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PART 1

BURNING PLASMA EXPERIMENT SPECIAL

TECHNICAL PAPERS

DEPARTMENTS

ON THIS COVER This month's cover shows a cross section of the Burning Plasma Experiment.

BURNING PLASMA EXPERIMENT PHYSICS DESIGN DESCRIPTION

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The Burning Plasma Experiment (BPX) [formerly called the Compact Ignition Tokamak (CIT)] is designed to determine the physics behavior of self-heated fusion plasmas and demonstrate the production of substantial amounts of fusion power. In order to maximize performance at a given cost, the BPX tokamak reactor is designed to operate at a high toroidal magnetic field (9.0 T) and plasma current (11.8 MA) within a device of relatively compact dimensions (major radius 2.59 m, minor radius 0.80 m, elongation 2.0, and triangularity sweeping between 0.25 and 0.45). This design is expected to product between 100 and 500 MW of fusion power (alpha particles plus neutrons from a deuterium-tritium fuel) and a ratio of fusion power to heating power (Q) between 5 and ignition. This report presents a description of the design, followed by predictions of the performance, alpha-particle physics issues, magnetohydrodynamic stability, ion cyclotron and electron cyclotron resonance frequency heating systems, pellet fueling, boundary physics, impurity and particle control, diagnostic systems, and the operational plan.

I. BURNING PLASMA EXPERIMENT PHYSICS DESIGN DESCRIPTION

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I.A. THE BPX MISSION AND ITS RELATIONSHIP TO THE WORLD FUSION ENERGY PROGRAM

The mission of the Burning Plasma Experiment (BPX) is to

> "determine the physics behavior of self-heated fusion plasmas and demonstrate the production of substantial amounts of fusion power."

Supporting objectives are to

1. demonstrate the production of fusion power in excess of 100 MW, at fusion-reactor-level power density
2. determine the confinement physics, operational limits, and alpha-particle dynamics of self-heated fusion plasmas with alpha power greater than auxiliary heating power
3. demonstrate heating, fueling, and plasma handling techniques necessary to produce reactor-level power density, self-heated fusion plasmas
4. optimize plasma performance in the range of $Q = 5$ to ignition, with fusion power up to 500 MW. (Here, Q is defined as the ratio of fusion power to externally supplied plasma heating power, including ohmic heating power.)

The missions of the existing Tokamak Test Fusion Reactor (TFTR) and Joint European Torus (JET) experiments can be summarized as demonstrating the "scientific feasibility" of fusion power. Planned deuterium-tritium experiments on these devices will demonstrate that there are no fundamental scientific impediments to the production of controlled thermonuclear fusion power, using magnetic confinement. The next step after these demonstrations will be to determine the optimum configuration and design of a magnetic confinement fusion power plant in order to be able to evaluate realistically the economic, safety, and environmental acceptability of fusion power. There are issues of both plasma physics and fusion engineering involved in this optimization process. The BPX is focused primarily on providing early information on the key short-pulse physics issues of this step through the objectives to "determine the confinement physics, operational limits, and alpha-particle dynamics of self-heated fusion plasmas with alpha power greater than auxiliary-heating power" and to "optimize plasma performance in the range of $Q = 5$ to ignition, with fusion power up to 500 MW." However, the objectives to "demonstrate the production of fusion power in excess of 100 MW, at fusion-reactor-level power density [and] demonstrate heating, fueling, and plasma handling techniques necessary to produce reactor-level power density, self-heated fusion plasmas" constitute substantial engineering challenges as well, which also provide valuable input to the devices beyond BPX.

The mission of the International Tokamak Experimental Reactor (ITER) is to "demonstrate the scientific and technological feasibility of fusion power [and in so doing] provide the data base in physics and technology necessary for the design and construction of a demonstration fusion power plant."[1] The relationship of the BPX and ITER devices was carefully considered by the Fusion Policy Advisory Committee (FPAC) and documented in their 1990 final report to the Secretary of Energy[2]:

> "The Committee believes that both CIT and ITER are essential to proceed into the 1990's with the confidence to meet our stated goal of an operating Demonstration Power Plant by 2025. These facilities are complementary. The goal of ITER is to produce about 1000 MW of fusion power at high gain ($Q > 5$) in long pulses, ultimately steady state. Once operational, ITER would be used for engineering testing including nuclear power components.
>
> The CIT would produce 100 MW or more of fusion power, also at high gain ($Q > 5$), but in pulses of about 10 seconds in duration. The smaller CIT could operate several years before ITER and provide valuable input on burning plasmas prior to ITER operation. This follows the successful development strategy of the past in which the smaller Princeton Large Torus developed neutral beam heating for the TFTR, then in construction; ASDEX revealed the H-mode of operation that ultimately doubled the performance of JET; and small U.S. facilities developed the lower-hybrid current drive now employed successfully on the Japanese JT-60. Sim-

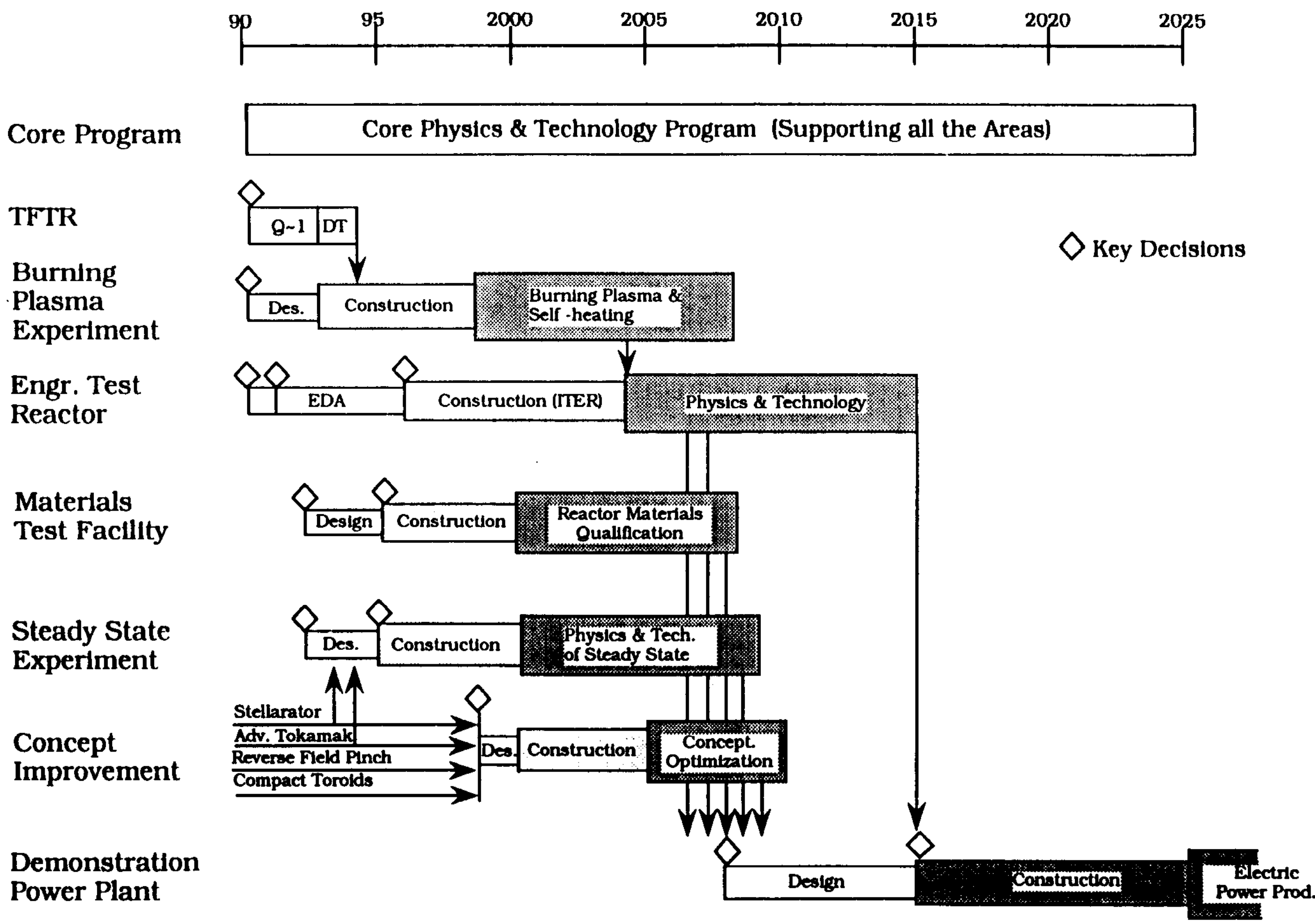

Fig. 1.1. Magnetic fusion energy development plan from FPAC final report.

> ilarly, CIT could provide advanced information that would avoid a prolonged and costly startup learning period in ITER.
>
> In this report the terms Burning Plasma Experiment and Compact Ignition Tokamak (CIT) are used interchangeably..."

BPX is anticipated to provide important information for the final design and construction of ITER internal hardware and external subsystems, remote maintenance, and diagnostic systems. Results from BPX will be especially valuable to help optimize the initial physics phase of the ITER device, in order to permit entry into the engineering phase in as timely a manner as possible. In the area of plasma engineering, measurements of the effects of high-current disruptions on internal components and measurements of the performance of high-heat-flux divertor elements should be especially valuable. Experience with remote maintenance on BPX will be highly informative as well. In the area of plasma physics, key issues are global confinement scaling, operational limits (especially plasma density), and the stability of the alpha-particle population in the high-temperature regime planned for current drive in ITER. The ITER design is required to be "sufficiently flexible to provide access for the introduction of advanced features and new capabilities, and to allow for optimizing plasma performance during operation."[1]

Figure 1.1 shows the time-line relationships between BPX and the other elements of the Magnetic Fusion Energy Development Plan recommended by the FPAC. The FPAC recognized, however, the uncertainty of these schedules:

> "The ITER project has identified a list of physics and engineering R&D issues that need to be resolved in order to support a decision for construction in 1996; their resolution will influence the ultimate ITER design and schedule. Irrespective of the evolution of the ITER program, a CIT

Burning Plasma Experiment under construction at that time puts the U.S. and the world fusion effort in a strong position to demonstrate significant production of fusion power, and to answer many key scientific questions about burning plasma physics, at about the turn of the century. If ITER construction does go forward on its currently projected schedule, research results from CIT will greatly reduce the risk that ITER could run into difficulties which would compromise its ETR mission."[2]

It should also be noted, as indicated in Fig. 1.1, that both BPX and ITER contribute to the design of the Demonstration Power Reactor (DEMO). Thus, complementarity between the BPX and ITER designs is extremely valuable. The BPX design focuses on high magnetic field as the approach to improved performance, while the ITER design emphasizes large size. BPX uses ion cyclotron range of frequency heating (ICRF) and can test fast-wave current drive, while the dominant heating and current-drive method in ITER is presently negative ion neutral beams. The data base provided by these two devices will thus permit an increased range of options for the DEMO design.

I.B. OVERVIEW OF THE PHYSICS DESIGN DESCRIPTION

The BPX Physics Design Description is structured approximately in parallel with the BPX physics organization (Fig. 1.2). The BPX physics group includes front-line working scientists, both experimentalists and theorists, from institutions all around the United States. Collectively, they bring to the project the institutional knowledge accumulated in the United States over the last 20 years of tokamak research, as well as ongoing involvement with present experiments and intimate knowledge of results from tokamak experiments all over the world. In most cases, they also have had extensive personal involvement with research abroad. The BPX project is overseen by the BPX Steering Committee (BPSC), composed of leading scientists from major U.S. fusion research centers, universities, and industry. Key physics design decisions are discussed in detail with them. The wisdom and experience of these two groups of distinguished scientists provides confidence that the BPX physics design is grounded in solid experimental fact and theoretical analysis.

The remainder of this document addresses the detailed description of the BPX physics design:

Chapter II presents an overview of the device design and describes how it satisfies the requirements laid out in the General Requirements Document (GRD). It describes the specific physics requirements for the three reference operating modes of BPX (double-null divertor, single-null divertor, and inner-wall limiter), as well as the requirements for a number of alternative modes developed to ensure adequate operational flexibility. This chapter provides the context for understanding the following chapters and also the linkages between the GRD and the detailed analyses that follow.

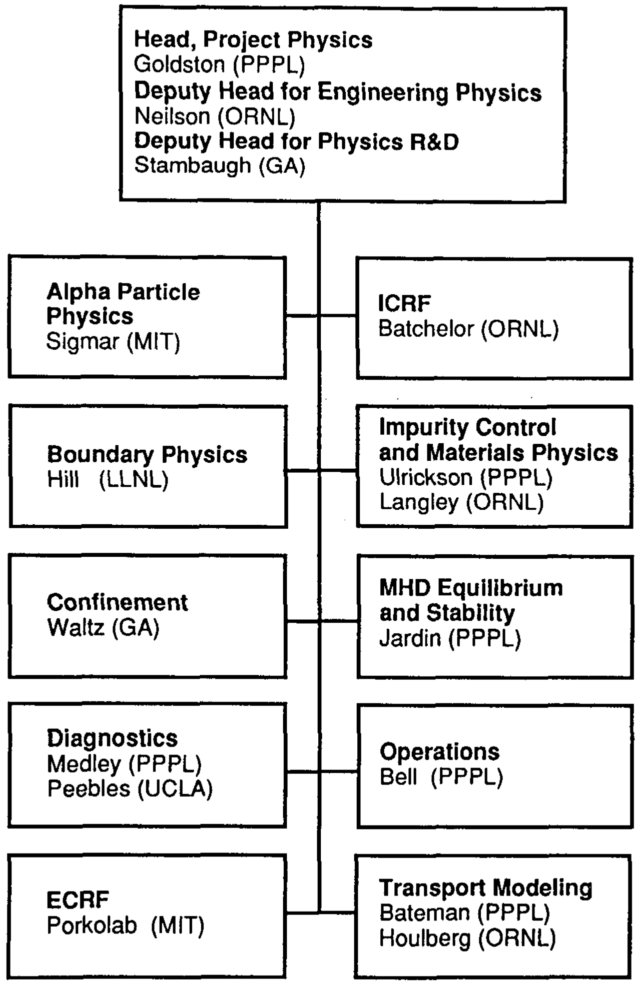

Fig. 1.2. BPX physics organization.

Chapter III describes the performance projections for BPX. Projections have been made on the basis of (a) dimensional extrapolation, (b) theory-based modeling calibrated against experiment, and (c) statistical scaling from the available empirical data base. While the results of all three approaches roughly coincide, we currently view the third as the most reliable. Based on study of the recently developed ITER H-mode data base, we take as our projected confinement performance τ_E = 1.85 times the ITER89-P L-mode scaling relation, with an estimated uncertainty of ±25%. The "center-of-the-error-bars" projected performance gives very high Q's (~ 25), close to pure ignition, and in the range anticipated for a tokamak reactor. The estimated uncertainty in confinement coupled with uncertainties in Z_{eff} and $n_e(r)$ are such that we can have high confidence in achieving at least $Q > 5$ in the BPX experiment. To have similarly high confidence in ignition, however, would require a much larger, and consequently much more expensive, tokamak device. Thus, in the judgment of the BPX Project and the BPSC, the present device is appropriately sized to meet the mission and supporting objectives described in Sec. I.A. It is important to take into account that uncertainties in projection are balanced by the likelihood that new understanding and operational techniques developed over the next 10 years will contribute to the ability to optimize plasma performance in BPX.

Chapter IV presents the alpha-particle physics issues that can be addressed in BPX and their relationship to the performance requirements. $Q > 5$ is required in order to be able to determine alpha-particle heating efficiency η_α with sufficient accuracy. The projected performance of $Q \sim 25$ and $P_{fus} \sim 500$ MW provides a wide range of physics parameters over which η_α can be measured. Theoretical calculations, for example, indicate that BPX can operate in regions both stable and unstable against the alpha-particle-driven toroidal Alfvén eigenmode (TAE). Slightly better confinement scaling or a heating power upgrade would be required, however, to provide studies of η_α in plasmas close to the beta limit at reduced magnetic field.

Chapter V addresses issues associated with plasma control and magnetohydrodynamic (MHD) stability in BPX. The values of q, κ, δ, and β planned for BPX are well within the ranges already explored experimentally in present devices and are projected to yield stable plasmas. (The planned density range is compared favorably to projected density limits in Chap. III.) The highly elongated BPX plasma is stable to free-boundary $n = 0$ displacements (e.g., axisymmetric, but not necessarily rigid, vertical motion) on the ideal time scale and can be held in place by the vertical feedback system on the field penetration time scale of the vacuum vessel. Theoretical and experimental investigations of the ranges of l_i and $\beta_0/\langle\beta\rangle$ expected in BPX indicate stability against $n \neq 0$ modes as well. By beginning plasma heating late in the current rise, the $q = 1$ radius can be kept rather small ($\sim$15% of the minor radius), and so the effects of sawteeth can be minimized. On the other hand, by waiting 2 seconds into the current flattop before beginning heating, plasmas with fully developed sawteeth ($r_{q=1}/a \sim 1/q_{95}$) can be studied.

Detailed simulations of standard BPX discharges have been performed to predict the evolution of the poloidal field (PF) coil currents and divertor heat loads for engineering design and analysis purposes. The plasma startup, growth, and shutdown phases have been optimized on the basis of MHD stability considerations. The PF system provides the required sweep of the separatrix field lines across the divertor surface and also the transformer flux swing needed to ensure full-current performance even under pessimistic plasma conditions. Position and shape control capabilities provide the operational flexibility needed to adjust the plasma's interface with the ICRF antenna (to optimize coupling) and with the divertor (to optimize power handling).

Detailed self-consistent calculations of a wide range of possible plasma disruptions, including the effects of so-called "halo" currents (which flow between open field lines in the torus and the vacuum vessel wall) are used in the design of the BPX vacuum vessel and first-wall components. Further studies are in progress to be certain that all "worst-case" conditions have been enveloped by the cases analyzed. The toroidal field ripple in BPX is found to cause negligible alpha-particle losses, and field errors are specified to result in minimal cross-field transport and asymmetries in heat loading.

Chapter VI presents the physics basis for the design of the ICRF system on BPX. Results from JET have been very encouraging in that excellent H-mode performance has been achieved with ICRF alone, including both confinement and impurity levels comparable to those achieved with neutral beam injection. Results from TFTR are encouraging as well, showing efficient heating at plasma densities approaching those planned for BPX. The single-pass absorption in the base mode of heating for BPX, D-T majority/^{3}He minority, is adequate even with very low levels of ^{3}He (<1%). The ICRF system on BPX should provide flexibility to study a number of other heating modes, such as hydrogen minority heating and fast-wave current drive. In general, a modest radio-frequency (RF) tail is created with ICRF on BPX, so that ion heating is dominant. In lower density regimes, however, it may be possible to study sawtooth stabilization using the RF-driven tail.

A key ICRF antenna design task has been determination of the minimum coupling resistance expected in BPX, in order to ensure the delivery of the full ICRF heating power to the plasma under all circumstances. Numerical studies validated against H-mode experimental results from DIII-D and L-mode results from TFTR indicate that a zero-width scrape-off layer and maximum plasma density provide the lowest coupling resistance, and therefore the highest voltages in the

RF system for a given absorbed RF power. The present design delivers the specified 20 MW of ICRF power absorbed into a worst-case plasma with zero scrape-off width, operating at the Greenwald density limit. The operating voltage is 32.5 kV, and the surface power density is 8.5 MW/m^2. While these numbers are acceptable, further optimization of the antenna design is anticipated.

Chapter VII describes the physics basis for the electron cyclotron heating (ECH) upgrade option on BPX. Single-frequency ECH would be adequate to provide heating during the current and field ramp to 9 T, and off-perpendicular injection could provide coupling to the plasma center during this process, if desired. Central heating at lower steady-state fields can be accomplished by off-perpendicular injection and/or through the use of step-tunable sources. Studies of the localizability of ECH power in BPX indicate that even in the presence of substantial density fluctuations at the plasma edge, the ECH beam should be tightly enough collimated to provide the possibility of controling MHD modes through local heating.

Chapter VIII describes the physics basis for the pellet fueling requirements for BPX. Present data and analyses indicate that 4 to 5 km/s pellets should provide substantial penetration into even high-temperature BPX plasmas. Simulation studies indicate that useful peaking of the density profile can be achieved in cases with fully developed sawteeth. Current ramp scenarios have also been developed in which the sawtooth is fully suppressed or the $q = 1$ radius is kept small. In these cases, one can anticipate large density peaking factors.

Chapter IX presents an overview of boundary physics and power handling in BPX. Power handling is an area of uncertainty comparable to confinement in its effect on projecting the performance of future tokamaks. The minimum requirement of handling 100 MW of fusion power at $Q > 5$ can be projected with high confidence in BPX, based on the swept-divertor concept. However, the maximum requirement of handling 500 MW of fusion power is at the projected capability of the BPX divertor system, with no additional margin provided for physics uncertainties. Present models of divertor physics, while mathematically sophisticated, are not thoroughly calibrated against a wide range of experimental data and do not include even all of the known physics, so the uncertainties are significant. On the other hand, some of the uncertainties are likely to play in favor of enhanced divertor performance (e.g., impurity radiation in the scrape-off layer), and development of new operational techniques over the next 10 years are very likely to contribute in this area. The PF system is capable of performing multiple sweeps (with a power-supply voltage upgrade) if required to compensate for a narrower-than-expected scrape-off width. Nonetheless, the divertor is an area where increased performance margin through physics R&D is expected and would be welcomed.

Chapter X presents the physics analyses associated with impurity and particle control. Calculations of the erosion indicate that the net erosion will be small and that impurity influx and redeposition of divertor material can be minimized. The pre-shot wall temperature of 350°C is also an important impurity control measure. Calculations of particle pumping and tritium retention indicate that even worst-case assumptions are consistent with an in-vessel inventory of 2 g of releasable tritium, as specified in the GRD. Helium and He/O glow discharge cleaning are provided to deplete and remove codeposited layers of carbon and D-T. The estimated effects of disruptions on plasma-facing components are acceptable, although further analysis and R&D in this area are still required.

Chapter XI presents the diagnostic system for BPX. In general, providing adequate diagnosis of high-Q, D-T plasmas is a major physics and engineering challenge. There are stringent requirements in the area of machine operations, such as accurate position measurement and the ability to view the divertor tiles in the infrared during a discharge. For physics-oriented measurements, the design philosophy is to provide diagnostics for all of the quantities that are normally measured on present devices, in order to understand BPX performance and to determine accurately any significant changes in scaling to be learned from the extrapolation to BPX parameters. In addition, a full set of alpha-particle physics diagnostics is required, and flexibility must be maintained to allow for the development of new diagnostic techniques that are made possible either by technological developments or by the new plasma parameters of BPX.

Chapter XII presents the operational plan for BPX. D-T operation is achieved about 2 years after first plasma, but all diagnostic and heating systems are fully exercised, and the remote maintenance equipment is fully checked out in realistic applications, before tritium is introduced into the machine. The equivalent of only 400 full-field pulses is used, out of the fatigue lifetime of 3000 such pulses, when full-field, high-Q, D-T operation begins about 3 years after first plasma. The BPX operational plan is designed to achieve the mission of BPX within 3000 full-field pulses. It should also be recognized that after this experimental program is completed, margin against fatigue lifetime is reduced, but machine operations can continue. In general, the conclusion of the BPX Physics Design

Description is that based on our best present understanding of tokamak physics, the BPX device has adequate performance in all areas to meet its mission and to thereby make a major contribution to the development of magnetic confinement fusion energy.

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