNS at high fluence.

The mission for facilities that follow the next stage of fusion R&D is a significant issue. The facility following the Next Steps is identified in Fig. 3.1 as a Advanced Fusion Integration Facility which would integrate and demonstrate all technologies and operating scenarios required for an attractive fusion power plant, and would therefore provide the scientific and technological foundation for a commercial fusion power plant. It is anticipated that the Advanced Fusion Integration Facility, which could be more advanced than ITER-RC, would demonstrate the entire fuel cycle and perhaps the production of electricity. The following Commercial Prototype would in turn be somewhat more advanced than the DEMO in the ITER-RC strategy and would have the important mission of demonstrating the high reliability needed for a commercial fusion power plant.

3.5 Advantages and Concerns for the Modular Program Pathway

The Modular Program Pathway follows from the viewpoint that magnetic fusion is still a research program and is not yet ready for a development step at the scale of a fusion power plant. The advantages of the Modular Program Pathway described in Section 3.1 include:

- The Modular Pathway allows the rapid development of toroidal ignition and burn physics in the only toroidal confinement device presently capable of reaching that physics regime -- the pulsed tokamak. This can be done less expensively and with greater assurance of success than in an integrated facility such as ITER because it does not have to carry out other elements of the program.
- The critical magnetic fusion issues are naturally modular and are maturing on different time scales. The Modular Pathway allows programs and facilities to be optimized with focus on specific technical issues and to have different time scales for resolution. The steps are more appropriate to our current level of knowledge than the "single step to DEMO" pathway.
- The Modular Pathway allows greater flexibility to explore and accommodate innovations and optimization of individual elements prior to a large scale integration facility. The Advanced Fusion Integration Facility (AFIT) would have more attractive

features and less technical risk than the single step approach. Additionally, should the AFIT be designed to produce electric power, it could be considered a DEMO.

- The technical basis exists now to proceed with the burning plasma, advanced tokamak and materials development initiatives described. The Modular pathway allows for early investigation and resolution of burning plasma physics, steady-state advanced confinement physics and fusion materials and technology issues.
- The "advanced" tokamak is a very cost effective research tool for advancing generic toroidal plasma physics (enhanced performance, steady-state and burning plasma physics) and enabling technology that will benefit other innovative toroidal magnetic concepts that are being developed in the Steady-State Advanced Confinement Initiative.
- The Modular Pathway has multiple less-costly facilities which substantially reduces the technical risk associated with a single facility. Multiple facilities also offer the flexibility of multiple international locations, and allow financial commitments to be made in smaller amounts and phased over time.

The concerns about the Modular Pathway include:

- The integration of burning plasma physics, steady-state operation and nuclear technology at the energy scale is delayed.
- The Modular Pathway is a departure from the present international plan and is presently inconsistent with the national fusion strategies of our ITER partners. An explicit part of the European and Japanese national fusion plans, at this moment, is the concept of a single step to DEMO from JET and JT-60U.
- Some design solutions and technologies developed for copper magnet ignition experiments and D-D steady-state experiments may not extrapolate to an integrated facility.
- The Modular Pathway has "two steps" to a DEMO, rather than the "one step" to a DEMO in the ITER-RC Pathway. Thus, the cost and time of getting to the DEMO stage in the Modular Pathway will both be larger than the corresponding numbers for the ITER-RC Pathway, albeit the design basis for the DEMO will be superior in the Modular Pathway with separate physics and technology facilities followed by an Advanced Integrating Facility.

3.6 Near Term Actions for the Modular Strategy

1. Determine the "must have" physics requirements for the burning plasma physics initiative.
2. Carry out a detailed assessment/review of the physics, engineering issues, and costs for Ignitor-like compact tokamak burning plasma devices. Engage/inform the U. S. fusion community during this assessment.
3. Hold an workshop with interested international participants on Burning Plasma Physics and Engineering Issues of compact tokamaks. There needs to be an assessment of Ignitor and similar compact tokamaks to evaluate what fraction of the goals of a burning experiment they will meet and how Ignitor-like compact ignition tokamaks fit into the longer range magnetic fusion program. This assessment should include a comparison to other burning plasma experiments such as BPX, the most similar design but a larger device with a divertor.
4. Determine the physics requirements for the steady-state advanced confinement initiative.

5. Carry out a detailed technical assessment/review of the physics and engineering issues for the steady-state advanced confinement element including an assessment of the relative capabilities of existing facilities, facilities now under construction and possible new facilities. Engage/inform the U. S. fusion community during this assessment.

6. Define the role and technical requirements for engineering testing of nuclear components in each of the three pathways.

These activities should be carried out with the goal of developing a plan within 6 months.