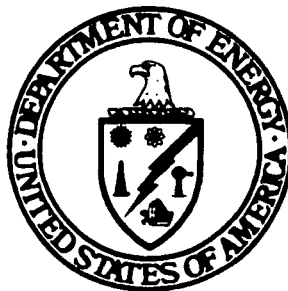# FUSION ENERGY ADVISORY COMMITTEE

*Advice and Recommendations
to the Department of Energy*

*In Partial Response to the Charge Letter
of September 24, 1991: Part E*

November 1992

U.S. Department of Energy
Office of Energy Research
Washington, DC   20585

FUSION ENERGY ADVISORY COMMITTEE
Advice And Recommendations To
The U.S. Department Of Energy.

In Partial Response To The Charge Letter
Of September 24, 1991: Part E

Members of FEAC

Robert W. Conn, Chairman
David E. Baldwin
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Marshall N. Rosenbluth
John Sheffield
Peter Staudhammer
Harold Weitzner

FEAC/92-IV

November 6, 1992

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Preface

This document is a compilation of the written records that relate to the Fusion Energy Advisory Committee's deliberations with regard to the Letters of Charge received from the Director of Energy Research, dated September 24, 1991, February 20, 1992 and June 22, 1992.

During its fifth meeting, held in September 1992, FEAC provided a detailed response to the charge contained in the letter of June 22, 1992. In particular, it responded to the sentence:

> "I am asking for your best technical judgement on how to structure the magnetic fusion program within these different funding assumptions, but without change in the basic goal of demonstrating fusion power and within the basic assumption of strong international collaboration."

The response was prepared in the form of a report entitled "Fusion Energy Advisory Committee Report on Program Strategy for U.S. Magnetic Fusion Energy Research", dated September 23, 1992. It has been published by the U.S. Department of Energy as a separate document.

To assist with their response to the charge, FEAC established a working group, designated Panel #4, which reviewed priorities in the U.S. intermediate confinement experiments in detail. This panel prepared background material which was provided to FEAC during its September 1992 meeting to help with its deliberations. The report of Panel #4 is included in this report as Appendix I.

CHARGE TO FUSION ENERGY ADVISORY COMMITTEE

Introduction

A year ago, the Fusion Policy Advisory Committee (FPAC) reported its findings and recommendations on fusion energy programs of the Department of Energy (DOE). The Secretary of Energy adopted FPAC's recommendations subject to existing budget constraints. This translated to terminating work on alternative confinement concepts and pursuing only the tokamak concept within the magnetic fusion energy program, as a precursor to a Burning Plasma Experiment (BPX) that would be integrated into a larger international fusion energy program. Fusion energy was highlighted in the National Energy Strategy, which mentioned both the International Thermonuclear Experimental Reactor (ITER) and BPX as major elements of the program. The Secretary travelled to Europe earlier this year to conduct personal discussions with the Italian government on their potential interest in a bilateral agreement on BPX.

Since that time, a number of events have led to a reexamination of the strategy being used to pursue an energy-oriented fusion program. The estimated cost of BPX has increased and foreign interest in substantial participation has not materialized. Last week, the Secretary of Energy Advisory Board Task Force on Energy Research Priorities was asked to review the relative priority of the BPX proposal among the programs of the Office of Energy Research and to recommend on the appropriate tasking to the Fusion Energy Advisory Committee (FEAC). The Task Force recommended that the DOE not proceed with BPX, but rather focus on ITER as the key next step after the Tokamak Fusion Test Reactor (TFTR) and the Joint European Torus in developing the physics of burning plasmas, along the lines currently being proposed by the European Community. The Task Force also recommended that the U.S. fusion energy program continue to grow modestly (even in an ER budget that is declining in constant dollars) and suggested that a more diverse program that included a less costly follow-on device to TFTR in the U.S. would be more effective in the long run.

Charge

I would like to explore seriously the programmatic implications of this recommendation under two budget scenarios -- a constant dollar budget for magnetic fusion through FY 1996 and a budget at 5 percent real growth per year through FY 1996. I am therefore charging the FEAC to advise me on the following questions.

1. Identify how available funds now used for BPX, as well as a modest increase (described above) could be used to strengthen the existing base program for magnetic fusion research.

2. Within the above envelope of funding, identify what follow-on experimental devices for the U.S. fusion program might be planned for use after the completion of experiments at TFTR and before the planned start of ITER operation. For such devices, indicate how they would fit into the international fusion program.
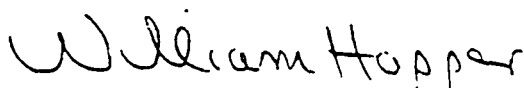
3.   What should be the U.S. position on the appropriate scope, timing, and
      mission of ITER if BPX does not go forward?

Although you will need some months to complete the work envisioned in this
charge, I would like to have your initial thoughts on the above three topics
in a letter report from your meeting of September 24-25, 1991.

Then, by January 1992, I would like to have your recommendations on the
appropriate scope and mission of ITER and any suggestions you can make to
lower its cost or accelerate its schedule. At the same time, I would like
your recommendations on the relative importance to the U.S. of the various
ITER technology tasks, on the role and level of U.S. industrial involvement in
the ITER engineering design activity, and on the balance between ITER project-
specific R&D and the base program.

By March 1992, I would like your views on how to fill the gap in the U.S.
magnetic fusion program between the completion of TFTR work and the planned
start of ITER operation. In addressing this issue, please include
consideration of international collaboration, both here and abroad.

By May 1992, I would like to have your recommendations on a U.S. concept
improvement program, including relative priorities and taking into account
ongoing and planned work abroad.

William Happer
Director
Office of Energy Research

Department of Energy
Washington. DC 20585

February 20, 1992

Dr. Robert W. Conn
Chairman, Fusion Energy
  Advisory Committee
University of California, Los Angeles
6291 Boelter Hall
Mechanical, Aerospace, and Nuclear
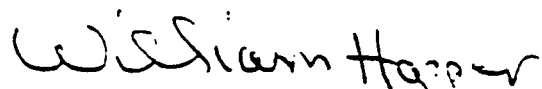 Engineering Department
Los Angeles, CA 90024-1597

Dear Dr. Conn:

I am writing to expand on the portion of the charge you received September 24, 1991, regarding concept improvement. Specifically, that charge asked "By May 1992, I would like to have your recommendations on a U.S. concept improvement program, including relative priorities and taking into account ongoing and planned work abroad." I understand that you discussed this charge element at your meeting on February 6 in California, forming a panel (#3) to develop information and requesting some points of clarification from DOE. I further understand that possible major program elements which address tokamak improvement, such as TPX and the ATF/PBX-M facilities, are already well along in your review process through Panel 2.

Given that tokamak reactor development will be the primary focus of the U.S. magnetic fusion program, it is reasonable to ask what activities are appropriate on non-tokamak concepts and on small-scale exploration of tokamak improvements. There are a number of ideas on alternate concepts and tokamak improvements, and the exploration of these ideas has historically added richness and innovation to magnetic-fusion development. It would be useful if you could recommend a policy and selection criteria to help guide our program choices on concept improvements within our goal-oriented program strategy. The overall policy question is whether, given the demands of the mainline tokamak program and current budget constraints, we should encourage and fund proposals on concepts other than tokamaks.

Within the concept improvements area, what priorities should be given to exploratory tokamak improvement proposals, like the compact toroid fueling and helicity current drive that are now under small scale investigation? Should the priority be higher for U.S. alternate concept activities that connect to major significant international programs or for unique U.S. activities? Under what conditions and within what criteria should concepts that have little connection to tokamaks, or to other major international programs, be considered?

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I know that these issues are of intense interest to some members of the U.S. fusion community. It is important to have your best judgment on these questions within the context of overall magnetic fusion program goals, strategies, and funding constraints.

Sincerely,

William Happer
Director
Office of Energy Research
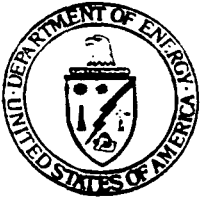
Department of Energy
Washington, DC 20585

June 22, 1992

Dr. Robert W. Conn
Chairman, Fusion Energy
 Advisory Committee
University of California, Los Angeles
Los Angeles, CA 90024-1597

Dear Bob:

The Fusion Energy Advisory Committee (FEAC) has now reviewed and reported on the primary elements of the magnetic fusion program. Given that background, it would be quite helpful if FEAC would provide recommendations on strategic program planning. Please provide your views for three different out-year funding assumptions: starting with the FY 1993 House Appropriation Mark of $331M for magnetic fusion, (A) 5 percent real growth; (B) level funding, i.e., with only inflation; (C) flat, without inflation. Of course, the FY 1993 budget process is still incomplete, and I will revise this guidance if we have better figures before you meet.

Within these assumed cases, which program elements should be enhanced, protected, reduced, or eliminated and on what schedule? In all cases the primary goal should be maximum progress toward a Demonstration Power Plant. I am asking for your best technical judgment on how to structure the magnetic fusion program within these different funding assumptions, but without change in the basic goal of demonstrating fusion power and within the basic assumption of strong international collaboration.

Please provide your recommendations by the end of September 1992. I know that all FEAC members have worked intensely to develop your recommendations on the individual program elements in my first set of charges. Therefore, I believe it is most useful to take this overview now while the contextual information is fresh. I realize that this will require additional dedication on top of your already extensive labors. I do appreciate your efforts.

Sincerely,

William Happer
Director
Office of Energy Research

UNIVERSITY OF CALIFORNIA, LOS ANGELES

UCLA

BERKELEY • DAVIS • IRVINE • LOS ANGELES • RIVERSIDE • SAN DIEGO • SAN FRANCISCO

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Dr. David Baldwin, LLNL   June 8, 1992
Dr. Harold Weitzner, NYU

Dear David and Harold:

Thank you for agreeing to be chair and vice-chair of FEAC Panel 4 on "Priorities in the Intermediate Confinement Experiments." Your report will provide important input to the FEAC workshop in July on priorities in the overall fusion program. In addition, it will assist the FEAC in reaching its specific recommendation in September on the operation of ATF.

The facilities in the toroidal program that you are asked to evaluate and prioritize are the ATF stellarator and the PBX, and C-Mod tokamaks. This should be done against the background of the DIII-D and TFTR capabilities, assuming that full D-T operation in TFTR beginning in mid-1993 and a strong DIII-D program are supported as recommended in the April 1 FEAC letter to Dr. Happer. As described below, I ask you to focus more on a factual evaluation for our July meeting, leaving for September a more complete determination of a basis for FEAC recommendations on priorities.

For the July meeting, please provide the following information for each of the identified mid-scale toroidal facilities:

1. The physics issues that are addressable in this class of facility and the completeness with which each of the identified devices can address these issues: and

2. For each device, the goals and objectives, additional hardware, the strengths, uniqueness, limitations, present status, projected costs and time required to achieve its objectives.

In addition, for the July meeting, please provide preliminary priorities and their time scale that your Panel would assign to the operation of these facilities, along with an indication of the reasoning behind these priorities.

At the July meeting, the full FEAC will make use of your evaluations and your draft priorities in its examination of the broader program. Later, in time for the September meeting, I would like your panel to reexamine its preliminary priorities in light of the FEAC's July workshop and feedback provided there. Further, this will provide an opportunity for your Panel to hear responses from the programs reviewed. Your revised priorities will then serve as input to the September meeting of FEAC. This two-step process will provide ample opportunity for each program to have a fair opportunity to answer questions and concerns.

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When this process has been completed, the FEAC must answer the following questions:

1. If the fusion budget is sufficient to do so, do all of the facilities warrant operation? If not, which ones do not warrant operation?

2. If the fusion budget is not sufficient to operate simultaneously all the facilities which warrant operation,

   a) Should their operation be phased, implying one or more machines would be mothballed, and if so how?
   b) Should all be operated at a reduced level? or
   c) Should one or more be closed down, and if so in what priority order?

The combination of your evaluations and priorities should be sufficient to permit FEAC to respond to Dr. Happer's request concerning the ATF and other priorities. I understand that this will not be an easy undertaking for your Panel, for FEAC, or for the programs involved since all are staffed by high quality groups. I will do all that I can to assist you in this endeavor.

Sincerely,

Robert W. Conn

ROBERT W. CONN
DIRECTOR AND PROFESSOR

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November 6, 1992

Dr. William Happer
Director
Office of Energy Research
U.S. Department of Energy
Washington, D.C. 20585

Dear Dr. Happer,

Recently, I forwarded to you the report of the Fusion Energy Advisory Committee relating to strategic planning for future activities in magnetic fusion energy research. As part of the process that led up to that report, FEAC established a panel that reviewed priorities in the intermediate confinement experiments and that provided background to FEAC during its deliberations. The background material was presented to FEAC both verbally and in the form of a written report. I am forwarding with this letter a copy of that panel report.

Sincerely,

Robert W. Conn
Chairman
on behalf of the
Fusion Energy Advisory Committee

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Appendix I

The report to FEAC of Panel #4,
dated September 22, 1992.

Report from Panel 4 of the

Fusion Energy Advisory Committee

on

Priorities in the Intermediate Confinement Experiments

David Baldwin (Chair)*
Lawrence Livermore National Laboratory

Harold Weitzner (Vice Chair)*
New York University

Steve Dean*
Fusion Power Associates

Richard D. Hazeltine
University of Texas, Austin

Neville C. Luhmann
University of California, Los Angeles

Stewart Prager
University of Wisconsin

Barrett H. Ripin*
Naval Research Laboratory

Marshall N. Rosenbluth*
University of California, San Diego

Richard E. Siemon*
Los Alamos National Laboratory

Alan Wootton
University of Texas, Austin

September 22, 1992

* Member of FEAC

INTRODUCTION AND BACKGROUND

In a letter dated June 22, 1992, Dr. Will Happer, Director, Office of Energy Research, requested that the FEAC address the priorities within the MFE program under several budget scenarios. As part of the preparation for answering this charge, Panel 4 was created by the FEAC chairman, Professor Robert Conn, whose charge letter to the Panel is contained in the Appendix. He asked the Panel to assemble background information on three mid-sized toroidal facilities, the Alcator C-Mod and PBX-M tokamaks at the Massachusetts Institute Technology and the Princeton Plasma Physics Laboratory, respectively, and the ATF stellarator at the Oak Ridge National Laboratory and to provide draft conclusions to the FEAC on the question of priorities. The Panel was later asked to provide similar background information on the proposed DIII-D Upgrade, in view of the fact that the Upgrade objectives encompassed many of those of the three smaller facilities.

The following four sections contain summaries of the four machines' capabilities and their places in the world programs. These summaries were prepared with the cooperation of the respective research groups. Draft positions of the Panel on priorities were communicated directly to FEAC as part of its deliberations, the conclusions of which will be available at the September 1992 FEAC meeting. Because the priority recommendations of the FEAC supersede those of Panel 4, the issue of priorities have not been revisited by the Panel, as had originally been planned.

I.
ALCATOR C-MOD

A.   Program Plan

The purpose of the Alcator C-Mod program is to address a range of critical issues confronting the development of the tokamak as a viable fusion reactor concept. These issues include power and particle handling, control, enhanced transport, and RF heating and current drive. The high magnetic field (9T) and strong shaping ($\kappa \approx 1.8$) of Alcator C-Mod result in plasma currents up to 3 MA, projecting to plasma performance comparable to the best so far achieved in any tokamak. The state of the art plasma diagnostic complement and ample port access combine with the unique characteristics associated with the high particle-, power, and current- densities of this relatively small size device to position Alcator C-Mod as a premier research facility in the world tokamak program.

Alcator C-Mod has a unique capability to address the problem of power handling in an ITER- (and reactor-) relevant divertor geometry. The surface power density (total power divided by plasma surface) in C-Mod is in the range of 0.5-1 MW/m^2, which exceeds the level required in ITER and is typical of a reactor. A major objective of the Alcator research program is to demonstrate a solution to the problem of divertor power handling at reactor-relevant power densities, in a manner consistent with clean, high performance core plasmas. Our initial approach focuses on reactor-relevant metallic plasma facing components, an inclined-plate, semi-closed divertor geometry, and the high-recycling, radiative modes of divertor operation.

Alcator C-Mod also offers unique opportunities for addressing problems of axisymmetric stability, disruption avoidance and control, as well as for characterizing disruption effects. The low vacuum vessel resistance and the conducting super-structure are particularly relevant to ITER, as is the magnetic configuration. Extensive diagnostics are installed to monitor heat deposition, halo currents (including toroidal variation), and power balance. The hybrid analog/digital control system can be used both for disruption avoidance and to implement optimized ramp-down techniques to minimize electromagnetic loads.

The physics and scaling of confinement remain vital topics determining the feasibility and attractiveness of a fusion reactor. Two general issues must be addressed: transport prediction and confinement improvement. The Alcator C-Mod program aims to advance

this area of fusion research both through fundamental studies of transport and by investigation and development of enhanced confinement *modes*. The unique parameter range accessible on Alcator C-Mod (high field, high density, small size) can be used to test dimensionless similarity scaling by comparison with larger, low-field devices, such as DIII-D and ASDEX-U, which operate with the same non-dimensional parameters. C-Mod will extend the study of enhanced confinement to unique densities and magnetic fields, of direct applicability to future devices of the IGNITOR class or reactors in the ARIES line. It will also establish the scaling of confinement and access to enhanced confinement regimes. Of particular interest is the possibility of enhancing confinement at reduced current (high q*, high β leading to an attractive reactor scenario with substantial bootstrap current and modest RF current drive requirements, as exemplified by the ARIES I study.

In addition to providing bulk heating power for the divertor and confinement studies, the ICRF program on Alcator C-Mod aims to optimize antenna performance at high RF power densities (≥ 10 MW/m^2) and to determine the heating effectiveness of various ICRF heating scenarios. These studies should lead to optimized antenna design for ITER and reactor type devices. ICRF will also provide heating power for long pulse, lower-hybrid current driven plasmas as part of the advanced tokamak studies program. The goal of these experiments will be to demonstrate noninductive current drive operation, including profile control, in combination with substantial bootstrap current, for time durations in excess of the L/R time, and to study the confinement in such discharges at reactor-like densities, magnetic fields, and q* values. These experiments will also address the compatibility of such non-inductively driven operation with high performance, low Z_{eff} core plasmas and high heat flux divertor operation.

The culmination of the C-Mod program would thus be a demonstration of the combination of essential features of an attractive tokamak reactor, namely high confinement, non-inductively sustained current, low impurity content, and reactor-relevant divertor power density.

B. Divertor and Edge Physics
B.1 Introduction
Alcator C-Mod is designed with a closed divertor (see figure I.2). This design is complemented by the very wide range in densities that can be achieved with a high magnetic field. The combination of these geometrical and plasma characteristics creates

an ideal environment in which to test the radiative and gaseous divertor concepts. Additional unique aspects of the Alcator C-Mod divertor are the high-Z divertor material, metal walls and a highly-inclined plate (poloidal angle ~ 15°, toroidal angle ~ 1°). It is an important contribution to divertor designs to test the compatibility of such features with ITER-level power- and particle-densities. Since the upper divertor has an open, flat-plate geometry, it will be possible to perform direct comparisons of closed and open divertor performance on a single tokamak. C-Mod has an extensive array of edge and divertor diagnostics which will be used to characterize the edge transport of plasma and impurities, and their divertor plate interactions.

A unique contribution of the Alcator C-Mod divertor program lies in our ability to provide results with plasma parameters, divertor geometry and divertor material which are significantly different from those being investigated elsewhere in the world program. Essentially all divertor tokamaks, other than C-Mod, operate with essentially similar plasma parameters (n_e, T_e, B). To make matters worse, the bulk of data from these experiments are at electron densities, power densities and magnetic fields which are significantly lower than those envisaged for ITER or ARIES. The consequence is that present modeling is based on a narrow set of data with geometries unlike those being considered for next step devices. With the data from Alcator C-Mod, it will be possible to make *interpolations* in a number of areas, rather than extrapolations, to ITER and ARIES.

B.2   Divertor Integrity

Control of the power deposited on the divertor plates, during normal operation as well as disruptions, is a central limiting factor in the present ITER design. For example, the predicted peak heat loads for the ITER CDA are 10-30 MW/m^2, levels which are beyond the steady-state heat removal capabilities of present technology in anything other than small laboratory experiments. However, included in the heat load specification is an uncertainty of a factor ~ 5 which is composed of an amalgam of uncertainties associated with such variables as scrapeoff lengths, inner-outer divertor asymmetrical loading (from plate to plate) and toroidal peaking factors. The uncertainties are due to the sparsity of data and the dissimilarity of the geometries of presently operating tokamaks compared to ITER. It is clear that no matter how the predictions of the divertor plate heat loads for ITER might be reduced, either through advanced divertor physics (e.g., radiation) or by reduction of uncertainties, any such decreases could have an enormous impact on the overall ITER design.

Alcator C-Mod will have maximum surface power density (total power divided by plasma surface area) in the range 0.5-1 MW/m^2, and divertor plate power densities in the range 5-30 MW/m^2. These values make the power handling problems prototypical of ITER and tokamak reactors.

An extensive array of edge diagnostics will provide information on the heat loads and the uncertainties associated with their prediction. These include 220 thermocouples measuring first-wall tile temperature distributions, a poloidal array of IR tile surface temperature measurements, a poloidal array of 48 Langmuir probes inset flush in the closed divertor surfaces, and a pneumatically-driven Langmuir probe in the SOL. Langmuir probes have also been placed in the outboard limiter and the ICRF antenna protection tiles. Comparisons of measured heat loads with those predicted from measured plasma parameters at the plate will provide important tests of the standard sheath transmission model at small field-line angles of incidence. Edge modeling is crucially dependent on the heat transmission factor boundary conditions and on a knowledge of particle and heat diffusivities in the SOL. The density and temperature e-folding lengths measured by the probes, and other diagnostics such as reflectometry and spectroscopy, will provide a measure of the diffusivities and insights into the underlying transport processes.

Disruptions also affect divertor integrity. In addition to the substantial heat loads generated during disruptions, another important disruption effect is the generation of *halo* currents flowing poloidally through the first-wall and vacuum vessel. These currents interact with the toroidal field, producing forces which have caused damage to the tiles and other in-vessel structures in some tokamaks. The details of these currents are not well understood. The outer closed divertor plates of Alcator C-Mod are designed so that they can be electrically isolated from the vacuum vessel; at present they are connected to the vessel through current shunts. The current flowing through each of the 10 toroidal divertor segments will be measured to determine the magnitude, time duration and toroidal variation of disruption-induced halo currents, and thus to infer the resultant forces. The divertor probe array will yield information about the poloidal profile of halo currents at the divertor surface, as well as about the local plasma evolution during disruptions.

B.3   Divertor Lifetime

The plasma-induced erosion rate of plate material is of great concern in determining first wall longevity. For ITER and ARIES, the erosion rate due to evaporation and sputtering is predicted to be large. However, the redeposition rate of the eroded material is predicted to compensate much of this, so that the net erosion rate is much smaller. The operational data base relevant to this question is small, particularly with respect to the effect of highly inclined divertor plates (poloidal angle ~ 15°) found in ITER CDA, ARIES and Alcator C-Mod. It is unknown whether the movement of material swept along a plate due to erosion/redeposition cycles will be an important factor or not.

Perhaps of greater import is the erosion which occurs during disruptions. Predictions for net erosion in ITER with carbon tiles imply that disruptions will lead to the need to replace the divertor anywhere from 1 to 17 times during the physics phase. The uncertainty of this prediction is due primarily to the unknown protection characteristics of the plasma vapor shield, and to the unknown toroidal and poloidal heat load distributions. There are similar uncertainties for tungsten as the divertor material.

The Alcator C-Mod staff is currently working with Sandia National Laboratory, Livermore (SNL-L) to characterize our molybdenum tile surfaces. After each period of operation, beta-backscattering will be used to monitor, in-situ, changes in the surface. Periodically, sample tiles at different poloidal locations will be removed for more in-depth analysis. In-situ measurement of the erosion rate (not net erosion) will be made utilizing 1 and 2-D spectroscopic imaging of the divertor surfaces at wavelengths corresponding to neutral molybdenum emission. The time integrals of these data can be compared with the net erosion/redeposition rate, and the redeposition inferred. The measured Mo source rates will also be compared to those calculated from knowledge of the hydrogen fluxes, sheath potential and sputtering rates, thus providing information on the effect of small field line incidence angles on sputtering rates. The above diagnostics can also be used to characterize disruption erosion and the resulting metal plasma vapor shield.

B.4   Particle Control

There are two stages at which the inflow of impurities can be controlled: (1) source reduction of impurities and (2) reduction of their transmission to the central plasma. The impurity source rate at the divertor plate can be reduced by lowering sputtering (lower T_e and T_i) or by reduction of evaporation (heat loads). The effect of the small field line

incidence angle on sputtering rates and ion sheath acceleration could be beneficial, but at present this is quite uncertain. The wall impurity source can likewise be reduced through reduction of the high-energy charge-exchange neutral flux from the central plasma (operation at high density) and through (better bonding of O_2 to wall; covering of metals). Once an impurity is generated, the second level of protection becomes important, namely impurity screening by reduction in neutral impurity transmission through the edge plasma and flow of the corresponding impurity ions into the divertor. The processes involved in SOL plasma flows are poorly understood, and there is little experimental data concerning flow patterns.

A number of periscope systems are being installed on C-Mod to provide spectroscopic views (1- and 2-D) of the walls, divertor plates and antennas. The source rates for impurities at these locations will be measured as functions of changing central and divertor plasma conditions. The impurity screening efficiency can be calculated through a model which uses measured divertor and SOL plasma parameters and known ionization and charge-exchange processes. This model will be compared to measurements of impurity densities in the central plasma (VUV and X-ray emission) and in the SOL (Omegatron mass-spectrum analyzer).

Flows in the SOL are a vital factor in determining the impurity screening efficiency. It appears that flow reversal, with resulting stagnation points, must exist, but this is not experimentally confirmed. It is important to understand these flows and, specifically, to compare different plasma geometries (open vs. closed divertor) under the same plasma conditions. JET and DIII-D have operated with open, flat-plate divertors that have been utilized for obtaining H-mode, not particle control, but the ITER CDA and ARIES divertor geometries are of a more closed nature.

The Alcator C-Mod divertor program emphasizes the diagnostic characterization of SOL flows. A Mach probe will be the initial flow diagnostic for the SOL. In addition, a multiple-point gas puffing system will be used to inject trace impurities at different poloidal locations and the flow of those impurities followed by their emission. Direct spectroscopic measurements of doppler shifts will also be made.

The plasma parameters in the divertor and the divertor geometry have important roles in impurity retention. It seems likely that the higher the density in the divertor (opaque to neutrals) and the more closed the divertor geometry (mechanical baffle), the better its

impurity retention properties. High divertor densities can also improve hydrogenic neutral exhaust (proportional to the divertor neutral pressure) with a concomitant improvement of density control in the central plasma. However, the neutral pressure in the divertor cannot be increased indefinitely; the neutrals can *blow-through* back to the main plasma and have effects on the density and confinement there.

The closed and open divertor geometries, combined with the high densities found in Alcator C-Mod, provide a unique ability to explore the allowable limits of neutral pressure (and pumping). Initial experiments will focus on the use of 3 gas gages to measure the divertor and general vessel pressures. The divertor neutral pressure can be varied utilizing some of the 28 capillary gas puffing tubes inset in the tiles. This will determine the limits of neutral pressure for a given set of divertor and plasma conditions. The effects on the divertor plasma characteristics will be monitored as well.

For helium exhaust, high pressure in the divertor is not enough. Methods have been proposed to pump helium preferentially, including semi-permeable membranes, material surfaces that pump helium preferentially (helium self-pumping) and optimized pump duct geometries. All of these methods need to be tested for high particle and heat flux conditions, with both high-Z and carbon first-wall tokamaks.

Alcator C-Mod and SNL-L are determining the applicability of using the helium self-pumping technique (developed by SNL-A/ANL and used on TEXTOR) to C-Mod. For Alcator C-Mod, nickel would again be a suitable material. However, other materials, including molybdenum, could be considered. This experiment would be significant in demonstrating the viability of the concept in a metal first-wall device.

B.5   Divertor Concept Improvement Studies

B.5.1   Radiative Divertor

The Alcator C-Mod divertor configuration is the most closed of any divertor now in use. This is due to the combined baffle-divertor plate structure at both the inner and outer divertor plates which minimizes the mechanical opening to the divertor. This unique geometry, combined with the high densities, and high power- and particle-fluxes, maximizes the entrainment of impurities and their resultant radiation, which in turn minimizes the heat load on the divertor plates. This capability is further enhanced by the existence of the 28 gas puffing tubes described in section B.3.3. Impurity gases can be puffed into the divertor at one or more poloidal locations to modify the radiation profile

in the divertor. A central goal of the Alcator program will be to explore the efficacy of this technique through probe, IR, spectroscopic, bolometric and pressure measurements. In addition, the effects of such a radiative plasma on the central plasma will be characterized. If the initial results warrant it, and with additional funding, pumping will be added.

B.5.2 Gaseous divertor

In the gaseous divertor concept the parallel ion heat flux into the divertor is converted, through neutral collisions, into a perpendicular neutral heat flux. No other tokamak has a divertor so ideally suited for the investigation of the gaseous divertor concept. The characteristics of the Alcator C-Mod divertor, which are so important for its use as a radiative divertor, are also advantageous for gaseous divertor studies. This work is a natural extension of the particle control studies of maximal divertor neutral pressure described earlier. It will be crucial to identify the dependence on divertor pressure of the various divertor power loss channels: radiation, peak heat load and distribution. Biasing of the outer divertor plate (which requires additional funding) could allow enhanced neutral retention.

B.5.3 Highly-inclined divertor plates

Single-null Alcator C-Mod plasmas can be operated with strike points on either the highly-inclined closed divertor or the more standard open flat-plate divertor. Studies will be performed to compare the two geometries, characterizing the differences in heat loads, impurity generation, impurity retention (magnetic sheath, plasma baffle and mechanical baffle with closed divertor), neutral particle retention and heat load profiles.

B.5.4 High-Z material tiles

There are clear gains to using a high-Z material in a reactor: density control, low tritium and deuterium retention, low disruption erosion and low sputtering (for low T_e). Alcator C-Mod will make unique contributions to our understanding of tokamak operation with high-Z materials, under reactor relevant heat and particle flux conditions.

C.   Control and Disruptions

Active control of the tokamak plasma is essential for proper operation. In circular cross-section plasmas control has usually been limited to the current, the radial and vertical position, and perhaps the electron density. In more modern, elongated, tokamaks the control of the shape is also important and, of course, the inherent axisymmetric instability

to vertical displacements requires much more careful implementation of feedback algorithms. As tokamak research moves forward, the control of increasingly more aspects of the plasma becomes important. Eventually, of course, stabilization of the operating point of a reactor through burn control will be essential. However, even before then, there exist many opportunities to influence, and hence control, the confinement and stability properties of the plasma by active control of heating, fueling and edge profiles. Perhaps the most important stability problem is the control and avoidance of disruptions, since these are often a driving factor in structural design.

Alcator C-Mod is an excellent facility in which to study various aspects of plasma control. Its thick conducting structure is prototypical of future large machines and makes it mechanically robust to disruptions. The ITER-relevant plasma shape and divertor provide a vital testbed for optimizing the control and stabilization of the axisymmetric configuration, while the modest size and excellent internal access around the plasma in Alcator makes possible the investigation of different plasma-stabilizing structures, for example 'twin loops'.

Control of plasma profiles can be obtained by control of the particle and heat sources. The 20-shot pellet injector provides an excellent means of density profile control. ICRF absorption calculations indicate rather localized absorption of RF power. We therefore anticipate quite good control of the heating profiles and hence, to the extent permitted by transport, the temperature profile. In combination, these two tools offer the opportunity to explore control of the pressure and to some extent the current profile. We therefore plan to pursue studies relating to control of the transport (discussed in more detail in the transport section) and also the stability. The planned LHCD experiments offer additional possibilities for direct control of the current profile.

Another important aspect of control relates to the effect of fast particles on MHD instabilities such as the sawtooth. Sawteeth can have a dominant influence on the prospect of ignition in experiments such as ITER, since they limit the peak temperature and may eject high-energy particles (e.g., alphas) from the plasma core. The ICRF heating on Alcator C-Mod offers the ability, at densities in the vicinity of 10^{20} m^{-3}, to explore the sawtooth stabilization effects observed on JET and elsewhere. Lower Hybrid current drive enables the pursuit of similar studies using fast electrons and in addition offers current profile control, whose effects on stability can be studied in combination with the fast particles. Present theoretical indications are that the ion tails to be

anticipated from ICRF are not likely to excite toroidal Alfvén modes, which are a concern in respect of fusion alpha confinement. However, the existence of a substantial ion tail is expected to offer the opportunity to contribute to the broader understanding of energetic particle transport and instabilities.

The dynamic loads experienced by a tokamak during a disruption depend on the decay rate of the plasma current and on the plasma motion, as well as the conducting properties of the surrounding structure. The factors determining current quench are not fully understood. However, the decay rate has been found to be strongly affected by the material composition of the first wall. Alcator C-Mod is unique among modern tokamaks in having a high-Z metallic first wall and it can therefore study the decay rate dependence on wall material as well as vessel conductance.

It appears extremely difficult to control directly the helical instabilities leading to disruptions. However the instabilities are themselves determined by the magnetic configuration, especially the profiles. We intend therefore to explore methods of disruption avoidance and mitigation through active monitoring and control of the profiles and plasma shape.

For this application, Alcator C-Mod's unique hybrid digital/analog real-time control system, which provides for a high degree of flexibility, will be programmed to identify disruption precursors and to respond appropriately. Initially the response would probably be a quick reduction of density, current, shaping, and/or ICRF power, i.e., a withdrawal from the boundaries of the operational space. If the withdrawal is insufficient to avoid a disruption, control of plasma position and other parameters during a disruption may be able to limit its severity. We shall study strategies for accomplishing this mitigation.

D.   Transport
D.1   Background
Studies of confinement and transport continue to be a vital part of the Alcator program. Two general issues will be addressed: transport prediction and confinement improvement. The first is crucial if we are to have confidence in the performance of future machines. As fusion devices get larger, the cost of compensating for inadequate understanding by conservative design can be enormous. The second will also be crucial in lowering the cost of future machines by reducing the demands on machine engineering required to

reach the necessary levels of performance. In mapping out our research program, we place our greatest emphasis on those areas and those parameters for which Alcator C-Mod represents a unique facility; most notably its very high toroidal field, current and plasma density capabilities, and divertor configuration. In addition to its contribution to the general state of knowledge of transport, this work can be directly relevant to machines in the IGNITOR, ARIES line. It should be noted however that with its high performance, wide range of operating parameters, and excellent set of diagnostics, C-Mod should be able to address all of the transport issues important to the fusion program short of those directly connected with a nuclear burn.

D.2 Transport Prediction

The goal of these transport experiments is to obtain sufficient understanding of energy and particle confinement in present day devices to confidently predict the performance of future devices. Empirical scaling laws are a minimal representation of this knowledge though much progress has been made on this basis alone; understanding of the underlying physics would be much more satisfactory. By itself, C-Mod allows the extension of scaling laws into new parameter ranges, increasing the accuracy of the scaling and/or revealing discrepancies. Compact, high field devices have shown themselves to be capable of running over a wide range of operating space, thus historically the Alcator tokamaks have made significant contributions in the area of transport prediction.

It has been proposed to improve the quality of the empirical scaling approach and to increase contact with the underlying transport physics by deriving scaling relations in terms of appropriate dimensionless parameters. A crucial test of this approach comes from comparing machines running with identical dimensionless parameters but very different dimensioned parameters. Specifically we have proposed to undertake such a study jointly with DIII-D and ASDEX-U. All three machines can run discharges with similar shape and aspect ratio (though the match is closest between C-Mod and ASDEX-U) and with nearly identical dimensionless parameters. Table I.1 lists machine parameters from C-Mod, ASDEX-U, and DIII-D, and compares them to those scaled exactly from the minor radius. Note that to make this comparison worthwhile, one of the machines must be a small high field, high current device; otherwise the dimensional differences are too small to be significant.

An important feature of compact high field devices is their ability to run high performance discharges over an extremely wide range of density. (For Alcator C, this

range was well over two orders of magnitude.) This translates into an ability to run over a wide range in the collisionality parameter with hot, thermalized plasmas. Such a collisionality scan will be important in elucidating the role of trapped electrons and ions in anomalous transport. The trapped electron mode is of course a leading candidate for anomalous electron losses in a wide variety of machines and regimes.

D.3   Confinement Improvement

There is special interest in regimes where confinement is enhanced, particularly when relative to scalings with plasma current, which could lead to practical driven-current/steady-state machines. With enhanced confinement regimes seen in such great variety and on virtually all existing tokamaks, it would seem reasonable to assume that future machines can operate with confinement well above L-Mode levels. Before this assumption can be made however, a number of important questions must be answered: What are the conditions necessary to access enhanced regimes? What are the mechanisms which lead to the enhancement? How does confinement scale in these regimes and how good can confinement get?

There is no consensus on the mechanism or the *trigger* to achieve H-Mode. The general observation is that the threshold goes up with device (plasma) size and toroidal field. Attempts to derive a scaling law for the power threshold have met only limited success; exceptions exist to virtually every rule. Because of its unique place in parameter space, C-Mod should make a substantial contribution to resolving this question. Extrapolations of scaling relations from different groups result in values of H-Mode threshold for C-Mod from 200 kW to over 10 MW. (Our *best* guess from dimensionless scaling arguments is a threshold in the range of 3-4 MW.) Clearly, we will be able to distinguish between these extremes. It may be that a simple scaling relation is not an appropriate approach to understanding the conditions to reach H-Mode, particularly as the role of atomic processes in governing plasma edge conditions may be of great importance. C-Mod experiments will be a crucial test of the extent to which scaling from plasma physics considerations is applicable.

Confinement within the H-Mode regime is not nearly as well characterized as in L-Mode or Ohmic plasmas. As described previously, C-Mod will extend the parameter range of the H-mode database into unique regimes and test the validity of the dimensionless parameter scaling approach. Particular attention will be given to parameter scans which promise to discriminate among competing theoretical models of H-mode transport.

Figure I.3 shows the projected performance for C-Mod in $n\tau$-T_i space, under 4 different confinement assumptions. An interesting feature of this plot is the drastic difference in results when a Neo-Alcator term is added in quadrature with H-mode confinement. Because of its very high current density, C-Mod will routinely run in regions of parameter space where the L- and H-mode scalings predict confinement times greatly exceeding those predicted for ohmic plasmas. The actual behavior of devices in this regime has never before been investigated.

The second generic method for improving confinement comes from peaking the plasma density profile. C-Mod will employ a 20-shot pellet injector which should be capable of accessing this regime over wide ranges of target density and applied power. The standard explanation for enhanced confinement in this regime is the suppression of ITG modes by the short density scale length. While appealing and consistent with much data, there are some serious discrepancies between experimental results and the ITG theory. Definitive tests may come by comparing experiment and predictions for both particle and impurity confinement. It should also be possible to observe the ITG fluctuations directly with the microwave reflectometer. With very peaked profiles, the ITG mode should be completely suppressed. At reasonable densities, the large plasma current in C-Mod should provide operation with $\chi_{nc} \ll \chi_e$. This would allow us to isolate the electron transport and possibly measure residual ion transport associated with electron modes.

D.4   Objectives of C-Mod Transport Program
The main objectives of the C-Mod transport physics program can be summarized as follows:

- Test dimensionless scaling concepts.
- Investigate the threshold and properties for enhanced confinement regimes (H, VH, P, IOC, Limiter Biasing/E_r and combined regimes).
- Measure transport with strong applied RF heating.
- Test ITG and other drift wave theories.
- Correlate energy, particle, and impurity transport and investigate the importance of off-diagonal transport matrix terms.
- Study the relationship between particle transport and the density limit.
- Attempt to determine whether electron transport is driven by electrostatic or electromagnetic modes, or both.

E. RF Heating Program

The C-Mod RF heating program consists of two key elements: (a) ICRF heating with fast magnetosonic waves at 80 MHz; (b) Long pulse (exceeding the L/R time) noninductive current drive with lower-hybrid waves at 4.6 GHz. In addition to the heating and current drive experiments, these systems will also be used to control temperature and current profiles, and to explore access to enhanced confinement regimes.

E.1   ICRF Heating Program

The primary role of ICRF power in C-Mod is to provide the bulk auxiliary heating power for carrying out the programs outlined elsewhere in this summary. Of particular importance are studying transport and divertor heat load issues with ITER and reactor relevant densities, magnetic fields, and plasma shapes. C-Mod can also test ICRF heating at ultra-high densities, such as may be encountered in compact Ignitor-type devices. ICRF heating will be carried out in both gas and pellet fueled discharges, and in both inductively and non-inductively (LHRF) driven discharges. The ICRF power density on the antenna surfaces will be high, up to 20 MW/m^2, which is in the reactor-relevant regime.

E.2   ICRF Equipment

The transmitters for the initial operating phases consist of two 80 MHz FMIT units, each operating at 20 MW, for a total source power of 4.0 MW. We also have two additional transmitters on site, which could be refurbished if more power were needed. Given that only about 75 – 80% of the source power is typically absorbed by the bulk plasma, at least one of these additional transmitters should be refurbished for a total absorbed power of 4-5 MW. The power from each transmitter is coupled to the plasma by a two-strap antenna attached to the vacuum chamber wall. The straps may be operated either in-phase (monopole) or out of phase (dipole), and the RF power density on the antenna is ~ 10 MW/m^2 if one 2 MW transmitter is connected to each two-strap antenna. The antennas can handle the expected RF and plasma heat loads for pulse lengths of at least 10 seconds. Based on code modeling, we expect good loading resistance for plasma densities in the range from 10^{20} to 10^{21}m^{-3}, allowing coupling of the 2 MW of RF power at RF voltages not exceeding 45 kV anywhere in the transmission line (including the antenna itself).

The design of the two-strap antenna is shown in Figure I.4. The Faraday shield elements are slanted and coated with TiC or B$_4$C. To facilitate operational flexibility in the initial

phase of plasma operations, a movable, single strap (monopole) antenna will be used for coupling studies. This antenna has already been fabricated and will be installed in the machine before Phase I operations recommence, thereby giving us experience with operating the ICRF system at the earliest possible date. The fabrication of the double-strap antennas is well under way, and the antennas will be available for installation into C-Mod prior to the start of Phase II.

E.3   ICRF Heating Regimes

Assuming a deuterium bulk plasma, the base operating scenarios will use toroidal fields of 5.3 Tesla (H minority regime) and 7.9 Tesla (He-3 minority regime). Single pass absorption is excellent for the H minority regime (~ 90%) whereas in the He-3 minority case it is typically about 20%. In past ICRF experiments, efficient heating has been observed for single pass absorption of order of 5% or higher. High single pass absorption is desirable for minimizing impurity generation. The slanted, coated Faraday shield design should minimize impurity production at the antenna surface. Ultimately, boronization may be necessary for effective impurity control in an *all metallic* environment such as in C-Mod.

In order to heat plasmas at even lower magnetic fields (higher β), we find effective heating scenarios at B = 3.95 Tesla (2nd harmonic He-3 minority, with approximately 20% single pass absorption) and B = 2.65 Tesla (2nd harmonic H minority; electron Landau/TTMP regime). In summary, while transmitters with continuously tunable frequency would be optimal, a wide variety of magnetic fields (2.65, 3.95, 5.30, 7.90 T) are suitable for effective heating of deuterium majority plasmas for transport, current drive and divertor heat load studies.

E.4   Advanced Tokamak Regimes with Lower Hybrid Current Drive

Utilizing the 4.0 MW, 4.6 GHz lower-hybrid system from the Alcator-C program, C-Mod can access non-inductive operation regimes which are unsurpassed by any other existing tokamak facility in the world. Of particular importance is the high *effective* current drive power (P/R $\propto$ I) and the long-pulse operational capability of the C-Mod magnet system at fields of B $\leq$ 5 Tesla, where $t_{pulse} \geq 7$ s. Owing to the small size of C-Mod, this pulse length exceeds the L/R time (t ~ 4 s), even at $T_e \simeq 5$ keV, $Z_{eff} \simeq 2$. To achieve the above temperatures at central densities of 10^{20} m^{-3}, an H-mode factor of 2 is assumed.

The main hardware requirements for achieving the LHCD options in C-Mod include an upgrading of the power supply/modulator systems to long pulse (10 s) operation, and fabrication and installation of a new grill/window array. The 16 klystrons are CW tubes, each with output power of 250 kW, for a total of 4 MW source power. At present, half of this system is on loan at PPPL and we expect that this equipment will be returned to MIT in FY 94, after completion of the PBX lower hybrid experiments.

Table I.2 summarizes several advanced tokamak regimes which have been identified by a combination of the ACCOME code and/or Nevins' Spread Sheet (based on the ITER-89 formulary). For the sake of completeness, in Table E.1 we also include a pellet-inductive scenario which assumes a peaked density (pressure) profile, and hence a high bootstrap fraction ($f_{BS} \geq 80\%$) at low currents (high β_p) with intense ICRF heating. The diffusion time of fast electrons is typically an order of magnitude longer than their collisional slowing-down time.

We thus expect efficient current profile control using the lower-hybrid driven fast electrons. Typical values of v^* at the 80% flux surface are from 0.1 to 0.5, so the plasma is sufficiently collisionless to test bootstrap current theories.

The following important scenarios may be noted in Table E.1:

- Confinement studies in noninductively driven plasmas with strong ICRF heating at ITER-like parameters ($n_e \simeq 10^{20}$ m^{-3}, $B \simeq 5$ T, $I_p \simeq 1.5$ MA, $q_\psi \simeq 3$, R/a ~ 3.3, $t_{pulse} \simeq 10\, t_{skin}$).
- Tokamak operation and confinement studies with High Bootstrap Fraction ($I_{BS}/I_p \gtrsim 0.6$) at high $\epsilon\backslash\beta_p$ and $t_{pulse} \simeq 10\, t_{skin}$.
- Current profile control (with LHCD and high I_{BS}) combined with ICRF heating, permitting C-Mod to operate in the 2nd stability regime ($\beta > \beta_{Troyon}$ for $t_{pulse} \simeq 10\, t_{skin}$
- Combined LHCD and ICRF provides for synergistic current drive studies.
- LHCD in C-Mod allows a quantitative study of fast electron diffusion ($\epsilon \leq 100$ keV) under a wide range of plasma operational modes.

In summary, C-Mod is an excellent vehicle for studying advanced tokamak scenarios of the type identified for TPX/SSAT. These studies would take place in the CY 1996 - 2000

time frame, thereby providing important information for optimizing the SSAT operating scenarios.

F.   Unique Features of Alcator C-Mod; Limitations, Schedule

F.1   Features

Alcator C-Mod is at present the only tokamak in the world with a closed divertor and with the capability to modify the divertor relatively easily. In view of the ITER relevant plasma shape and heat fluxes, the device is ideal for studies of ITER divertor design, including questions of the effectiveness of radiative and gaseous divertors. The high Z divertor walls and metallic first wall also provide unique ITER relevant information.

The combination of high magnetic field, current, and mass density will allow unique testing of the validity of the scaling laws used to project machine performance. The availability of substantial amounts of ICRF power is essential in this task.

With the availability of LH current drive, Alcator C-Mod could also explore enhanced performance tokamak regimes for about 10 current penetration, or skin, times. This capability, although dependent on upgrades for LHCD, is unique in the period in which it is planned to be done.

F.2   Limitations

The budget described in the next section does not include increased engineering or technical staff, which might be necessary, or at least highly desirable, as extra hardware and diagnostic capability becomes available.

The ICRF system does not, at present, include tunable power supplies, so that scans of machine performance as a function of B will require only different heating scenarios. In the earlier device, Alcator C, ICRF heating was not effective for plausible reasons. Thus, final success of the ICRF heating on a high density device is not absolutely assured.

The small size of the device and divertor will cause some difficulties in accurate diagnostic analysis of the scrape-off layer and divertor regions

F.3   Schedule

Since the preparation of the body of this report (June 1992) the date for the completion of the repairs to the poloidal coil feeds has slipped one or two months. The start date of October 1992 shown in Figure A.2 could slip to late November or December of 1992.

G.   Budget

The group has layed out a program shown in Figure I.1 that costs $16.8 M per year. The personnel costs of $11 M include roughly 18 physicists, 20 engineers, 30 technicians and 18 graduate students, plus support staff. With operating costs of $3.3 M and $.9 M for small scale acquisitions approximately $1.6 M per year is proposed for capital equipment. Over a six year period it would be allocated as follows: upgrade of ICRF systems $2.0 M, LH systems $4.8 M, divertor biasing $.5 M, and testing of ITER prototype divertor $2.3 M.

1992	1993	1994	1995	1996	1997	1998

B ≤ 5T   B ≤ 9T   B ≤ 9T   B ≤ 9T   ICH ≤ 8MW

$B \leq 5T$ $B \leq 9T$ $B \leq 9T$ $B \leq 9T$ $ICH \leq 8MW$
$I \leq 0.8MA$ $I \leq 2.5MA$ $I \leq 3.0MA$ $I \leq 3.0MA$ $LH \leq 4MW$
$\kappa \leq 1.5$ $\kappa \leq 1.8$ $\kappa \leq 1.8$ $\kappa \leq 2.2$
$ICH \leq 2MW$ $ICH \leq 4MW$ $ICH \leq 6MW$ $ICH \leq 8MW$
$LH \leq 2MW$

I — Ohmic, SOL & ICH Studies

IIA — IIB — Full-field, Radiative divertor &ICH Physics

IIIA — IIIB — Hi Power ICH, Gaseous divertor & H-mode Studies

IVA — IVB — Opt. Divertor &Tokamak Improvement Experiments

Repair Coil; Install ICH Ant.

Install Flywheel, Dipole Ant.'s

Install FMIT #3

Install FMIT #4, LH grill

Maint. & Insp., Mod. Divertor

Limiter Ops, Shaped control, Shaped Div. ops, Ohmic Conf., ICH coupling, Pellet fuelling

Radiative Divertor, Power loading char., Compare open & closed divertors, ICH phyics, Transport scalings, L&H-mode studies

Gaseous divertor, H-mode confinement, Div. bias/pumping β-limit studies, Shaping - high κ

Design/build optimized divertor, Advanced shaping LH current drive Quasi-steady state ops. with divertor cooling

Figure I.1
Schedule and physics program highlights for Alcator C-Mod, through CY 1998
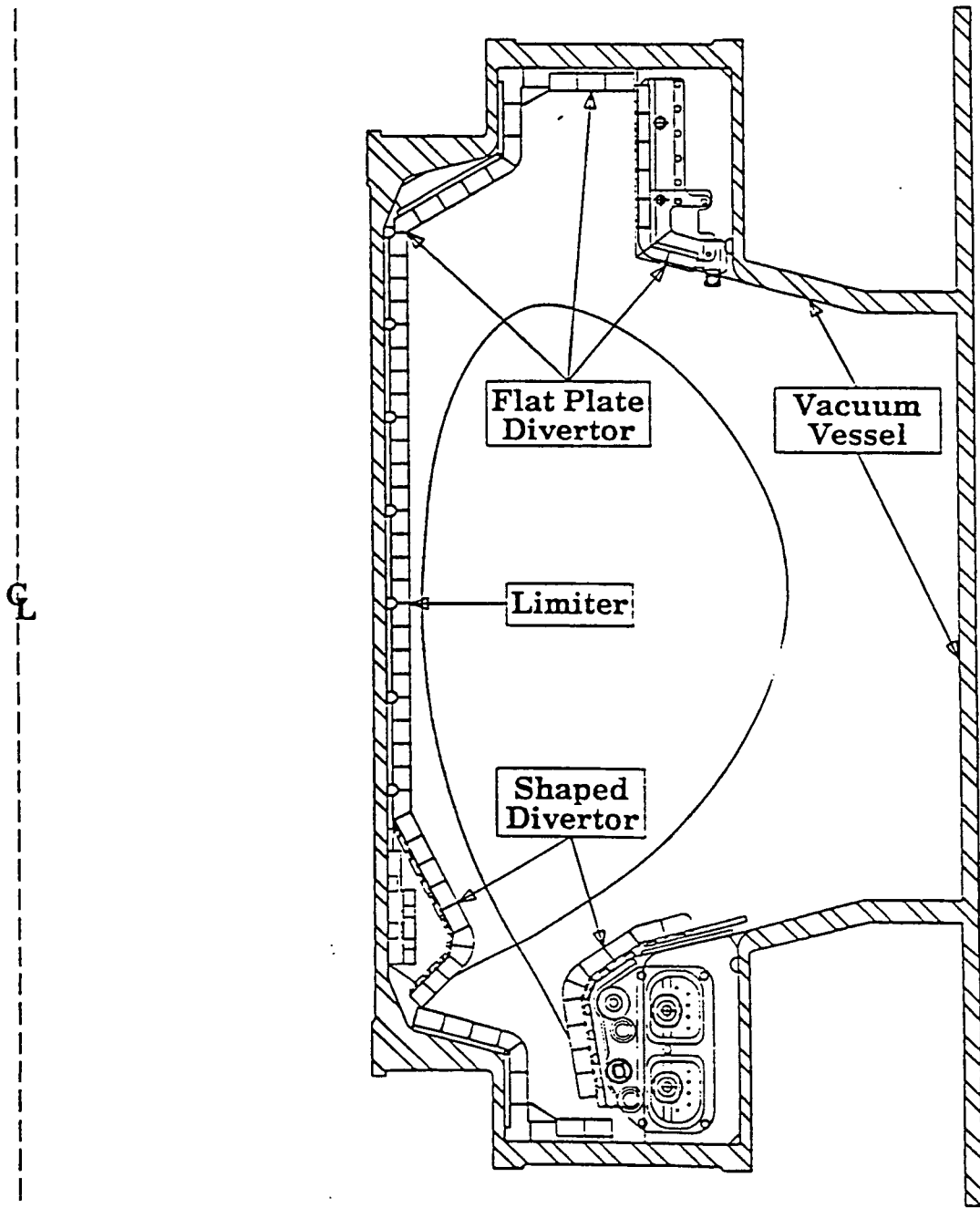
Figure I.2
Alcator C-Mod vessel and first wall hardware

	C-Mod Test Point	ASDEX-U Actual	ASDEX-U Scaled from C-Mod	DIII-D Actual	DIII-D Scaled from C-Mod	Machine X
a (m)	.21	.5	.5	.67	.67	1.7
b (m)	.38	.8	.9	1.3	1.2	3.1
R (m)	.67	1.65	1.59	1.67	2.14	5.5
B (T)	8.0	2.7	2.7	2.0	1.9	.6
Ip (MA)	2.0	(<1.6)	1.6	(<3)	1.5	1.2
P (MW)	4	(<12)	2	(<20)	1.7	.8
ne (M^3) (x 10^{20})	6.0	(<1)	1.0	(<1)	.6	.09

Table I.1
Values for nominal machine parameters and a proposed operating point for dimensionless scaling experiments
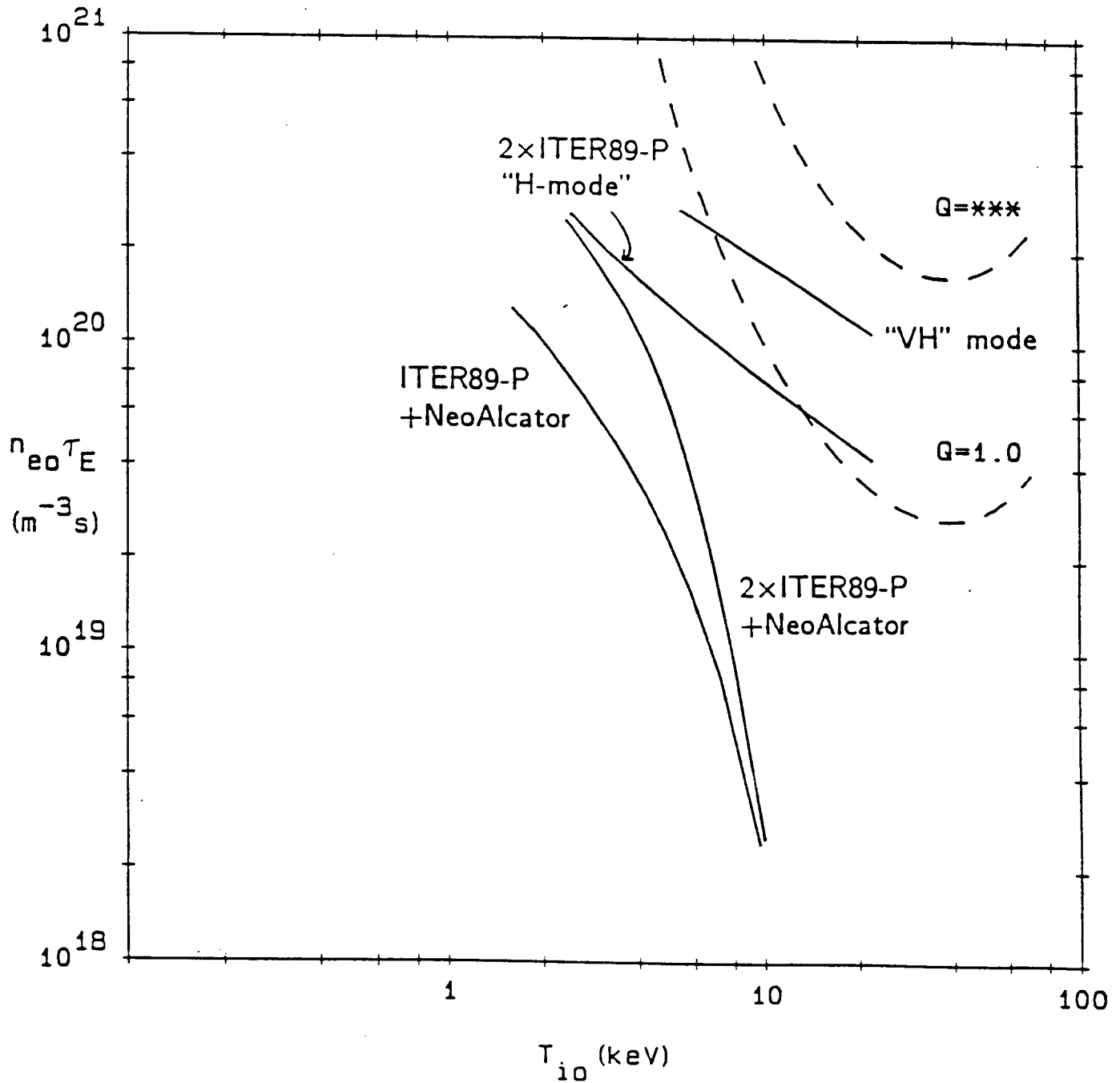
Figure I.3

Predicted plasma performance for Alcator C-MOD under different confinement assumptions. Parameters are I_p = 2.5 MA, B_T = 9 T, (q_ψ = 3), auxiliary heating power 4 MW. Each curve is a density (n_{e0}) scan from 1 to 20 x 10^{20} m^{-3}. For the cases including NeoAlcator losses, they are added in quadrature with the (scaled) ITER89-P transport. Whether or not NeoAlcator confinement can be exceeded makes a very great difference at the lower densities.

SECTION-A-A
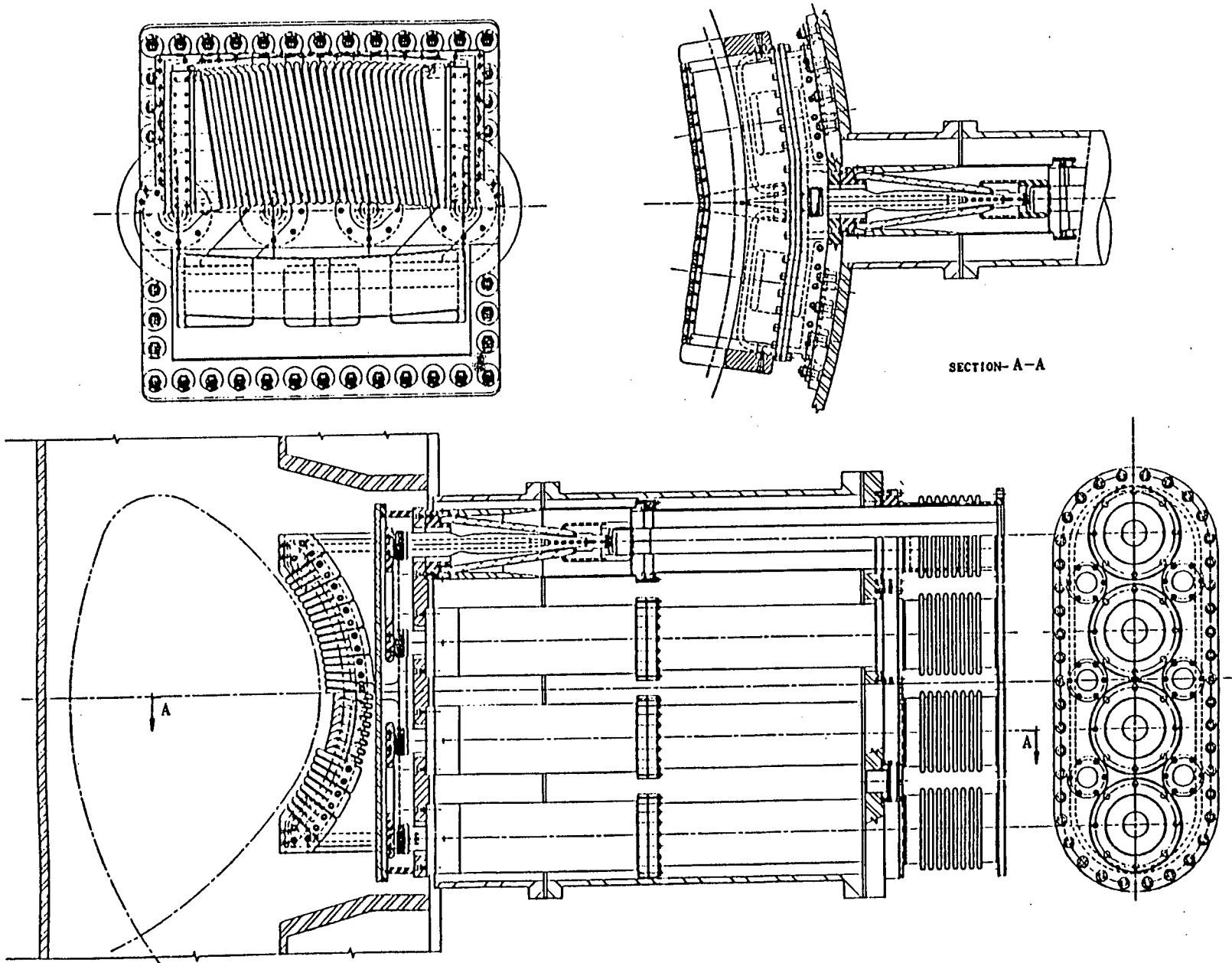
Figure I.4   Engineering drawing of the two-strap ICRF antenna.

	ITER-like (LHCD)	High Bootstrap, 2nd Stable (LHCD)	High Bootstrap (OH-pellet)
B_o (T)	5.3	4.0 (5.3)	5.3
I_p (MA)	1.4	0.39	0.33
I_{BS}/I_p	0.03	0.63	0.81
$<n_e>$ (10^{20}m^{-3})	0.68	0.70	0.51
t_{pulse} (s)	7.0	10.0	1.0
$q(0)/q_{95}$	0.7/2.3	2.5/7.5	2.5/12.3
τ_p/τ_{skin}	11	13	0.5
P_{LH} (MW)	4.0	0.5	0
P_{ICRF} (MW)	1.2	4.3	6.0
$\beta(\%)/\beta_{troyon}(\%)$	0.09	1.20	0.80

Table I.2

Advanced Tokamak Scenarios (From ACCOME code and Nevins' Spread Sheet)
(Assumes H-mode, f_H = 2; k = 1.6–1.8)

II.
ORNL Advanced Toroidal Facility

A.   Background and Overview

A.1   Mission

ATF was designed to demonstrate high-β, steady-state, disruption-free operation that leads to a reactor-relevant configuration. The ATF program goals for FY 94 – FY 97 are:

- demonstration of $\langle \beta \rangle > 4\%$, low collisionality, and improved confinement for pulse lengths up to 30 s;
- optimization of the stellarator configuration and operational techniques for improvement of LHD performance and the design of a better D-T stellarator;
- development of steady-state power and particle handling.

Only ATF has the combination of configuration flexibility, β capability, pulse length, access for power and particle handling, and ICRF heating capability to accomplish these goals. It is cost effective because ATF and most of its heating additions already exist. It is timely because ATF would provide data >5 years before the large ($400 M) Japanese LHD stellarator, and could thus determine the next generation of reactor development. There is a window of opportunity in 1994-97 for ATF to make crucial contributions to the development of the stellarator concept and to allow the U.S. to capitalize on the strong world program at relatively low cost.

A.2   Rationale

The world stellarator program provides the only credible, timely alternative to the toka-mak in reactor development. In the tokamak area, a steady-state, advanced tokamak (SSAT) has been proposed in parallel with ITER as essential to the development of an attractive tokamak DEMO. In the stellarator area, ATF will perform a similar role in demonstrating a reactor-relevant mode of operation. In particular, ATF protects against failure of the tokamak program and offers a promising route to fusion development through development of a lower-aspect-ratio, sheared-magnetic-field stellarator reactor (the decision points shown in Fig. II.1).

Stellarators have demonstrated energy confinement times, scaling and keV temperatures similar to those in tokamaks at the same stage of development (Figure II.2) and Wendelstein VIIAS recently (June, 1992) achieved H-mode operation. Stellarators have achieved beta ($\langle \beta \rangle$) up to 2% and have operated for up to 20 s. Now, it is essential to

validate projections of $\langle \beta \rangle \geq 4\%$ and improved confinement at low collisionality. In particular, dimensionless parameter modulation studies in ATF showed that confinement improves with increasing β and decreasing collisionality v as $\tau_E/\tau_{grB} \propto \beta^{0.3} v^{-0.2}$.

A.3   Stellarators as Reactors

Stellarators require no plasma current. This eliminates disruptions and their associated engineering problems, greatly reduces the recirculating power requirements of a reactor, and removes a serious physics constraint on optimization.

New stellarator reactor assessments (Lyon et al, presented at ANS Meeting in June, 1992) using the same costing algorithms and unit values as in the ARIES-I tokamak reactor study show that a stellarator reactor with same power output as ARIES-I could have

- similar major radius
- lower cost of electricity
- significantly higher mass utilization
- much lower magnetic field

as shown in Table II.1.

The magnet systems and associated structure are smaller for the stellarator reactor because the helical coils are closer to the plasma, and both the helical and poloidal field coils have smaller cross sections. The stellarator coil set, structure, and blanket can be constructed of ≥ 10 modules, with additional central access, thereby easing the maintenance problem. Coil configurations can be designed to have low power density external divertors and to avoid the accumulation of helium ash, while retaining acceptable alpha particle heating.

B.   Technical Objectives

The ATF program focuses on performance improvements and physics understanding needed for developing an attractive stellarator reactor: (1) confinement improvement; (2) reactor-relevant levels of beta; (3) low bootstrap current; (4) exploration of a configuration and transport data base for extrapolation to larger devices; (5) development of an efficient long-pulse heating scheme (ICRF); (6) acceptably low fast-ion losses; and (7) steady-state capability (power, particle, and impurity handling). *ATF is the only existing stellarator that allows an integrated test of these physics issues.*

B.1   Short-term program (<5 years)

The proposed 3½ year experimental program for ATF in FY 1994–1997 is based on 1½ years of preparation at low budget levels in FY 1993-94, as discussed in Section VI. The main objective for ATF in FY 1994-1997 will be to: (1) demonstrate reactor-relevant parameters ($\langle\beta\rangle \geq 4\%$, low collisionality [$v^*_{helical} \approx 10^{-2}$, $v^*_{toroidal} < 1$], and ≈ 2 improvement in confinement) in long-pulse operation (≈ 30 s); and (2) optimize the stellarator configuration (developing a reactor-attractive magnetic configuration and a corresponding modular coil set for a possible D-T test step). Key elements of the physics program are:

- stellarator optimization: physics of ICRF, ripple-induced transport, minimization of the energetic-particle loss region, effect of electric fields on τ_p and τ_E, and dependence of beta limits on configuration properties;

- tests of neoclassical theory: bootstrap current and Ware pinch, transport from stochastic fields and magnetic islands;

- anomalous transport: gyro-reduced Bohm scaling and separation of helically-trapped and toroidally-trapped particle instabilities; and

- second stability: tests of the predicted high-β confinement enhancement ($\tau_E/\tau_{grB} \propto \beta^{0.3} v^{-0.2}$) and exploration of the second-stability regime under equilibrium conditions; stability of broad pressure profiles.

This program would build on ATF's demonstrated physics base: configuration control, β-self stabilization; the neoclassical nature of the bootstrap current and its control (and reversal); and the correspondence between tokamaks and stellarators in edge fluctuations, velocity shear layer, and transport. These contributions are indicated graphically in Fig. II.3.

B.2   Longer-Term Program (>5 years)

If successful, the next step (FY 1998-) would be true high-power steady-state operation of ATF to demonstrate steady-state power and particle handling and plasma control. If this step were successful (and the data from LHD, W VII-X, ATF, etc. are sufficiently encouraging), and the tokamak were not ready to proceed to a DEMO, then the stellarator program would be ready to proceed with the development of a D-T Stellarator.

C. Special Features

C.1 Uniqueness of ATF Contributions

ATF will remain the world's largest stellarator until the end of this decade when LHD starts producing results, and is the only large stellarator in the U.S. program. ATF has >2× the plasma volume and much greater interior access and port size than any other stellarator and >2× the heating power and >10× the pulse length of any other sheared stellarator. Even when LHD is in operation, ATF will have >2× the volume power density in pulsed operation and could have >2× the heating capability and 10× the surface power density of LHD in steady-state operation .

C.2 Place in the World Program

There are basically two routes to stellarator development: sheared systems such as Heliotron-E, CHS, and ATF leading to LHD; and low-shear systems such as W VII-AS leading to W VII-X. The sheared system is optimum at lower aspect ratio. Each system has important reactor-relevant features, such as $\langle \beta \rangle \geq 5\%$, modularity, and a natural divertor. The best system for development of the reactor must arise from the experimental program. ATF was designed to complement, not duplicate, other stellarators in the world program by testing MHD optimization principles. It has a magnetic configuration similar to that in LHD, but with extra flexibility for optimization.

C.3 Other Special Features

ATF can access the widest variety of magnetic configurations of any toroidal experiment, allowing independent control of shear, magnetic well, and trapped particle fraction for fundamental toroidal physics studies. Dynamic configuration control allows access to configurations not otherwise accessible. In addition, ATF is the only experiment capable of exploring steady-state operation in the second-stable regime and the only U.S. toroidal experiment capable of true high-power steady-state operation before SSAT.

D. Parameters and Limitations

Because of budget reductions, auxiliary heating levels (≈ 1 MW) were far below the designed level of 6–8 MW during the initial period of ATF operation (1988-91), and development of particle and power-handling systems were delayed. Nevertheless, good progress was made in achieving relevant plasma parameters and in demonstrating the 20-s pulse-length operation, with detailed, time-dependent control of the magnetic configuration important to the next phase of the program. Maximum parameters achieved (not simultaneous) were $T_i(0) \approx 1$ keV, $T_e(0) \approx 1.5$ keV, $\bar{n}_e \approx 2 \times 10^{20}$ m^{-3}, $\tau_E \approx 30$ ms,

and $\langle \beta \rangle \approx 1.7\%$. Sets of simultaneous plasma parameters for four different operating regimes are given in Table II.2.

ECH plasma startup and busbar cooling presently limit ATF to operation at $B \approx 1.9$ T (5 s), $B \approx 0.95$ T (30 s), and $B \approx 0.67$ T (steady state). Using ICRF for plasma startup, as has been done on other stellarators, would allow ATF to operate at fields between 0.5 T (for maximum beta) and 2 T (for maximum confinement). Cooling of the main busbar from the helical field power supply would allow steady-state operation at B up to 1 T.

ATF has an uncooled vacuum vessel, which limits the power to the vacuum vessel walls to ≈ 100 kW in steady-state operation and to 3-MW 20-s pulses every 10 minutes. There is room to install water-cooled panels inside the vacuum vessel for steady-state operation at higher power, but this is not planned for the 1994-1997 operating period.

The present levels of 0.4 MW of long-pulse ($\leq$30-s) heating and 1.8 MW of short-pulse (0.3-s) heating are inadequate for an experiment of the size of ATF (3 m³ plasma volume). ATF plans call for short-pulse heating of 3.4 MW, long-pulse heating of 1.4 MW, and steady-state (>1 hour) heating of ≈ 0.1 MW at the start of operation in mid FY 1994. By October 1995, the short-pulse heating would be 8.4 MW and the long-pulse heating would be 4.4 MW.

Figures II.4 and II.5 show how ATF can reach the high-β and low collisionality regimes required to meet its physics objectives. Figure II.4 shows the plasma β as a function of heating power for gyro-Bohm confinement and the expected enhancement $\tau_E/\tau_{grB} \propto \beta^{0.3} \nu^{-0.2}$. Values of $\langle \beta \rangle > 4\%$—sufficient to test β self stabilization—are accessible even with gyro-Bohm scaling. Note that for this plot we choose to operate at the maximum density attainable in ATF, $\bar{n}_{max} = 1.3 \, \bar{n}_{Sudo}$, where $\bar{n}_{Sudo} = 0.25(PB/a^2R)^{1/2}$. Figure II.5 shows the central temperature and electron collisionality as a function of power. The helical trapping regime (where instabilities related to helically trapped particles may occur) is accessible even with gyro-Bohm confinement; with enhanced confinement, the toroidal trapping regime (where instabilities related to toroidally trapped particles may occur) also becomes accessible.

E.   Current Status

ATF is not operational at present. At the end of May 1991, an electrical short occurred between two of the segments in one of the two large helical-field windings. A temporary

repair allowed ATF to operate in the Fall of 1991, but at reduced field and lower repetition rate. Two spare helical segments are on hand to replace the damaged segments, but ATF must be disassembled to install them. This disassembly was underway when it was halted for lack of funds. ORNL estimates that it will take 4-6 months to finish the repair in FY 1993.

The ATF group is now heavily involved in collaborations on different U.S. tokamaks (DIII-D, PBX-M, TFTR) and Tore-Supra. Most of these people could return to ATF in mid-FY 1994 without seriously compromising these collaborations.

F. Time and Cost to Complete Objectives

F.1 ATF Program, FY 93 – 97

Figure II.6 shows the budget and schedule needed to carry out the research program outlined in Section II. A detailed budget breakdown is shown in Table II.3. This program allows for $3^1/_2$ years of physics operation beginning in mid-FY 94. Prior to that, resources are required to restore ATF to full operation. A more aggressive funding profile would accelerate the ATF program by six months, as is shown in italics in Table II.3.

The Fabrication budget provides for improved plasma heating capability (using heating supplies already at ORNL) and installation of a module for edge particle and power control tests for development of a stellarator divertor. A significant portion of the ATF Operations budget, 3–5 M$/year, could be provided for outside collaborators. In the past, US and foreign collaborators have contributed significantly to the ATF program, supported both by their own funding and ORNL direct support of 0.5-1 M$/year.

F.2 Steady-State Upgrade

ATF was designed for steady-state operation. The coil systems and power supply can operate steady-state at $B = 1$ T, but additional plasma heating and cooling is required. The ATF Steady-State Project would cost an additional 25 M$ in capital investment (plus ~20M$ operating for three years) and have two phases. In the first phase (with 2-MW heating capability): the power supply buswork and the helical-winding joint cooling would be upgraded for 1 T operation; new, cooled panels would be added to the vacuum vessel to handle 8-MW in steady state and the full divertor and pumping system would be installed; and cooling would be added to the ICRF antennas and the BBC transmitters. In the second phase, an additional 4-6 MW of steady-state ICRF heating would be added.

Each 2-MW increment would require an antenna, steady state transmitter, and transmission line.

G. Appendix. Historical Background

During the past 10-15 years, there have been tremendous advances in stellarator research. Notably, during the early 1980's experimental research on the sheared Heliotron-E (Kyoto) and shearless W VII-A (Garching) showed confinement and beta similar to that in tokamaks at the same stage of development. Subsequently, further experiments and theoretical work, including an increased effort in the U.S. and the former Soviet Union, showed that there could be attractive modular stellarator reactors.

The ATF was developed at ORNL with extensive national and international collaboration to demonstrate these capabilities. During its operational phase from January 1988 through October 1991, ATF amply demonstrated its potential to provide important answers for stellarator development and the understanding of the fundamental toroidal physics. Broad collaboration was an important feature of the program. However, exploitation of its full capabilities was limited by successive reductions in budgets owing to changes in DOE policy, as shown below in unescalated M$.

FY 1987	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992
19.3	17.7	16.2	12.8	8.2	0.4
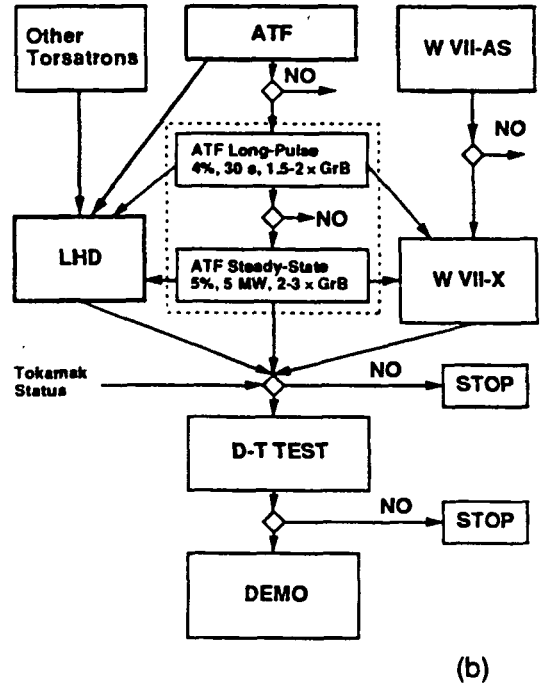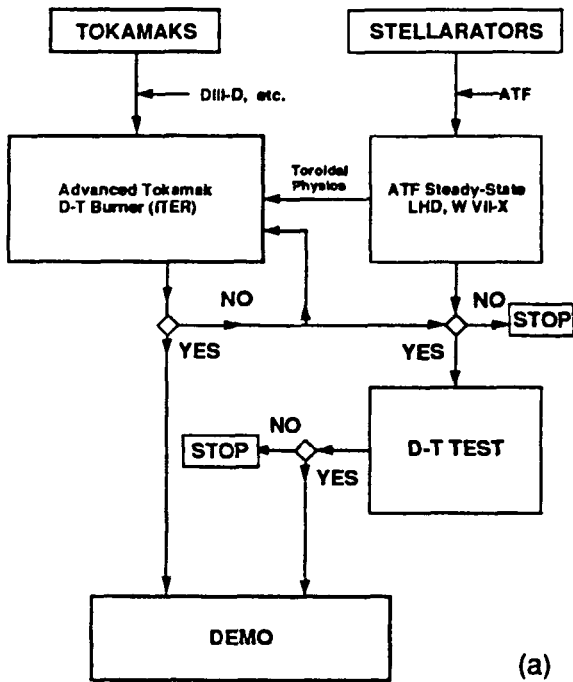
Fig. II.1. Plans for development of (a) steady-state toroidal fusion reactor,
and (b) stellarator reactor option.

$$\tau_{gB}^{stell}\text{(s)} = 0.25 \cdot B_T^{0.8} \cdot n_{20}^{0.6} \cdot P_{MW}^{-0.6} \cdot a_m^{2.4} \cdot \kappa \cdot R_m^{0.6} \cdot A^{-0.2}$$

$$\tau_{gB}^{tok}\text{(s)} = 1.94 \times 10^{-2} \cdot I_{MA}^{0.8} \cdot n_{20}^{0.6} \cdot P_{MW}^{-0.6} \cdot a_m^{2.2} \cdot (R/a)^{1.02} \cdot A^{-0.2}$$
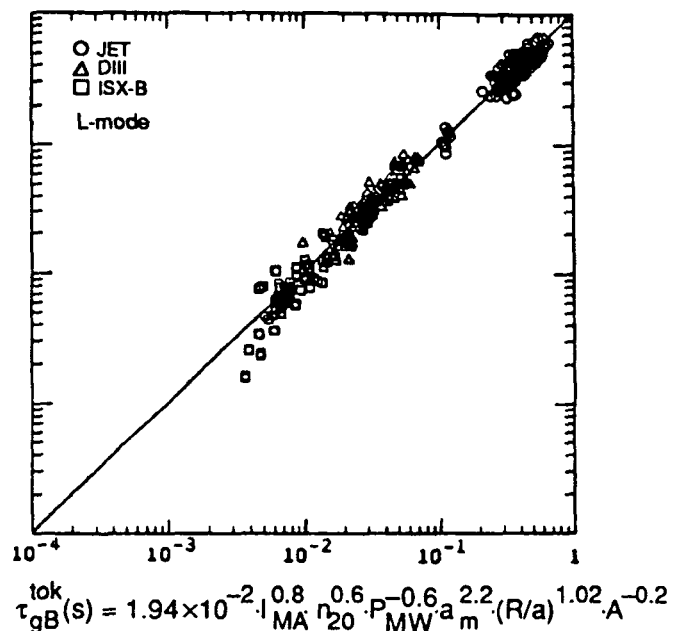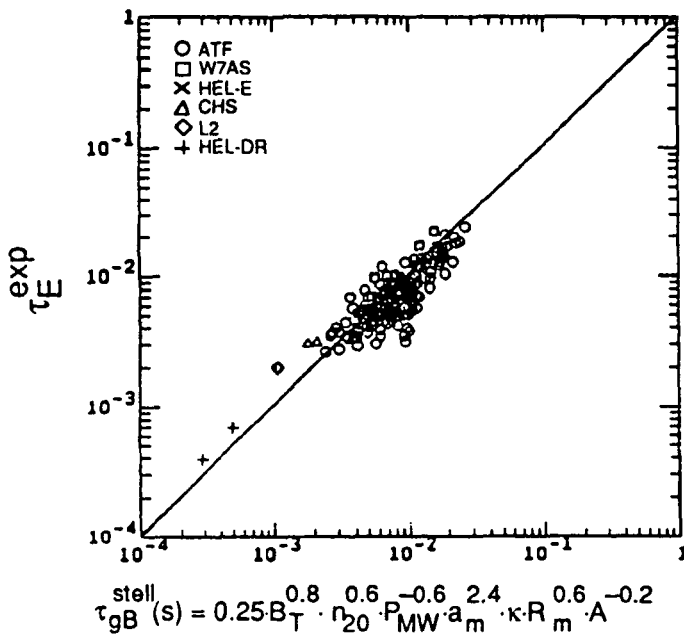
Fig. II.2. Comparison of energy confinement in stellarators and tokamaks (L-mode) using gyro-Bohm scaling. For $\kappa = 1.4$, $R/a = 3$, $q_{cyl} = 3$, (an "average" tokamak geometry), $\tau_E^{stell} = \tau_E^{tok}$.

Key Characteristic	Reference stellarator	ARIES-I
Net electric power, GW	1.0	1.0
Major radius, m	6.65	6.75
Cost of electricity, mill/kW(e)h	67.8	87.4
Mass utilization efficiency, kW(e)/tonne	197	100
Magnetic field on axis/at coil, T	5.61/12.0	16.6/19.9
$\langle \beta \rangle$	5.4%	1.9%
Recirculating power fraction (to plasma)	0	9%

Table II.1. Comparison of reference stellarator and ARIES-I reactor designs.

Program Element	Demonstrated	Objective
• Plasma performance	enhancement factor $\tau_E \propto \tau_{gyro\,Bohm} \left(\dfrac{\beta^{0.3}}{v^{*0.2}} \right)$ $\langle\beta\rangle \le 1.5\%$ 20-s pulse lengths @ 0.4-MW	≈ 2 enhancement in τ_E $\langle\beta\rangle \ge 4\%$, 30-s pulse lengths @ 4-MW
• Configuration control	static & dynamic *control* of $q(r)$, $q'(r)$, $V''(r)$, $f_{trapped}$, $p(r)$	static & dynamic *optimization* of $q(r)$, $q'(r)$, $V''(r)$, $f_{trapped}$, $p(r)$
• Verify neoclass. theory	control of bootstrap current	bootstrap current, Ware pinch @ lower v^*
• Fluctuations & anomalous transport	edge like tokamak resistive interchg. in gradient $\Rightarrow \tau_E^*$ DTEM $\Rightarrow$ constrain n(r) ?	separate helically-trapped and toroidally-trapped particle instabilities, gyro-Bohm scaling
• Role of electric field	biased limiter $\Rightarrow$ velocity shear decouple energy & particle conf.	optimize at finite β
• Second stability	access to Mercier 2nd stability with narrow p(r) ($\langle\beta\rangle \approx 0.5\%$)	equilibrium access and τ enhancement at $\langle\beta\rangle \ge 4\%$
• Particle & power handling	long pulse (20 s) with ECH	particle and power distribution, install divertor test modules
• ICRF heating	modest heating at low power	heating optimization using inside- and outside-launch antennas

improved confinement, high β, low v transport physics*

stellarator optimization

ICRF power and particle handling

integrated test: $\beta = 4\%$ $\approx 2 \times \tau$ **improvement** duration ≥ 30 s

Fig. II.3. ATF program elements, achievements, and future objectives.

	High stored energy	_High β_	_Low ν*_	_Long ECH_
shot	#11740	#11186	#14514	#16654
B (T)	1.9	0.45	1.9	0.95
$\bar{n}_e$ (10^{19} m^{-3})	11	4.3	0.53	0.5
P_{abs} (MW)	0.96	0.98	0.79	0.25
τ_E^* (ms)	26	6.1	6.4	5
$T_e(0)$ (keV)	0.4	0.3	1.0	1.0
$T_i(0)$ (keV)	0.4	0.3	1.0	0.2
$\langle \beta \rangle$ (%)	0.4	1.7	0.08	0.15
duration (s)	0.25	0.2	0.15	20

Table II.2. ATF parameters for four different operating regimes
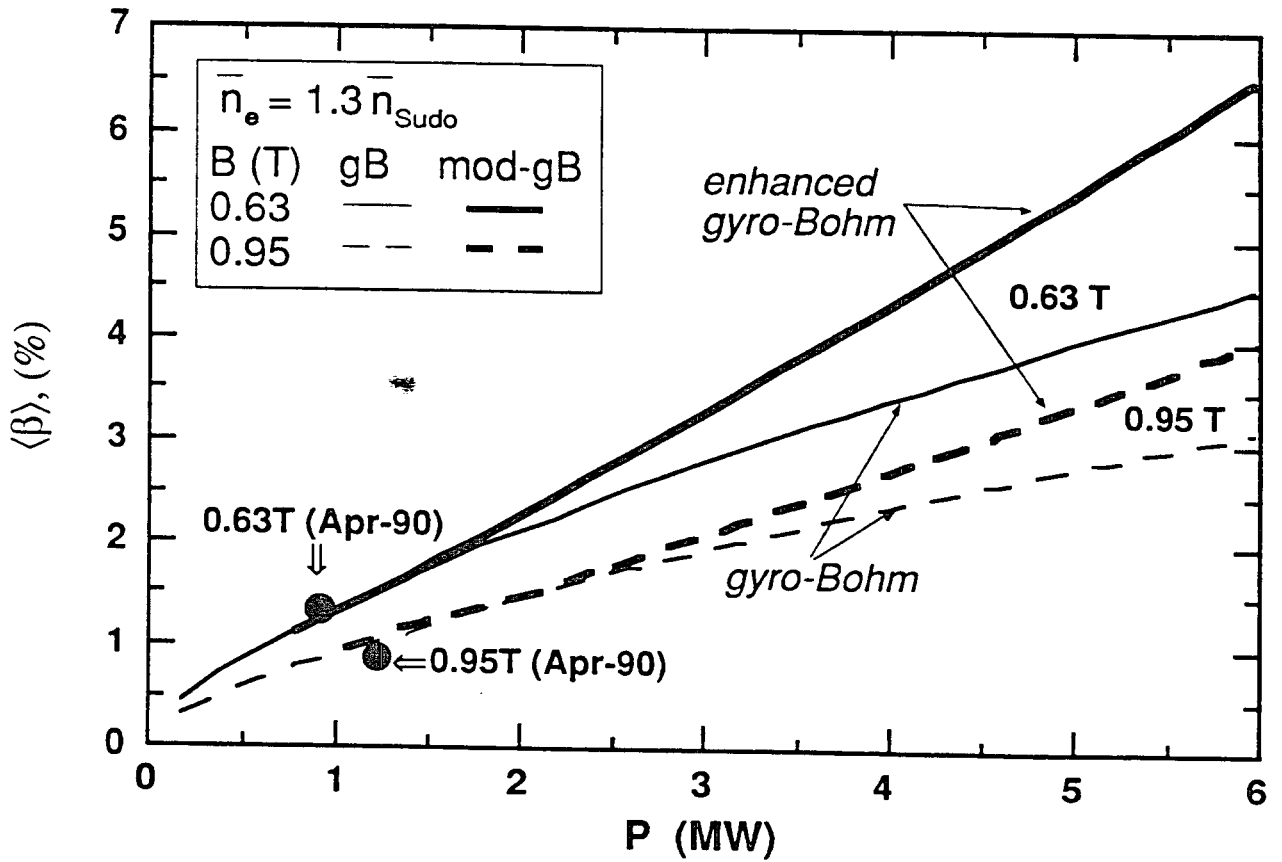
Fig. II.4. Expected plasma β in ATF as a function of heating power for gyro-Bohm confinement (thin lines) and for predicted enhanced confinement $\tau_E/\tau_{grB} \propto \beta^{0.3} v^{-0.2}$ (thick lines). The solid points show achieved experimental values.

-35-
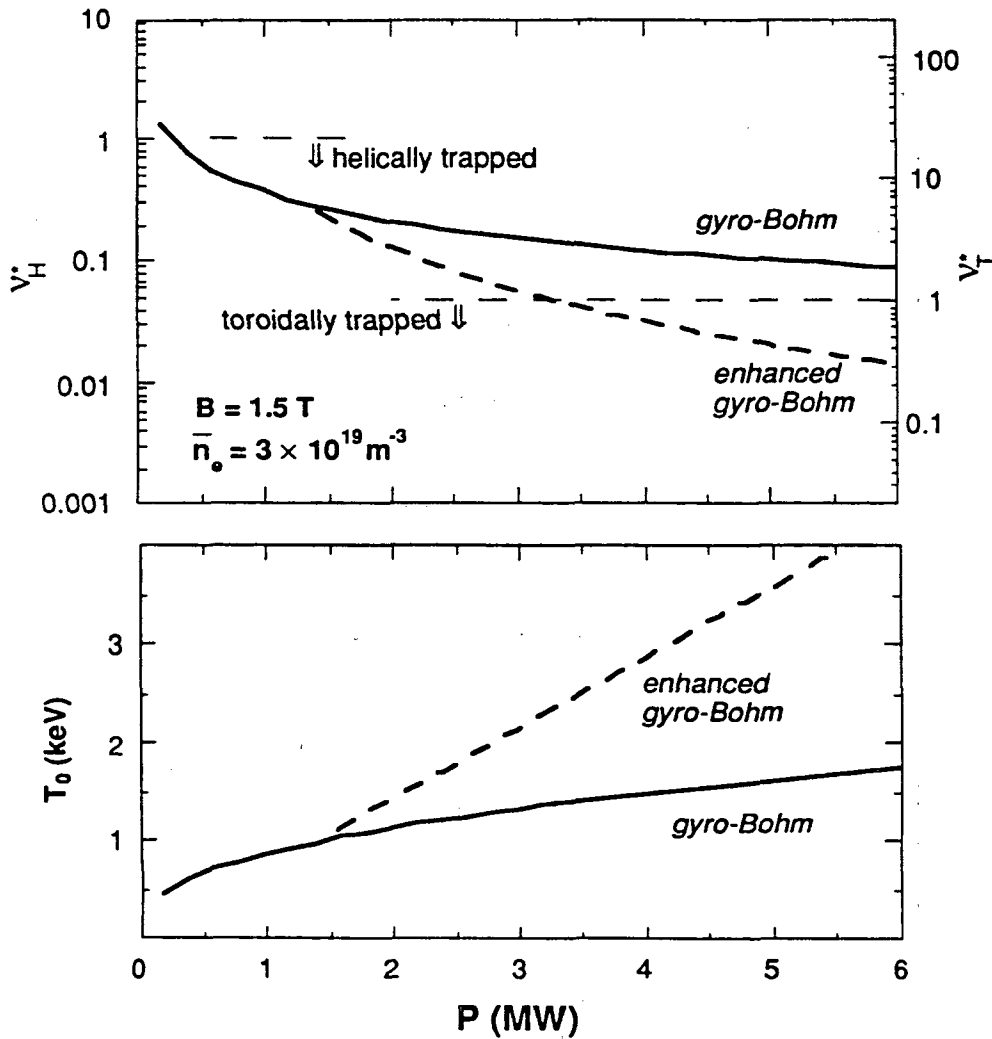
Fig. II.5. Expected central temperature and electron collisionality for helically trapped ($\nu_H{}^*$) and toroidally trapped ($\nu_T{}^*$) particles as a function of heating power in ATF for gyro-Bohm and enhanced confinement.
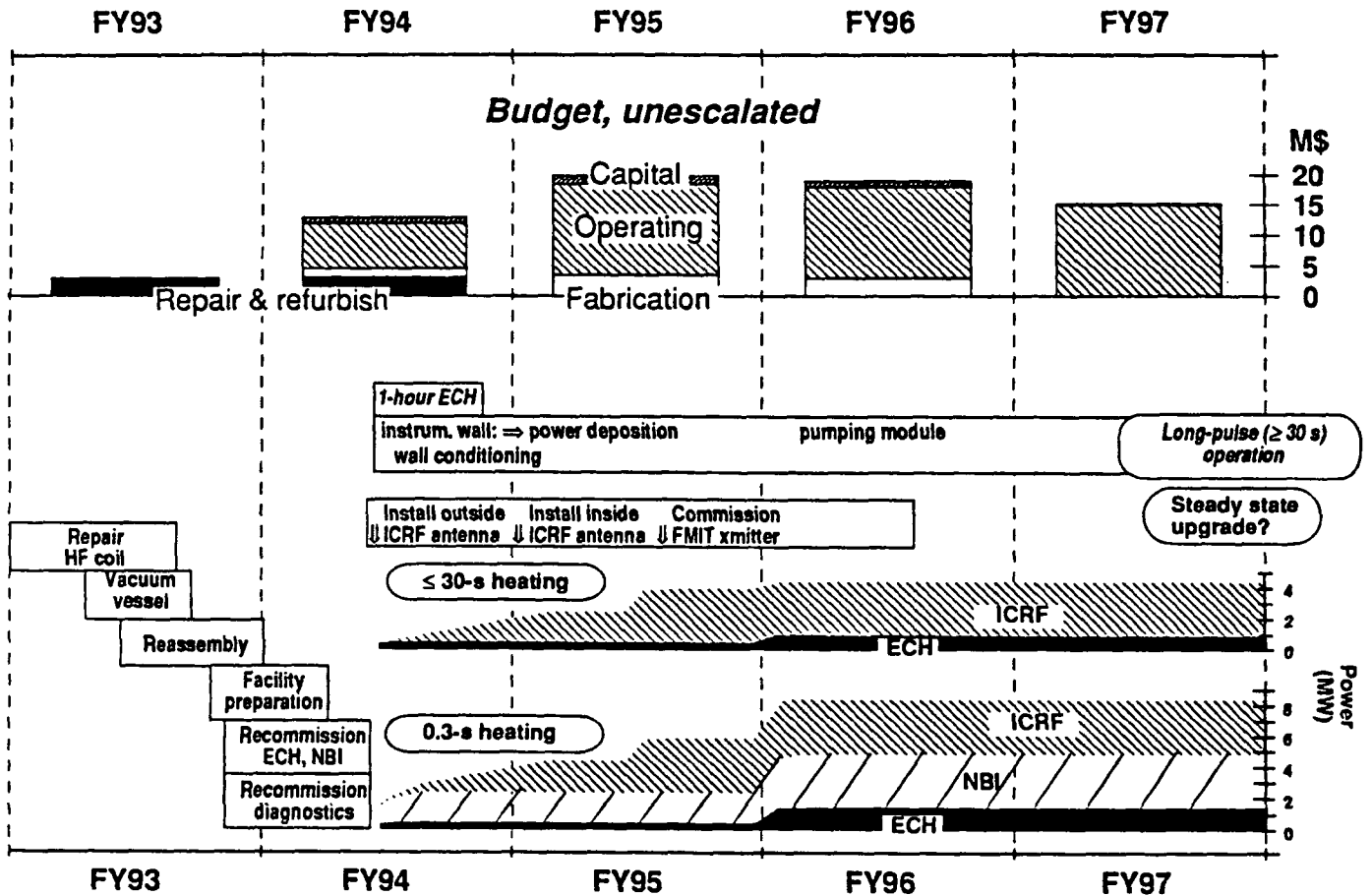
Fig. II.6. ATF budget and schedule, FY 1993–FY 1997

	FY93	FY94	FY95	FY96	FY97	Total
Repair & Refurbish	2.7	3.2	--	--	--	5.9
Fabrication	0.3	1.4	3.5	3.0	--	8.3
Operations	--	7.4	15.0	15.0	15.0	52.4
Capital	--	1.0	1.5	1.0	0.2	3.7
Total	3.0	13.0	20.0	19.0	15.2	70.2
Total (accelerated)	7.2	19.4	20.0	17.0	7.6	71.2

Table II.3. ATF budget required to meet objectives
(in M$, unescalated)

III.
DIII-D

A.   Summary

The DIII-D program proposes to address most of the key issues now emphasized in tokamak fusion research: high-beta, high-confinement operation, noninductive current drive with detailed profile control, enhanced pulse length, and elaborate divertor application and testing. In other words, it intends to explore "advanced" tokamak operation in a serious and extensive manner. A distinctive feature of the program is its intent to attack these technical challenges in an integrated fashion, eventually applying the various operational advances simultaneously. The point is not only to take advantage of synergism, but also to provide the experimental information most pertinent to future, larger tokamak devices.

These are obviously ambitious plans that cannot be presumed to succeed. Yet it is notable that, in a nonintegrated and sometimes transient fashion, many of the planned achievements have been approached or even exceeded in previous DIII-D experiments: high beta and current profile control are two prominent examples. Indeed, the widely admired DIII-D accomplishments make its program plan appear well grounded.

B.   Background and Overview

DIII-D, the fifth generation of magnetic fusion experiments operated by General Atomics, is the largest non-circular tokamak in the US. It is operated with extensive national and international collaboration, including a long standing Japanese collaboration and multi-disciplinary collaborations with, among other institutions, LLNL, ORNL, and UCLA. The DIII-D program is being called upon to provide R&D results to the engineering designs of the International Thermonuclear Engineering Reactor (ITER) and the proposed toroidal plasma experiment (TPX). The DIII-D tokamak program is now demonstrating, transiently, advanced tokamak operation in the form of high confinement modes, second stability and large bootstrap current operation. Specifically, DIII-D has achieved:

- high performance plasma operation with plasma betas exceeding those required for a reactor;
- high-quality plasma confinement regimes (VH-mode, high inductance);
- increased first stability regime limits and tokamak operation in the second stability regime;

- radio frequency heating and current drive that could provide for reactor current profile control and steady-state operation;
- divertor heat load reduction and methods of impurity control that bear on steady-state reactor operation.

The DIII-D long range research program focuses on divertor and advanced tokamak issues. Its goal is to provide an *integrated* demonstration of well-confined high-beta plasma with non-inductive current drive and effective divertor operation. Implementation of this plan would obviously bear on ITER R&D issues, as well as possibly opening new ITER operating options. In addition, it should identify engineering design concepts pertinent to an electrical demonstration plant (DEMO).

The DIII-D experimental plan and the schedule for implementation of new research tools is indicated are Fig. III.1. The current and proposed parameters are shown in Fig. III.2.

The proposed DIII-D program would use theoretical and technology advances to extend the tokamak experimental physics database. Results from experiments at DIII-D and other tokamaks indicate that plasma and divertor performance can be increased beyond the baseline engineering design of ITER. A simultaneous demonstration of such improved tokamak plasma and divertor operation could open a path to an attractive DEMO, provide new operating options for ITER, and establish an advanced physics foundation for the TPX program.

C. DIII-D Technical Objective and Goals

The DIII-D objectives and goals are based on extensions of DIII-D transient results. They can, and should, be tested with enhancements to the DIII-D tokamak capability, especially with regard to an extension of plasma duration beyond current profile and particle/wall time constants. To assure that the near-term research results will benefit ITER and TPX, it is planned to conduct experiments in an integrated fashion. That is, the various advanced operation features are to be demonstrated simultaneously, at sufficiently high temperatures to obtain reactor-relevant values of the appropriate dimensionless plasma parameters.

The above issues provide the motivation for the DIII-D program objective: to carry out an integrated, disruption-free long-pulse demonstration of a well-confined, high-beta plasma. Confinement will be based on the simultaneous use of noninductive current drive and an advanced plasma and divertor configuration. The reference scenario has an

ITER-relevant quantitative goal: to sustain a 2 MA plasma with 5% beta for 10 sec. This goal is a step toward a steady-state tokamak with improved divertor operation.

Quantitative DIII–D advanced tokamak goals are guided by the ARIES reactor study, which has quantified tokamak performance parameters that could lead to an attractive reactor concept. These parameters—confinement quality, stability factor, divertor peak heat-flux reduction factor, and bootstrap current fraction—serve as targets for DIII–D advanced tokamak goals, and provide a basis to chart technical progress. Target parameters for DIII–D research are specified in Table III.1. The goal is to simultaneously achieve and sustain such parameters in DIII–D for times longer than the tokamak current relaxation time.

Present results and understanding indicate that this goal can be achieved with upgrades to the DIII-D current drive and tokamak capabilities.

D.   DIII-D Research Plan

The DIII–D tokamak is well suited to carry out the challenging divertor and advanced tokamak research outlined above. DIII–D has operated reliably, in part because it is relatively modern: in 1986 it was outfitted with a new vacuum vessel, new outer poloidal field coils, and 5 sec pulse-length neutral beams. The DIII–D poloidal magnetic field system can generate a wide range of plasma shapes. A digital control system allows these shapes to be modified in real time, giving efficient access to a broad range of experiments. The neutral beam system operates reliably with power levels up to 20 MW. The DIII–D magnetic field is limited to 2.1 T, but the magnet set provides good access for heating and diagnostic systems. The plasma diagnostic set is among the best in the world, particularly in the plasma edge and divertor region. Thus, the existing DIII–D facility is well suited for the proposed research and complements research being carried out at other tokamak facilities.

Improved divertor and advanced tokamak results have already been obtained from DIII–D, making prospects for the proposed research appear promising. Advanced divertor experiments, which have demonstrated high pressures in the divertor plenum chamber, bode well for particle control with divertor pumping. Simple gas puffing in the present advanced divertor region has transiently reduced peak divertor heat loads by a factor of five from an ITER-relevant level of 5 MW/m^2. Hence, it seems plausible that a ten-fold reduction could be sustained with a more optimally designed radiative divertor

configuration. Fast wave electron heating and current drive experiments at the 1 MW level, as well as 110 GHz ECH microwave heating experiments at the 0.3 MW level, show efficient power coupling. These results all bode well for the proposed divertor and advanced tokamak experiments at higher power levels.

The objective of the proposed research in advanced tokamak development is to establish advanced operation through significant improvements in both the stability and confinement limits, using localized rf current profile control, rf and neutral beam heating for pressure profile control, as well as control of plasma rotation and of the plasma boundary conditions. There are 3 goals: (1) building on recent experimental results, achieved under transient and dynamic conditions, which show advanced tokamak modes (including VH–mode, high inductance H–mode, high second stable core, and second stable high bootstrap configurations); (2) experimental validation of the physics of rf current drive and efficiency optimization, in order to develop and prove the tools needed for advanced tokamak optimization; (3) combination of these efforts for active control and optimization of the advanced tokamak modes, leading to a demonstration of fully noninductive, high beta, actively controlled operation. The proposed rf systems (8 MW of fast wave power plus 10 MW of ECH/LH microwave power) will provide the local deposition and control needed for this research program. With the DIII–D neutral beam system, the rf systems will provide full noninductive operation for 10 sec at the target 2 MA, 5% beta condition. In addition, the current profile control systems will provide the flexibility to optimize advanced tokamak configurations and carry out fundamental tokamak physics research.

The proposed divertor program concentrates on divertor power and particle control under conditions relevant to ITER and future devices. The initial divertor research program will be carried out with the existing advanced divertor configuration that includes particle pumping and electrical bias capability. Experiments will concentrate on hydrogen pumping, helium removal, impurity control, edge plasma, and divertor physics. Divertor predictive code modeling and supporting theory will be benchmarked with detailed diagnostic measurements. This should provide a reliable model for design of future divertors. In 1995, a new configuration, called a radiative divertor, will be installed. It is designed to disperse the divertor power over a broader surface area, reducing peak divertor power flux by up to an order of magnitude when compared to present conventional divertors. The first radiative divertor will be passively cooled and operate for up to 10 sec. A later actively cooled configuration will allow experiments up to 60

sec duration. It is assumed here that ITER would pay for that upgrade. The rf power levels and plasma conditions for the 2 MA, 5% beta, 10 sec integrated demonstration will also provide a relevant test of divertor performance.

The experimental plan is shown in Fig. III.3. Each year's operating campaign is characterized by simple physics objectives that will provide key information to next-generation tokamaks (ITER, TPX, DEMO, Reactor). The cross-hatched periods indicate the periods of research operations. Figures III.4 and III.5 show the experimental programs and research objectives for the Advanced Tokamak and Divertor Development programs. These objectives will be carried out with close coupling between theories, experiments, and technology development to advance magnetic fusion research in such critical areas as transport, stability, plasma-wall interaction, toroidal Alfvén eigenmode suppression, divertor physics, rf heating and current drive, and disruption control.

E. Advanced Tokamak Research

The DIII-D tokamak operates with thermonuclear performance approaching that of larger tokamaks with many of the dimensionless plasma parameters equaling or exceeding those required for ITER and the ARIES power reactor. However, these parameters are not currently achieved simultaneously. Therefore, the goal of the future DIII-D high power rf and advanced divertor programs is to achieve many of these parameters simultaneously, and thereby demonstrate an integrated physics-engineering solution to ITER divertor, disruption, and current drive issues.

A long-time thrust of the General Atomics fusion program has been aimed at production of high beta plasmas and the development of theoretical understanding of plasma stability. A measure of beta enhancement is called normalized beta $\beta_n=\beta/(I/aB)$, which increases as the current profile is peaked.

The highest beta values achieved at DIII-D exceed those anticipated for ITER. In these 11% beta discharges, the central beta is 44% and the plasma center is calculated to be in the second stability region, with the outer edge at the first stability limit. Future experiments using high power rf current drive to control the current profile will aim at broadening the central high beta second stability regime and maintaining this second stability configuration for longer duration.

Using the neutral beam system, which injects up to 20 MW in one toroidal direction for neutral beam current drive, DIII-D plasmas that enter the second stability regime do so over most of the plasma radius. Up to 0.4 MA of non-inductive current drive has been driven and poloidal betas of 5.2 have been reached.

The high power density DIII-D neutral beam system (1 MW/m^3) is also well-suited to investigate toroidal Alfvén eigenmodes (TAE modes). These dangerous modes are predicted to be driven unstable by fusion alpha particles or high energy beam ions whose speed exceeds the Alfvén speed. Operating DIII-D at reduced magnetic fields decreases the plasma Alfvén speed to that of the 80 keV deuterium ions injected by the neutral beams. These beam ions thereby simulate alpha particles. When the beam ion beta reaches a few percent, TAE-mode magnetic fluctuations at 20 to 100 kHz are excited, beam ions are ejected, and the thermonuclear fusion power production is reduced. This provides an excellent opportunity to develop and demonstrate engineering design solutions to stabilize TAE-modes. Theory predicts that stabilization can be implemented on DIII-D and ITER by controlling the plasma current profile, the plasma shape, the fast ion pressure profile, and the density profile. DIII-D already has plasma shape control and is developing rf systems for plasma current profile control.

Significant progress has also been made in understanding plasma transport and developing enhanced confinement modes. High performance VH- or H-modes are regularly achieved and studied at DIII-D. The joint ASDEX/DIII-D JET H-mode study has found that the plasma H-mode thermal energy confinement time increases proportionally with plasma current, with major radius, and decreases with the square root of heating power. DIII-D has shown that the confinement improvement in H-mode arises from shear in the E x B driven edge plasma rotation suppressing turbulence. Tools to control plasma rotation and to further increase confinement improvements are being considered.

While the H-mode is considered the standard operating mode for ITER, recently DIII-D has operated in the very high performance, or VH-mode, with confinement that is nearly twice that of H-mode. The VH-mode was discovered in DIII-D following boronization. It is characterized by a broader edge temperature pedestal. The DIII-D VH-mode results in two times higher thermonuclear triple product (nTτ) than their H-mode discharges. Otherwise many of the VH-mode characteristics such as current scaling and fluctuations are similar to H-mode.

Development of localized rf heating and current profile controls to provide new tools for heating, controlling, optimizing, and sustaining plasma stability and confinement is a key element of the DIII-D program. Recent DIII-D results (as well as those from TFTR and JET) have indicated the important role that the plasma current density profile plays in plasma performance as measured by normalized beta or normalized plasma confinement. A centrally peaked current profile is better than the broad current profile that is normally generated with inductively driven ohmic current drive.

Four advanced tokamak operating modes have been generated in DIII-D by control of the current profile. The pulsed methods employed in the current DIII-D experiments are limited in duration by relaxation of the current profile to the inductively-driven profile. Current profile control using rf waves can be employed to sustain the advanced tokamak configurations. Estimates of the required rf power levels are in the range of 10 to 20 MW.

DIII-D is developing two innovative and attractive rf heating and current drive technologies; 110 GHz microwave electron cyclotron heating (ECH) and 30-120 MHz radio frequency fast wave heating and current drive. The aim is to develop a physics and engineering database for technologically-attractive and cost-effective systems for ITER and as well as the tokamak DEMO reactor.

A 1.6 MW 60 GHz ECH system is being used for basic physics studies, such as modulated transient transport, sawtooth and edge localized mode (ELM) stabilization, plasma pre-ionization and startup, ECH current drive, and electron heating to aid fast wave current drive experiments. The ECH-driven currents are consistent with theory and extrapolate to a current drive efficiency of 0.2 to 0.3×10^{20} A/W m^2 for ITER conditions. Future DIII-D ECH experiments at 10 MW ECH power levels and higher frequency are expected to drive MA-level currents at higher densities.

The next step in the DIII-D ECH program is construction of a 2 MW 110 GHz prototype system using four 0.5 MW gyrotrons. Currently, one 0.5 MW system is under test at DIII-D .

The second element of the DIII-D rf current drive and profile control program uses a 2 MW fast wave system, operating at the ion cyclotron resonance frequency (ICRF). ICRF ion heating has long been recognized as efficient and effective, especially in large

tokamaks. A second important application of this available rf technology is being pioneered in DIII-D, fast wave electron heating and current drive. The antenna is designed to launch waves at a frequency for which no ion cyclotron resonances exist in the tokamak rather than directly coupling to plasma ions. In this case, the fast waves heat electrons by Landau damping and transit-time magnetic pumping. Fast waves are the chosen current drive techniques for ARIES, and are identified as an alternative current drive scheme for ITER.

Fast waves have long been recognized as attractive from an engineering viewpoint, albeit lacking a physics database. DIII-D is now establishing this database using a 2 MW 30-60 MHz system and a four-element phased array directional antenna. These experiments are succeeding because DIII-D has sufficiently high electron temperature (boosted by ECH) for the fast waves to be absorbed by the electrons. These new results now open an additional engineering option for an ITER current drive system. In contrast to neutral beam current drive, less technology development is required and fast waves do not excite TAE-modes as do high energy ion beams.

The DIII-D fast wave current drive antenna was designed and fabricated by ORNL. It has been extremely effective at heating electrons with heating efficiency equal to neutral beam heating. In DIII-D experiments, H-mode plasmas are easily produced by fast wave heating alone. Fast wave current drive experiments have yielded promising results. Directional antenna phasing has been demonstrated and changes in sawtooth behavior are as expected. Loop voltage drops are consistent with 0.1 MA of driven current. Future fast wave experiments will be conducted at higher fast wave power with the addition of a second 2 MW transmitter and with increased electron heating power from the 2 MW 110 GHz ECH system. On a longer time scale, the fast wave power will be increased to 8 MW for advanced tokamak and divertor research.

F.   Divertor Development

Presently the ITER divertor is patterned after the DIII-D divertor configuration. While satisfactory for DIII-D, the design is inadequate for ITER power handling. To address these shortcomings, a DIII-D advanced divertor program is underway.

Two techniques to reduce the divertor power flux have been demonstrated on DIII-D. The first is sweeping the divertor strike point to reduce the time average peak divertor heat flux. The second involves injection of gas into the divertor. This technique has

reduced the peak heat flux by a factor of five, with little degradation of plasma performance.

The present DIII-D advanced divertor consists of a continuous toroidal ring that can be electrically biased to enhance plasma flow into its entrance slit. The baffled divertor chamber will house a cryopump, to be operational in early 1993. Divertor pumping will allow DIII-D to control the density of H-mode plasmas, operate at higher temperatures where rf non-inductive current drive is more efficient, operate with a more severe and realistic divertor environment, and carry out H-mode helium transport studies.

Plasma erosion of divertor materials is already measurable in DIII-D and is feared to be critical to ITER. The Divertor Materials Exposure System (DIMES) being implemented on DIII-D will allow in-situ experiments with various materials to be tested in ITER-like conditions, including time-varying power pulses from edge localized modes and disruptions.

The present DIII-D advanced divertor was built to maximize divertor flexibility while minimizing the impact on the flexibility or capability of DIII-D. As a next step, a radiative divertor will be implemented. This concept, demonstrated in Doublet III, aims at ITER and reactor relevance by radiating the power to reduce divertor heat loads and by decreasing the electron temperature to reduce the energy of particles striking the divertor plates. A demonstration in a high power current driven tokamak would provide relevant design data for ITER and DEMO. The radiative divertor will provide radiative heat dispersal, impurity entrainment, fuel recycling control and pumping, helium exhaust, and materials qualification. With adequate diagnostics, code development, modeling, and benchmarking, DIII-D would become an integrated test bed for ITER divertor development.

The ultimate step in the present DIII-D divertor program plan is installation of an actively cooled divertor. It is assumed that ITER would fund this step.

F.1   FY 92-95 Program

During the period FY 92 - 95 the DIII-D research program will have completed:
- Transient studies of advanced tokamak operating regimes (VH-mode, peaked-current profile, high $e b_p$, second stability.)

- Validation of current drive physics of fast wave and electron cyclotron with medium power.
- Initiated active control and optimization of advanced tokamak configurations.
- Completed the advanced divertor program demonstrating deuterium and helium pumping, plasma density and impurity control.
- Developed and benchmarked improved theories and models of divertor and advanced tokamak operation.
- Completed construction of a radiative divertor.

F.2   Outlook for FY 96 - 98

During the period FY 96 - 98 the DIII-D research program will have completed:
- Completed demonstrations of active control and optimization of advanced tokamak configurations.
- Demonstrated high performance non-inducting sustained plasma (5% beta at 2 MA for 10 sec.)
- Optimized non-inductively driven advanced tokamak operating modes.
- Completed radiative divertor physics development and initiated actively cooled divertor operation.

G.   Upgrade and Modifications Required

Completion of the DIII-D FY 95 - 98 program will require several upgrades and modifications. In cost estimating, DIII-D assumed that the base program would continue at the current level. This will allow the present ongoing projects to be completed, aging components to be replaced for modifications to diagnostics, modernization of the computer system, and establishing a remote experimental site. These are essential regardless of any upgrades or modifications. These expenditures together with research operating costs consume the base budget. To add new capability requires additional funds. The DIII-D cost estimates are provided in FY 92 dollars. Here, it should be noted that a continuation of the base program at the current funding level implies that the present 18 week (per year) run schedule will be maintained. While the recent DIII-D advances in plasma control and improvements in diagnostics and data acquisition have permitted them to make optimal use of the available machine time, it is clear that answers to the important physics and technology issues being addressed by DIII-D could be provided on a greatly accelerated time scale if additional run time is provided. When queried on this issue, the DIII-D senior staff responded that a 36 week schedule would result in a doubling of the productivity and research output while still permitting the

required maintenance, upgrades and experimental planning to be conducted. Concerning cost, the addition of a few additional weeks would require approximately $300 k/week. An expansion to the full schedule would necessitate the hiring of additional staff resulting in a yearly cost of $15 M. One view that was emphasized by the DIII-D staff (and shared by the Reviewers) is that the proposed upgrades should *not* be compromised or deferred in order to provide increased machine time.

G.1 Radiative Divertor Installation

Demonstration of ITER and DEMO relevant divertor power dispersal and reduction in the incident plasma temperature requires installation of a radiative divertor. This passively cooled modification would incorporate particle baffling and pumping as well as electrical bias. A high triangularity double null configuration appropriate for advanced tokamak performance is under consideration. Implementation will be as a joint LLNL/GA effort incorporating other advanced divertor program participants. The total estimated cost is $16.2 M, including costs of relocating existing diagnostics.

G.2 Fast Wave Power

The upgrade of the fast wave power for advanced tokamak central current drive and increased divertor power loading will necessitate the addition of two 2 MW commercial rf transmitters (30-120 MHz). The total estimated installed cost, including transmission lines, and tuners, is $6.5 M. To increase the pulse duration capability of the present 4-element 2 sec ORNL antenna to a 6-element 10-20 sec antenna would cost an estimated $2.8 M. To increase the pulse length to 60 sec will require water cooled Faraday shields estimated to cost $4.6 M.

G.3 Increased Microwave Power

To increase the microwave power for advanced tokamak current profile control to 10 MW will require installing ten 1 MW 110 GHz gyrotrons that are currently under development at Varian. This is an upgrade of the original 60 GHz gyrotron system comprised of ten 0.2 MW. Currently, four units are being modified for 0.5 MW 110 GHz gyrotrons. DIII-D is now examining the possibility of using 3 MW of lower hybrid power and 6 MW of ECH gyrotron power. A decision point occurs at the end of FY 95 as to the advisability of adding more ECH power or adding lower hybrid power. The total estimated cost ranges from $11.5 M to $25.4 M, depending on the mix of ECH and LHH and the power level. The final decision will take account of cost, physics issues,

and the state of the technology development (primarily a gyrotron issue). Currently, DIII-D is performing a detailed study of the various options.

G.4 Diagnostics

New diagnostics are essential for the radiative divertor and advanced tokamak programs. The estimated costs are $2.0 M and $3.0 M, respectively.

G.5 Tokamak Long Pulse Upgrade

Several tokamak subsystems need to be modified in order to increase the tokamak pulse length to 10-15 sec at 2 MA plasma current and 2T magnetic field. This upgrade will also allow 60 sec operation at half current and field. This pulse duration will equal several current diffusion time constants for advanced tokamak studies and the 60 sec pulse length will equal several particle/wall time constants. Needed are additional poloidal field power supplies, improved magnet cabling, utility transformers, and increased water cooling capacity. The total estimated cost is $7.1 M for 10 sec and $1.4 M additional for 60 sec.

G.6 Neutral Beam Long Pulse Extension

To increase the neutral beam system pulse length beyond 3.5 sec at 20 MW or 5 sec at 16 MW required upgrades of the beamline internal heat handling, water cooling, and power supplies. Upgrading the neutral beam system to operate at 20 MW for 10 sec will cost $8.1 M, and $3.8 M additional for 60 seconds.

G.7 Radiation Shielding

Additional neutron shielding is required to carry out the envisioned active long pulse divertor and high performance advanced tokamak research program. Currently, DIII-D is enclosed in a shield that reduces the radiation site-boundary radiation by a factor 300. It is proposed to add an additional factor of five, to raise the neutron shielding factor to 1500. Based on experience with the installation of the present shield, DIII-D estimates the additional shielding will cost $3.5 M.

G.8 Building Modifications

General Atomics will provide the building expansion required to accommodate the improvement to the DIII-D capability described above.

H.   DIII-D Special Features
Unique Capabilities of DIII-D

H.1   The largest Operating US Tokamak in the Post-TFTR Era
The DIII-D divertor and advanced tokamak upgrades together with the strong national collaborative program would provide the US with an internationally competitive magnetic fusion tokamak facility until the operation of TPX and ITER.

H.2   Plasma Shape
The large number of individually controlled poloidal field coils has allowed studies of plasma shape (elongation and triangularity with indentation to be explored) on confinement and stability. This capability has also been exploited by positioning small plasmas within different regions of the neutral beam injection footprint.

H.3   Variable Aspect Ratio
The flexible DIII-D magnetic coil set together with the large vacuum system allows a range of aspect ratios $2.5 < A < 5.3$ to be studied. This feature has been exploited for confinement studies and could be used for advanced tokamak second stability studies. Most experiments are performed at small aspect ratio with strong shaping.

H.4   Digital Plasma Control
DIII-D possesses the world's most advanced operating real time digital plasma control system. Already plasma shape, neutral beam power, and fast wave antenna loading are under active control. In addition, more than one experiment can now be performed on a single tokamak discharge. Future work will include active current profile control and disruption avoidance and control.

H.5   Biased and Pumped Divertor
The electrically biased divertor has been effective for particle control, divertor strike point variation, non-inductive startup, and H-mode threshold control. Installation of cryopump is to be completed in January 1993 for density control.

H.6   ECH Power
The present 2 MW 60 GHz ECH system has been effective in electron heating, sawtooth and ELM suppression, non-inductive startup, electron heating, modulated transport studies, heat pinch and transport studies, H-mode studies, current drive and profile

control. The 110 GHz upgrade to 10 MW will extend these studies to higher density, temperature and plasma current regimes. Pressure and current profile control will be applied to sustain and optimize advanced tokamak configurations.

H.7 Fast Wave Heating and Current Drive

The DIII-D fast wave ICRF system has pioneered direct electron heating and current drive. This work is in collaboration with ORNL who built the 4-element phased antenna. Experiments at the 1 MW level have shown heating efficiency equal to that of neutral beams and have resulted in driven currents of 0.1 MA. This technique looks attractive for TPX and needs to be demonstrated at power levels well above Ohmic.

H.8 Diagnostics

A diagnostic system with exceptional edge and divertor diagnostics is operational. Particularly noteworthy is the multi-pulse, multi-laser Thomson scattering system, the high-resolution, high-speed ion profile charge exchange recombination system, the multi-point motional Stark effect current profile instrument and five fluctuation diagnostics systems.

H.9 Collaborators

DIII-D is a highly collaborative effort. Half the scientific staff are from institutions other than GA. They participate in program planning, experiment development and leadership, and in communicating the results. The most long standing collaboration is with JAERI. The three major US partners are LLNL, ORNL, and UCLA. Collaborating institutions at DIII-D include: International Laboratories (JAERI, JET, ASDEX, TEXTOR, Tore Supra, T-10, TSP, and Compass), National Laboratories (LLNL, ORNL, SNLA, SNLL, ANL, PPPL, INEL), and Universities (UCLA, UCSD, UCI, UCB, MIT, RPI, Cal Tech, Johns Hopkins, U. Md, U. Illinois., U. Paris, U. Washington, and U. Wisconsin).

I. Uniqueness of Contributions of DIII-D

DIII-D is dedicated to developing critical information pertinent to ITER, TPX and a commercially attractive tokamak reactor, through an integrated program of advanced tokamak and divertor operation. Its extensive poloidal field coils and digital plasma control system provide exceptional flexibility to address a wide range of research topics. Its advanced divertor and current profile control capability is unique: DIII-D is the only divertor tokamak with ECH and fast wave current drive capability. The high power neutral beam system allows TAE-mode studies to be carried out with shape and current

profile control. The vacuum and magnet system, together with unique diagnostics, provide conspicuous opportunities for plasma disruption and mitigation studies.

I.1   Place in the World Program

DIII-D is the only device in the world proposing to simultaneously tackle the divertor and advanced tokamak mission in an integrated manner. Simultaneous demonstrations will be of major benefit to commercial fusion development; it is not clear that addressing these issues separately offers more than academic interest.

DIII-D has been the pacesetter of ITER design through its active contributions to the ITER Physics R&D. Because many elements of ITER are patterned after DIII-D, results from DIII-D are directly applicable to ITER.

DIII-D is also exploring various operating modes of TPX: improved confinement, second-stability, efficient steady state current drive and profile control, divertor power dispersal and plasma cooling.

Two additional points deserve emphasis:

- The US has no other machine offering DIII-D's capability in advanced tokamak and divertor operation;
- DIII-D can seriously compete with JET and JT-60U on the world front, in spite of its smaller size and cost.

J.   DIII-D Limitations

The current and proposed machine parameters of DIII-D are given in Fig. III.2. Its size and current are roughly half that of JET and JT-60U. However, its flexibility allows innovative confinement, stability, current drive and divertor experiments to be conducted. The main limitation is the present low rf power levels (2 MW ECH and 2 MW fast wave power), which are inadequate for a tokamak the size of DIII-D. Because these levels are comparable to the ohmic heating levels, rf experiments, while of scientific interest, are not sufficiently definitive for extrapolation to ITER or TPX; nor are they adequate to sustain advanced tokamak configuration or to stress the divertor.

Another important DIII-D limitation is that the pulse length is at present limited to 3.5 to 5.0 seconds, although 10 second low plasma current (0.7 MA) and low power (3 MW)

experiments are possible. The limited pulse length thus limits DIII-D divertor studies; the proposed upgrades will fix this.

At the present time the outer wall is only partially graphite-covered, resulting in some limitations on plasma operating mode and plasma purity. The outer wall will be covered with graphite tiles in early FY 93.

K. DIII-D Current Status

Operational Status (as of 7/7/92)

K.1 Pre-FY 93 Operation

The DIII-D is a flexible tokamak facility powered with neutral beams and rf systems capable of addressing a wide range of physics and engineering research issues. DIII-D began operation in 1986 and has had modest increases in capability in the past seven years.

Graphite tiles provide armor on the inside wall, floor, and ceiling. The outer wall presently is mainly armored with Inconel tiles. In early FY 93, the outside wall will be armored with graphite to reduce metallic impurities in the DIII-D plasma. Recent experiments have used boronization wall conditioning, resulting in a fivefold reduction of carbon and oxygen impurities and a thirty-fold reduction in metallic impurities.

In order to develop techniques for active disruption avoidance, magnetic error compensation coils and a high speed (0.1 msec) digital control system, to feedback control currents in plasma shape and position magnets, are being implemented.

The DIII-D tokamak has been operating typically 17 to 26 weeks per year since 1986. In FY 92, without reprogramming, DIII-D operations have been curtailed to 14 weeks. Experiments run were in support of the long range DIII-D program goal as well as in support of next generation tokamaks. Physics experiments include: transport, H-mode physics, high-beta, rf heating and current drive, divertor, advanced tokamak development, TAE mode control, disruption control, and operational issues of importance to ITER and TPX. These experiments are carried out using the 20 MW NBI, 2 MW FWCD, and 2 MW ECH heating systems, the 400° C bakeable vacuum vessel, and the advanced divertor configuration, with real-time digital plasma control, intershot boronization and helium glow discharge wall conditioning.

The first 0.5 MW 110 GHz gyrotron system is undergoing testing at short pulse lengths (30 msec). Pulses injected into DIII-D show electron heating with the expected efficiency and at the expected location. Longer pulse (10 sec) gyrotrons are being developed at Varian. Sockets, magnets, transmission system, and antennas are all completed at DIII-D. These units are compatible with future 1 MW gyrotrons. Fabrication of mode converters is awaiting Varian completion of the production 0.5 MW gyrotron.

The advanced divertor baffle and bias system is operational and showing promising particle control results. A 1 m^2 helium cryopump will be installed in early FY 93.

The 2 MW fast wave system is operational and a tilted Faraday shield is being installed. A second 2 MW fast wave transmitter is being purchased and ORNL is building four element antennas to be installed in the 0 and 180 degree midplane ports, presently occupied by outboard limiters that will not be needed once the outer wall is armored with graphite in the beginning of FY 93.

The neutral beams are operating reliably, although some high voltage transformers need to be rebuilt, and the control system needs to be modernized. Minor beamline melting occurs but serious damage is being avoided by careful monitoring and control.

The JET pellet injector is being modified by ORNL for installation in Summer 1993.

A new 150 liter/hour helium refrigerator will become operational in early 1993 to provide helium for the neutral beamlines, advanced divertor, ECH gyrotrons, and pellet injector.

The computer systems are nearly saturated but handle the present data load (60 MBytes/slot). The hardware and software are becoming obsolete and therefore must be modernized. A remote experimental site is being developed by LLNL to open DIII-D up to off-site DIII-D users and to off-load data analysis from DIII-D computers

The diagnostic system is generally mature and operational. Preliminary data have been obtained from a Divertor Institute Materials Exposure System (DIMES) for divertor materials testing and a lithium beam emission measurement. These diagnostics funded by D&T and APP will become fully operational in FY 93.

K.2   Repairs Necessary

No repairs are necessary

L.   DIII-D Schedules and Budgets

The DIII-D program plan for 1992-98 is given in Fig. III.1. The more detailed divertor and advanced tokamak plans are given in Figs. III.4 and III.5. The projected budget estimates assume a $41 M budget in FY 93 for GA, LLNL and ORNL. The cost of upgrades and modifications are summarized in Table III.2 together with a funding profile that is consistent with the program plan schedule of Fig. III.3. These costs are in FY 92 dollars. The $10.1 M cost to increase the pulse duration to 60 seconds is indicated by the parentheses, and is not included in the $74.6 M total cost.

Figure III.1

DIII–D CAPABILITIES ALLOW A WIDE RANGE OF RESEARCH AND TECHNOLOGY ISSUES TO BE ADDRESSED
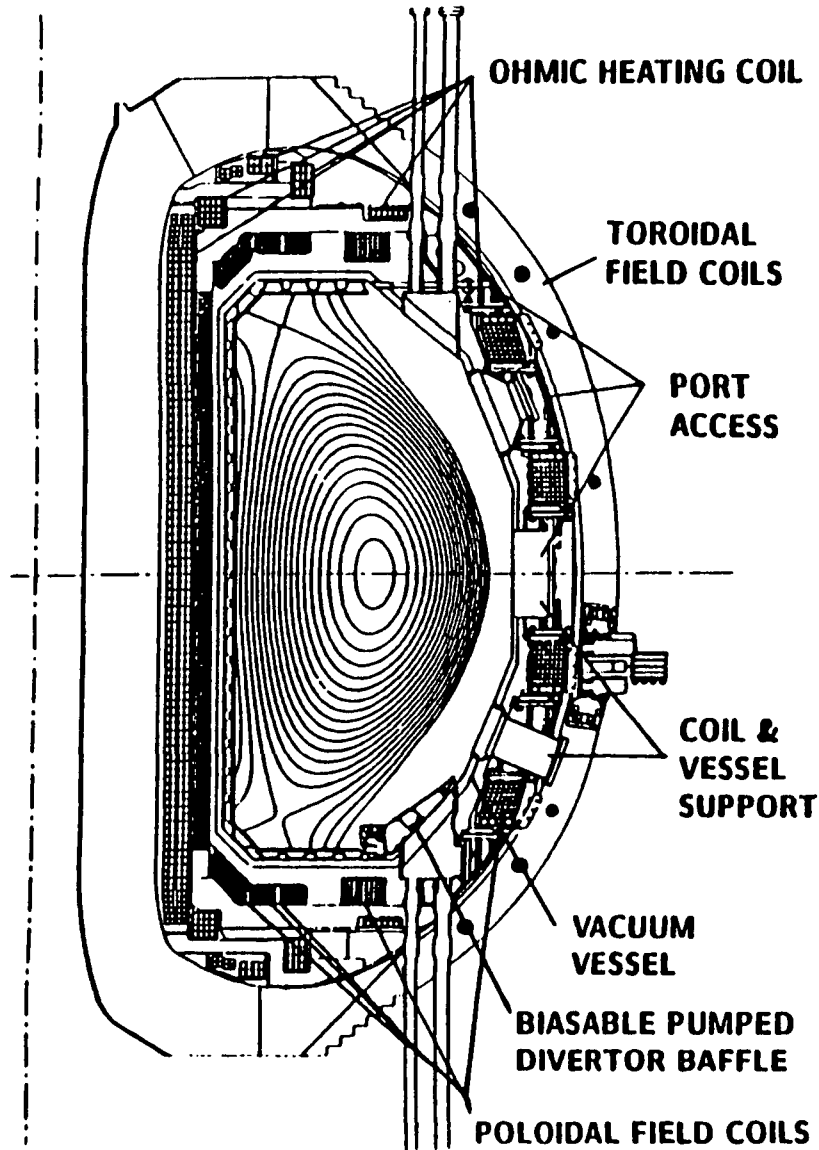
	Present	Proposed
Major radius	1.67 m	
Minor radius	0.67 m	
Maximum toroidal field	2.2 T	
Available OH flux	12 V-sec	
Maximum plasma current*	3.0 MA	3.5 MA
Neutral beam power (80 keV)	20 MW	28 MW
RF power (60 GHz)	2 MW	
RF power (110 GHz)		10 MW
RF power (30–120 MHz)	2 MW	10 MW
Current flattop (divertor at 2 MA)	5 sec	10 sec

* Divertor operation; Limiter operation of 3 MA achieved.

Labels on diagram: OHMIC HEATING COIL, TOROIDAL FIELD COILS, PORT ACCESS, COIL & VESSEL SUPPORT, VACUUM VESSEL, BIASABLE PUMPED DIVERTOR BAFFLE, POLOIDAL FIELD COILS

Figure III.2

Table III.1

TARGET PARAMETER GOALS FOR DIII–D ADVANCED TOKAMAK OPERATION

	ITER CDA Design	Best Achieved (Independently)	DIII–D Target (Simultaneously)
Confinement quality ($H = \tau_E/\tau_{ITER\text{-}89P}$)	2	3.4	4
Stability factor [$\beta_N = \beta/(I/aB)$]	3	6.5	6
Divertor plate heat load dispersal (reduction factor)	1	5	10
Bootstrap current fraction	0.3	0.5–0.8	0.5

GENERAL ATOMICS

DIII-D LONG RANGE PROGRAM OBJECTIVES

FY	93	94	95	96	97	98	99
CY	93	94	95	96	97	98	99

◆ Develop physics of Advanced Tokamak (AT) regimes (transient)

◆ Demonstrate particle and impurity control techniques (ADP)

◆ Establish configuration for Radiative Divertor

◆ Assess confinement and stability limits of AT plasmas

◆ Evaluate and optimize bootstrap current

◆ Validate FWCD physics and efficiency

◆ Validate second harmonic ECCD

◆ Demonstrate active current profile control

◆ Demonstrate sustained advanced tokamak configurations

◆ Initiate Radiative Divertor operation

◆ Initiate 8 MW FWCD operation

◆ Demonstrate noninductive p and j profile control

◆ Demonstrate divertor power dispersal

◆ Demonstrate AT operation with high current

◆ Demonstrate 2 MA, 5% β

◆ Demonstrate long-pulse power and particle control

Figure III.3

DIII-D DIVERTOR PROGRAM PLAN

FY	93	94	95	96	97	98	99
CY	93	94	95	96	97	98	99

ADVANCED DIVERTOR PROGRAM (ADP)

Test of techniques for future:
Blasing, Pumping, Particle Control
Broad Collaboration

- Cryo-pump
- Density control
- τ_E improvement
- Impurity entrainment
- Bias-enhanced pumping
- Erosion & redeposition
- Helium exhaust

Input from Improved codes - LEDGE, DEGAS, NEWT1-D, DIVIMP

Input from core physics - τ_E vs δ, SN, DN

Set plasma shape, divertor configuration, diagnostic modifications

RADIATIVE DIVERTOR

Geometry Optimization
Atomic Physics

- Radiative performance
- Code validation
- Pumping tests
- Bias tests
- Geometry change

2-D code development (incl. impurities)

design   construct

ACTIVELY COOLED DIVERTOR

ITER Technology
Long Pulse issues

ITER Team input

R&D input: Helium Cooling Development
Materials
Brazing, Clamping

design   construct

- Verify cooling scheme
- High heat loads
- Long pulse

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GENERAL ATOMICS

Figure III.4.

DIII-D ADVANCED TOKAMAK (AT) PROGRAM PLAN

FY	93	94	95	96	97	98	99
CY	93	94	95	96	97	98	99

TRANSIENT STUDIES OF IMPROVED REGIMES

VH-mode, Peaked-Current, High e bp, 2nd Stable + ...

- Shape dependence
- Explore 2nd stable regime
- Evaluate & optimize bootstrap current
- Assess confinement & stability limits of AT discharges

Understanding & Validation Phase

Demonstration Phase

ACTIVE CONTROL & OPTIMIZATION OF AT CONFIGURATIONS

Integration of stability, confinement, and current drive physics

- Use rf to modify p and j profiles
- Use external coils to control rotation
- Full noninductive operation at low current
- Disruption avoidance
- Optimization of scenarios with integrated active control

OPTIMIZED NONINDUCTIVE AT DISCHARGES

For DEMO design, ITER & TPX operation

- Demonstrate AT scenarios with fully noninductive current drive
- Evaluate improvements in stability and confinement relative to reference discharge
- Integrate & optimize AT scenarios

Demonstration Phase

VALIDATE CURRENT DRIVE PHYSICS

Medium power tests of FW & EC physics

- Validate FWCD physics
- Validate second harmonic ECCD physics
- Assess impact of trapped electrons on CD efficiency
- Evaluate bootstrap current interaction with rf
- Demonstrate density control

DEMONSTRATE HIGH PERFORMANCE PLASMA

High beta & high current reference discharge

- 2 MA plasma current
- 5% beta
- 10 seconds

Figure III.5

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Table III.2

GENERAL ATOMICS

COST SUMMARY FOR DIII–D UPGRADES IN FY92 $M

	Total Cost	FY93		FY94	FY95		FY96	FY97	FY98
Radiative divertor	16.2	2.0	*	7.0	7.2		–	–	–
Tokamak	7.1	1.0		2.0	2.0		2.1	–	(1.4)
Neutral beam	8.1	1.0		1.0	3.0		3.1	–	(3.8)
Fast wave	9.3	2.0		3.0	4.3		–	–	(4.9)
ECH	25.4	0		1.0	5.9	*	9.3	9.2	–
Diagnostics	5.0	1.0		1.0	1.0		2.0	0	–
Neutron shield	3.5	0		0	0		1.5	2.0	–
	74.6	7.0		15.0	23.4		18.0	11.2	(10.1)

( ) = cost to upgrade to 60 seconds.

* = decision point.

IV
PBX-M

A.   PBX-M Background and Overview

The PBX-M program is aimed at exploring two of the physics elements most important to making an attractive tokamak reactor: an attractive operating regime characterized by high beta, high confinement, and high self-driven currents; and avoidance of disruptions.

In their designs and experimental programs, PBX-M and its predecessor PBX have been strongly driven by theoretical considerations, esp., in focusing on the second MHD-stability regime as a potential reactor operating regime. Theory has suggested two approaches to second stability: (i) providing strong inboard cross-section indentation for $q_0 \sim 1$ (as in "bean shaping"); or (ii) employing non-indented configurations and holding $q_0 > 2$. The former approach would be more attractive from a reactor viewpoint in allowing higher beta, more bootstrap current and (possibly) reduction of certain transport-inducing microinstabilities. The latter approach would be attractive in requiring a simpler poloidal-magnetic coil arrangement. Having the capability to explore mixes of the two approaches, PBX-M is positioned to determine the optimum.

Suppression of kink modes becomes more difficult at higher beta. Internal kinks can be suppressed by increased triangularity, as in bean shaping. External kinks in PBX-M can be controlled for moderate time scales by its close-fitting conducting   ll. Longer-time external kink avoidance will require control of the current and pressure profiles, together with other active/passive techniques. For example, use the ponderomotive force of the ion Bernstein wave has been suggested and will be explored. Similarly, disruptions brought on by MHD activity should in principle be controllable by the same techniques of profile modification, possibly combined with active/passive means, although a detailed scenario has not been demonstrated.

Avoidance of MHD modes can be explored in PBX-M only for pulse lengths of 1-3 sec. (the latter with minor facility modifications). At best, these only approach the L/R time scale for current relaxation, or several (~ 5) times scales for profile adjustment.

In experimental program, the PBX-B has benefited from a number of productive collaborations with other institutions. Both in degree of these collaborative efforts and the degree to which the groups are integrated into the PBX-M team and its program planning,

PBX-M is prototypical of the "national-facility" type of organization anticipated for TPX/SSAT.

B.   PBX-M Mission, Technical Objectives and Results

The PBX-M mission focuses on the second stability regime and disruption-free operation. In the short term (FY 92 - 95), the goal is to explore stability, confinement, and self-driven current in the second stability regime. Capabilities to achieve these goals include current and pressure profile control, plasma shaping, passive stabilization, and high aspect ratio operation. The longer term (FY 96 - 98) program is disruption-free, second-stability operation for times longer than the profile relaxation time. Both missions would provide timely input into SSAT and other future advanced tokamaks. Below are presented results from pre-FY 92 and plans for FY 92 - 95 and FY 96 - 98.

B.1   Pre-FY 92 Operation

The original PBX performance was limited by external kink-driven disruptions, resulting in a maximum β_t of 5.5% and defining the basis for the ensuing upgrade, PBX-M. The major upgrade changes were to increase major radius to allow greater indentation (30% in PBX-M vs. 20% in PBX) and higher aspect ratio (from 4.7 to 5.5), to provide for a closed divertor region for impurity control and optimized H-mode operation, and to install a close-fitting conducting wall for stabilization of the external kink. Coarse current-profile modification was to be affected by current ramping, neutral beam injection, L-mode to H-mode transitions, and pellet injection. The first stage of the PBX-M program examined transient high-β_t equilibria with regard to internal modes and conducting shell stabilization of external kinks. Highlights of the pre-FY 92 operating period of PBX-M include:

- Achievement of second stability in the outer 40% of the plasma volume with $q_0 < 1$
- Internal kink (fishbone) suppression by shaping and current profile control
- Passive plate stabilization of the external kink at high β_t.
- Simultaneous achievement of high $\tau_E/\tau_{E-ITER-P}$ (=3-3.5) and $\beta_t/(I/aB)$ (=4-4.5)
- High value of $\beta_t(R/a) = 37\%$ for tokamaks, signifying a high level of stability at high current density (low q).
- Suppression of low-n MHD modes by pellet injection
- Control of q_0 with counter-neutral-beam injection.

During this period, however, plasma performance was limited by low-n internal modes at high β_t and by low-n external modes at high β_{pol}. Theory indicates that both limitations can

be overcome by pressure and current profile control, indicating the need to enhance these capabilities.

B.2   FY 92 - 95 Program

The primary objective of PBX-M for the FY 92 - 95 period remains the demonstration of the existence and accessibility of the second regime of stability to ideal ballooning modes in high pressure plasmas. To this end, current and pressure profile control will be used to obtain second stability across the entire plasma for a resistive skin time. A 2 MW lower hybrid current drive (LHCD) system will permit edge current drive and associated control of q_0. Fast wave current drive (FWCD) will be considered for further q_0 control. A several-MW ion Bernstein wave (IBW) system will extend β well beyond the first stability limit and provide ion pressure profile control. Localized electron Landau heating by IBW may also provide seed electrons to enhance LHCD localization and efficiency. Radial diffusion of fast electrons generated by LHCD will be studied.

Confinement will be studied (including possible improvement at second stability) with profile and fluctuation diagnostics. The profile modification tools will permit controllable variation of the scale lengths of the shear, density and temperature.

The flexible shaping and heating systems will permit study of the influence of energetic particles on stability. ITER-relevant issues for PBX-M include toroidal Alfvén eigenmodes, kinetic ballooning modes, and energetic particle stabilization of sawtee...

Innovative diagnostic development and comparison with theory rema... major elements of the program. Strong national and international collaboration will continue in all aspects, from planning to data analysis. Presently four Ph.D. thesis students and four pre-thesis students participate in the project. A national Program Advisory Committee advises on issues and priorities.

B.3   Outlook for FY 96 - 98

After preliminary assessment of second regime operation, experiments will focus on maintaining second regime and other high reactivity, low collisionality advanced tokamak plasmas for pulse lengths greater than several skin times. The shape can be optimized within one shot. For example, one approach would be to access second stability with $q_0 \sim 1$, utilizing indentation, but then to sustain $q_0 > 1$ second-stability in a non-indented cor.figuration, including single null.

Complete disruption avoidance will be a major goal. Techniques include an integrated passive shell/active coil feedback scheme, and fine current and pressure control on short time scales. Both coil and profile feedback requires development of an advanced active plasma control system, perhaps using a neural network. The long pulse length (~3 sec) is possible with the present facility, although LHCD and IBW systems must be upgraded. The increased duration will facilitate confinement and power handling studies, although the emphasis will remain on MHD issues.

C. Upgrades and modifications Required

Completion of the PBX-M FY 96 - 98 program will require the following upgrades and modifications in order to test long-pulse, disruption-free operation:

C.1 Long-Pulse Capability: Plasma Operation

Flattop currents up to 600 kA are presently accessible for up to 3 sec (the shortest plasma skin time, from the Mikkelsen '89 formula, is 0.3 - 0.5 sec). To study self-driven currents a low collisionality requires a toroidal field of 2 T, at a cost of $0.4 M for minor repairs of some motor generators.

C.2 Long-Pulse LHCD

To upgrade the 2 MW LHCD system from 0.5 sec to 2 sec (for a 3 sec plasma duration) requires component modifications and exploitation of additional PPPL power supplies at an estimated cost of $0.5 M. For increased flexibility in profile control an additional 2 MW is desired, costing $4 M using existing MIT/LLNL components or $7 M if completely new.

C.3 Long-Pulse NBI

Upgrading the NBI from 0.5 sec to 2 sec requires installation of active cooling of the power handling surfaces at an estimated cost of $5.0 M (including various ion source and instrumentation improvements).

C.4 Advanced Active Plasma Control System

The existing analog plasma control system is now being replaced by a fast digital system, modeled after a new DIII-D system. This will greatly increase the number of input signals used to analyze and control the plasma. Finer control of the plasma shape and divertor geometry will be possible. Further feedback control upgrades will use parameters

determined in real time to interface to profile control and stabilization hardware. This will guide the plasma along the path of stable operation. Total system cost is about $1 M.

C.5   Active/Passive Stabilizing Shell ("Power Shell")

The combined passive shell/active coil system will be used to control low-n external modes. This integrated approach has been successful for n=0 vertical position control. Estimated cost is $2.5 M.

C.6   IBW Upgrade

With the availability of the TFTR ICRF power supplies and sources, the total power of 14 MW will yield the highest RF power density in presently operating tokamaks. The pulse length capability is already 2 sec. The power can be used for current drive (FWCD, IBW-LHCD synergy, bootstrap current), pressure profile control (FW minority heating, IBW off-axis bulk ion heating) and plasma performance improvements (IBW mode stabilization and ICRF/IBW sawtooth stabilization). Central FWCD may reduce Ohmic electric fields to aid LHCD. Anti-current drive can be used to increase q_0. Fast wave minority heating can heat central electrons to increase FWCD efficiency. Antenna modifications for the increased power would cost approximately $1.0 M.

C.7   Divertor/Biasing Upgrade

A divertor upgrade is crucial to high performance in long-pulse, high-power discharges. For 3-sec. operation, real-time water cooling must be installed. Divertor pressurization ("gaseous divertor") could be applied to minimize divertor heat loads and impurity outflow. Special divertor electrode structures could be installed if suggested by present m=1 divertor bias experiments. The estimated divertor upgrade cost is $2 M.

C.8   Diagnostics

The long-term diagnostic emphasis is to improve spatial and temporal resolution and to develop techniques to support divertor physics and long-pulse operation. A tangentially-viewing hard x-ray imaging diagnostic has measured the suprathermal electron distribution during LHCD. A duplicate perpendicularly viewing system is proposed. A tangential CO_2 phase contrast imaging diagnostic is being prototyped on CDX-U to image the outer 40% of the plasma. It will be upgraded to cover the full radial (horizontal) extent of the plasma. More fast framing cameras will be acquired for fluctuation studies. Divertor diagnostics will be enhanced with more density and power loading measurement capabilities. New ECE systems will supplement the time-resolved electron temperature measurements with the

multi-pulse TVTS diagnostic during long pulses. The neutral probe beam pulse length will be extended for current profile measurements with the motional Stark effect system. Extra channels are planned for the beam emission spectroscopy and poloidal rotation diagnostics. More CAMAC hardware and workstations will be obtained to accommodate these enhancements. The total cost is about $2.8 M.

D. PBX-M Special Features
Unique Capabilities of the PBX-M Experiment
D.1 Dedicated Experiment
PBX-M has the dedicated mission of advanced-tokamak, second-stability studies. It can test different "trajectories" from first to second stability (e.g., low q_0/high-indentation to high q_0/no-indentation). Three new diagnostics are being developed to study the distribution and transport of fast electrons produced by LHCD.

D.2 High Plasma Indentation
PBX-M plasmas can be produced with inboard indentation of up to 30%, predicted to lead to enhanced MHD stability. Strong indentation should allow access to second stability with naturally occurring current densities ($q_0 = 1$), as opposed to the approach through strong current profile control with $q_0 > 2$. It has already been demonstrated that bean shaping suppresses low-n "fishbone" oscillations.

D.3 High Aspect Ratio
The PBX-M aspect ratio of 5.5 is the highest of presently operating tokamaks. High aspect ratio has two advantages with regard to second stability. First, the β value and power needed to reach second stability decreases with increasing R/a (β at R/a = 5.5 is about half of that at R/a = 3). Second, the unstable gap between the first and second regimes decreases with aspect ratio. High aspect ratio also yields improved confinement and stability in the first regime (see *Pre-FY 92 Operation*).

D.4 Lower Hybrid Current Drive and Profile Control
The LHCD system (2 MW @ 4.6 GHz) is the primary tool for current profile control. To do so, it is necessary to tailor the deposition profile of LH waves. This is achieved by a unique LH system which can change dynamically the phase velocity of the waves on a millisecond time scale. Each of 32 wave guides is phased independently, allowing a wide range of wavelength spectra. Experiments on time-varying phase velocities have already been carried

out in order to confirm the coupling predicted by theory. In contrast to JET results, recent experiments on PBX-M have not shown LH coupling to be affected by H-mode conditions.

D.5 Low-n Passive Stabilization

A 2.5 cm thick, aluminum, conducting shell surrounds 70% of the plasma surface and can be as close as a few cm ($r_w/a < 1.1$). In other tokamaks, the wall distances are typically $r_w/a > 1.5$. The growth times of disruption precursor modes have been observed to increase from 100 ms to 10-20 ms as the plasma surface is brought closer to the passive plates. The increase (to the L/R time of the shell for n=1 modes) is significant in that it offers the opportunity to control the current profile to suppress the modes within that time. Results with pellet injection for profile modification indicate success in suppressing low-n activity. The possible passive stabilization of ELMs (believed to be caused by external kinks) is also an important H-mode issue.

D.6 Diagnostics

Innovative diagnostics which have already originated in the PBX/PBX-M programs include charge exchange recombination spectroscopy (for ion temperature), a fast ion diagnostic (for high space and time resolved fast ion loss measurements), beam emission spectroscopy (for internal density fluctuations), and motional Stark effect polarimetry (for current profile measurements). For example, the 10-channel polarimetry diagnostic with 1 cm and 3 ms space and time resolution is sufficient for detailed comparisons between experiment and MHD theory. New diagnostics operating or under development include 2D-tangential hard x-ray imaging for suprathermal electron distribution measurements during LHCD, a fast reciprocating edge probe for fluctuations (with UCLA), 2D phase contrast imaging, reflectometry (with CIEMAT, Spain), third-harmonic ECE for ballooning mode fluctuations (with MIT), and two ECE diagnostics for suprathermal electron disruptions and transport. Several of these require the good tangential access available.

D.7 Passive Plate Biasing

In PBX-M, the electrically isolated, double-null divertor permits a potential drop from the outside to the inside edge of the plasma. Developed with UCLA, five electrically separated passive plates allow electric fields to be applied across different plasma regions to study effects on H-mode transitions, ELM control, second stability access, and transport of impurities into the divertor region.

D.8 Ion Bernstein Wave Heating

The present system is 2 MW at 40-80 MHz for 2 sec., upgradeable to 14 MW using the TFTR ICRF power supplies. Uses of IBW include bulk ion heating (without the fast tail produced by ICRF), off-axis ion heating for pressure profile control, electron pre-heating for increased LHCD efficiency, ponderomotive force stabilization of low-n edge modes, and core transport reduction through ponderomotive-induced velocity shear. Experiments on PBX are designed to overcome difficulties with IBW encountered elsewhere. Preliminary low power results indicate effective ion heating. Activity in IBW/ICRF antenna improvement includes boron-nitride-coated Faraday shields to reduce RF edge currents, and a folded waveguide launcher to reduce RF electric fields near the plasma edge. Conversion from IBW to ICRF (for heating or FWCD) requires a new antenna.

E. Uniqueness of PBX-M Contributions

PBX-M is a tokamak dedicated to, and uniquely positioned to focus on, the attainment of second stability through the combination of bean-shaping and current profile control. Although others are proposing second-stability studies employing the unindented, high-q_0 approach, no machine in the world today offers the PBX-M flexibility, especially in its exploration of extreme indentation and the q_0/triagularity tradeoff.

The PBX-M aspect ratio of 5.5 positions it outside the range of 2.5-3.0, common in today's tokamaks having similar performance. Higher aspect ratios have been identified as a promising improvement path for tokamaks, requiring lower current for the same confinement, possible relieved divertor heat load issues, and easier access to second stability.

The very flexible lower-hybrid system in PBX-M permits dynamic preprogrammed or, with modifications, fed-back control of the current drive power, e.g., to evolve the current profile with the equilibrium or, on a faster time scale, to respond to emerging MHD activity. As a technique for current drive, LHCD is thought to be limited in its reactor applications to surface-driven current, owing to issues of penetration. However, as a physics tool, LHCD provides localized power deposition and the means for determining the limitations to profile control inherent in possible transport of locally-driven current. These issues are critical to successful operation of the TPX/SSAT, as well as follow-on machines like a DEMO.

E.1 Place in the World Program

PBX-M's place in the world program derives from the uniqueness of its ability to address many of the issues of high beta and second stability. High-beta operation, especially in

second stability, has been identified in studies such as ARIES as necessary, or at least very desirable, for a reactor. The U.S. program has placed high priority on improvements such as these for the tokamak through its DIII-D program and its proposed TPX/SSAT program. Its timing in relation to TPX/SSAT and ITER are shown in Fig. IV.1.

F.   PBX-M Limitations

The current machine parameters of PBX-M are given in Table IV.1, together with planned improvements. Its current is modest in absolute value by larger tokamak standards. However, this value is believed to be enhanced by the aspect ratio in its effect on confinement. As a test-bed for steady-state, second-stability operation, PBX-M is limited primarily by its pulse length of 1-3 sec. (the longer times with some facility modifications). However, these pulse lengths are several times the current-profile relaxation time. Positive results from PBX-M would provide important information on second-stability access and hope for the prospects of even longer-time current-profile control, e.g., in TPX/SSAT.

G.   PBX-M Current Status

G.1   Operational status (as of 6/12/92)

The PBX-M device is operational for ~30 weeks/year and is performing experiments with the NBI, IBW, and LHCD heating and current drive systems. At present, there are no engineering issues for the field coils and busswork, poloidal field power supplies, and available motor generators (toroidal field supply).

The first MW of the LHCD power system is complete, and power levels exceeding 500 kW into the plasma have been achieved. The second MW of LHCD power is expected to be ready in December 1992. The procurement of the second phase shifter/power splitter will be in FY 92 (with reprogramming). The hard X-ray diagnostic, in support of the LHCD experiments, is functioning, and the foil drive mechanism, enabling energy discrimination of the hard X-ray signal, is operating. The full 2 MW capability for IBW is available, controls testing is complete, and development for the design of a "folded" waveguide is presently being performed in collaboration with ORNL. Two of the four neutral beams are operational; minor cryopanel repairs are in progress for the remaining two. The UCLA fast reciprocating probe is installed on the machine, all controls installation activities are complete, and the probe is in routine use. Dummy load testing of the UCLA biasing power supply was successful. Bias experiments will commence in late June.

G.2 Repairs necessary

Eight motor generators currently power the PBX-M TF system. An axially-connected set of four additional generators exists for the support of PBX-M, but requires electrical insulation restoration. The increase from eight to twelve generators would permit an increase in the TF from 1.6 to 2.0 T, and would cost approximately $0.4 M.

H. PBX-M schedules and budgets

The broad PBX-M research plan for 1992-98 is given in Fig. IV.2. The more detailed second-stability, enhanced-confinement and disruption-avoidance plans are given in Figs. IV.3-5. The projected budget profiles (in FY 92 $) are given in Table IV.2. When the budgets of the collaborators are included, the total FY 93 PBX-M budget was ~$15 M, to which can be added certain protected benefits from TFTR, e.g., relief from electric demand charges.

As shown in Table IV.2, the FY 92 level of funding must increase through FY 94 - 96 to cover the cost of these benefits when TFTR shuts down, to cover the addition of an assumed 10-12 TFTR physicists and their support staff, and to cover the ~$19 M costs associated with the planned upgrades. The profile given in Table IV.2 includes all of these costs.

PBX-M PROVIDES TIMELY INPUT FOR DESIGN AND OPTIMIZATION OF SSAT AND ITER

| 92 | 93 | 94 | 95 | 96 | 97 | 98 | 99 |

SSAT

DESIGN   PROCUREMENT & FABRICATION   ASSEMBLY   OPS

PROFILE CONTROL, ADVANCED REGIME OPERATION

DISRUPTION-FREE, LONG-PULSE ADAVANCED REGIME STABLE OPERATION

PBX-M

2ND STABILITY WITH PROFILE CONTROL FOR $\tau \sim \tau_{skin}$

CONTROLLED, DISRUPTION-FREE ADVANCED REGIME OPERATION WITH $\tau > 5\tau_{skin}$

H-MODE OPERATION WITH LHCD

DISRUPTION-FREE OPERATION, DIVERTOR ISSUES

ITER

ENGINEERING DESIGN   CONSTRUCTION

Figure IV.1

PBX-M Research Plan

	FY92	FY93	FY94	FY95	FY96	FY97	FY98

Goals

Profile Control: High-βp, High-β/ε for $\tau/\tau\text{-skin} \sim 1$

Confinement Enhancement

Disruption Avoidance by equilibrium-control

High-βp & High-β/ε for $\tau/\tau\text{-skin} \sim 1$

Initial studies of $\tau/\tau\text{-skin} \sim 5$

SSAT prototype and upgrade studies

High-β/ε and High-βp non-inductively driven plasmas for $\tau/\tau\text{-skin} \sim 5$

Disruption Avoidance by feedback

Tools

LH 1MW/ 0.5s   2MW/ 0.5s   2MW/ 2s   4MW / 2s

IBW/ICH 2MW/ 2s   3MW/ 2s   4MW/ 2s   7MW/ 2s   FWCD 4MW / 2s

NB 6MW/ 0.5s   NB 6MW / 2s

Advanced Control   Partial Helical Coil + Feedback

TF 1.0s   3.0s   Higher-power Divertor

Figure IV.2

PBX-M Machine Capabilities

	Present	FY95	FY97
R (m)	1.65	1.65	1.65
a(m)	0.30	0.30	0.30
I_p (MA)	≤0.6	≤0.6	≤0.6
B_T (T)	1.5	2.0	2.0
elongation	2.2	2.2	2.2
indentation	0.3	0.3	0.3
pulse length (sec)	1	3	3
P_{NBI} (MW)	6 (0.5 sec)	6 (0.5 sec)	6 (2 sec)
P_{LHCD} (MW)	1 (0.5 sec)	2 (2 sec)	4 (2 sec)
P_{IBW} (MW)	1 (2 sec)	4 (2 sec)	7 (2 sec)
P_{FWCD} (MW)	-	-	≤4 (2 sec)

Table IV.1

PBX-M FY-92/98 Budget Plan

(in $ M)

	FY92	FY93	FY94	FY95	FY96	FY97	FY98
Operations	9.2	10.7	11.3	12.3	10.8	12.6	12.6
Enhancements	2.9	2.3	4.8	6.3	7.3	2.3	0.4
Total	12.1	13.0	16.1	18.6	18.1	14.9	13.0

Table IV.2

Exploration of the Physics in the High β and High βp 2nd Stability Regime

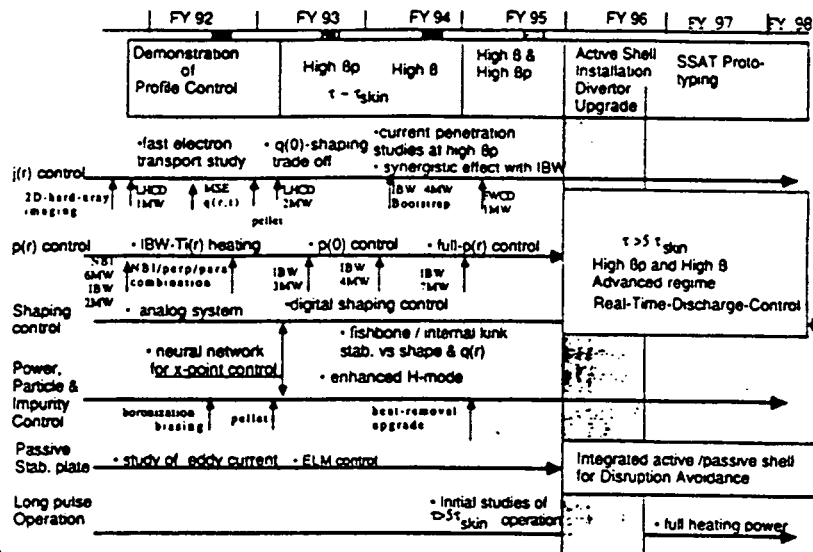|	FY 92	FY 93	FY 94	FY 95	FY 96	FY 97	FY 98

Demonstration of Profile Control | High βp   High β $\tau \sim \tau_{skin}$ | High β & High βp | Active Shell Installation Divertor Upgrade | SSAT Proto-typing

j(r) control
- fast electron transport study
- q(0)-shaping trade off
- current penetration studies at high βp
- synergistic effect with IBW

2D-hard-x-ray imaging — LHCD 1MW / IBW 2MW — MSE q(r,t) pellet — LHCD 2MW — IBW 4MW Bootstrap — FWCD 1MW

p(r) control
- IBW-Ti(r) heating
- p(0) control
- full-p(r) control

NBI 6MW IBW 2MW — NBI/perp/para combination — IBW 3MW — IBW 4MW — IBW

Shaping control
- analog system
- digital shaping control

Power, Particle & Impurity Control
- neural network for x-point control
- fishbone / internal kink stab. vs shape & q(r)
- enhanced H-mode

borosization biasing | pellet | heat-removal upgrade

$\tau > 5 \tau_{skin}$
High βp and High β
Advanced regime
Real-Time-Discharge-Control

Passive Stab. plate
- study of eddy current
- ELM control

Integrated active /passive shell for Disruption Avoidance

Long pulse Operation
- initial studies of $\tau > 5\tau_{skin}$ operation
- full heating power

Figure IV.3

Physics Studies Of Enhanced Confinement in PBX-M

	FY 92	FY 93	FY 94	FY 95	FY 96	FY 97	FY 98

Demonstration of Profile Control | High βp   High β $\tau \sim \tau_{skin}$ | High β & High βp | Active Shell Installation & Divertor Upgrade | SSAT Proto-typing

- studies of fast electron transport during LHCD
- IBW off-axis heating and confinement improvement
- transport studies (bias, high aspect ratio)
- q(0) control with pellet
- enhanced H-mode with shaping, q-control, biasing, ELM control
- q(r) control in high βp
- studies of high βp 2nd regime transport and turbulence
- bootstrap current with IBW
- transport studies at high β and high βp
- current control for $\tau > 5\tau_{skin}$
- assess FWCD with LHCD for full q(r) control

LHCD 1MW
2D-hard-x-ray imaging
IBW 2MW / MSE q(r,t)
p(0)-control
pellet 8 barrels
neural network for x-point control
pellet borosization
NBI/perp/para combination
IBW 3MW
digital control
LHCD 2MW
IBW 4MW / IBW 7MW
FWCD 1MW

$\tau > 5\tau_{skin}$
High βp High β
Advanced Regime
Real-Time-Discharge-Control

Integrated active/passive shell for Disruption Avoidance

Figure IV.4

Disruption Avoidance in PBX-M

	FY 92	FY 93	FY 94	FY 95	FY 96	FY 97	FY 98

Demonstration of Profile Control | High βp   High β $\tau \sim \tau_{skin}$ | High β & High βp | Active Shell Installation, Divertor Upgrade | SSAT Proto-typing

Passive Stabilizing Plate
- Stabilization by forced plasma rotation
- ELM control
- design of integrated passive/active shell

Plasma Control
- analog shaping system
- neural network for x-point control
- digital shaping control
- initiate Real-Time-Discharge-Control
- shaping control at high βp

Power, Particle, Impurity Control
- m=1 biasing
- boronization
- closed divertor and enhanced divertor action
- divertor upgrade

IBW 2MW LHCD 1MW | Biasing | LHCD 2MW | IBW 4MW | IBW 7MW

$\tau \sim 5\tau_{skin}$
High βp High β discharge
- Real-Time-Discharge-Control
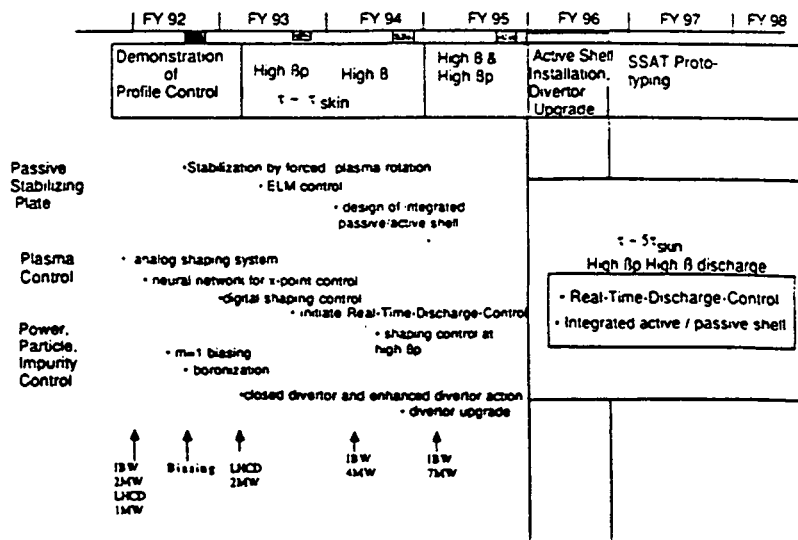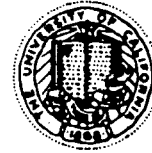- Integrated active / passive shell

Figure IV.5

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Dr. David Baldwin, LLNL
Dr. Harold Weitzner, NYU

June 8, 1992

Dear David and Harold:

Thank you for agreeing to be chair and vice-chair of FEAC Panel 4 on "Priorities in the Intermediate Confinement Experiments." Your report will provide important input to the FEAC workshop in July on priorities in the overall fusion program. In addition, it will assist the FEAC in reaching its specific recommendation in September on the operation of ATF.

The facilities in the toroidal program that you are asked to evaluate and prioritize are the ATF stellarator and the PBX, and C-Mod tokamaks. This should be done against the background of the DIII-D and TFTR capabilities, assuming that full D-T operation in TFTR beginning in mid-1993 and a strong DIII-D program are supported as recommended in the April 1 FEAC letter to Dr. Happer. As described below, I ask you to focus more on a factual evaluation for our July meeting, leaving for September a more complete determination of a basis for FEAC recommendations on priorities.

For the July meeting, please provide the following information for each of the identified mid-scale toroidal facilities:

1. The physics issues that are addressable in this class of facility and the completeness with which each of the identified devices can address these issues: and

2. For each device, the goals and objectives, additional hardware, the strengths, uniqueness, limitations, present status, projected costs and time required to achieve its objectives.

In addition, for the July meeting, please provide preliminary priorities and their time scale that your Panel would assign to the operation of these facilities, along with an indication of the reasoning behind these priorities.

At the July meeting, the full FEAC will make use of your evaluations and your draft priorities in its examination of the broader program. Later, in time for the September meeting, I would like your panel to reexamine its preliminary priorities in light of the FEAC's July workshop and feedback provided there. Further, this will provide an opportunity for your Panel to hear responses from the programs reviewed. Your revised priorities will then serve as input to the September meeting of FEAC. This two-step process will provide ample opportunity for each program to have a fair opportunity to answer questions and concerns.

When this process has been completed, the FEAC must answer the following questions:

1. If the fusion budget is sufficient to do so, do all of the facilities warrant operation? If not, which ones do not warrant operation?

2. If the fusion budget is not sufficient to operate simultaneously all the facilities which warrant operation,

   a) Should their operation be phased, implying one or more machines would be mothballed, and if so how?
   b) Should all be operated at a reduced level? or
   c) Should one or more be closed down, and if so in what priority order?

The combination of your evaluations and priorities should be sufficient to permit FEAC to respond to Dr. Happer's request concerning the ATF and other priorities. I understand that this will not be an easy undertaking for your Panel, for FEAC, or for the programs involved since all are staffed by high quality groups. I will do all that I can to assist you in this endeavor.

Sincerely,

Robert W. Conn

Appendix II

Minutes of FEAC Meeting of
September 22-23, 1992.

MINUTES

Meeting of Fusion Energy Advisory Committee
Ramada Renaissance Hotel
13869-71 Park Center Road
Herndon, VA 22071

September 22 - 23, 1992

Present:

Dr. Robert W. Conn, Chairman, UCLA
Dr. David E. Baldwin, LLNL
Dr. Klaus H. Berkner, LBL
Dr. Ronald C. Davidson, PPPL
Dr. Stephen O. Dean, Fusion Power Associates
Dr. Daniel A. Dreyfus, Gas Research Institute
Dr. John P. Holdren, UCB
Dr. Robert L. McCrory, Jr., University of Rochester
Dr. David O. Overskei, General Atomics
Dr. Ronald R. Parker, MIT
Dr. Barrett H. Ripin, NRL
Dr. Marshall N. Rosenbluth, UCSD
Dr. John Sheffield, ORNL
Dr. Richard E. Siemon, LANL
Dr. Peter Staudhammer, TRW, Inc.
Dr. Harold Weitzner, NYU

Tuesday, September 22, 1992

Welcome and Opening Remarks

Dr. Conn called the meeting to order and welcomed the committee members to the Ramada Renaissance Hotel at Herndon. He informed the committee that Dr. Richard E. Siemon's membership on FEAC, replacing Dr. Rulon Linford, had recently been approved by the Secretary of Energy. Dr Conn extended a warm welcome to Dr. Siemon on behalf of the committee.

Dr. Conn stated that the main purpose of the meeting was to work on, and finalize, the report that members had been preparing following the one-week meeting of the Panel 5 ad hoc committee at Crested Butte, Colorado, where potential future strategies for the U.S. magnetic fusion program had been reviewed within the framework of a variety of budget scenarios. Dr. Conn indicated that he had received a request from the Secretary of Energy Advisory Board (SEAB) to present to them FEAC's recommendations concerning the design and construction of TPX, for each of the budget scenarios that had been considered at Crested Butte. The meeting was scheduled for the morning of Thurs-

day, September 24, 1992, which was the day immediately following the projected close of the current FEAC meeting.

Dr. Conn informed the meeting that two new charge letters had been received by FEAC. One concerned undertaking a review of the Inertial Fusion Energy (IFE) program, and the other asked for advice on fusion-related materials and associated matters. Dr. Conn indicated that several members of the public had already made known their intentions to speak at the meeting during the time set aside for public comment, and suggested that others who wished to speak should contact the committee secretary.

Up-Date from DOE

Dr. N. Anne Davies announced that Dr. James F. Decker was unable to attend the meeting and that he sent his apologies to the committee. She stated that she would therefore make the entire presentation on behalf of the Department of Energy, and that she would cover three major activities, viz. TFTR, ITER and TPX.

TFTR D-T Program

Dr. Davies began by summarizing the TFTR timetable; she indicated that the Office of Energy Research intended using the TFTR D-T campaign as a test case upon which to base the way in which all fusion-related energy research should be handled. Dr. Davies informed the committee that the PPPL Operational Readiness Review for the 1000 Ci Test was in progress. A nine-member external review team had visitied PPPL and prepared a report on its findings. In turn, PPPL had developed an action plan aimed at implementing suggested improvements. DOE's own Operational Readiness Review for the 1000 Ci Test, which would be conducted by an eleven-member team, was scheduled for January 1993. Tests of the tritium handling systems, using small quantities of tritium, would begin after completion of the DOE Operational Readiness Review. All of the outstanding hardware required for D-T operation is scheduled to be installed by July 1993, and both PPPL's and DOE's Operational Research Reviews for full D-T operation were scheduled to be completed by September 1993. The TFTR D-T program is scheduled to start in September 1993, and to be completed by the end of September 1994.

ITER Activities

Dr. Davies reviewed the status of ITER. The International Agreement to proceed with the Engineering Design Activities had been signed, formally, on July 21, 1992. The first Council Meeting had been held early in September, when the nominations for Director and for Deputy Directors had been approved. The structure of the Joint Central Team had also been approved along with the chairmanship and membership of the major working committees. Dr. Davies presented viewgraphs that listed the key ITER personnel.

The rules of procedure that ITER will use for conducting business had been reviewed and approved. The size of the Joint Central Team had been discussed; this could number 150 persons by the end of Protocol I. The Council had requested that design and construction cost estimates be prepared by July 1993. The selection of the Joint Central Team division directors was planned for the end of September 1993, and the relocation of Joint Central Team personnel was already underway. Dr. Paul Rebut, Director of ITER, had requested that a magnet testing facility be established. Since such a facility had not previously been contemplated, the matter was being investigated by a technical panel.

Status of TPX

Dr. Davies reported that an ESAAB meeting had been held in July to approve the mission need (Key Decision #0). No decision had yet been made authorizing the start of the conceptual design of TPX. The Secretary of Energy, Admiral Watkins, had asked that a task force of SEAB review the matter before making a decision. The meeting was scheduled for September 24: FEAC had already heard that Dr. Conn would be addressing that meeting.

Dr. Davies stated that the TPX National Council had been formed in August and had already made recommendations to the Director of Princeton Plasma Physics Laboratory concerning basic design parameters and the limit that should be placed upon total project cost. The selection of a Project Director for TPX was in progress. Dr. Davies reported that the DOE had set a cost limit for the construction of TPX of about $400 million in FY92 dollars, expected to be equivalent to about $500 million in as-spent dollars. The DOE had also agreed that the project should start by focusing upon the conceptual design of a 2.25m, 3.35T/1.87MA steady-state advanced tokamak with superconducting magnetic coils. Early objectives are aimed at completing a report on the conceptual design, together with a detailed cost estimate, by March 1993. At that time it was intended that an international panel of experts should conduct a thorough design review.

Dr. Davies pointed out that the language used in the Congressional Appropriation Conference Committee's budget supports undertaking TPX conceptual design activities in FY 1993. She provided the committee with copies of the language: This is reproduced below.

CONFERENCE COMMITTEE REPORT
OF THE FISCAL YEAR 1993
ENERGY AND WATER DEVELOPMENT
APPROPRIATIONS BILL, H.R. 5373

SEPTEMBER 15, 1992

MAGNETIC FUSION

The conferees provide $339,710,000 for the magnetic fusion program. The conferees direct the Department of Energy to apply this reduction in a manner that is cost effective and least disruptive to the mission and priorities of the magnetic fusion program.

The conferees note with approval the recent agreement to proceed with the engineering design activity phase of the International Thermonuclear Experimental Reactor (ITER). The conferees provide funds to meet fully the U.S. commitment to ITER and direct the Department to provide a plan for selection of a

U.S. candidate site for future construction of ITER.

The conferees recognize the technical progress that has been achieved in magnetic fusion energy here and abroad. The conferees direct the Department to assure that ITER, D-T in TFTR, and the DIII-D program receive the highest funding priority within the magnetic fusion program. The conferees have also provided funds for design work leading to a steady-state advanced tokamak planned for operation as a national facility.

The FY 1993 Budget

Dr. Davies presented details of the initial budget proposed by the Office of Fusion Energy for FY 1993 and drew comparison with anticipated final expenditures for FY 1992. She pointed out that the FY 1993 numbers were still uncertain, for a number of reasons. First, the total fusion energy budget that DOE had worked with for FY 1993 was $335 million; she explained that for reasons of conservatism the lower of the original House and Senate mark-ups had been used. The conference committee had agreed upon the higher of the two figures, viz. $339.7 million, and the budget spread now needed adjusting to reflect this. Second, a new expenditure category termed "General Reduction/Reserve" had been established, which affected every division within the Office of Fusion Energy, and which amounted to $10.6 million in the budget presented; this reserve effectively reduced the funding available to programs by that amount. Dr. Davies explained that the "General Reduction/Reserve" figures represented estimates only and that since this was the first year of their inclusion, she did not know what the exact amounts would be.

Status of 1992 Reprogramming Request

Dr. Davies reviewed the status of the reprogramming request that had been made for FY 1992. Although the majority of the reprogramming had been approved, $7.6 million was still awaiting approval. Dr. Davies felt it was possible that this funding would yet be released.

Industrial Participation in Fusion

Dr. Bennett Miller presented a summary of his final report concerning industrial involvement in the fusion program. The report had been written as a "White Paper" in August, 1992 and was entitled "From Patronage to Partnership - Toward a New Industrial Policy for the Fusion Program".

For ease of comprehension, Dr. Miller indicated that he had broken his task up into four main sections: the purpose; the charter; the problem; and a proposal. "The purpose" was to focus attention on industrial involvement in fusion with a view to meeting the objectives called out in FEAC's letter of February 14, 1992 to DOE:

"The role of industry in the U.S. fusion program should be strengthened in order to prepare industry for the major ITER-construction tasks. The international competition in ITER will require the U.S. to develop a clear strategy for U.S. industry involvement. Such a strategy should take into account the different relationships between government and industry of the different ITER parties. As well, DOE procurement practices should be examined to assure a leadership role for U.S. industry.

To provide U.S. industry with the knowledge of fusion requirements and to secure the maximum benefit from industrial involvement, the DOE should develop a plan that deliberately includes a broader and more integral industrial participation in the fusion program. This plan should encourage the development in industry of both technical and programmatic expertise and should allow for the continuity of this expertise over the long term."

Dr. Miller indicated that "the charter" had permitted him to adopt an informal approach to the study. While he had sampled, reviewed and evaluated the opinions of very many experts, the conclusions contained in the white paper were non-consensual; the views that were expressed represented one person's opinion, his own.

Turning to "the problem", Dr. Miller pointed out that there was no government policy relating to industrial involvement in fusion. Further, there was no significant investor-owned or public utility interest in the fusion program. He stressed that the fusion program's potential customers were saying that they did not want the product. Worse, traditional vendors and suppliers to fusion activities were losing interest. Dr. Miller emphasized that the fusion program would have no future if it remained disconnected from its ultimate customers and its potential suppliers. He stated that industry must "become" the program and indicated that now was the time for the fusion program to change from a scientific one to an industrial one.

Dr. Miller pointed out that significant changes in the way that utilities did business would take place over the next ten years. Many cogeneration facilities were being constructed and many small energy producers

would enter the marketplace. Dr. Miller felt these changes may work to fusion's advantage. However, he emphasized that the practices of the past must not be repeated. Traditionally, when funding for fusion's core program had increased, industry's interest and involvement in the program had increased. Conversely, when the core program had shrunk, industry's interest had waned.

Dr. Miller explained the rationale for a new approach to the fusion program. He stated that fusion was feasible and emphasized that engineering, not science, was the important next step. He pointed out that industry must be involved as a full-fledged partner for this to succeed.

Dr. Miller reviewed the dominant traditional modes of government-industry interaction. He explained that he termed one mode the "vendor option"; this mode worked well when the government was purchasing existing products, which implied that there was a steady commercial market for them. The second mode he termed the "partnership option"; this mode worked well when a research program was needed in order to develop the required product. The partnership option could be commercially-linked or not. In the commercially-linked version, the government procured novel defense systems by having industry develop a product that DOD laboratories tested, and that was then refined and retested in an iterative process, at goverment expense, until a viable, economic product was developed that balanced the demands of defense with the realities of commercial manufacturing practices. In the non-commercial version, the government recognized that there would never be a ready market for the end product of the needed development program and, on a competitive basis, selected and made a long-term commitment to an industrial partner to develop products based upon a unique technology.

Dr. Miller concluded by describing "a proposal". He suggested that DOE should develop a policy towards industry that embodied some form of non-commercial partnership option as the next step in the fusion program. He stated that even though critics would object to the proposal, a number of features commended it:

- It presented an accepted mode of government-industry interaction, albeit not a very common one.
- It addressed industry's two major concerns: a major role in program definition; and a stable, predictable business environment.
- It would benefit the taxpayer in both the short and the long term. In the short term, it would ensure that real corporate commitments were made to fusion as a business

area. In the long term, it would ensure that the persons who must ultimately carry the technology into the commercial phase were the same ones who were planning its deployment.

Dr. Conn reminded the committee that the vehicle that FEAC had suggested to DOE for bringing industry back into fusion was ITER, via the U.S. Home Team's program. This would also prepare U.S. industry for the construction phase of ITER. Dr. Miller commented that while this approach was good, it still left program leadership and control in the hands of government and not with industry. Dr. McCrory stated that while he felt that the partnership approach was good, he felt that the timing was inappropriate: DEMO was too far into the future and commercial reactors were even further away. Dr. Miller responded that such sentiments became self-fulfilling prophesies. He conceded that a time delay of 40 years to DEMO might present a problem but suggested that a delay of 20 years would not. He pointed out that the actual timing would depend almost entirely upon the level of effort that was put into the program. He noted that another factor would soon come into play that might help matters: EPRI controls the energy market now, but it will not in 10 - 15 years' time.

Dr. Siemon asked if, in Dr. Miller's scenario, it was intended that universities and national laboratories be a part of the partnership. Dr. Miller responded affirmatively but added the qualification that he saw them being "little" partners. Dr. Siemon pointed out that since the technology currently resided at the universities and national laboratories, they should play a large role during the transition period. Dr. Miller agreed with this suggestion.

Dr. Dreyfus emphasized that large construction companies, large equipment vendors, and large utilities must be involved in future fusion developments since these were the entities that were going to construct, equip and operate the fusion facilities. He stressed that the new legislation relating to public utilities would not result in the formation of a large number of "mom and pop" energy generation concerns. Dr. Overskei suggested that the committtee should clarify in their minds who or what the actual customer for their product was likely to be. He emphasized that making the assumption that the utilities would be the customer was incorrect. Rather it was the high-technology companies that would be the customers for fusion technology.

Dr. Baldwin returned to the question of timing, stating that although components associated with the ITER project needed to be developed now, there would be a

long period between ITER and the next fusion machine. He questioned what would happen to the transfer program in the meantime and, worse, what would happen even earlier if ITER itself did not get built. He stated that if the commitment to transfer was made at the wrong time, and if the U.S. fusion program was unable to continue to supply support over a significant period of time, then the transfer program would fail. Dr. Miller agreed that starting the transfer program now involved risk, but it was his opinion that it was the right thing to do.

Dr. McCrory referred to his earlier question and stated that while he agreed that one could get industry involved and interested in the fusion program by changing the anticipated time-scale, there was far more to the problem than that. The industries that would be involved in fusion were those that were involved in fission. However, whereas the fission/fossil fuel trade-offs were well known, it was not even known whether fusion would proceed to commercial reality. A government policy was needed that endorsed fusion as a future energy source. Dr. Overskei stressed that the issue of commitment was of paramount importance. The national laboratories had carried forward a commitment to fusion for years simply because they were paid to do so. Likewise, industry only does what it is paid to do. Industry will not carry fusion forward unless it is paid to do so.

Dr. Conn stated that there appeared to be great interest on the part of FEAC in trying to move the program in the direction that Dr. Miller had suggested but that committee members were concerned with minimizing the risk involved and controlling it on the "down side". He reiterated that he saw the opportunity as revolving around ITER. Dr. Staudhammer stressed that industry would not make a commitment to the program unless there were profits to be made. He emphasized that commitments and large investments were made not on feasibility but on perceived commercial value. The stumbling block impeding industry's entry into the fusion program was that, in the present economic environment, some existing participating group would have to give something up.

Dr. Parker stated that the question of identifying the true customer for fusion was of vital importance and should not be overlooked. The possibility existed that the fusion program could be aimed at satisfying the wrong customer and in so doing end up with the wrong product. Dr. Rosenbluth drew attention to the fact that, at ignition, alpha particle effects would become pronounced and that the program was entirely lacking in experience in this area. He added that work on low-activation materials had yet to show commercial feasibility. As a result, it was his opinion that now

would be the wrong time to start a transfer program to industry. Dr. Rosenbluth further stated that the approach suggested by FEAC, which would involve using ITER as the transfer vehicle, was the correct way to proceed at present.

Dr. Staudhammer said that he wished to make a statement for the record. He stated that he wished to compliment Dr. Miller for a very fine piece of work. He expressed concern that FEAC might leave the report unattended for a very long time. Dr. Staudhammer pointed out that the work had raised many questions to which no answers had been provided. He suggested that DOE generate a charge, possibly to FEAC, to look into the matter and provide some answers. Dr. Conn responded that Dr. Staudhammer's recommendation had been noted.

Dr. Dean pointed out that FEAC had made a recommendation to DOE that DOE provide a policy concerning industrial involvement in the fusion program. DOE had acted upon that recommendation and had asked Dr. Miller to evaluate the situation. In turn, Dr. Miller had provided a report on the matter. Dr. Dean stated that DOE now needed to take the input provided by Dr. Miller and by FEAC and draft their industrial policy. Dr. Conn agreed and suggested that DOE should come back to FEAC with its policy whenever it was ready.

Dr. Sheffield endorsed Dr. Staudhammer's remark that FEAC should not now discard Dr. Miller's report and do nothing about it.

Report from Panel 4

Dr. Baldwin reviewed the activities of Panel 4. He reminded FEAC that Panel 4 had been asked to provide input to Panel 5 at Crested Butte in July. Panel 4 was therefore forced to respond very rapidly to its charge. Also, the charge given to Panel 5 overlapped to some extent that of Panel 4, and this overlap had affected the manner in which Panel 4 had functioned.

Dr. Baldwin reminded the meeting that the charge to Panel 4 had been to evaluate and prioritize ATF, Alcator C-Mod, and PBX-M. Later in the process the charge had been expanded to include the DIII-D upgrade. The report that the panel had prepared contained background information on the four facilities and included program plans and programmatic roles, facility limitations, and projected costs. The panel's priorities had been presented at the FEAC workshop in Crested Butte as input to the strategic program planning exercise.

Dr. Baldwin stated that the problem that had emerged

for the panel when determining priorities, lay in the facts that all four facilities were staffed by excellent groups of scientists, and all four programs were addressing vitally important programmatic issues. Dr. Baldwin reviewed, briefly, the major features of each facility and the tasks each was particularly suited to address. He stated that, as was customary with FEAC panel reports, the recommendations of the panel had not been included in it: They had, instead, been prepared in viewgraph form for presentation at the current meeting. The viewgraph showed:

<div align="center">

Higher Priorities

ATF

DIII-D Upgrade

Lower Priorities

Alcator C-Mod

PBX-M

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Dr. Baldwin explained that the panel had placed ATF in the higher priority category since it felt it was important for the U.S. to maintain a sound position internationally in this technology.

Dr. Conn pointed out that essentially the same data had been presented to Panel 5 at Crested Butte. He suggested that FEAC accept the report and thanked the panel for a job well done. Dr. Dean asked how FEAC was intending to handle publication of this report. He pointed out that until recently, UCLA had been responsible for publishing them. Dr. Conn replied that DOE had expressed a desire to publish the FEAC reports themselves and in fact had already republished the first four as official DOE reports. Dr. Baldwin suggested that FEAC forward the Panel 4 report to Dr. Happer with a simple letter of transmission explaining that it had been used as background for Panel 5.

Dr. Anne Davies informed the committee that OFE had prepared a document on TPX for presentation at the SEAB Task Force meeting that was scheduled for the day following the end of this FEAC meeting. Copies were made available to committee members. Dr. Davies stated that the other main topic for the SEAB meeting was the advanced neutron source (ANS).

Change of Agenda

Dr. Conn stated that the outcome of Panel 5 would be a FEAC report and not a Panel 5 report. He indicated that much of the balance of the present meeting would be devoted to reviewing and modifying the draft docu-

ment that everyone had worked on since the end of the workshop. He pointed out that changes to the report would be made in "real time" and that it would be necessary to change the agenda in order to complete the process before the meeting ended.

Review of Charge to FEAC

Dr. Conn drew the committee's attention to the Letter of Charge of June 22, 1992 and reviewed the reasons why the letter had been written. He reminded the committee that during the last twelve months they had reviewed several important elements of the fusion program but never the program as a whole. FEAC had felt that it needed to ensure that the sum of the "parts" that it had proposed did not amount to more than the total program could afford, and so had suggested that DOE prepare a charge asking that FEAC undertake a review of the strategy underlying the entire magnetic fusion program. Dr. Conn briefly reviewed the letter itself.

Review of FEAC Program Strategy Report

Dr. Conn explained the order in which he proposed that FEAC should deal with the program strategy report. He suggested that the committee deal with the Executive Summary last. He indicated that Chapters 2 and 3 should be discussed and finalized first, since the material contained in these was essentially non-controversial. This discussion should be followed by one on Chapter 4, which related to program elements. Dr. Conn stated that Dr. Weitzner had agreed to lead the discussion on that chapter. Chapter 5, which contained the committee's recommendations, should be reviewed next; Dr. Conn said that Dr. Baldwin had agreed to lead that discussion. Finally, the discussion should return to the executive summary when every effort should be made to ensure that it was consistent with Chapter 5.

Throughout the remainder of the day, the committee reviewed as much of the report as time permitted.

Wednesday, September 23, 1992

Review of FEAC Program Strategy Report

Throughout the majority of the morning session, the committee continued to review the program strategy report but did not complete the task.

Public Comment

Dr. Bogdan C. Maglich, Advanced Physics Corporation, presented a brief paper aimed at ways of broadening support for fusion and significantly increasing OFE's

budget. He stated that the DOE's fusion program had no constituency among the nation's industrial, environmental or public interest groups that, logically, one would expect to be supportive of environmentally attractive power technologies. He suggested that the program should be modernized, broadened and increased through government-industry partnership projects and through the incorporation of smaller devices into the program that would permit hundreds of universities and R&D companies to undertake fusion research.

He stated that his company had examined the underlying causes behind the stagnation of the U.S fusion program by canvassing the attitude toward fusion of all the special interest groups that would be expected to support fusion research. They had reached the surprising conclusion that, almost universally, such groups either were not supportive of fusion or were directly opposed to it. Dr. Maglich quoted the following examples:

- The energy independence movement is indifferent to fusion because it believes that fusion will not replace oil. It estimates that oil consumption would be reduced by only 5% through fusion.

- The electric power industry has no interest in fusion since it believes that fusion's use of radioactive fuel will make it environmentally unacceptable. In addition, the industy believes that fusion will be excessively capital intensive and unlikely to be economically viable.

- The electric power equipment industry considers fusion unacceptable simply because it is a "nuclear" technology. Currently envisioned fusion power plants would replace coal-fired plants which provide about 50% of today's power generating capacity. The powerful coal lobby is intent on stopping fusion.

- The university community is opposed to the present approach of constructing multi-billion dollar devices since this policy effectively excludes all but a few from participation in fusion research.

- The responsible environmental movement, centered around the Union of Concerned Scientists, opposes fusion because it is "nuclear" and there is no exclusively peaceful nuclear energy: Any nuclear reactor can be converted for weapons use

or for breeding weapons fuel.

- The non-proliferation lobby opposes fusion because any D-T fueled reactor can be used to breed plutonium and tritium.

Dr. Maglich stated that much of the opposition to fusion would dissipate if a study were initiated to develop an environmentally attractive fusion power reactor that was based upon the use of non-radioactive fuels. He suggested that a tokamak that used ^{3}He-D as the fuel would make a good starting point.

Dr. Maglich suggested that a study be undertaken regarding how to apply the time-sharing principles agreed for the high-energy collider to the major tokamaks, thus to attract many more universities and small research groups into the fusion program. He also suggested that OFE hold a contest relating to new concepts for compact, non-radioactive fueled fusion systems, and that DOE actively promote a government-industry partnership program.

Dr. Sheffield pointed out that perhaps more universities were involved in fusion than one might think. He stated that 45 universities had participated in ORNL fusion projects in recent years. Dr. Conn agreed that replacing oil with fusion in power generation would have little effect on overall oil consumption, but pointed out that 70% of all oil was used in transportation. If transportation were to convert to electric traction power, then the potential effect of fusion on oil consumption would be much greater. Dr. Maglich accepted this point but emphasized that it might be coal, and not fusion, that would benefit from this change. He stressed that the coal industry and its allies comprised the main anti-fusion lobby.

Review of FEAC Program Strategy Report

During the early part of the afternoon session, the committee continued with its review of the program strategy report until it had completed Chapter 5. It was agreed that the report would be modified to reflect the changes that the committee had agreed upon and sent to members for final review before it was forwarded to the Director of Energy Research.

New Charges to FEAC

Dr. Conn informed the committee that FEAC had received two new charges from Dr. Happer. He proposed to establish two new panels, Panel 6 and Panel 7, to review these charges and to provide input to FEAC. The first charge, which would be reviewed by Panel 6 and for which a response was requested by February 1993, concerned the Neutron Intereactive

Materials Program of the Office of Fusion Energy. The second, which would be reviewed by Panel 7 and had a requested response date of April 1993, concerned the Inertial Fusion Energy Program of Energy Research. Dr. Conn added that he would like both panels to provide FEAC with an interim report on progress at their next meeting and suggested that a date in January might be suitable.

After some discussion, FEAC agreed that a better response to the charge concerning the materials program would result if it could be delayed until March 1993, provided that this delay would be acceptable to DOE.

Dr. Conn led a discussion concerning possible membership of the two panels and the availability of members of FEAC to serve on them.

Establishment of Panel 6

Dr. Conn suggested that Dr. Berkner be the chairman of Panel 6, which would review materials matters. Members of FEAC who will sit on Panel 6 were agreed as follows:

> Dr. Klaus Berkner, Chairman
> Dr. Richard Siemon
> Dr. Stephen Dean
> Dr. Harold Weitzner
> Dr. Marshall Rosenbluth
> Dr. John Holdren
> Dr. Peter Staudhammer

The committee agreed that Dr. Berkner should work with Dr. Conn to select and invite appropriate scientific experts to join the panel and expand its range of expertise.

Establishment of Panel 7

Before selection of the membership of this panel, a discussion took place on the breadth of the charge. Dr. Staudhammer pointed out that the panel might experience difficulty regarding access to classified material. The selection of the chairman was also reviewed, bearing in mind that conflict-of-interest issues involving the apportionment of funding between the magnetic and inertial fusion programs needed to be seen to be avoided. It was agreed that the matter of the chairman for this panel would be discussed with the inertial fusion program directors before any decision was made. The members of FEAC who will sit on Panel 7 were agreed as follows:

> Dr. Stephen Dean
> Dr. Ronald Davidson
> Dr. Barrett Ripin
> Dr. David Baldwin
> Dr. John Sheffield
> Dr. Robert McCrory
> Dr. David Overskei

The committee agreed that Dr. Conn should work with whomever was chosen as chairman to select and invite to join this panel appropriate scientific experts that would include representatives of the inertial fusion community.

Terrence A. Davies
IPFR/UCLA
October 12, 1992