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The Fusion Science Research Plan for the Major U.S. Tokamaks

Advisory Report
prepared by
Major Facilities Review Panel,
Scientific Issues Subcommittee,
Fusion Energy Advisory Committee

for submission to
Dr. Martha A. Krebs,
Director
Office of Energy Research
U.S. Department of Energy

May, 1996

May 21, 1996

Dr. Martha Krebs
Director
Office of Energy Research
U.S. Department of Energy
Washington, DC 20585

Dear Martha:

In your letter of March 25, 1996, you asked FEAC, now FESAC, to address issues related to the major facilities in the fusion energy sciences program of the Department. You specifically enclosed a charge to the committee. I in turn asked the FESAC Scientific Issues Subcommittee (Scicom) chaired by Prof. Jim Callen of the University of Wisconsin to address the charge and prepare a set of findings and recommendations which FESAC could transmit to you in response to your charge.

I transmit here on behalf of FESAC the letter from Scicom chair Jim Callen to me as the findings and recommendations to you. Also attached is the report prepared by a Scicom panel to inform the Scicom review process. The questions you asked in your Charge were given a very thoughtful, intensive, and extensive examination. The findings and recommendations are direct and to-the-point, and were arrived at through community-wide input and debate. It is also worth noting that the members of Scicom themselves have diverse scientific backgrounds and levels of experience in fusion research. The report describes for you the specific scientific objectives of the programs in each major fusion facility (TFTR, DIII-D, and Alcator C/Mod), and the letter and report both make clear the priority scientific issues each facility will address. I find the recommendations in the Scicom letter of Prof. Callen to be fully consistent with the restructured program and focus recommended by FEAC in its report to you of last January.

I trust you will find the recommendations to be responsive and helpful to the Department as we move forward to maintain a vigorous fusion energy sciences research program in the United States.

Sincerely,

Robert W. Conn
Chair, Fusion Energy Sciences
Advisory Committee

May 17, 1996

Dean R.W. Conn, Chair, FEAC
University of California, San Diego
Office of the Dean, School of Engineering
9500 Gilman Drive
La Jolla, CA 92093-0403

Dear Professor Conn:

You faxed to me in March a charge to the Fusion Energy Advisory Committee (FEAC) from the Department of Energy regarding the major U.S. fusion facilities, asking then that the Scientific Issues Subcommittee (SciCom) begin to address the issues involved in order to prepare a report to FEAC with findings and recommendations. The attached report responds to this major facilities charge and to your formal letter of May 6, 1996 to me. This charge was one of two initiated by the March 25, 1996 letter from Dr. M.A. Krebs to you.

In order to carry out a review of the major U.S. fusion facilities and "produce an optimum plan for obtaining the most scientific benefit from them," the FEAC-SciCom created in March a Major Facilities Review Panel, chaired by Dr. George H. Neilson, Jr. The panel interacted with the research teams at the major facilities through a combination of written input and a day-long roundtable discussion. The FEAC-SciCom wishes to officially thank the members of the panel for their work, and the members of the research teams from the major facilities who provided the voluminous input in such a short time.

There are tremendous opportunities for the advancement of fusion science and improvement of the tokamak concept through the vigorous pursuit and execution of the research programs for the three major U.S. fusion facilities (TFTR, DIII-D, C-Mod), as part of "A Restructured Fusion Energy Sciences Program" (FEAC Report, January 27, 1996). These research programs are outlined and prioritized (for the next two years) in the attached panel report. The scientific goals for the remaining life of TFTR, which was specifically requested in the charge, are also addressed in this report. The FEAC-SciCom voted unanimously to accept the attached report of its Major Facilities Review Panel.

The key findings of the Major Facilities Review Panel, which are strongly endorsed by the FEAC-SciCom, are:

> Finding #1. The major tokamaks and their scientific teams provide the U.S. program with a strong set of capabilities for addressing major physics issues for fusion plasmas and for improving the tokamak concept. The FEAC's favorable assessment of the facilities' capabilities and their potential to contribute in the restructured program is confirmed.
>
> Finding #2. The research plan for the major tokamak facilities will produce impressive scientific benefits over the next two years. The plan is well aligned with the new

mission and goals of the restructured fusion energy sciences program recommended by FEAC.

Highlights of the exciting scientific progress anticipated over the next two fiscal years (for the reference FY 1997 major facilities budgets of: TFTR, $54M; DIII-D, $46M; and C-Mod, $13M) include (not in priority order):

- Improved characterization and control of "transport barriers" (primarily from TFTR and DIII-D).
- Increased understanding and control of high pressure (beta) plasmas (primarily from DIII-D and TFTR).
- Significant increases in D-T fusion power in advanced confinement regime plasmas which will facilitate improved understanding of self-heating by fusion alpha particles and alpha-particle-driven instabilities (TFTR).
- Increased understanding and scenarios for removal of particles and heat from the plasma periphery (primarily from DIII-D and C-Mod).

In addition, in response to the request in the charge letter which stated "In the case of TFTR, if the resources are available to permit operation of TFTR through FY 1997, what are the specific scientific objectives that would merit continuing operations through FY 1997 and into FY 1998? How would you measure progress toward such objectives in a review in mid FY 1997?" the Panel provided the following information which the FEAC-SciCom strongly endorses:

Finding #3. Assuming that the resources are available to permit operation of TFTR through FY 1997 (as appears to be the case under the proposed budget), with the possibility of operation into FY 1998, its highest-priority scientific goals are (not in priority order):

- Evaluate the response of the plasma to alpha heating in advanced-tokamak regimes.
- Characterize the physics of the transport barrier and demonstrate techniques for their control, including application of ion Bernstein waves in reactor-relevant plasmas.
- Evaluate the heating and current drive effectiveness of radiofrequency heating in the ion cyclotron range of frequencies in D-T plasmas.
- Establish the cooling and radial transport of alpha particles using two waves, in support of the concept of alpha-channeling.
- Evaluate alpha confinement and stability in advanced-tokamak regimes.

If a review of TFTR were to be held in mid-FY 1997, the following set of objectives should be used to measure progress toward the above goals:

- Perform an initial evaluation of the response of the plasma to alpha-heating in advanced-tokamak regimes at fusion power levels of about 10 MW.
- Perform an initial characterization of the physics of the transport barrier.
- Couple at least 2 MW of ion Bernstein wave power into the plasma and perform an initial evaluation of its effects on transport.
- Establish the cooling of alpha-particles as part of an initial evaluation of alpha-particle interaction with ion Bernstein waves.

In order to produce an optimized plan, the Major Facilities Review Panel makes the following two recommendations:

Recommendation #1. DIII-D operating time in FY 1997 should be increased by ~50% (within their reference budget level) in order to increase the scientific output in all research areas, and to foster DIII-D's role as a major national collaborative research facility. In achieving this, the DIII-D program should consider reducing the downtime for and/or delaying the divertor upgrade installation.

This recommends an increase from 12 to about 18 weeks of scientifically productive operation of DIII-D.

While the Major Facilities Review Panel Recommendation #1 focused on enhancing the scientific productivity of DIII-D in the near term, the FEAC SciCom does not endorse this recommendation (by a vote of 6 in favor and 9 opposed) due to our concerns over the larger issue of maximizing scientific productivity over the full transition from 3 operating major U.S. tokamaks in FY 1997 to only 2 operating major tokamaks after mid FY 1998. Since the need for additional operation time on DIII-D will become even more critical after TFTR shuts down, it is not clear that expansion of operation time on DIII-D in FY 1997 at the expense of deferring upgrades is the best long term strategy.

Recommendation #2. Additional resources (~$1M) should be applied to the Alcator C-Mod program to increase its near-term scientific output and to build up scientific capabilities needed for the long-term:

- Diagnostic neutral beam and associated diagnostics.
- Completion of the 8-MW ion cyclotron range-of-frequencies (ICRF) heating system.
- Divertor cryopump.

Assuming a fixed total budget for the major facilities, the resources should be obtained through equal reductions in the TFTR and DIII-D programs (~$0.5M each).

The FEAC-SciCom endorses this recommendation by a vote of 13 in favor, 1 opposed and 1 abstention.

The process of developing this plan has been valuable in shaping the tokamak research program. The information provided by the panel's report will guide development of an "optimum plan for obtaining the most scientific benefit" from the three major facilities over the next two years.

We look forward to the impressive fusion science results that should emerge from the scientific programs proposed in the attached plan developed by our major facilities Review Panel. Please let me know if you need any further assessments or information to assist FEAC in its development of a formal response to Dr. Krebs' major facilities charge.

Sincerely,

James D. Callen, Chair, FEAC-SciCom

On behalf of the Scientific Issues
Subcommittee of the Fusion Energy
Advisory Committee (FEAC-SciCom),

and its Major Facilities Review Panel

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for the
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prepared by
Major Facilities Review Panel,
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Dr. Martha A. Krebs,
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Executive Summary

The Fusion Energy Advisory Committee (FEAC) report, "A Restructured Fusion Energy Sciences Program," called for a major facilities review as one of the immediate actions needed to re-align the fusion program for a smooth transition into FY 1997. In its assessment of the three major U.S. tokamak facilities, the Tokamak Fusion Test Reactor (TFTR), DIII-D, and Alcator C-Mod, the FEAC found that all three could contribute to advancing the new goals of the restructured U.S. fusion science program. The purpose of the recommended review of these facilities was to "examine and evaluate [their] progress, priorities, and potential near-term contributions," and to produce "an optimum plan for gaining maximum scientific benefit from their operation, at a funding level not exceeding the FY 1997 President's Budget Request for fusion." In March, 1996, the FEAC was formally charged by the Department of Energy with carrying out the review, and the President's FY 1997 budget request, providing $113M for the operation of the major facilities, was submitted to Congress.

The re-alignment of the major facilities' programs with the goals of the new Fusion Energy Sciences Program effectively began with the release of the FEAC report in January, 1996. During February and March, all three research teams, including their collaborators and their separate Program Advisory Committees, carefully re-evaluated their priorities and their research, operation, and upgrade plans in light of the FEAC report and their FY 1997 budget guidance. The budgets proposed for FY 1997, while lower than those for FY 1995, provide increases for all three facilities over the current fiscal year budgets. This will have the very gratifying effect of increasing the rate of scientific progress in all research areas. The updated plans were communicated to the review panel in written responses to a set of questions posed by the panel, and in a day-long roundtable discussion between the panel and representatives of all three facilities on March 25. Our review of these plans produced the following major findings:

> **Finding #1.** The major tokamaks and their scientific teams provide the U.S. program with a strong set of capabilities for addressing major physics issues for fusion plasmas and for improving the tokamak concept. The FEAC's favorable assessment of the facilities' capabilities and their potential to contribute in the restructured program is confirmed.
>
> **Finding #2.** The research plan for the major tokamak facilities will produce impressive scientific benefits over the next two years. The plan is well aligned with the new mission and goals of the restructured fusion energy sciences program recommended by FEAC.

The major physics issues for the major facilities' programs are,

- Transport and Transport Control
- Magnetohydrodynamic (MHD) Equilibrium, Stability, and Control
- Heating, Current-Drive, and Fueling for Profile Control
- Divertors, Boundary Physics, and Plasma-Wall Interactions
- Alpha and Fast-Particle Physics
- Advanced-Tokamak Scenarios and Integration

(These are the issues for fusion plasma physics listed in the FEAC report, Appendix A by slightly different titles.) The plans in place at the conclusion of this review will lead to substantial scientific accomplishments by the major facilities in all these areas during the next two years. The expected accomplishments are described in detail in the body of our report and in the question responses submitted by the three facilities. Some highlights of the exciting scientific progress that we anticipate are as follows:

- We will increase our understanding of the physical mechanisms governing "transport barriers" – regions of significantly reduced transport that result in improved confinement – and will demonstrate techniques for controlling them.
- We will increase our understanding of how plasma properties influence high-pressure stability limits and will use that understanding and new control techniques to increase pressure limits and sustain the plasma near its pressure limits for longer periods of time.
- We will use transport and stability control techniques to significantly increase the fusion power output from deuterium-tritium plasmas, and use this improvement to increase our understanding of how plasmas operating in high-performance regimes respond to self-heating by fusion alpha particles.
- We will increase our understanding of the physical processes that govern the removal of heat and particles from the plasma periphery and will use that understanding to develop operating scenarios to ensure compatibility between a high-performance plasma and a practical first-wall structure in tokamak reactors such as ITER.

The FEAC report recommended that the TFTR facility should be the first of the three to be shut down, after a period of operation (about 2 years) to extract the remaining scientific benefit from it. With a deadline for TFTR shutdown now established (during FY 1998 at the latest), its planning has been adjusted so as to maximize operating time while foregoing all but a few critical upgrades. We were charged with determining the highest-priority scientific objectives for the remaining operating life of TFTR, and reviewed those developed by the TFTR team in conjunction with its Program Advisory Committee.

> **Finding #3.** Assuming that the resources are available to permit operation of TFTR through FY 1997 (as appears to be the case under the proposed budget), with the possibility of operation into FY 1998, its highest-priority scientific goals are (not in priority order):
>
> - Evaluate the response of the plasma to alpha heating in advanced-tokamak regimes.
> - Characterize the physics of the transport barrier and demonstrate techniques for their control, including application of ion Bernstein waves in reactor relevant plasmas.
> - Evaluate the heating and current drive effectiveness of radiofrequency heating in the ion cyclotron range of frequencies in D-T plasmas.
> - Establish the cooling and radial transport of alpha particles using two waves, in support of the concept of alpha-channeling.
> - Evaluate alpha confinement and stability in advanced-tokamak regimes.
>
> If a review of TFTR were to be held in mid-FY 1997, the following set of objectives should be used to measure progress toward the above goals:
>
> - Perform an initial evaluation of the response of the plasma to alpha-heating in advanced-tokamak regimes at fusion power levels of about 10 MW.
> - Perform an initial characterization of the physics of the transport barrier.
> - Couple at least 2 MW of ion Bernstein wave power into the plasma and perform an initial evaluation of its effects on transport.
> - Establish the cooling of alpha-particles as part of an initial evaluation of alpha-particle interaction with ion Bernstein waves.
>
> (Adopted by the review panel by a vote of 13 in favor, 0 opposed, and 1 absent.)

The DIII-D and Alcator C-Mod facilities will continue operating for several more years after the shutdown of TFTR. The planning for these facilities therefore seeks a balance between near-term scientific output and investment in facility capabilities for the long term. This balancing is complicated by the reality that their budgets continue to be very tight, despite the welcome increases over this year's budgets. Thus while these two facilities will move forward with their scientific programs in FY 1997, they will remain seriously under-utilized. Within a total budget of $113M for the major facilities, which we take as a constraint, it is not possible to correct this situation entirely. However, based on our review of the program's scientific priorities, we make two recommendations which, if adopted, will produce an improved plan for the operation of the major facilities.

> **Recommendation #1.** DIII-D operating time in FY 1997 should be increased by ~50% (within their reference budget level) in order to increase the scientific output in all research areas, and to foster DIII-D's role as a major national collaborative research facility. In achieving this, the DIII-D program should consider reducing the downtime for and/or delaying the divertor upgrade installation. (Approved by the review panel by a vote of 8 in favor, 5 opposed, and 1 abstention.)

We found that more experiments in the DIII-D facility is a priority need for all research areas, including transport, MHD, and divertor physics. There has been a substantial investment in DIII-D hardware capabilities over the years; recent ones include new divertor diagnostics and a long-awaited profile control system that will soon be operational, representing a quantum leap in capability for physics studies. Because the DIII-D research program is conducted as a multi-institutional collaboration (about half the research staff are from outside the host institution), it is important to maximize the opportunity for collaborating scientists to conduct experiments. These considerations combine to place a high premium on operating time (along with the associated planning and analysis effort to make it scientifically productive). However, only 12 weeks of operating time is currently planned for DIII-D in FY 1997, the same as in FY 1996. In order to obtain the most scientific benefit from DIII-D, we think it is important to increase the scientific output in FY 1997, so we recommend this increase in operating time. We also recognize the radiative divertor upgrade as a high priority for divertor, boundary physics, and plasma-wall interaction research (although substantial progress can be made with the present configuration), as well as for high-performance core plasma studies. Its installation has already had to be delayed and split into two phases due to budget reductions, and ideally one would prefer to avoid further delays. However, installation of the first phase of this upgrade in FY 1997 requires a vent of approximately half a year, in which time the machine is obviously inoperable. Thus we face a conflict of priorities. Delaying the divertor installation to increase operating time is a painful tradeoff, if it must be made, but one that we consider warranted by the need to increase the rate of scientific progress next year.

> **Recommendation #2.** Additional resources (~$1M) should be applied to the Alcator C-Mod program to increase its near-term scientific output and to build up scientific capabilities needed for the long-term:

- Diagnostic neutral beam and associated diagnostics.
- Completion of the 8-MW ion cyclotron range-of-frequencies (ICRF) heating system.
- Divertor cryopump.

Assuming a fixed total budget for the major facilities, the resources should be obtained through equal reductions in the TFTR and DIII-D programs (~$0.5M each). (Approved by the review panel by a vote of 7 in favor, 2 favoring reductions in DIII-D only, 2 favoring reductions in TFTR only, and 3 abstentions.)

The development of Alcator C-Mod capabilities has been hampered by tight budgets for its entire operating life. We believe it is necessary to speed up the investment in this facility to ensure that it will be competitive in the long term, since it will be one of only two major U.S. tokamaks operating after 1998. The diagnostic neutral beam will support diagnostics to measure the current profile, ion temperature, rotation velocity, and fluctuations. Completing its basic auxiliary heating complement of 8 MW will enable Alcator C-Mod to operate near the beta (pressure) limit, and is cost-effective because it will make use of source capacity already installed. Both upgrades are critical for Alcator C-Mod's long-term advanced-tokamak program, and both involve collaborations with other institutions, an approach which we believe should be encouraged. The divertor cryopump is needed to improve particle control flexibility for the divertor physics program, currently the main emphasis on Alcator C-Mod. Besides these upgrades, the additional resources recommended will allow modest expansions in research staff and operating time, resulting in immediate increases in scientific output. We would prefer it if the additional resources for Alcator C-Mod could be made available without impacting other parts of the fusion program. However, under the assumption of a constrained total budget for the major facilities, there is no alternative but to offset the increase with reductions in DIII-D and TFTR. We recommend it be shared equally to avoid making an excessive impact on either one and to make clear that there is no adverse judgment against either one implied by this recommendation.

In summary, the community has developed a research plan for the major tokamak facilities that will produce impressive scientific benefits over the next two years. The plan is well aligned with the new mission and goals of the restructured fusion energy sciences program recommended by FEAC. Budget increases for all three facilities will allow their programs to move forward in FY 1997, increasing their rate of scientific progress. With a shutdown deadline now established, the TFTR will forego all but a few critical upgrades and maximize operation to achieve a set of high-priority scientific objectives with deuterium-tritium plasmas. The DIII-D and Alcator C-Mod facilities will still fall well short of full utilization. Increasing the run time in

DIII-D is recommended to increase the scientific output using its existing capabilities, even if scheduled upgrades must be further delayed. An increase in the Alcator C-Mod budget is recommended, at the expense of equal and modest reductions (~1%) in the other two facilities if necessary, to develop its capabilities for the long-term and increase its near-term scientific output.

I. Introduction

Background

The Fusion Energy Advisory Committee (FEAC) report, "A Restructured Fusion Energy Sciences Program," called for a major facilities review as one of the immediate actions needed to re-align the fusion program for a smooth transition into FY 1997. The "major facilities" refer to the three large U.S. tokamak experimental programs,

- The Tokamak Fusion Test Reactor (TFTR), sited at Princeton Plasma Physics Laboratory,
- DIII-D, sited at General Atomics, and
- Alcator C-Mod, sited at the Massachusetts Institute of Technology.

In assessing these facilities for its January 27, 1996, report, the FEAC concluded that all three could contribute to advancing the new goals of the restructured U.S. fusion science program. The review was recommended to "examine and evaluate [their] progress, priorities, and potential near-term contributions," and to produce "an optimum plan for gaining maximum scientific benefit from their operation, at a funding level not exceeding the FY 1997 President's Budget Request for fusion." At that time, the fusion budget request had not yet been determined.

In March, 1996, the FEAC was formally charged by the Department of Energy with carrying out the review, and the President's FY 1997 budget, requesting $255.6M for the Fusion Energy Sciences program, was submitted to Congress. The charge (Appendix A) formally requests the optimum plan that was called for in the FEAC report, and further requests that three specific points be addressed:

- The highest-priority near-term (~2 years) scientific objectives for the facilities,
- Recommended actions to more effectively use the facilities to address these objectives, and
- Scientific objectives for TFTR, assuming the resources were available to operate it into FY 1998, and objectives by which to measure its progress in mid-FY 1997.

The FY 1997 budget allocation proposed by the Office of Fusion Energy Sciences (OFES) provides budget increases for all three facilities over the current fiscal year, which will have the very gratifying effect of increasing the rate of scientific progress in all of them. The major facility groups have used the FY 1997 budgets shown in Table I.1 and assumed constant budgets for FY 1998 in planning the scientific programs presented at this review. Our review charge specified that we should consider changes in these budget allocations (assuming a fixed total for the facilities) as a possible action for producing an optimum plan.

Table I.1. Proposed FY 1997 budget allocation (in millions of dollars) for the major facilities, compared with actual FY 1995 and 1996 budgets.

	FY-95	FY-96	FY-97
TFTR	65	51	54
DIII-D	50	38	46
Alcator C-Mod	16	10	13
Total Major Facilities	131	99	113

During February and March, all three research teams, including their collaborators and their separate Program Advisory Committees, carefully re-evaluated their priorities and their research, operation, and upgrade plans in light of the FEAC report and their budget guidance. Their updated plans were presented to the review panel in written submissions, as described below. Collectively they describe an excellent scientific plan for the next two years, which we found to be well-aligned with the new goals of the restructured Fusion Energy Sciences Program.

Review Process

The review was conducted by the FEAC's new standing subcommittee, the Scientific Issues Subcommittee (Scicom). The charge from FEAC to Scicom is included in Appendix A. The Scicom chair appointed a review panel consisting of six Scicom members, six other scientists from the U.S. fusion community, and two scientists from abroad, with one of the Scicom members designated as panel chair. The panel membership included one senior scientist from each major facility who had a working knowledge of all aspects of his facility's program. Besides being full voting members on all issues, these members helped the others to accurately understand their programs by providing clarification and detail when needed. The panel members and current Scicom members are listed in Appendix B.

The program heads of the three major facilities met with the Scicom at its February 26-27 meeting in Austin, TX and provided overview briefings on their programs. On March 6, the panel issued a list of questions (Appendix C) seeking information on the facilities' research, upgrade, and operations plans; on suggested actions and impacts of budget reductions; and on assessments of facility utilization and the programs' contributions to various fusion program goals. We found that the answers, received March 18, were responsive to the panel's questions, and provided a detailed description of the research plans for the next two years and the expected scientific benefits. These submissions thus went a long way in producing the plan requested in our charge. A day-long roundtable meeting of the panel and facility representatives on March 25 at the Department of Energy, Germantown, MD, headquarters served to answer follow-up questions from the panel and provide clarification. This completed the input phase of the review.

The panel deliberated in executive session on March 26-27. We discussed the scientific plans in each of six topical areas and attempted to determine priorities among specific research issues, upgrades, and operating time, as well as various tradeoffs among these. In this report, we emphasize priorities, since we consider the assessment and establishment of priorities to be the panel’s main contribution to an optimum plan. Throughout, we make numerous detailed recommendations concerning priorities and emphases. These may be identified as recommendations or suggestions, or by use of the word “should”. We believe that the overall program will be improved by following these recommendations and that it should be the responsibility of the individual program leaders to determine how best to do so. We have also made two recommendations that call for more visible programmatic changes, one an increase in operating time for the DIII-D facility, the other a reallocation of budget resources to benefit the Alcator C-Mod program. These are identified as Recommendations #1 and #2, respectively, in a separate section (Section V) of the report. While most panel decisions were reached by consensus, these two were decided by panel vote, the results of which we report. The lack of unanimous support for these recommendations is not surprising, given the difficulty of the choices involved. Nevertheless they are the result of careful consideration and vigorous discussion after looking at the issues in some detail. On balance we believe they are necessary to “obtain the most scientific benefit” from the major facilities.

A summary of the panel’s conclusions, including its major recommendations, was presented in an oral briefing by the panel chair at the March 28-29 Scicom meeting in Germantown. The meeting was attended by numerous members of the community, including representatives of all three major facilities, and OFES staff.

Report Organization

In Chapter II, Sections A through F, we discuss the research plans by topic area, combining the plans of all three facilities. The topic areas (which correspond to the FEAC and National Research Council issues for fusion plasma physics) are:

A. Transport and Transport Control
B. Magnetohydrodynamic (MHD) Equilibrium, Stability, and Control
C. Heating, Current-Drive, and Fueling for Profile Control
D. Divertors, Boundary Physics, and Plasma-Wall Interactions
E. Alpha and Fast-Particle Physics
F. Advanced-Tokamak Scenarios and Integration

Each of these sections describes the scientific issues and current status in the respective area; the facilities' research plans, including upgrade and operating time requirements; our assessments of the priorities; and detailed recommendations. These sections address the first point of the review charge. In Section II.G we provide the goals of TFTR for its remaining operating life, in response to the third point of the charge.

Chapter III deals with the facilities' technical plans, including hardware upgrades and facility operation. The panel's assessments of these plans are provided, based on the priority needs of the research program from Chapter II. We also discuss the issue of facility utilization, in view of the FEAC call for the facilities to be fully utilized.

Chapter IV provides assessments of the major facilities' plan from the point of view of various goals of the U.S. fusion program: advancement of FEAC scientific goals, promotion of U.S. leadership in concept innovation, resolution of ITER Physics R&D issues, and contribution to materials and technology development.

In Chapter V we present our major recommendations, in response to the second point of the charge. Our summary conclusions are provided in Chapter VI.

II. Research Plans

The research program on the major tokamaks is focused on tokamak concept improvement, consistent with the national program goal to "Develop fusion science, technology, and plasma confinement innovations as the central theme of the domestic program." It also advances the field of plasma science and, through its contributions to the ITER physics basis, supports the international program to develop fusion energy. The program is divided into six research areas, which correspond to the six Fusion Plasma Physics Issues defined (by slightly different names) in the "Fusion Program Scientific Goals" appendix (Appendix A) of the FEAC report.

A. Transport and Transport Control

Scientific Issues

The transport of energy, momentum, and particles across magnetic surfaces is one of the critical factors that will determine the attractiveness of any magnetic confinement system. Tokamak experiments with intense auxiliary heating around 1980 established a baseline level of confinement performance, which has come to be known as the "Low Mode" or more commonly the L-mode. Since then many regimes of "enhanced" confinement have been studied, starting with the "High Mode," or H-mode in the early 1980's. Besides their potential for improving tokamak performance, these regimes provide important insights on the physical mechanisms that govern transport. Understanding these mechanisms is critical to the long-term goal of obtaining a predictive capability for transport, so this is the theme for much of the current transport research.

Core Transport Barriers

A central scientific issue for the facilities is the elucidation of the physics of core transport barriers, thereby facilitating the development of a unified perspective on experimentally observed enhanced-confinement regimes, including Reversed Shear modes, encompassing the Enhanced Reverse Shear (ERS), Negative Central Shear (NCS), and Weak Negative Shear (WNS) modes; Super Shots; Pellet-Enhanced Performance (PEP) modes and the Very High (VH) mode. This issue may be sub-divided into the analysis of:

a) Physical processes and constituents of barriers, such as:
    i) magnetic geometry: effects of safety factor profile, shaping, and bootstrap current,
    ii) electric field shear: particle density and ion temperature gradients, toroidal velocity shear (influenced by co- and counter- neutral beam injection), and poloidal velocity shear (RF and self-generated).
b) Transition mechanisms into and back-transitions out of enhanced regimes. Transitions can be triggered by heat and momentum sources and facilitated by deposition and current profile control. A related issue here is

reaching an understanding of the ontogeny and phylogeny of the Super Shot and PEP modes and placing these in context with Reverse Shear confinement regimes. Another critical issue is obtaining a quantitative understanding of the hysteresis factor for core barrier back-transitions.

c) Bifurcation thresholds, specifically:
   i) characterization of relevant dimensionless ratios governing local gradient threshold behavior, scaling, and clarifying the local or non-local behavior of the transition dynamics. Note that some interpret the Joint European Tokamak (JET) VH-mode as a non-local transition.
   ii) optimization of threshold requirements via shaping, etc.,
   iii) identification of fluctuation precursors to the transition.

d) Transport channel (i.e., electron or ion) asymmetry and discrimination, specifically understanding the relationship between the range of behavior exhibited, which extends from hot ion barriers (e.g., the TFTR ERS and Super Shot modes and the DIII-D NCS mode where the electron thermal diffusivity (χ_e) drops only slightly) to PEP modes (e.g., as in Alcator C-Mod, where the electron and ion temperatures are about equal) to electron heating regimes (e.g., DIII-D with electron cyclotron heating and JT-60U) where a reduction in χ_e, symptomatic of an electron transport barrier, is observed and, in the case of JT-60U, is dominant.

e) Neoclassical transport in regimes of steep gradients and strong electric field shear close to the magnetic axis, such as in Reverse Shear barriers. This is crucial to the goal of barrier control in long pulse advanced tokamak scenarios.

f) Role of wall conditioning in transport barrier formation. Here, it is crucial to understand the mechanisms through which wall conditioning (including neutrals, impurities, Li pellets, etc.) favors barrier formation.

Radial Electric Field Control by Radiofrequency Waves

An important component of a long pulse advanced tokamak system is rotation profile control for electric field shear control of pressure profiles. Currently rotation is controlled by neutral beam injection, but the utility of this method in a burning plasma is questionable. An additional significant issue, which any control scheme must confront, is the nature of its interaction with alpha particles. To this end, the TFTR group has planned to study ion Bernstein wave (IBW) control of poloidal rotation via the suggestion of generating a localized shear layer through the RF-wave-induced Reynolds stress. This is a flexible control tool, allowing variation of the strength, location and width of the transport barrier. A poloidal rotation diagnostic (based on charge-exchange recombination spectroscopy) will be installed as an integral part of the planned program. A unique capability of TFTR is the capacity of studying IBW profile control in an environment with significant alpha particle heating. Investigation of IBW rotation profile control is a high priority issue, and one which is central to the TFTR program.

Core Transport Physics

In addition to barriers, several other scientific issues remain in the realm of core transport physics. These include:

a) The relation of fluctuations to transport. Here, information from novel electron and ion temperature fluctuation measurements would be especially useful in regimes where there is suppression of ion transport but not electron transport.

b) The validation and relation to microphysics (i.e., fluctuation measurements, scaling, etc.) of dimensionless scaling arguments.

c) The isotope scaling of transport coefficients.

d) Electron transport. Indeed, it seems safe to say that ion, particle and momentum transport seem much better understood and much more readily described by quasi-linear micro-instability theory than electron transport is. Curious anomalies abound, from the "hang-up" of χ_e in the ERS mode, to the electron transport barrier formation observed in the electron-cyclotron-heating-driven NCS mode to the "disappearance" of neo-Alcator scaling in Alcator C-Mod. Possible explanations for these anomalies include an electron-ion coupling anomaly, transport due to magnetic stochasticity, and short wavelength sub-critical bifurcations.

e) Transient transport phenomena and their relation to general transport issues. Here, a few central issues are:
    i) non-locality phenomena, such the edge heating-center cooling experiments,
    ii) the implications of transient phenomena for electron transport,
    iii) the possible manifestation of self-organized criticality phenomena in transient experiments, and
    iv) the study of ion heat pulse propagation, using high frequency charge-exchange recombination spectroscopy and other techniques.

f) Exploration of transport physics in high-internal-inductance (l_i) modes.

g) The relationship of particle, momentum, and energy sources to transport.

h) Increased exploration of regimes with $T_i \approx T_e$. Most present studies of enhanced confinement have T_i substantially greater than T_e; this is a particularly interesting limit from the ion transport point of view, but may not be relevant to ignited/burning plasmas.

L-to-H Mode Transitions, Edge Turbulence and Transport Physics

The physics of the L→H transition is a critical issue for ITER and for advanced tokamak scenarios. Moreover, the L→H transition is inexorably coupled to more general issues bearing on edge turbulence, both inside the separatrix and in the scrape-off layer. These matters enter the physics of edge density limits, as well. Specific issues include:

a) The parameter scaling of L→H and H→L transition thresholds.

b) Identification of critical parametric relations pertinent to the L→H transition.

c) Understanding the role of neutrals in L→H physics.

d) Understanding slow L→H and H→L transitions, their dynamics, and the relation of "slow" to "fast" transitions.

e) Edge Localized Modes (ELMs):
    i) basic physics and characterization,
    ii) identification of critical parameters controlling access to grassy behavior regimes,
    iii) elucidation of the physics of Type III ELMs.

f) Characterization of L-mode edge turbulence, both inside the separatrix and in the scrape-off layer.

g) Documentation of the effect of barriers on particle and energy transport using measurements of density and temperature fluctuations.

Plans

All three major tokamaks plan research programs in the transport area. This is quite appropriate, given the diversity of issues involved and the range of control and diagnostic capabilities that the three machines bring to bear. A summary of the program plans is presented in Table II.A.1, with our assessment of the priorities noted.

Table II.A1. Summary of research plans in Transport and Transport Control. Priority assessments are denoted as High (H) or Medium (M). The asterisk (*) for Alcator C-Mod core physics issues denotes the opinion that diagnostic upgrades which are planned but not yet available will be necessary for effective contribution.

	Core Barriers	**RF Momentum Control**	**Core Transport**	**L-H Transitions**
TFTR	Elucidate: -V_ϕ shear profile effect. -Variable heat, particle deposition on RS physics and threshold. - Exploration of interaction of α-particles with transport barriers. **H**	-Exploration and study of transport barrier formation and control. -Study transport barrier control in α environment -IBW and V_θ diagnostic are critical. **H**	Elucidation of physics of: -isotope effects on transport fluctuation. -T_i and $V_{\|\|}$ fluctuations, and relation to transport. **M**	
DIII-D	Elucidation of effects of: -Shaping. -Variable heat and particle deposition. -Boundary control or RS dynamics and threshold. -Test steep gradient neoclassical predictions. **H**		-Clarification of ρ* scaling relation to microphysics. -Progress on electron heat transport processes (using ECH). -Elucidate transport in high-l_i mode. -Exploration of $T_i \approx T_e$ regime, i.e., VH-Mode. **H**	-Explore physics of slow L-H transition and relation to standard transition. -Identify critical parametric dependence in transition threshold. -Elucidate physics of ELMs, especially grassy regimes -Study L -mode edge turbulence. **H**
Alcator C-Mod	-Elucidate physics of RF and pellet driven core barriers (including thresholds). -Explore non-NBI RS regimes **M***		- Validation and clarification of ρ* scaling studies. - transport in $T_i \approx T_e$ regime. - absence of particle and momentum sources (no NBI). **H***	-Explore physical processes and parametric dependencies -Identify critical parametric dependence of threshold -Study L-mode edge turbulence -Explore ELM dynamics and classification -Study VH-mode dynamics, non-NBI access to VH mode. **H**

To carry out the planned program, a number of hardware upgrades are critical. From the transport perspective, these are the upgrades that should be accorded priority.

TFTR Upgrades

a) Exploitation of the ion Bernstein wave antenna and the poloidal rotation diagnostic to study transport barrier control in an α-particle environment. The installation of these systems is expected to be completed in the current fiscal year.

b) Improving the high frequency charge-exchange recombination spectroscopy system for ion temperature fluctuation studies, particularly at the transition to the Enhanced Reverse Shear (ERS) mode.

DIII-D Upgrades

a) Core density profile measurements are essential. We suggest Thomson scattering be explored as an alternative to reflectometry.

b) Fluctuation diagnostics upgrades, including
   - Upgrade of beam emission spectroscopy from 32 channels to 64 channels.
   - Development of a two-dimensional beam emission spectroscopy system and ion-temperature fluctuation measurements.
   - Electron cyclotron emission system for measurement of electron-temperature fluctuations.
   - Improvement of probes for edge electron temperature fluctuation measurements.

c) Installation of the first 3 MW of the planned 6-MW electron cyclotron heating (ECH) system. Physics studies using this complement should be performed before deciding to proceed with the remaining 3 MW. The objective is to control the heat deposition profile and allow heating in the electron as well as the ion channel which permits proper two-fluid transport studies. Heat pulse propagation experiments will be facilitated also. Initial physics results are expected by March, 1997.

d) Assessment of the physics capabilities of a proposed heavy ion beam probe system, should be pursued. The purpose of this system would be to measure electrostatic potential fluctuations and radial electric fields.

Alcator C-Mod Upgrades

a) Installation of a proposed diagnostic neutral beam and the following associated diagnostics:
   - Beam emission spectroscopy to measure density fluctuations
   - Charge-exchange recombination spectroscopy to measure ion temperature and rotation
   - Motional Stark effect diagnostic to measure the safety factor profile.

b) Electron cyclotron emission diagnostic for measurement of electron temperature fluctuations.

c) Installation of the X-mode and imaging reflectometry systems for density fluctuation measurement.

d) Completion of the 8-MW ICRF heating system to explore transition thresholds for reverse shear regimes.

Actions

In summary, the panel recommends the following actions as high priority to provide a strong transport research program in the next two years:

a) The Alcator C-Mod core fluctuation diagnostic upgrades (beam emission spectroscopy, electron cyclotron emission) and the required diagnostic neutral beam should be implemented.
b) The ion Bernstein wave antenna to drive rotational shear flow drive and the poloidal velocity diagnostic should be implemented as scheduled on TFTR, and aggressively exploited next year.
c) Increased operating time and associated analysis resources for transport studies on DIII-D.

B. Magnetohydrodynamic Equilibrium, Stability, and Control

Scientific Issues

The operating space for tokamak plasmas is bounded by magnetohydrodynamic (MHD) stability limits. Through control of the plasma equilibrium properties, such as the shape of the cross section and the profiles of current density and pressure, it is possible to modify some of these limits. This is important to the achievement of an attractive reactor operating scenario.

Increasing the plasma beta (β, the ratio of plasma pressure to magnetic field pressure) increases the fusion power density and thus allows a more compact (smaller in size and/or toroidal field) machine. Higher beta also increases the drive mechanism for the bootstrap current and thus contributes to the attainment of efficient steady-state current drive. Other performance-limiting phenomena considered here are edge-localized modes (ELMs), disruptions, and density limits. Ideal MHD stability limits, which place an upper bound on operating limits, are fairly well understood with good agreement between experiment and theory. Operational scenarios to optimize beta within the ideal limits have been identified and are continuing to evolve; many of these remain to be tested experimentally. Many of the limiting MHD mechanisms observed in experiments are non-ideal (neoclassical tearing modes, double tearing modes, resistive interchange modes, resistive kink or sawtooth, resistive wall mode, locked modes etc.). Significant progress is needed in both the experiment and theory/modeling to understand these.

Operating at or near the known MHD limits increases the risk of plasma disruption, in which the plasma stored energy and plasma current decay rapidly. In elongated plasmas, the thermal quench is often followed by vertical instability and the combined current decay and plasma motion results in large halo currents in the scrape-off plasma. The large thermal and electromechanical loads during disruptions set the most stringent design criteria, and the characterization and understanding of disruptions is a critical research element for the design of tokamaks. The goal is to operate near the known MHD limits while at the same time avoiding disruptions and, for those disruptions that can not be avoided, to mitigate as far as possible the high heat flux and high electromechanical loads. The research in this area is aimed at characterizing and understanding the instability mechanisms that limit tokamak performance and controlling them to achieve reliable, sustained high-performance operation.

MHD Stability and the Beta Limit

The main scientific issues for beta limits are 1) characterizing the MHD modes whose instability thresholds determine the limit and 2) determining the influence of various plasma properties on the limits. The specific issues are:

a) Limiting instabilities:

- The ideal limit is described as a value of $\beta_{N,max}$, where $\beta_N=\beta/[I_P/aB_T]$ (%, MA, m, T). Theoretical analyses based on MHD stability to ideal ballooning and ideal kink modes predict $\beta_{N,max} \approx 2.8$ to 4.5. The dependence of the beta limit on the normalized current (I_P/aB_T) is well established.
- Resistive and neoclassical MHD tearing-type modes with low poloidal and toroidal mode numbers (m,n) are present in plasmas with monotonic safety-factor (q) profiles.
- The Mercier instability limits the duration of the VH-mode when the value of the central safety factor (q_0) is less than unity.
- Resistive interchange, double-tearing, and ideal modes are present in reverse-shear scenarios (with non-monotonic q profiles).
- Sawtooth behavior is typically observed for $q_0<1$ in inductively-driven discharges, causing periodic release of energy (in the case of "monster" sawteeth, large amounts of energy) from the central region of the discharge.
- Mode locking occurs when rotating helical perturbations near the edge of the plasma couple to stationary magnetic asymmetries caused by field errors, causing the rotation to slow down or stop. Locking often causes the mode to become unstable, resulting in a disruption. This behavior sets limits on the allowable field error magnitude.
- Significantly higher values of beta are predicted if the plasma is stabilized by a resistive wall. This stabilization is confirmed by several experiments, but the detailed requirements of plasma rotation and dissipation mechanisms still require evaluation and understanding.

b) Influence on beta limits of various plasma properties:
  - Profiles of pressure, current-density, and toroidal rotation velocity. Two high-beta regimes of particular interest are the high-internal-inductance (l_i), with peaked current profiles, and the reverse-shear, with non-monotonic safety-factor (q) profiles and current-density profiles. Toroidal rotation plays an important role in wall stabilization of external modes when there is a large bootstrap current component near the plasma edge.
  - Plasma shape. Strong shaping, characterized by high elongation (κ) and triangularity (δ), is favorable for high-beta stability. The optimum pressure and current density profile for stability is dependent on the plasma shape.
  - Plasma collisionality, particularly through its effect on the bootstrap current via neoclassical MHD.
  - Kinetic effects, due to the presence of fast particles such as alpha particles or beam ions. A reciprocal issue is the influence of unstable modes on fast-particle losses.
  - Non-ideal effects, manifested, for example, in a dependence of $\beta_{N,max}$ on the toroidal magnetic field observed in TFTR.

c) Mode saturation behavior, linear and non-linear evolution.

Stability of Edge-Localized Modes (ELMs)

This issue is closely coupled to issues of transport and divertor physics. Edge-localized modes are a signature of H-mode operation, which features a steep pressure gradient at the edge. Energy transport out of the core plasma causes the pressure gradient to build up until it reaches a local stability limit. The resulting instability causes the energy to be suddenly released into the scrape-off layer and the pressure gradient to relax, and the process repeats itself. The energy release alters the power balance in the divertor and can temporarily cause reattachment and excessive heat flux to the target structures. Thus, ELMs represent a potential performance limitation for high-power tokamaks and their characterization is a high-priority research need for ITER. Three types of ELMs have been identified:

Type I	caused by ideal ballooning instability,
Type II	high radiation, occurs with strong plasma shaping,
Type III	possible resistive ballooning, but not clearly identified as the cause; other possibilities need to be identified.

In general, a more complete characterization and physics understanding of ELMs is needed. Specific issues are the characterization of the local plasma parameters at the onset of instability, and a clear identification and characterization of the responsible instability, or precursor.

Disruptions

Plasma disruptions cause sudden discharge termination and are potentially damaging to the machine structures. Their accurate characterization is essential for the design of ITER and any other tokamak. Their mitigation and, ultimately, elimination are required in the long term for improving the tokamak concept. The specific issues in this area are:

a) Evolution of disruption events:
   - Island formation in the plasma column.
   - Coupling of modes, including stochasticization of the magnetic field in the region between the islands.
   - Magnetic reorganization of the plasma configuration.
   - Thermal energy decay.
   - Plasma current (magnetic energy) decay.
   - Vertical displacement events (VDE), and generation of large halo currents.

b) Effects of disruptions:
   - Heat loads (2D models needed).
   - Electromagnetic forces (2D models needed).
   - VDE effects, especially halo currents (2D models needed).
   - Poloidal and toroidal asymmetries in electromagnetic forces and halo currents (3D models needed).
   - Runaway electron generation. Large runaway currents are predicted for high plasma current disruptions (for example in ITER), but the generation process is not yet confirmed in the experiments.

c) Characterization of disruptions, especially in high-performance plasmas. Scalings of key characteristics such as decay rates and halo currents are needed. Contributions of tokamak disruption data to a worldwide ITER data base is the immediate need.

d) Mitigation and avoidance of disruptions:
   - Identification of impending disruptions by precursors or approach to operational limits.
   - Pellet-induced pre-emptive disruption (to reduce severity).

Density Limits

The existence of density limits in tokamaks has long been recognized. Experimentally, density limits with cold gas-injection fueling are well characterized by the Greenwald limit, $\bar{n}_e < I_P / \pi a^2$ (10^{20} m^{-3}, MA, m); however the physics governing the limit seems to be a combination of MHD and transport phenomena and is not well understood. The issues are to gain the necessary understanding and to establish control techniques for modifying the limit. This is an important research need for ITER, which must operate well above the Greenwald limit in its reference mode. Specific issues are:

a) Understanding the edge density limit, in particular its relationship to divertor detachment, and its dependence on power and local plasma parameters.

b) Understanding bulk particle transport, and the use of deep fueling to obtain peaked density profiles.

c) Effectiveness of different fueling methods in controlling density, including gas injection at the edge, pellet injection, and compact-toroid injection.

d) Demonstration of scenarios for operation above the Greenwald limit.

Plans, Actions, and Priorities

All three major tokamaks have research and hardware upgrade plans in the MHD area.

TFTR Plans

In order to increase the fusion performance of TFTR, and hence the ability to study additional alpha particle effects, an understanding of beta limits and methods to increase stability margins is critical. The plans are:

a) Beta limits:
   i) Evaluate the beta-limiting MHD phenomena in collisionless plasmas and their dependence on the pressure profile, current density profile, collisionality (ν^*), normalized gyroradius (ρ^*), and toroidal magnetic field.

    - ii) With its D-T capability, TFTR will be able to study the influence of alpha particles on stability limits as well as the effects of MHD modes on the alpha population and, in particular, alpha losses.

b) Disruptions:
    - i) Since disruption evolution up to the thermal quench is relatively insensitive to shape details, TFTR's studies in this area will contribute to ITER characterization needs.
    - ii) Alpha losses during disruptions.
    - iii) Krypton-pellet-induced pre-emptive disruptions.

Beta-limit control experiments will be aided by the upgrade of the motional Stark effect system to improve current-profile measurement resolution, and by the ICRF four-strap antenna and frequency change to provide off-axis profile control. The lower hybrid system would also be useful for off axis profile control but has been placed on hold due to budget constraints. The infrared periscope upgrade would provide better measurements of disruption heat loads on the first wall.

DIII-D Plans
Control of stability limits and extending the duration of high-beta regimes are central to DIII-D's overall goal of demonstrating an integrated advanced-tokamak scenario. Its flexible, strong shaping and current-profile control systems provide valuable capabilities for high-beta and MHD studies. The plans are:

a) Beta limits:
    - i) Stability limits in high-performance regimes (VH, reverse shear, and high-l_i modes) will be investigated and compared with theory. The dependence on plasma shape, pressure profile, and current density profile, and the impact of the resistive interchange instability, will be studied. Evaluation and understanding of the role of rotation in wall stabilization and locked modes is an important objective. The impact of the neoclassical tearing mode stability on the beta limit in long pulse discharges, including the effects of details of the profiles and the collisionality, will be studied.
    - ii) A high priority goal is to extend the duration of high-performance scenarios operating near the stability limits. The dependence on edge conditions will be investigated, in particular the importance of local versus global parameters. Methods of controlling the edge pressure gradients (reduction of power fluxes, control of transport) will be investigated. A major emphasis will be placed on profile control using non-transient methods, especially the FWCD and ECCD systems for on- and off-axis current profile control, in combination with divertor pumping for density control.

b) Disruptions
   - i) A priority is to study the evolution of VDEs and associated halo currents, including asymmetries. Two- and three-dimensional models will be developed, scalings will be determined, and there will be a strong contribution to the ITER database in this area.
   - ii) Physics of disruptions, measurement of plasma profiles during disruption, and measurement of heat and electromagnetic force loads.
   - iii) Disruption avoidance with a neural net system, mitigation with pellet-induced pre-emptive disruptions and other scenarios.

c) Density Limits:
   - i) Fueling with gas puff and pellets, understanding of density limit mechanisms.
   - ii) Investigation of scenarios for operationally exceeding the Greenwald limit.

The highest-priority upgrade for MHD studies is the installation of the first 3 MW of the planned ECH system. This will be used for off-axis current drive, predicted to be about 300 kA. Important diagnostic upgrades include improvements to the motional Stark effect (current profiles) and electron-cyclotron emission radiometer (electron temperature) systems, and addition of the capability to measure the central density. Operating time is a priority to accommodate MHD experiment with these new capabilities.

Alcator C-Mod Plans

Characterization of disruptions, particularly halo currents, is currently a major emphasis of the Alcator C-Mod program. An extensive set of diagnostics is available for this purpose. At present, Alcator C-Mod is limited by ICRF heating power in its ability to approach beta limits. Significant high-beta studies will await completion of the full 8-MW ICRF heating system.

a) Disruptions:
   - i) A priority is to study the spatial and temporal evolution of halo currents, determine how they scale, and understand the driving mechanisms. Two-dimensional models will be developed. A particular emphasis is halo current asymmetries; three-dimensional models will be used to aid understanding. Fast and spatially resolved diagnostics are available.
   - ii) Physics of disruptions until the thermal quench.

b) Density Limits:
   - i) Although not of high priority, there is some interest in exploring the density limits at the high magnetic fields available in Alcator C-Mod.

Hardware upgrades will emphasize the development of long-term capabilities. Completion of the 8-MW ICRF system, which includes a new four-strap antenna and recommissioning of tunable sources, is important for two reasons: being able to heat the plasma to the beta limit, and providing off-axis current-profile control by mode-conversion current drive. The installation of a planned diagnostic neutral beam and associated diagnostics, particularly those for measuring current-density, ion temperature, and rotation profiles, is also critical for future research in this area. High-priority should be given to these upgrades. In the longer term, the planned lower hybrid current drive system, unfunded at present, should be installed to augment off-axis current-profile control capability.

A summary of the program plans is presented in Table II.B.1, with our assessment of the priorities noted.

Table II.B1. Summary of research plans in MHD Equilibrium, Stability, and Control. Priority assessments are denoted as High (H), Medium (M), or Low (L). The asterisk (*) for Alcator C-Mod stability / high beta issues denotes the opinion that heating and diagnostic upgrades which are planned but not yet available will be necessary for effective contribution.

	Stability / Beta Limits	**Edge Localized Modes (ELMS)**	**Disruptions**	**Density Limits**
TFTR	Limiting instabilities **M** Profile influence **M** B_T, ν^* influence **H** Alpha interactions **H**		Evolution to thermal quench **M** Current quench, VDE, halo **L** Mitigation **L** Alpha losses **H**	
DIII-D	Limiting instabilities **H** Profile influence **H** Shape influence **H** Rotation, edge influence **H**	**M** for MHD (addressed by radiative divertor program)	Evolution to thermal quench **M** Current quench, VDE, halo **H** Mitigation **M**	**H**
Alcator C-Mod	Limiting instabilities **M*** Profile influence **M*** Shape influence **M*** Other influence **M***		Evolution to thermal quench **M** Current quench, VDE, halo **H** Mitigation **M**	**L** (High B)

C. Heading, Current Drive, and Fueling for Profile Control

Scientific Issues and Plans

Auxiliary heating is currently utilized in tokamaks to achieve high plasma temperatures and pressures and is included in most designs for future tokamak devices as a means of achieving burning-plasma conditions. Noninductive current drive, obtained with specially-configured auxiliary heating systems, is required in combination with the self-generated bootstrap current for steady state tokamak operation. The application of auxiliary heating power to tokamak plasmas also enables operation in enhanced-confinement regimes, such as H-modes, Super Shots, VH-modes and, more recently, a variety of reverse shear variants (see Section II.A). Control of the fuel deposition using various external fueling techniques is also seen to have a direct influence on plasma performance. Theoretical studies have shown fundamental roles for power deposition, fuel deposition, and rotation-profile and current-profile modification in the formation of internal transport barriers that often characterize these enhanced confinement regimes.

Methods for delivering power to plasmas include neutral beam injection (NBI) and the launching of radiofrequency (RF) electromagnetic waves into the plasma. The most successful radiofrequency heating schemes to date involve waves in the ion cyclotron, lower hybrid and electron cyclotron range of frequencies (ICRF, LHRF, and ECRF, respectively). Over the next few years, the U.S. program will concentrate most heavily on ICRF and ECRF, with the experimental program in LHRF deferred for a few years due to budgetary constraints. The scientific issues underlying the success of any of these methods are power deposition localization and power partitioning among plasma species, efficiency, parasitic absorption, plasma profile response, and edge interactions. Fueling can be accomplished through the injection of neutral beams, solid pellets, compact toroids, or cold gas.

Neutral Beam Injection (NBI)

Neutral beam injection at energies up to about 120 keV (based on positive-ion acceleration) is currently the scheme used most often to achieve high performance plasmas in tokamaks, both in the U.S. and worldwide. Moreover, the fast particles from NBI are used as a tool to facilitate the study of Alfvén eigenmode instabilities. While the physics of heating, fueling, current-drive and rotation-drive by NBI is well understood, its effects on plasma transport are a subject of vigorous research. In particular, the role played by the NBI-induced plasma rotation in the formation of transport barriers via sheared flow is under investigation on both DIII-D and TFTR. Recently, 500-keV neutral beams (based on negative-ion acceleration) have been installed on JT-60U. This technology provides greater penetrating power and may lead to a system suitable for reactors. Because of the differences in deposition profiles, comparisons of neutral-beam-driven advanced-tokamak scenarios in JT-60U with those of DIII-D and TFTR may lead to increased understanding of the fundamental processes which underlie these regimes.

Ion Cyclotron Range of Frequencies (ICRF) Waves

Heating by ICRF waves is reasonably well understood, particularly minority heating scenarios. However, there is a wide variety of ICRF heating and current drive scenarios with a number of outstanding issues which need resolution.

Most ICRF studies have been performed in deuterium (D), hydrogen (H) and helium (He) plasmas at moderate densities. Experiments in Alcator C-Mod are extending the database to high-density regimes. The first 4 MW of a planned 8-MW system based on existing RF sources is now operational; the remaining 4 MW is needed to support Alcator C-Mod's long-term needs for heating and current drive. The use of deuterium-tritium (D-T) plasmas with fusion alpha particles adds complexity to wave-particle interactions, but also makes new operating regimes accessible. Second-harmonic tritium heating, the favored ICRF heating scenario for ITER, has already been demonstrated on TFTR. Fundamental D and minority T heating are also under consideration for ITER, the latter in particular for low reactivity operations; wave propagation and absorption in these two regimes will be explored in TFTR.

Fast-wave electron heating (FWEH) and current-drive (FWCD) processes, which rely on the power being well coupled to electrons, are important research tools for plasma control and are of interest for ITER. However, parasitic absorption by ions (deuterium, tritium, and alpha particles) may degrade the efficiency of these techniques. Deuterium plasma experiments on JET have indicated that significant parasitic absorption by third harmonic D occurs, though the actual power split between D ions and electrons remains an open question. The DIII-D program will continue to study parasitic absorption at harmonic D resonances during FWCD experiments. The installation of a new launcher based on folded waveguide technology on TFTR will allow operation in a FWEH regime where parasitic absorption effects by both D and T harmonics can be explored experimentally. Theoretical studies suggest it may be possible to avoid these parasitic absorption effects by driving current at frequencies below the fundamental cyclotron frequency of any ion species in the plasma. Alcator C-Mod, and perhaps TFTR, will explore this possibility in order to assess its viability for use in ITER.

Efficient heating and current drive via mode conversion of fast waves, launched from the outer edge of the discharge, to ion Bernstein waves (IBW) has been demonstrated on TFTR and Alcator C-Mod, with results that agree reasonably well with one-dimensional ICRF modeling code predictions. This may be a useful technique for localized electron heating and current-profile control, especially for D-T operation, since efficient mode conversion requires high concentrations of two different ion species. However, some of the basics physics issues can be studied in D-^{3}He, H-^{3}He or D-H plasmas. The TFTR will use its capability to sweep the D-T ion-ion hybrid layer (where the mode conversion occurs) from the inner edge to the magnetic axis to study the localization of mode-conversion heating (MCH) and current drive (MCCD) in D-T plasmas. A new four-strap antenna to be installed this fiscal year will provide more control over the launched wave spectrum for current drive. Cooling of the alpha population by mode-converted waves will be tested as part of an exploration of the concept of alpha-channeling. Alcator C-Mod will study mode conversion at high plasma density using different ion species, using its capabilities to vary RF source frequency (when upgrades are completed) and toroidal magnetic fields over wide ranges. On both TFTR and Alcator C-Mod, MCCD will provide the only off-axis noninductive current drive system in the next two years. On DIII-D, while the possibilities for MCCD are restricted by limits on its magnetic field and source frequency, accessible scenarios exist in either D-H or H-^{3}He plasmas. Modeling predictions of the mode conversion efficiency for these scenarios are somewhat discouraging, but their exploration on DIII-D (as well as on Alcator C-Mod) will provide key tests of the existing models.

It should be remarked that initial theoretical studies have indicated that mode conversion may become negligible in large, hot, and dense devices such as ITER. However, reasonable efficiencies are predicted for the more compact sizes envisioned for advanced tokamak reactors. It should also be noted that existing two dimensional ICRF field codes often yield power deposition profiles that differ significantly from experimentally measured profiles in strong mode conversion regimes. Continued theoretical effort in these areas is needed.

Direct-launch ion Bernstein wave (IBW) heating was found to induce internal transport barriers in the PBX-M device. Theoretical analysis is consistent with the observations and indicates that the process scales favorably to reactors, requiring only about 10 MW of power in an ITER-class plasma. An IBW launcher will be installed in TFTR this fiscal year to further investigate this effect and, if successful, to use it for enhancing performance. Localization of the power deposition is critical, since the transport barrier formation is believed to be driven by highly localized gradients in the wave electric fields. Since alpha particles are co-resonant with D ions, barrier formation driven by IBW absorption at a D resonance may be seriously degraded by absorption on energetic alphas. TFTR will explore these effects by comparing barrier formation at T harmonic resonances against barrier formation at D harmonic resonances in DT plasmas.

An important physics area related to efficient coupling of ICRF or IBW waves to the core plasma is the study of antenna-plasma interactions including ponderomotive force effects, RF sheath formation, parametric instabilities, scattering from edge fluctuations, and coupling to parasitic surface waves. These issues impact the accessibility of the launched wave to the core plasma, the dissipation of power in the edge plasma, the scrape-off layer (SOL) density profile, and the generation of impurities. Important contributions to the understanding of antenna-plasma interactions have been made by all three U.S. machines and their predecessors. Valuable future contributions would include 1) expanding the database of measurements of RF-induced SOL density profile modifications using antenna-mounted reflectometers on DIII-D (for FWH and FWCD) and TFTR (for IBWH) in a variety of RF scenarios, 2) quantifying the effect of boron wall conditioning in reducing RF-specific high-Z impurities on Alcator C-Mod, and 3) comparing the edge interactions of conventional antennas with those of the planned folded waveguide launcher on TFTR.

The ICRF provides a wide variety of possible plasma control scenarios, as well as numerous scientific issues. Table II.C.1 provides a summary of the ICRF research plans for the next two years in the major U.S. tokamak facilities.

Table II.C.1. Summary of ICRF research plans for the next two years. Our priority assessments are denoted as High (H) or Medium (M).

TFTR	**DIII-D**	**Alcator C-Mod**
• MCCD for off-axis current-profile control in D-T ERS scenarios. • MCH, IBW to explore alpha channeling possibilities. • ITER-relevant heating scenarios in D-T plasmas. • Direct-launch IBW for transport barrier control. **H**	• FWCD for q(0) control to sustain AT scenarios. • Effects of parasitic absorption effects by D on FWCD. **H**	• ICRF heating and current drive at high density: – efficiency – H-mode power threshold and confinement time scaling • MCH / MCCD for profile control and sustainment of PEP modes. • FWCD at $\omega < \omega_{ci}$ to determine viability. **H**
• FWCD at $\omega < \omega_{ci}$ to determine viability. • Effects of parasitic absorption by D and T on FWEH. **M**	• MCH / MCCD in H-^{3}He and D-H plasmas to determine viability for off-axis control. **M**	• Parasitic absorption by shear Alfvén mode conversion and ion resonances. • Wave propagation and damping in the core using phase contrast imaging **M**

Electron Cyclotron Range of Frequencies (ECRF) Waves

Electron cyclotron resonance heating (ECH) and current drive (ECCD) have been successfully demonstrated on a number of tokamaks at modest power levels in low density plasmas with low toroidal magnetic fields. Theoretical calculations of heating efficiency based on ray tracing and Fokker-Planck techniques are in agreement with the experimental observations. An important remaining issue for ECCD is the effect of electron trapping in the magnetic field well on the current drive efficiency. Some of the key characteristics of ECH and ECCD which may make it favorable for reactors include controllable, highly localized strong single pass absorption and high power density compact launchers which need not be located very close to the plasma edge. In the near term, localized electron heating, in particular, may help in distinguishing between electron and ion transport channels in different operating regimes, and localized current-profile control may be able to increase the stability limits and duration of advanced-tokamak scenarios.

Progress in ECH and ECCD has been paced by the development of the required high-power microwave technologies: sources, transmission systems, and windows. The first 3 MW of a planned 6-MW, 110-GHz system is currently being installed on DIII-D and will be completed by March 1997. This will be the primary off-axis current drive system on DIII-D and is therefore critical for DIII-D’s advanced-tokamak program. The 3-MW system should be fully exploited as a high priority on DIII-D; if successful, purchase of the additional 3 MW should be considered.

Lower Hybrid Range of Frequencies (LHRF) Waves

Efficient off-axis noninductive current drive via lower hybrid waves (LHCD) has been demonstrated experimentally on a number of tokamaks and is reasonably well-supported by theory and modeling. It is under consideration for use on ITER, in order to enable advanced tokamak operations on that device, and is being pursued on JET, JT-60U, Tore Supra, and other tokamaks. Outstanding scientific and technical issues include developing an understanding of the observed efficient current drive despite the gap between phase velocity of launched waves and maxwellian distribution of electrons in plasmas, of the role played by the ohmically driven electric field in current drive efficiency, of synergisms with other RF heating and current drive schemes, of effects due to interactions with fusion-generated alpha particles, and design of couplers appropriate to fusion reactors. For the next two years, there will be a minimal lower hybrid wave physics program in the U.S., due to budgetary constraints which have forced the cancellation of the LHCD program on TFTR and a delay of the proposed program on Alcator C-Mod. In the longer term, Alcator C-Mod will study off-axis current profile control with pulse times longer than either the skin time or the current diffusion time, thereby providing information on steady state current profile control issues.

Pellet Injection

Pellet injection deposits the fuel inside the plasma boundary and is therefore more efficient than gas injection and offers better localization control. Turbulent transport can be reduced by control of particle input, either by pellet injection or by neutral beam fueling (already discussed). Pellet injection rapidly steepens the density gradient, thus:

a) increasing the gradient in the radial electric field, and enhancing electric-field shear stabilization, and
b) reducing the drive for ion-temperature-gradient modes.

Both of these act to quench anomalous transport. Indeed, perhaps the first example of a "core transport barrier" was the post-pellet Alcator-C plasma, in which impurity transport was reduced to the neoclassical level, leading to particle accumulation and increased radiation losses. The JET Pellet-Enhanced Performance (PEP) mode was an early example of a reverse shear regime. Pellet injection, combined with off-axis current drive, such as provided by mode conversion processes or electron cyclotron waves, provides unique opportunities for studying the dynamics of purely RF-supported transport barriers. Alcator C-Mod, with its capability of operating with pulse lengths longer than the current relaxation time, will be particularly well-suited for these studies. Pellet injection can also be used to lower power thresholds for transport bifurcation, as recently demonstrated on TFTR with Li pellet injection. All three of the major facilities have pellet injectors and actively pursue programs which utilize them. TFTR is primarily concerned with exploiting pellets to control density profiles and to enhance access to Enhanced Reversed Shear regimes. DIII-D will use pellet injection to evaluate and understand density limits near the Greenwald limit. They will use pellet fueling in conjunction with divertor pumping to control the density profile and evaluate its effect on transport and the bootstrap-current profile. Alcator C-Mod is actively pursuing ICRF-sustained PEP regimes, and DIII-D and TFTR could contribute to this area using ECCD and MCCD, respectively.

Compact Toroid Injection

Injection of compact toroids (CT) offers a possible means of achieving core fueling in hot, dense burning plasmas, such as those anticipated for ITER. Simple theoretical estimates indicate that a CT will penetrate into the plasma until all of its kinetic energy density has been used to displace the local magnetic field energy density. Experiments on the Tokamak de Varennes and the Davis Diverted Tokamak have demonstrated fueling up to 30% of the total plasma density with no impurity pollution. Further development of the technique requires installation of CT injectors on large tokamaks with higher magnetic fields and, ultimately, in a burning-plasma device. The ITER program has, very recently, called for CT injection fueling experiments to explore the compatibility of operation above the Greenwald limit with H-mode confinement.

Actions and Priorities

The panel recommends the following actions as high priority to provide a strong research program in heating, current drive, and fueling in the next two years. We note that the priorities in this area are to a large extent driven by plasma control needs for research in transport, MHD, alpha physics, and scenario integration.

Neutral Beam Injection (NBI)

- Increased collaboration with JT-60U on negative-ion-based NBI experiments; comparison with positive-ion results in U.S. tokamaks.

Ion Cyclotron Range of Frequencies (ICRF) Waves

- Exploitation of TFTR's unique capabilities for studying ICRF in DT plasmas, and its effects, prior to shutdown. Installation and exploitation of the TFTR ion Bernstein wave antenna to drive rotational shear flow control transport barriers is especially critical.
- Exploitation of DIII-D's fast-wave current-drive system to control the on-axis current-density profile in negative central shear regimes.
- Completion of the planned 8-MW ICRF system on Alcator C-Mod, including a four-strap antenna in collaboration with PPPL, to provide basic heating and current-drive capabilities for long-term advanced-tokamak research.
- Testing of antenna designs and the study of antenna-plasma interactions on all three tokamaks to provide information needed for the development of launchers for the next generation of tokamaks, such as ITER

Electron Cyclotron Range of Frequencies (ECRF) Waves

- Installation and exploitation of DIII-D's 3-MW, 110-GHz ECH/ECCD system. This is critical to the advanced tokamak program on DIII-D and has the potential to make unique contributions to the world fusion program. Success at the 3-MW level should be demonstrated before purchasing the additional 3-MW upgrade.

Lower Hybrid Range of Frequencies (LHRF) Waves

- Collaboration with JET, JT-60U, and Tore Supra until implementation of the planned LHCD system on Alcator C-Mod.

Pellet Injection

- Exploitation of the pellet injection capabilities of all three tokamaks for fueling, wall conditioning, and transport control. Pellet fueling is a strength of the U.S. fusion program and should continue to play an important role in tokamak concept improvement.

D. Divertors, Boundary Physics, and Plasma-Wall Interactions

Scientific Issues

The major research goal in this area is to understand the scientific issues associated with power and particle exhaust in tokamaks. Successful power and particle exhaust involves achieving plasma conditions at the tokamak edge and in the divertor which ensure that 1) the peak power flux on the plasma facing components is within acceptable levels (less than ~5 MW/m^2); 2) the energetic ion and neutral fluxes on the plasma facing components are low enough to avoid excessive erosion of the components; and 3) the neutral pressure (especially He) in the pumping ducts is sufficiently high that adequate particle removal rates are achieved, the core density is controlled, and impurities are removed. These conditions must be achieved simultaneously with the required levels of core plasma performance. In particular, good energy confinement must be maintained and impurity contamination must be within acceptable limits. The plasma and neutral-particle densities at the edge are critical parameters for compatibility of good core performance and a successful exhaust scenario.

The main concept for power exhaust using divertors is to use atomic processes 1) to reduce the peak heat and particle fluxes to the divertor target structures by transferring most of the power losses (both steady state and transient, e.g. ELM’s) from the edge plasma to the main chamber walls and divertor chamber side-walls and 2) to localize the recycling in the divertor to increase the neutral density there. The relevant mechanisms are 1) line radiation from hydrogen, from intrinsic impurities such as carbon, and from injected impurities such as Ne or Ar; 2) bremsstrahlung radiation from the main plasma; and 3) charge exchange and ionization.

The scientific issues associated with achieving these conditions include (Figure II.D.1) :

- Parallel and perpendicular transport of ion energy, momentum, and particles, both in the main plasma and in the scrape-off layer (SOL). Included are classical and anomalous processes, drifts, SOL currents, error fields, and enhanced transport near the X-point.
- Neutral atom and molecular transport in complex geometries. Included are ion-neutral and neutral-neutral collisions, both elastic and inelastic. Also wall collision mechanisms, including reflection, thermalization, recombination, absorption, and desorption.
- Plasma-wall and neutral-wall interaction processes. Included are physical sputtering, chemical reactions, radiation enhanced sublimation, erosion, and re-deposition.
- Impurity atomic physics. Included are ionization, recombination, and electron impact excitation collisions; enhancements due to rapid transport; and charge-exchange recombination.

- Atomic processes involving hydrogenic atoms and molecules. Included are charge-exchange; collisional radiative effects on ionization, excitation, and recombination; molecular processes (including vibrational excitation); and radiation transport recombination.

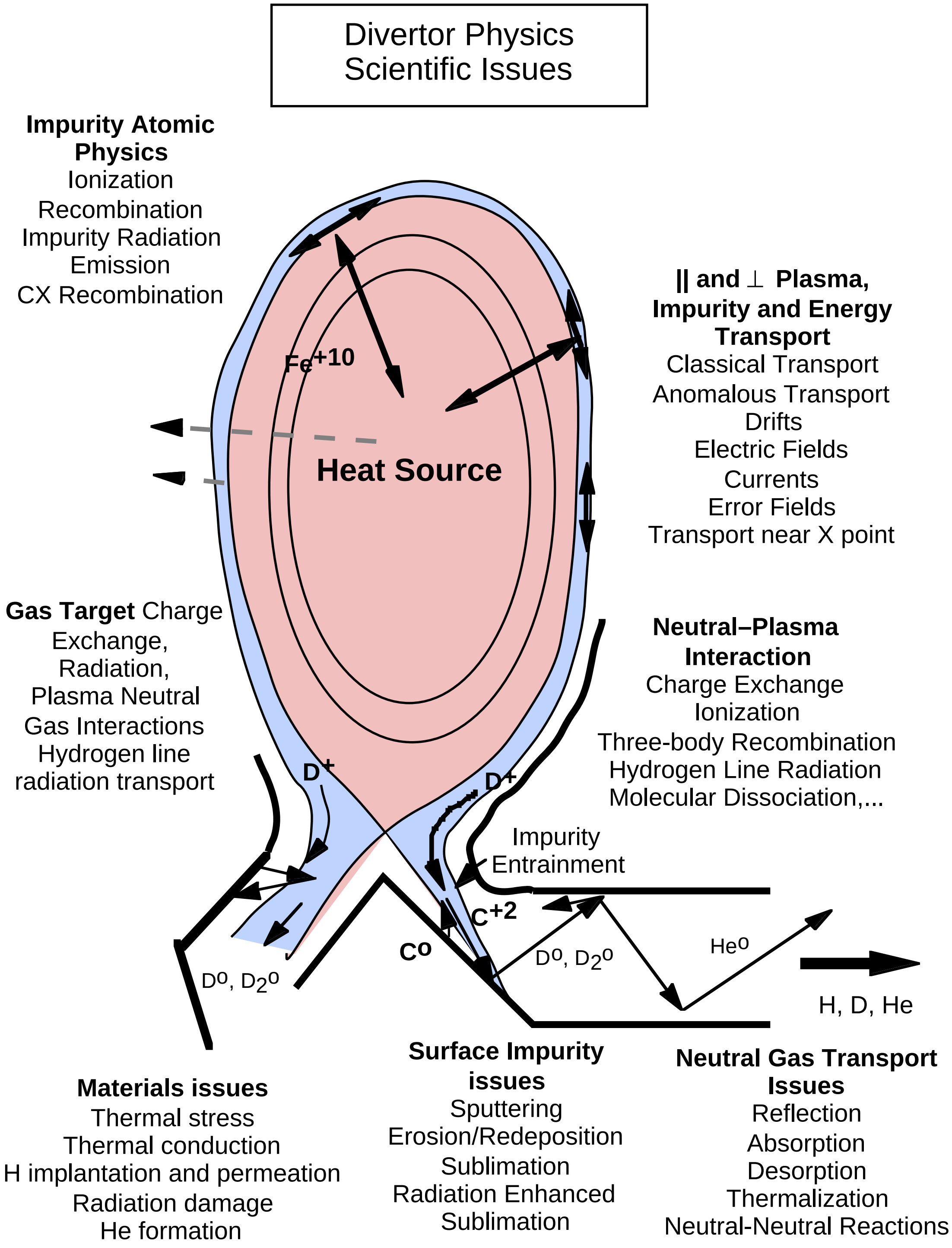

Figure II.D.1. Schematic illustration of divertor physics scientific issues

Since the radial decay length of the power channel is short, the plasma volume available for radiating in the divertor is relatively small. The impurity emissivity thus needs to be made very large while keeping the impurity contamination of the main plasma to a minimum. The emissivity can be enhanced in several ways:

- By increasing the impurity concentration in the divertor plasma. This is done by controlling the recycling, fueling and pumping and relying on transport processes to modify the particle flows.
- By charge-exchange recombination and impurity recycling.
- By achieving high density, low temperature regions in the edge and divertor plasma.

If the radiation losses are sufficiently high, and the resulting heat flux to the divertor targets sufficiently low, then charge exchange collisions, volume recombination, and radial diffusion will combine to reduce the plasma pressure at the divertor plate and cause it to “detach”. In this regime, the heat loads and particle flux on the plates can be lowered by an order of magnitude or more, thus reducing both the peak heat flux and the erosion rate. The achievement of these conditions is strongly dependent on the divertor configuration, the peak power fluxes, the impurity levels, and the plasma edge conditions.

Proper wall conditioning is an important element for achieving good plasma performance. Wall conditioning can reduce the hydrogen recycling and the impurity levels, especially oxygen. Conditioning methods include glow discharge cleaning, electron cyclotron resonance discharge cleaning, and the introduction of chemical getters and coatings.

Particle control is accomplished by fueling with gas puffing and/or pellet injection and pumping in the divertor chamber. Wall pumping is also important in present experiments, particularly with graphite plasma facing components. The divertor concentrates the particle fluxes, so that the neutral pressure is increased in the divertor chamber resulting in enhanced particle removal rates for a given pumping system. Because of the deleterious effects on confinement of high neutral pressures in the main chamber, the high neutral pressure region should be confined to the divertor chamber.

Finally, achieving high levels of plasma performance has always gone hand-in-hand with advances in controlling the plasma-wall interactions. For example, He glow discharge cleaning and divertor pumping have aided the achievement of advanced confinement regimes on DIII-D by providing better control of the density and recycling. Lithium pellet conditioning has been a strong element in the achievement of enhanced reverse shear discharges on TFTR. Recent implementation of boronization on Alcator C-Mod has resulted in substantial improvements in plasma performance. Progress in controlling plasma-wall interactions is thus critical for being able to study transport processes, energetic particle physics, MHD stability, and other issues requiring high performance plasmas.

Status of Understanding

Recognition of the importance of power and particle control issues to present experiments and to ignition physics experiments such as ITER has led to a strong emphasis on these issues in the U.S. fusion program. Both the DIII-D and Alcator C-Mod have divertors, differing in materials and configuration. An in-situ pump is installed on DIII-D and one is planned for Alcator C-Mod. Both facilities now have extensive divertor diagnostic capabilities, including some important recent upgrades in the case of DIII-D. Further diagnostic upgrades are planned (especially in Alcator C-Mod), as are divertor structure modifications (especially in DIII-D). Complementing the extensive measurement capabilities that are now available are sophisticated computational modeling capabilities for analyzing the divertor data. The impurity transport is analyzed with Monte Carlo impurity transport codes. The divertor and SOL parameters are analyzed with multi-species fluid codes such UEDGE and Monte Carlo neutral transport codes such as DEGAS. As a result of these advances in experimental and computational capabilities, substantial progress has been made in the understanding of the scientific issues.

Experiments on DIII-D and Alcator C-Mod have demonstrated that the peak power levels in the divertor can be reduced by impurity radiation from the edge and divertor plasma, and that localized recycling can result in very large neutral pressures in the divertor chamber. The planned experimental programs are oriented toward developing a detailed understanding of the mechanisms involved in this type of divertor operation and developing the plasma control capability needed to achieve such conditions simultaneous with high levels of central plasma performance.

Experiments on DIII-D and Alcator C-Mod have studied and characterized the roles of 1) impurity radiation with either intrinsic impurities such as graphite or seeded impurities such as neon, and 2) momentum and particle losses due to charge exchange, volume recombination, and radial diffusion in “detached” and “partially detached” operation. In particular, the prediction that gradients in the plasma pressure along the field lines are important signatures for detachment has been verified. Both facilities have successfully achieved conditions with high levels of radiation losses in the divertor and SOL and with low peak power fluxes on the divertor plates. Thomson scattering measurements on DIII-D and probe measurements from Alcator C-Mod indicate that the electron temperature is around 1 eV in detached plasmas, implying that volume recombination processes can play a large role in the particle balance. Spectroscopic measurements from Alcator C-Mod have demonstrated the existence of volume recombination.

Detailed measurements of the plasma parameters and heat fluxes on the divertor plates in DIII-D and Alcator C-Mod have allowed an initial characterization of divertor operating regimes with different recycling levels. Both have identified three such regimes: sheath-limited, high-recycling and detached. The variation of the divertor plasma behavior with the grad-B direction, divertor geometry and configuration, density, heating power, impurity level, pumping speed, and fueling rate are being studied. Experiments on Alcator C-Mod comparing three different divertor configurations have demonstrated that detachment can be achieved with vertical target and slot geometries with densities one-half of that required in an "open" geometry. Substantial neutral compression in the private flux region due to "reflection" of neutrals from the cold plasma near the separatrix was another feature of high density, vertical target plate divertors in Alcator C-Mod. Studies of the transport with Ohmic heating and with up to 3.5 MW of ICRF heating show that the radial power decay length decreases with auxiliary heating power. High to very high neutral pressures (up to 100 mTorr in the Alcator C-Mod vertical target divertor chamber) in the divertor chamber have also been achieved, verifying the importance of localized recycling predicted by divertor modeling codes. The DIII-D group has also extensively documented and characterized several divertor operating regimes which have enough detailed information to test and validate the divertor modeling codes.

Transient heat load effects such as those due to edge localized modes (ELMs) are being characterized and studied. A database of ELM characteristics is being assembled and analyzed. Classification of the different types of ELM's is being done, and simple theories and models are being constructed. Further discussion is provided in Sections II.A and II.B.

The data from both divertor experiments is being modeled using 2 D divertor modeling codes. The agreement between the codes is improving as more data is collected and analyzed. The recent addition of two dimensional profiles of the density and temperature, and of the impurity emission, has been especially important for testing the models and suggesting improvements.

Both experiments are well coupled to the international program in divertor physics. This increases the effectiveness and efficiency of the DIII-D and Alcator C-Mod programs because they are able to share and compare results and codes, and take advantage of developments on the other experiments. DIII-D has strong bi-lateral connections to JET and ASDEX Upgrade, and Alcator C-Mod has a strong connection to JET. Both experiments are very active participants in the divertor physics research and development (R&D) program coordinated by the ITER Divertor Physics Expert Group and the ITER Divertor Modeling and Database Expert Group.

Plans

The diagnostic upgrades recently installed on or planned for both U.S. divertor experiments and the improved modeling capability offer the opportunity to greatly improve our understanding of divertor plasma behavior and the underlying mechanisms.

DIII-D Plans

The DIII-D group will continue to emphasize the study and analysis of divertor plasmas to develop a physics understanding of divertor operation and embody that understanding in a divertor modeling code. They will study plasma and energy transport in the SOL and divertor plasma, impurity transport, radiation losses, the physics of detachment and momentum removal, the role of recombination and molecular physics, the role of pumping and fueling in affecting plasma transport, and He exhaust physics. The three main issues to be studied for the next two years are:

a) Radiative Divertor Physics: study impurity enrichment, impurity transport, and detachment.
   i) Enrichment of impurity concentration in the divertor relative to the core, including He transport studies.
   ii) Understand physics determining the optimum geometry for simultaneous radiative divertor and high-performance core.
   iii) Understand the physics of divertor/SOL heat/particle flows, and control them to enhance radiative divertor performance.
   iv) Understand the physics of highly-radiating (“detached”) regimes and control for maximum emissivity and radiating volume.
b) Scrape-off Layer Transport Physics and Model Validation: Obtain sufficient SOL and divertor measurements to test the capability of numerical models for predicting SOL width and plasma conditions at the divertor target.
c) Plasma-Wall Interactions: Obtain sufficient measurements on erosion processes to benchmark numerical models for divertor erosion rates.

A new “radiative divertor” structure designed to operate with the high-triangularity plasma shapes favored for core-plasma performance will be installed in DIII-D in two stages. The new divertor configuration includes baffles and cryopumps on both the inner and outer “legs”, and should be much better for confining the neutral recycling within the divertor chamber, and for controlling the recycling in the divertor and in the main plasma, the plasma density in the SOL, and the particle flows in the SOL. It should also provide for tests of the compatibility of good divertor operation with enhanced core confinement regimes.

The first stage of the upgrade (planned to be completed by March 1997) involves replacing the upper divertor structure only. Installation of the lower structure and the full diagnostic set for the new divertor will be delayed until the second stage (Feb. 1999). In the meantime, the full diagnostic set for the lower divertor will be retained to allow continued study of divertor physics with the present configuration. This is essential to accomplish the research plans described above. Performance issues and some new divertor studies will be possible after the first stage installation, but detailed studies of the new configuration will be postponed until 1999. The full upgrade was originally planned for 1997, but has been postponed due to funding limitations.

Alcator C-Mod Plans

The Alcator C-Mod group will study hydrogen and impurity transport in the divertor plasma. They will vary the divertor geometry, impurity level, heating power, plasma density and grad-B direction. They will continue to study the physics of detached and partially detached operation; the role of momentum removal by charge-exchange, volume recombination and radial diffusion; radiation transport; and the role of neutral-neutral and neutral-plasma interactions. While simple analytic models will be applied to interpret the data locally, collaborations with in-depth 2-D numerical modeling activities will be used to benchmark codes at these ITER-like divertor densities and power loading conditions. The three main issues to be studied for the next two years are:

a) Understand underlying parallel and perpendicular transport physics controlling SOL profiles of parallel power flow, density, and temperature.
b) Further understand the physics of a dissipative divertor through
    i) Determining whether recombination plays an important role for either ion current or momentum loss. Understand which recombination paths are important.
    ii) Comparison of the maximum allowable divertor volumetric loss rates with that predicted by non-coronal radiation and parallel transport. Determine whether other loss paths (e.g., charge-exchange) are important.
    iii) Development of impurity feedback techniques to optimize divertor losses and minimize core dilution.
c) Understand the relative importance of different aspects of impurity transport (source location, thermal vs. friction forces, etc.) in setting the relative levels of impurities in the core and divertor.

Three classes of diagnostic upgrades relevant for divertor physics studies are planned by Alcator C-Mod in the near term:

- Energy transport studies: probes for fluctuation measurements, divertor Thomson scattering, and diagnostic neutral beam (DNB) for edge ion temperature measurement.
- Impurity transport studies: divertor residual gas analyzer (RGA), Omegatron, SOL spectroscopy, and diagnostic neutral beam (DNB) for impurity density measurements
- Dissipative divertor studies: low-energy neutral analyzer, visible and ultraviolet spectroscopy.

Installation of a divertor cryopump is planned for mid-1998, though a partial installation might be done a little sooner. The pump will be useful for controlling the density, and essential for studying issues associated with He and impurity removal. It will add greatly to the flexibility of machine operation. An earlier installation was planned but funding limitations have forced a delay in the pump installation.

In addition, there are plans to modify the baffle for the lower divertor on the inner wall. This is to permit greater plasma shape flexibility.

In the longer term, a planned upgrade to 8-MW of ICRF heating will increase the parallel power flow and thus is important for divertor studies.

TFTR Plans

TFTR achieves better confinement with Li conditioning, but the mechanisms for this are not well characterized or understood. TFTR plans to devote run time to studying and characterizing the relationships among Li pellet injection, wall recycling, impurity levels, transport, and plasma performance.

The Li pellet injector will be upgraded to increase the number of pellets. Laser ablation of Li will also be investigated.

Actions and Priorities

Both DIII-D and Alcator C-Mod should proceed with their presently planned program, aimed at the development of a radiative divertor concept and an understanding of the physics involved with divertor operation. Under constrained budgets, however, some tradeoffs between high-priority items may be necessary and we discuss these in Section V.

DIII-D

- Given the reduction in funds for the radiative divertor upgrade, the DIII-D group plan for proceeding with the upgrade is a good compromise between getting as much of the upgrade installed as early as possible and maintaining their ability to study divertor physics issues with their present excellent diagnostic capability. The upgrade should be given a high priority.
- The DIII-D group should proceed with their plan to keep the present divertor diagnostics installed until it is feasible to replace them with the full diagnostic set for the radiative divertor. The group should exploit the present diagnostic capability, while proceeding with the planned diagnostics upgrade for the radiative divertor as a high priority.
- The DIII-D group should continue with their code development and validation program. In particular, they should continue development of a coupled 2-D fluid-Monte Carlo neutrals code (e.g., UEDGE coupled with DEGAS or EIRENE).
- The DIII-D group should consider ways to increase the run time devoted to divertor studies.

Alcator C-Mod

- The Alcator C-Mod group should proceed with their planned diagnostic upgrades, some of which rely on a proposed diagnostic neutral beam, as expeditiously as possible.
- The Alcator C-Mod group should make a strong effort to install the cryopump as early as possible. The upgrade of ICRF power to 8 MW will allow divertor studies at higher powers (similar to ITER power levels) and is important. These are candidates for additional funds if they become available.
- The Alcator C-Mod group should promote a closer interaction with divertor modeling groups. The present strategy of filling this need with collaborations is reasonable, but the level of activity in this area needs to be increased. This type of effort will be particularly important as more detailed data become available.

TFTR

- The proposed effort on TFTR to identify the mechanisms responsible for the beneficial effects of Li pellet injection and conditioning should continue.

E. Alpha and Fast-Particle Physics

Scientific Issues

Understanding the physics of burning plasmas, that is plasmas heated mainly by fusion alpha particles, is a critical goal for establishing the feasibility of fusion reactors. Though they will only be fully addressed in the next generation devices, many alpha physics issues can be studied in the present generation of devices, in particular TFTR. The main scientific issues in alpha-particle physics are:

1. The interaction of the alpha particles with intrinsic plasma instabilities. Such interactions will ultimately determine the effectiveness of alpha-particle heating in a reactor, such as ITER. The interactions of alpha particles with low mode-number MHD activity are of continuing interest since such activity occurs occasionally in all tokamaks and alpha losses caused by such modes could affect ignition margins and, if spatially localized, pose a threat to first wall components. One major goal is the development of a model to explain the spatial and energetic redistribution of alphas observed during sawtooth relaxations in TFTR.

2) Alpha-particle-driven instabilities. Experiments will also be conducted to evaluate the stability of the alpha-driven toroidal Alfvén eigenmode (TAE) in the core of plasmas with weak shear. Although a purely alpha-driven TAE has not yet been observed in D-T plasmas, the core TAE is predicted to be driven unstable by alpha particles in certain weak-shear D-T plasmas in TFTR. These experiments will provide an important test for the predictive capability of existing theory, which has so far been validated only for TAEs excited by externally driven fast ions. The experiments simulating alphas by externally driven fast ions have been very useful in exploring different regimes of alpha particle instabilities.

3) The interaction of the alphas with externally driven waves. Experiments on TFTR will attempt to demonstrate the cooling of alpha-particles through their interaction with the IBW. This is a crucial element in developing the physics of alpha-particle channeling, i.e., the ability to control the flow of power from energetic alphas to the fuel ions. In a reactor, alpha channeling could permit more precise control of plasma profiles and better utilization of plasma pressure in the fuel ion component. The scientific issue is to check the model which predicts cooling and radial transport of alphas due to IBW interaction. Experiments to date have shown a strong interaction between energetic beam ions and IBW waves generated in D-^{3}He plasmas. Such an interaction was predicted by alpha channeling theory. Using newly commissioned capabilities to generate the IBW by mode conversion in TFTR D-T plasmas, it will be

possible to examine interactions of the fusion alphas using the full array of diagnostics for both lost and confined alpha particles.

4) Magnetic field ripple effects on alpha particle confinement. Experimental results from TFTR have shown that alpha particles are well confined as expected from classical modeling of alpha particle orbits and collision thermalization. Using scintillator detectors mounted on the vessel wall, it has been possible to study in detail the alpha losses. The alpha loss to the vessel bottom was identified with classical first orbit loss and the loss to the outer mid plane with toroidal field ripple loss.

The theory development in alpha particle physics has anticipated many of these issues with clear predictions for transport losses and instability generation. The research in this area has been one of the most fruitful in terms of comparisons between theory and experiment.

Plans

The alpha and fast-particle physics program is centered at TFTR with its unique DT capabilities. No program plan in this area was presented by the other facilities, although within their proposed research they may encounter regimes with important fast-particle driven instabilities. The TFTR plan is focused on the compatibility of advanced-tokamak regimes with self-heated plasmas. The main scientific goals for the next two years are:

1) Characterize the interaction of alpha particles with ion Bernstein waves (IBW). This has a double scientific objective:
   i) to determine the relevance of IBW for internal barrier control in an alpha-particle environment.
   ii) to develop the physics basis for the alpha-channeling scheme.

   These experiments rely on the installation of an IBW antenna to be complete this fiscal year. The barrier control experiments also rely on a poloidal rotation diagnostic being installed in the same time frame.

2) Evaluate the response of the plasma to alpha-heating in advanced-tokamak regimes (high-l_i and/or reverse shear plasmas) at or beyond the power levels established in Super Shots (fusion power >10 MW).

   Here the basic goal is to test MHD effects on the alpha distribution and the compatibility of advanced-tokamak regimes with good alpha confinement. These experiments will provide further tests of the toroidal Alfvén eigenmode (TAE) theory at weakly reversed magnetic shear, particularly after the turn-off of NBI.

   Evidence for alpha particle heating has been observed in TFTR, but the effect is small. Experiments should continue in order to demonstrate significant alpha particle heating and confirm the basic heating mechanism for a burning-plasma such as that in ITER, and to discover any possible differences between alpha particle heating and auxiliary heating.

3) Evaluate alpha confinement and stability in advanced-tokamak regimes with alpha particle parameters comparable to present projections for ITER.

   The experiments would be aimed at characterizing alpha-particle effects on the stability of MHD modes, including TAEs, sawteeth, and low-n modes, in these regimes. They will also study toroidal-field ripple loss, which may be large in the ITER reversed shear scenario, and the redistribution of alphas by sawteeth in reverse shear plasmas. The sensitivity of the TAE stability to profiles, including the q-profile, will also be investigated. Also, benchmarking theoretical codes should be continued.

Priorities

Establishing the compatibility of the internal transport barrier with alpha confinement is the highest priority, together with the goal to complete the study of alpha particle instability at high performance. Another priority is to establish the active cooling and radial transport of alpha particles as part of an evaluation of alpha particle-wave interaction in support of the concept of alpha-channeling. Finally it is important to evaluate helium ash buildup in the core of enhanced reverse-shear D-T plasmas. In DIII-D, some priority should be given to the study of fast particle instabilities in the weak shear regime.

Actions

The proposed program is well focused and directed to a central issue and scientific niche of the U.S. fusion program, namely the compatibility of the advanced tokamak regime with alpha heating. To ensure success, the following actions are recommended:

1) Install and exploit the IBW launcher and poloidal rotation diagnostic as soon as possible in TFTR. Make available sufficient run time on TFTR to characterize alpha particle behavior in the advanced tokamak regimes.
2) Give some priority to the study of fast particle instabilities in the weak shear regime in DIII-D.

F. Advanced-Tokamak Scenarios and Integration

Scientific Issues

The concept improvement goals for tokamaks are to establish the scientific foundations for steady-state high power density operation, with a low frequency of disruption, and high power density operation in a burning-plasma environment. In the long term, these improvements could significantly enhance the attractiveness of tokamaks, by making them less expensive and more reliable. Steady state operation is expected to be achieved only in discharges with significant bootstrap current. Significant reductions in the size of the tokamak power plant core, the capital cost of the reactor core, and the net cost of electricity requires, in addition, that both the confinement time and the beta limit be increased relative to standard-performance scalings. For example, energy confinement times greater than H-mode (approximately two times ITER-89P) scaling and normalized beta (β_N, defined in Sect. II.B) exceeding ~3 %-m-T/MA are indicative of improved, or so-called "advanced-tokamak" performance. We would add to these conditions effective steady state heat and particle exhaust, and steady-state power deposition consistent with alpha particle self-heating.

To achieve these conditions requires a complete integration of the physics discussed in each of the previous sections: A)transport control; B)MHD equilibrium and stability control; C)heating, current drive, and fueling; D)divertors, boundary physics, and plasma wall interactions; and E)alpha physics. The highly nonlinear coupling among these elements makes their integration a scientific challenge. To date, advanced-tokamak regimes have been studied transiently. To be useful in future ignition devices and/or power plants, they must be shown to be compatible with steady state operation; hence their duration must be increased substantially.

The high-priority scientific issues are:

1) Stability at very high beta, including non-ideal effects; resistive wall stabilization, passive and active; double tearing modes, resistive interchange modes, infernal modes, etc.
2) Compatibility of current and pressure profiles of improved confinement regimes with high beta stability;
3) Alignment of the bootstrap current profile, and its control via density and pressure profile control. The bootstrap current profile must be similar in shape to that of the total current, otherwise external current drive requirements will be excessive.
4) Control of the pressure profile through a)transport and transport barrier control, b)heating power deposition control (including alpha channeling), c)control of the current profile and d)radiative heat loss.
5) Control of the current density profile, especially through off-axis current drive.

6) The impact of plasma shape on the advanced regimes; especially the relationship between strong shaping and the shape of the pressure profile.
7) Compatibility of current and pressure control schemes with high power density and D-T operation.
8) Impurity and He transport in advanced regimes.
9) Compatibility of high performance regimes with effective heat and particle exhaust. For example, low density operation may be preferred for current drive and current profile control, but high edge and divertor densities are preferred for heat removal, impurity shielding, He ash removal , and particle pumping.

Status of Understanding

By 1992, a number of tokamaks had experimentally observed improved confinement regimes with confinement significantly above that given by H-mode confinement scalings. As was clearly illustrated in the presentations at the 1992 IAEA Conference, the performance in these high confinement regimes was limited by stability at high beta. More recently, the fusion power output in the TFTR Super Shot regime was shown to be stability-limited. To make use of improved confinement regimes, it is necessary to identify improved stability regimes that are compatible with them.

Two advanced regimes have been identified as potentially useful for steady-state high-performance operation: 1) the Negative Central Shear (NCS) or Enhanced Reverse Shear (ERS) regime, and 2) the high internal inductance (l_i) regime. The studies indicate that plasmas with bootstrap-aligned profiles require wall stabilization of the external kink to achieve the highest beta values. In addition, self consistent steady state scenarios for both the reverse shear and the high-l_i regimes require broad pressure profiles.

In the high-l_i scenarios, the current is concentrated in the core and large magnetic shear is located in the outer portion of the plasma. Both theory and experiment show that the maximum stable beta scales in proportion to the internal inductance parameter l_i, which is a measure of current-profile peaked-ness. In the experiment, the high beta values are obtained with relatively broad pressure profiles, consistent with the theoretical expectations. Experimentally, this scenario is consistent with both high beta and high confinement; the issue, however, is consistency with steady state, i.e., bootstrap current alignment. High-beta scenarios with consistent pressure profiles and aligned bootstrap current in shaped plasmas have been identified theoretically, but such fully penetrated high-l_i profiles have not yet been achieved (central current drive in shaped plasmas is required) and it is not clear what the resultant confinement will be.

A potentially more attractive advanced-tokamak scenario is the reverse magnetic shear (identified as the NCS or ERS in different experiments with similar characteristics). This scenario has a hollow current density profile (and negative magnetic shear) in the center, and the current profile is naturally aligned with bootstrap current for sufficiently broad pressure profiles. Stable high-beta scenarios with 70% bootstrap current, well-aligned profiles, and strong shaping have been identified theoretically. It is clear from the ideal stability calculations that relatively broad pressure profiles are needed for the high beta with bootstrap current alignment, and that the stability limit increases with increasing plasma triangularity. It is also clear that these high beta values require wall stabilization.

Good evidence of wall stabilization has recently been obtained experimentally. Results are consistent with recently developed theory which requires plasma dissipation and rotation to obtain stability with a resistive wall. A major remaining issue for wall stabilization is the magnitude of rotation required for stabilization. Different theories require rotation frequencies from as low as a few times the inverse of the wall L/R time constant to as high as fraction of the Alfvén rotation frequency. Some experiments indicate the lower frequency of rotation is adequate for stabilization, but additional detailed experimental data are required.

Enhanced confinement in the NCS/ERS regime has recently received significant experimental attention. In the negative shear region, a transport barrier forms and the ion and particle transport are observed to be at or below the standard calculated neoclassical values. The electron transport is also observed to decrease, by approximately a factor 5 to 10; this is clearly demonstrated in cases with direct electron heating. The region of negative shear alone is not sufficient to explain the reduced transport. Significant progress has been made on understanding the transition into such regimes as well as the reduced transport. As a consequence of the very low transport in the core, the pressure profile becomes very peaked, so additional control methods are required to ensure compatibility with high beta. Also, the optimum safety factor profile (related to the magnetic shear) has not yet been determined, since the combination of strong negative shear with a strong pressure gradient is destabilizing to some modes. Additional experimental, theoretical, and modeling work is required to determine the optimum profile.

To make progress in extending the duration of these high performance discharges toward steady state requires plasma control. The major operational tools used to control the plasma are current profile control and wall conditioning (whose physics is believed to be tied to the core ionization and the change in particle and power deposition as a result of lower density). In the future, the current profile will be controlled by noninductive current drive; a number of techniques for off-axis current drive were discussed in Section II.C. Control of the transport and transport barriers will also be important for advanced-tokamak scenarios; IBW is a potential tool for this, as discussed in previous sections. Control of the density and core ionization will be accomplished by fueling systems and pumped divertors. Understanding the performance of these control techniques and their effect on the plasma is clearly an important research area for tokamaks.

Plans

The U.S. tokamaks over the next two years will further develop the physics basis of advanced-tokamak regimes. They will develop the tools necessary to control the pressure profile and the current density profile, and to exhaust the heat and particles. They will begin to extend high-performance scenarios into the burning-plasma regimes and toward steady state operation.

TFTR Plans

The main emphasis of the TFTR program is the understanding and control of the internal transport barrier and physics underpinnings of alpha channeling. These are important goals for advanced-tokamak integration. Transport-barrier control will be used to increase the fusion power output above present levels and enhance the alpha-particle population. This will allow the confinement and stability of the alpha particles and the plasma response to alpha heating to be evaluated. Alpha channeling is potentially a means to control the pressure profile through modification of the alpha particle heating profile. Although the alpha-heating power will be less than auxiliary heating power in TFTR experiments (except possibly close to the axis), they will be an important step toward the understanding of advanced-tokamak regimes under burning-plasma conditions. Specific TFTR goals are:

1) Evaluate the use of IBW as a control tool for internal transport barrier control.
2) Establish the cooling and radial transport of alpha particles in support of the concept of alpha channeling.
3) Evaluate mode-conversion current drive as an off-axis current drive tool in a DT plasma.
4) Evaluate the response of the plasma to alpha heating in advanced-tokamak regimes with the highest achievable ratio of alpha-heating to auxiliary-heating power density on axis.

The most important upgrade for TFTR is the IBW antenna and poloidal rotation diagnostic, needed for the transport control and alpha channeling experiments. To do the MCCD experiments, the antenna and the frequency change of the source are needed. In addition, the upgrade of one of the beam lines to 5-second duration is highly desired to begin to look at longer pulse issues.

DIII-D Plans

The integrated demonstration of advanced tokamak operation is a major goal of the DIII-D program. DIII-D plans include the demonstration of a self-consistent advanced-tokamak scenario with moderately enhanced performance parameters, and density and power exhaust with a pumped, high-triangularity, radiative divertor. Several specific goals that will contribute to achieving the DIII-D advanced-tokamak demonstration are:

1) Evaluation and understanding of off-axis electron-cyclotron current drive.
2) Control of the edge pressure gradient in H-mode and VH-mode discharges to extend the duration of the high performance regime, including high performance with ELMing discharges.
3) Development of a physics understanding of the ability to control the internal transport barrier by controlling the location of the minimum of the safety factor (q) in NCS discharges.
4) Evaluation of the role of plasma shape in establishing the internal transport barrier and in the stability limit of self-consistent high-bootstrap-fraction plasmas.
5) Understanding of the role of plasma rotation, and the magnitude of rotation required, for stability to the resistive wall mode at high beta.
6) Evaluation and development of heat removal scenarios compatible with low-density advanced tokamak operation.
7) Evaluation of high-l_i scenarios with central current drive as a potential advanced tokamak operation scenario.
8) Development of real time feedback control of the q-profile, using an advanced digital plasma control system.

This demonstration will require, as a high priority, the addition of the planned upper high triangularity pumped divertor to provide the density control required to obtain the needed off-axis current drive with the limited current drive power. The panel notes the conflict between the time required to install this divertor and the desirability and value of increasing the experimental operation of DIII-D. The ECRF system is crucial for off-axis current profile control, so exploitation of the initial 3-MW is of high priority. For the longer term program on DIII-D, the upgrade of the ECH power from 3 to 6 MW may be needed, but is not presently funded. In addition, diagnostic upgrades for improved central density measurements and better-resolved current-profile measurements should have high priority.

Alcator C-Mod Plans

The emphasis of the Alcator C-Mod program over the next two years will be to develop heat and particle control schemes for advanced tokamak scenarios, and to develop the heating and current drive and diagnostic capabilities for the future. The panel notes the potential for Alcator C-Mod to make significant contributions in this area in the long term; in particular, the tokamak has the capability to operate pulse lengths that are several times the current penetration time, 7 sec at 5 T toroidal fields. In the near term, Alcator C-Mod's plans are:

1) Evaluation of heat and particle control for high confinement regimes.
2) Investigation of pellet-enhanced performance (PEP) modes.
3) Evaluation of mode-conversion current drive for off-axis current profile control.

Over the next two years, it is important that Alcator C-Mod build up the capability for advanced-tokamak research in the future. The needed upgrades are:

1) Completion of the 8-MW ICRF system.
2) Installation of the divertor cryopump for H-mode control.
3) Diagnostic neutral beam and associated diagnostics for current density, ion temperature, and rotation velocity measurements.

Long-term program needs for Alcator C-Mod include lower hybrid current drive for off-axis current profile control. Currently this is not funded.

Actions

We strongly recommend the addition of the upper closed, pumped, high triangular divertor in DIII-D; this should remain a very high priority upgrade. However, we recognize the importance of increasing the experimental operation and analysis time on DIII-D to make progress in a number of scientific areas. So a balance in experimental time and upgrades is necessary (see Recommendation #1 in Section V). If a delay in the installation of the upper divertor is required to provide more experimental time, this should not lessen the commitment to complete the upgrade.

The modest upgrades planned for TFTR should be completed expeditiously and exploited, especially the IBW launcher and poloidal rotation diagnostic.

The indicated upgrades on Alcator C-Mod should be pursued aggressively (see Recommendation #2 in Section V). These are essential for Alcator C-Mod to develop a world class tokamak concept improvement program.

G. Summary of TFTR Research Plans and Measures of Performance

The FEAC report recommended that the TFTR facility should be the first of the three to be shut down, after a period of operation (about 2 years) to extract the remaining scientific benefit from it. With a deadline for TFTR shutdown now established (during FY 1998 at the latest), its planning has been adjusted so as to maximize operating time while foregoing all but a few critical upgrades. We were charged with determining the highest-priority scientific objectives for the remaining operating life of TFTR. Assuming that the resources are available to permit operation of TFTR through FY 1997 (as appears to be the case under the proposed budget), with the possibility of operation into FY 1998, its highest-priority scientific goals should be as described here. These were proposed by the TFTR team, who developed them in conjunction with their Program Advisory Committee. They were adopted by the review panel by a vote of 13 in favor, 0 opposed, and 1 absent.

Evaluate the response of the plasma to alpha heating in advanced tokamak regimes with ~20 MW of fusion power (4 MW of alpha-heating power) and the highest achievable ratio of alpha-heating to auxiliary-heating power density on axis.
Simulations of high auxiliary-heating-power (33 MW) advanced-regime plasmas show that ~20 MW of fusion power would be achieved in TFTR provided the transport barriers can be appropriately controlled. In these conditions, alpha-particle heating becomes a significant fraction, possibly as much as one-third, of the power flow in the core of the plasma and its effects should be readily apparent. The achievement of such conditions would represent both a significant validation of our understanding of the physics governing the core of a future reactor and a demonstration of our ability to integrate this understanding into the creation of a very high performance plasma.

Characterize the physics of the transport barrier, including the effects of the deposition profiles. Demonstrate techniques to control transport barriers using sheared rotation, current profile modification, and application of IBW in reactor relevant plasmas.
The planned ion Bernstein wave launch capability should provide control of the transport barrier location allowing us to control the pressure profile independently. By combining such control of the pressure profile with established techniques for transiently modifying the current profile, it should be possible to increase plasma stability and realize significant gains in the peak performance of D-T plasmas.

Evaluate the heating and current drive effectiveness of radiofrequency heating in the ion cyclotron range of frequencies in deuterium-tritium plasmas.

As a result of its combination of magnetic field, radiofrequency source parameters, and its deuterium-tritium capability, TFTR is uniquely positioned to study several issues in the physics of ICRF waves. Some of these have been identified as high priority ITER research and development needs, including deuterium-fundamental heating and mode-conversion heating and current-drive in deuterium-tritium plasmas. The folded waveguide ICRF coupler, developed in collaboration with ORNL would provide the first test of this potentially reactor relevant technology in a deuterium-tritium environment. Heating and current drive using coupling to either tritium or the fusion alphas is being investigated in a collaboration with the University of Wisconsin.

Establish the cooling and radial transport of alpha-particles using two waves as an evaluation of alpha particle-wave interaction in support of the concept of alpha-channeling.
After initial experiments planned for 1996-7 (see below), the next step in developing the physics of alpha channeling will be to demonstrate cooling and removal of a substantial fraction of the alpha population using a 2-wave scheme. Such schemes have already been proposed theoretically.

Evaluate alpha confinement and stability in advanced tokamak regimes with alpha particle parameters comparable to present projections for ITER.
The experiments would be aimed at characterizing alpha-particle effects on MHD stability in these regimes. Each advanced regime will have a different threshold for alpha-induced instabilities which will depend sensitively on the safety-factor and pressure profiles. The goal is to insure that the potential advantages of these regimes for a reactor such as ITER are not offset by decreased alpha-particle stability.

If a review of TFTR were to be held in mid-FY 1997, the following set of objectives should be used to measure progress toward the above goals:

Perform an initial evaluation of the response of the plasma to alpha-heating in advanced-tokamak regimes at fusion power levels at or beyond those already established in Super Shots (~10 MW).
Experiments in advanced-tokamak regimes at fusion powers of about 10 MW will enable us to evaluate the effects of the intrinsic MHD activity on the alpha distribution. While such regimes are attractive for reducing the size and cost of a future reactor, the compatibility of these regimes with good alpha-particle confinement and their stability to alpha-driven modes must be established. A critical test of the existing toroidal Alfvén eigenmode theory will be carried out in a configuration predicted to have the best chance of being excited by the alpha particles for modest fusion powers in TFTR.

Perform an initial characterization of the physics of the transport barrier including effects of electric field shear, local and global magnetic shear, and rotation.
Experiments will be conducted to manipulate the plasma current, pressure and rotation profiles to modify and control the transport barrier. Using the capability to vary the toroidal momentum input at constant neutral-beam heating power, it should be possible to distinguish the effects of sheared poloidal rotation from changes in the magnetic equilibrium. Changes in plasma turbulence associated with barrier formation will be measured and related to changes in the transport and to theoretical predictions. Fueling and diagnostic capabilities exist which will allow the particle transport of fuel ions to be studied directly. Through the development of a predictive capability for the transport barrier, improvements in the confinement and performance of deuterium-tritium enhanced reverse shear plasmas should be possible, thereby extending our knowledge of alpha-particle physics in this potentially important regime.

Couple greater than 2 MW of ion Bernstein wave (IBW) power into the plasma and perform an initial evaluation of its effect on transport barrier formation and transport suppression.
The initial experiments with the IBW system will be aimed at measuring the poloidal flow shear generated by the waves and the establishment of a transport barrier. This will allow us to test current theories for the mechanism of turbulence suppression and barrier formation.

Establish the cooling of alpha-particles as part of an initial evaluation of alpha-particle interaction with IBW.
Experiments in deuterium-tritium plasmas will be conducted to measure the effects of the IBW on the energetic and spatial distributions of the confined alpha particles. The IBW will be excited by mode-conversion from the ICRF fast wave in the mixed-species plasma. These experiments would test the basic theory on the interaction alpha-particles with RF waves which is an important element of the alpha-channeling scheme.

III. Technical Plans

To make progress in the research program described in Section II clearly requires that the facilities be operated and produce experimental data. It is equally clear from past experience that regular improvements in the experimental hardware, i.e. facility upgrades, are necessary to sustain a high level of scientific productivity. Establishing the optimum balance between operation and upgrades is an important challenge. In this section we summarize the plans for upgrades and experimental operation of the major facilities.

We also note that in a sophisticated experimental research program, theory and modeling are crucial in the design of experiments and the interpretation of measurements. Theorists and modelers are now actively involved in the programs of all three facilities, in many cases through collaborative arrangements. This trend is very healthy, and is conducive to making rapid progress in the understanding of tokamak physics and incorporating this understanding in predictive models. We encourage both the experimental and the theory/modeling communities to continue working together toward this common objective and to strengthen their ties on a continuing basis.

A. Facility Upgrades

Throughout Section II we pointed out the facility upgrades needed to make progress in each research area. Some of the planned upgrades were mentioned more than once because they support progress in multiple areas. Here we summarize the upgrade plans and our assessment based on research priorities in four categories: 1) in-vessel components, including wall conditioning systems; 2) profile control systems; 3) diagnostics; and 4) ancillary systems, including power supplies, computers, etc. In each category, the upgrade plans are presented in the order: TFTR, DIII-D, and Alcator C-Mod. The additional funding required to support some of the high-priority Alcator C-Mod upgrades is addressed in Section V under Recommendation #2. Summaries of the planned upgrade with their costs are provided in Tables III.A.1-3.

In-Vessel Components

TFTR lithium injectors. A modification to the existing lithium pellet injector to increase the number of pellets is being developed to increase the deposition of lithium on the limiter. An injector based on laser ablation of lithium is funded in FY 1996. Assessment: both upgrades are inexpensive and are supported.

DIII-D radiative divertor. This modification will provide DIII-D with a high-triangularity baffled and pumped divertor structure. It will provide density control for advanced-tokamak shaped plasmas and will reduce neutral backflow to the core plasma to decouple the divertor and core regions. The different magnetic geometries that are possible by varying the separatrix position will enable study of impurity enrichment in the divertor by flow generation. Because of past budget reductions, the upgrade will proceed in two phases. The upper structure will be installed in the first phase, and the lower structure installation and associated diagnostic reconfigurations will be completed in the second phase. Assessment: high priority, but may need to be balanced against high-priority needs to increase experimental operating time.

Alcator C-Mod cryopump. A full divertor cryopump system will be added to enable particle control for: H-Mode studies; puff and pump divertor flow and impurity shielding studies; and investigation of AT scenarios with quasi-steady-state pellet fueling and pumping. Assessment: high priority and a candidate for additional funding.

Alcator C-Mod divertor reshaping. The lower divertor modification (the so-called "nose job") would allow for greater shape flexibility, including higher triangularity. Assessment: lower priority in the near term.

Profile Control Systems

TFTR ICRF and Neutral Beam Systems: The planned upgrades, all to be completed in FY 1996, are as follows:

1) Ion Bernstein waves (IBW) launcher. Needed for transport barrier control research. Assessment: high priority.
2) Two modified 4-strap ICRF antennas. Needed to study mode-conversion current drive for off-axis profile control, and alpha cooling for alpha channeling research. Assessment: high priority.
3) Folded waveguide ICRF launcher. Test of an enabling technology developed for possible use in ITER, in collaboration with Oak Ridge National Laboratory. Assessment: supported.
4) Neutral beam pulse extension. One of the four neutral beam injectors will be modified to extend its pulse length to 5 s pulse at full acceleration voltage (120 kV). Needed to improve the control flexibility for access to enhanced reverse shear modes. Assessment: supported.

TFTR Lower Hybrid System. Plans to install a 1.3-MW system for off-axis current-profile control are presently on hold in order to support increased experimental operations. Assessment: concur.

TFTR tritium pellet injector. A planned modification of the existing D pellet injector to operate in tritium (TPI) for studying this approach to fueling of ITER has being put on-hold by the TFTR group due to budget constraints. Assessment: concur.

DIII-D Electron Cyclotron RF System. The first 3 MW of a planned 6-MW system is scheduled to be available in 1997, with the upgrade to 6 MW (at an additional cost of $9.4M) scheduled to begin in 1998. The ECH system is the primary off-axis current drive system on DIII-D and is needed for profile control for integrated, high-performance scenarios. Assessment: high priority for the initial 3 MW system.

Alcator C-Mod Ion Cyclotron RF system. The addition of 4 MW in variable-frequency (40-80 MHz) sources, a new four-strap antenna, and associated transmission components will increase the ICRF heating power to 8 MW. This is cost-effective because the sources are pre-existing and only need to be recommissioned. The project is being carried out in collaboration with the Princeton Plasma Physics Laboratory. It is planned to meet long-term needs for divertor and advanced-tokamak research. Assessment: high priority and a candidate for additional funding.

Diagnostics

Numerous diagnostic upgrades, many of them comparatively small, are planned. See Tables III.A.1-3 for complete lists. Here we discuss the major items.

TFTR Diagnostics. The planned upgrades are summarized in Table III.A.1. The only one not completed in FY 1996 is the infrared (IR) periscope system. This is based on a technique originally developed for JT-60U to study the spatial distribution of alpha-particle loss at the first wall. Assessment. The poloidal rotation measurement has highest priority, due to its importance for transport barrier research, followed by the motional Stark effect (MSE) system for current profile measurements. The IR periscope is supported.

DIII-D Diagnostics. There are several diagnostic upgrades proposed that would enhance the physics program on DIII-D. These are listed in approximately the order of priority in Table III.A.2. These are mostly modest in cost except for the Heavy Ion Beam Probe (HIBP), which would provide the capability to measure the profiles of radial electric fields and electrostatic potential fluctuations in the plasma core. This would involve the transfer of an existing HIBP system from the TEXT tokamak to DIII-D. Assessment. The HIBP is potentially important to the DIII-D program, but a physics evaluation is required to determine what measurement capabilities could actually be provided. The DIII-D program, including collaborators, should conduct such an evaluation of the HIBP before committing significant resources to its implementation.

Alcator C-Mod Diagnostic Neutral Beam and Associated Diagnostics. A plan has been developed to move an existing diagnostic neutral beam from the TEXT tokamak to Alcator C-Mod. Three diagnostic systems that rely on this beam will also be implemented under this plan. These are 1) a charge-exchange recombination spectroscopy system to measure ion temperature and rotation profiles, 2) a beam-emission spectroscopy system to measure fluctuations, and 3) a motional Stark effect system to measure current density profiles. These diagnostics are needed to develop state-of-the-art measurement capabilities for advanced-tokamak research in the long term. Assessment: high priority. Alcator C-Mod program costs are a candidate for additional funding within the major facilities budget (see Recommendation #2, Section V).

Ancillary systems

TFTR Plasma Control. The digital plasma control system from the PBX-M tokamak is being implemented on TFTR to provide an improved system for disruption avoidance and real-time feedback control of the q profile. Assessment: supported.

Table III.A.1. Summary of TFTR Upgrade Plans. Total FY 1997 plus FY 1998 cost is listed; on-hold items are in brackets. Relevance of the upgrade to the six research areas is indicated by: H (high priority for that area), M (medium priority for that area), or X (relevant to that area, priority not assessed.)

		TFTR					
	FY-97/98 Cost (K$)	**A. Transport and barrier control**	**B. MHD equilibrium, stability, and control**	**C. Heating, current drive and fueling for profile control**	**D. Divertors, boundary physics, plasma-wall interaction**	**E. Alpha and fast particle physics/ instabilities**	**F. Advanced tokamak scenarios and intergration**
In-Vessel Components							
Li Laser Ablation	0	X	X		X		
Li Pellet Upgrade	200	X	X		M		
Tritium Pellets (ON HOLD)	[800]	X				M	X
Profile Control Systems							
IBW Antenna	0	H	X	H		H	M
ICRF Ant. & Freq. Mods	0		X	H			X
Folded Waveguide	0			M			H
NBI pulse -->5s	0	X	X	X			
LH system 1.3 MW (ON HOLD)	[1300]		X	X			M
Diagnostics							
Poloidal Rotation Diag.	0	H	X			H	
MSE Upgrade	0	X	M	X			X
IR Periscope	300		H	X		X	
Oblique ECE for LH (ON HOLD)	0			X			
Ancillary Systems							
Digital Plasma Control	100	X	X				X

Table III.A.2. Summary of DIII-D Upgrade Plans. Total FY 1997 plus FY 1998 cost is listed. Relevance of the upgrade to the six research areas is indicated by: H (high priority for that area), M (medium priority for that area), or X (relevant to that area, priority not assessed.)

		DIII-D					
	FY-97/98 Cost (K$)	**A. Transport and barrier control**	**B. MHD equilibrium, stability, and control**	**C. Heating, current drive and fueling for profile control**	**D. Divertors, boundary physics, plasma-wall interaction**	**E. Alpha and fast particle physics/ instabilities**	**F. Advanced tokamak scenarios and intergration**
In-Vessel Components							
Upper Rad Div Struct	2900	X	X	X	M		H
Lower Rad Div Struct	1100	X	X	X	M		H
Profile Control Systems							
ECH (first 3 MW of 6-MW system)	600	H	X	H			M
Diagnostics (Core)	1100						
ECE radiometer		X	M	X			
MSE upgrade (more chan)		X	H	X	X		X
Core Density		H	H	X			X
Correlation reflectometer (Te & fluctuations)		H					
Beam emission spectroscopy (Ti fluctuations)		H					
Beam emission spectroscopy (2D & ne fluctuations)		H					
Collective ion scat				X			
Electron cyclotron scattering system				X			
Phase contrast imaging		X					
Lithium BES		X					
Heavy Ion Beam probe (with Univ of Texas)		X					
Diagnostics (Edge/Div)							
Upper div modifications	600				H		
Lower div modications	1400				H		
Div TS upgrade (increased pulse rate)					X		
Div Refl upgrade (higher ne)					X		
Div RGA					X		
Div Ti meas					X		
Lang Probe upgrades		X			X		

Table III.A.3. Summary of Alcator C-Mod Upgrade Plans. Total FY 1997 plus FY 1998 cost is listed. Relevance of the upgrade to the six research areas is indicated by: H (high priority for that area), M (medium priority for that area), or X (relevant to that area, priority not assessed.)

		Alcator C-Mod					
	FY-97/98 Cost (K$)	**A. Transport and barrier control**	**B. MHD equilibrium, stability, and control**	**C. Heating, current drive and fueling for profile control**	**D. Divertors, boundary physics, plasma-wall interaction**	**E. Alpha and fast particle physics/ instabilities**	**F. Advanced tokamak scenarios and intergration**
In-Vessel Components							
Divertor Cryopump	1025				H		X
Divertor Reconfiguration	385		M		M		
Profile Control Systems							
ICRF Upgrade to 8 MW	250+850		H	H			H
LowerHybrid	——						X
Diagnostics (Core)				X			
Diagnostic Neutral Beam	750	H					
ECE Radiometer	0	X	X				
Core Thomson	60	X					
Reflectometer	0	X					
Tangential Interferometer	80	X	X				
Collimated Neutron Array	12	X					
Phase Contrast Imaging	30	X					
Sieve fluctuation diagnostic	0	X	X				
Tangential XUV array	5	X					
Charge-Exchange Recombination Spectroscopy	200	H	M				
Beam Emission Spectroscopy	200	H	X				
Motional Stark Effect	235	H	M				
Diagnostics (Edge/Div)							
Edge Thomson	0				X		
Divertor RGA	0				X		
Omegatron	0				X		
IR Imaging	0				X		
Vis/UV fibers	0				X		
Probes	10				X		
X-point spectrograph	15				X		
Low-energy NPA	0				X		

B. Facility Operation

Operating time is limited by several factors: costs of energy and consumable items, availability after allowing time for maintenance and upgrades, and the scientific and technical staff needed to operate productively and safely. These must be carefully balanced to optimize scientific output; the proper balance will depend on the details of the research program, the remaining life of the experiment, and other factors. In the current fiscal year, operating time on all three facilities has been restricted by budget cuts. This problem will persist to some degree in FY 1997 and FY 1998, though it is alleviated by the proposed budget increases. Here we summarize and assess the operation plans for each facility, based on the budget guidance described in Section I.

TFTR

The proposed TFTR operating schedule has been developed to maximize operating time within the budget guidance constraints in order to optimize the research output of the facility. This has been accomplished by eliminating several significant upgrades. The planned shutdown of TFTR by FY 1998 is a major factor in this decision. A vent is planned for the end of FY 1996 and the first two months of FY 1997 to install the remaining upgrades (i.e., the IBW launcher, poloidal rotation diagnostic, etc.). In FY 1997, the TFTR will operate for 30 weeks, which is comparable to the run time in FY 1995 of 33 weeks. The FY 1998 plans call for six months of experiments totaling 18 weeks of operating time prior to shutdown. Assessment: We concur with the TFTR plans to give priority to operating time over upgrades, once critical upgrades are completed in early FY 1997.

DIII-D

The plans proposed by the DIII-D team under the budget guidance call for a vent at the beginning of FY 1997 of approximately six months duration. Several upgrades and maintenance tasks will be accomplished during this period but its duration is determined by the main task, the installation of the top radiative divertor structure. Twelve weeks of operation are planned for the last two quarters of FY 1997 and 15 weeks are planned for the first three quarters of FY 1998. A vent is scheduled beginning with the last quarter of FY 1998 to install the lower radiative divertor structure and reconfigure the associated diagnostics. Assessment: We found that there is a compelling need to increase the operating time planned for DIII-D in FY 1997 by approximately 50% to support the needs of the research program. The extensive downtime required for the radiative divertor installation appears to have a major impact on the facility’s availability for operation and is thus a concern. We recommend that this downtime be shortened as much as possible to make more operating time available. While the divertor upgrade is of high priority, consideration should be given to delaying its installation, if necessary, as a means to increase run time. Further details are provided under Recommendation #1 in Section V.

Alcator C-Mod

The Alcator C-Mod will operate for 15 weeks in both FY 1997 and FY 1998. A vent is planned for the winter of 1997, during which the 4-MW variable frequency ICRF, for heating and current drive, will be commissioned and the inner divertor may be modified. Operation would resume in the spring of 1997, with experiments in the transport and divertor areas taking advantage of the new facility capabilities. An extended maintenance interval is scheduled at the end of FY97 through the first quarter of FY98. The diagnostic neutral beam and its associated diagnostics will be installed during this interval. While the full complement of ten cryopump modules will not be available under the reference budget, a partial installation consisting of a few additional modules will be undertaken. Assessment: We support the Alcator C-Mod plans assuming present budget guidance. However, additional resources could advance the schedule for installation of the diagnostic beam and associated diagnostics and of the full divertor cryopump system.

C. Facility Utilization

The FEAC report refers to the "utilization" of operating facilities, and in particular calls for "full utilization of DIII-D and C-Mod" after TFTR shuts down. The major facilities represent a substantial resource for producing scientific gains and, under constrained budgets, their cost-effective use is clearly important. We attempted to assess the degree to which the facilities will be fully utilized under the proposed budgets but found that, at best, only a subjective assessment was possible. There are several factors that contribute to high facility utilization:

- Time spent on planning and data analysis, which generally involves research staff members also required for experimental operation. Increasing the time spent preparing carefully designed experiments may produce better science than extending an experimental run. However, too little run time will result in lack of data on important physics.

- The use of collaborations for planning, diagnostics, and data analysis. Collaboration makes the wide range of skills and capabilities of the U.S. program available for the experimental programs on the major tokamaks. However, effective experimental operation requires a significant core of on-site personnel (which can include collaborators) who are available essentially full time.

- The role of theory and modeling. Maximizing scientific productivity requires close interaction with theory and modeling groups, as noted in the introduction to this Chapter.

- The importance of upgrades and the effective utilization of vent time. Tradeoffs between upgrades and experimental operations are critical; the time and resources spent on upgrades impacts operation but contributes to maintaining state-of-the-art capabilities and leads to important new data in future operations.

- Staff availability and skill mix. Staff reductions forced by budget cuts have generally reduced the flexibility of the experiments to respond to new ideas or data requests which require additional run time. Collaboration, use of contract employees, temporary personnel loans from other institutions are means of optimizing the skill mix which have been exploited.

- Financial resources. Clearly the rate of scientific progress on a facility, however measured, can be increased by increasing operating time, facility upgrades, and staff. These require increased funding, whether to the host institution or to collaborators.

Assessment: All three facilities will continue to be less than fully utilized under the proposed budgets. The TFTR program has had to forego potentially valuable upgrades to gain operating time. The DIII-D operating time is severely limited, forcing a difficult re-examination of the tradeoffs including the possible delay of critical upgrades. The Alcator C-Mod has been slowed in its development by a history of tight budgets throughout its operating life to date. All three programs have struggled to plan the best possible utilization of their facilities under difficult constraints.

IV. Summary Assessments of the Major Facilities Plan

The major facilities' scientific plans can be assessed from several points of view, corresponding to the various goals of the U.S. fusion program: advancement of FEAC scientific goals, promotion of U.S. leadership in concept innovation, resolution of ITER Physics R&D issues, and contribution to materials and technology development. These aims are mutually compatible and complementary. For example, the research needed to resolve ITER concerns about disruptions and power and particle exhaust will also advance scientific goals in MHD and divertor physics, respectively. The plasma control needs for studying transport and stability in the major tokamaks are synergistic with the need for test beds for enabling technology development (e.g., high-power, steady state microwave sources). Thus, the program's science focus provides an excellent framework for advancing a range of goals that are important for fusion development.

In summary, we find that the research plan for the major tokamak facilities will produce impressive scientific benefits over the next two years, in spite of difficult budget circumstances. The plan is well aligned with the new mission and goals of the restructured fusion energy sciences program recommended by FEAC.

A. Advancement of Fusion Scientific Goals

The first key policy goal of the U.S. Fusion Energy Sciences Program is to "Advance plasma science in pursuit of national science and technology goals." Among the program's scientific goals, the FEAC report (Appendix A) identified the following key fusion plasma physics issues:

1) Magnetohydrodynamic Equilibrium, Stability, and Dynamics (Plasma Control);
2) Transport Processes (Plasma Confinement);
3) Plasma-Wall Interactions (Limiters, Divertors);
4) Wave- and Particle-Plasma Interactions (Plasma Heating, Fueling, and Current Drive);
5) Burning Plasma Physics (Alpha Physics, Burn Control);
6) Composite Issues (Systems Integration).

The research plans of the major U.S. tokamak facilities have been carefully constructed to examine each of these topics in an effective manner. These plans were described in detail in Section II. In the central areas of MHD, transport, and boundary interactions, where all tokamaks can contribute, the three facilities have complementary features and strengths. Each has a strong research program. Likewise, each is exploiting techniques for heating, current drive, and fueling, from which much will be learned. DIII-D and Alcator C-Mod have longer-term research programs in these areas. TFTR has a unique capability for alpha physics experiments and emphasizes those studies in its program.

Although each of the devices could accomplish more if greater resources were available, the program recommended here permits each to pursue a strong and productive scientific research program. The program will maximize scientific progress for fixed resources. Since each of the facilities supports the research of a corps of strong, creative physicists, one can reasonably expect some important new discoveries in many of these areas. The discoveries may easily be the most important result of the research.

Through the combination of sound planning and unexpected discoveries, research on the major facilities will make excellent progress in advancing the scientific goals of the U.S. Fusion Energy Sciences Program.

B. Leadership in Concept Innovation.

The second policy goal for the restructured program is to "Develop fusion science, technology, and plasma confinement innovations as the central theme of the domestic program." Plasma confinement innovations, high-performance tokamak operating modes, and diagnostics are among the selected areas in which the U.S. program seeks to maintain leadership. In the case of tokamaks, concept innovation involves developing the scientific understanding needed to make them less expensive and more reliable. The understanding gained will increase our ability to predict tokamak performance and to control it. As we have seen, the major tokamaks have vigorous research programs in all the critical areas, with an emphasis on understanding, control, and integration.

While no existing tokamak in the world program is capable of addressing all of these issues, each of the three major U.S. tokamaks can contribute significantly. The TFTR is the only tokamak in the world currently studying D-T plasmas in reactor-like conditions. It will use advanced transport controls to improve performance and study alpha heating and will explore a novel technique for controlling the alpha power deposition. The DIII-D tokamak, with its flexible strong shaping capability, profile control tools, and pumped divertor, is unique in its ability to evaluate the relationship among plasma stability, transport, and power and particle exhaust that form the foundation for steady-state, high-performance operation. It will use these capabilities to demonstrate an integrated advanced-tokamak scenario. The Alcator C-Mod tokamak is unique in the world program in its capability to operate at high magnetic fields, high density, and high divertor power densities. It will make critical contributions to divertor and transport physics understanding and, in the long term, to the development of advanced-tokamak scenarios.

After reviewing the programs for the three major facilities, we can confirm the FEAC conclusion that all three are "well positioned to make further scientific advances" in the concept innovation arena and will help the U.S. to maintain a leadership position in this area.

C. Resolution of ITER Physics R&D Issues

The third policy goal of the U.S. fusion program is to "Pursue fusion energy science and technology as a partner in the international effort." Currently the central element in the world fusion program is the International Thermonuclear Engineering Reactor (ITER), an international collaboration to construct an experiment to operate with burning plasmas. As a partner in this collaboration, the U.S. has a responsibility to contribute to the ITER physics basis, one of several key deliverables that will be used to justify a decision on whether or not to construct ITER after completing the current phase, the Engineering Design Activity (EDA).

By way of defining the needed contributions, the ITER program has defined a set of Physics Research and Development (R&D) needs. Demonstrations, databases, and predictive model developments are all needed, but the fundamental requirement in all cases is data from tokamak experiments. The major U.S. tokamaks have all contributed to the ITER Physics R&D program in the past and, with the research program plan presented here, will continue to do so in the next two years. To illustrate this, we provide a few examples of how the U.S. tokamaks will contribute to critical ITER physics needs.

Confinement

There is an urgent need for data related to H-mode operation, the envisioned operating mode for ITER. Data are needed to improve the scalings for the threshold heating power for the L-Mode to H-mode transition, and for H-mode energy confinement times. A long-term need is to expand the database of plasma profiles in H-mode operation. The H-mode is characteristic of diverted tokamaks, so the DIII-D and Alcator C-Mod programs will contribute toward these needs. In particular, Alcator C-Mod's transport plans emphasize H-mode studies, in response to ITER needs.

Divertors

Major advances in divertor power and particle handling performance are required for successful operation of ITER. An urgent need is to significantly expand the database on plasma edge parameters, including density, temperature, power and particle fluxes, and scrape-off decay lengths. In addition ITER has a high-priority need for a more complete characterization of edge-localized modes (ELMs) which could deliver large intermittent bursts of heat to divertor structures. The DIII-D and Alcator C-Mod facilities are now equipped with extensive divertor and edge-plasma diagnostics, with additions planned. Their planned divertor, boundary physics, and plasma-wall interactions programs will make important contributions to the ITER database needs in this critical area. In addition, the close collaboration of these programs with divertor model development groups will contribute to the development of a predictive capability for ITER divertor performance.

Disruptions

Tokamak disruption events are characterized by a sudden loss of stored energy and rapid discharge termination, with the potential for major damage to the first wall and other structural components. Designing the ITER tokamak to withstand disruptions requires a database that can be used to predict their characteristics. Of particular interest are the thermal energy decay rate, the plasma current decay rate, scrape-off plasma parameters during disruptions, and “halo” currents which flow from the disrupting plasma directly into the first wall and cause large electromagnetic forces. In addition, techniques to mitigate the severity of disruptions are needed. All three tokamaks plan programs for characterizing disruptions and testing pre-emptive methods of triggering an incipient disruption to reduce its severity.

D. Development of Enabling Technologies and Low-Activation Materials

Although not their primary purpose, the major tokamaks contribute to the development of enabling technologies and low-activation materials for fusion. They thus support these other parts of the program. The main contributions are in plasma technologies and low-activation materials. Some examples are provided.

Plasma Technologies

The DIII-D advanced-tokamak program requirement for electron-cyclotron RF heating has driven the development of high-power, steady-state microwave sources and transmission systems. The folded waveguide is a technology for launching ion-cyclotron RF waves at much higher power densities than are available with conducting-element launchers. The planned test of a prototype folded waveguide launcher on TFTR will be a critical step in its development. The D-T operation on TFTR has provided important practical experience in tritium safety issues, tritium system operation, and in-vessel inventory management.

Low Activation Materials

The DIII-D radiative divertor will use vanadium-alloy components in its in-vessel support structure. This will provide some useful experience and data for this potentially important low-activation material. In TFTR, the availability of a D-T fusion spectrum is facilitating tests to determine the activation and daughter products of a wide range of materials. The resulting activation levels are compared with predictions using the nuclear excitation cross-sections to provide a strong test of our ability to predict the activation.

There has historically been a strong synergy between enabling technology development programs and the U.S. tokamak programs. The results have been an impressive development of technology with significant impact on the physics results. Indeed, the continuing effort to upgrade the facilities (e.g., with new RF antennas, new divertor structures, and new diagnostics) is a clear demonstration that enabling technology development is continuing with considerable energy and ingenuity.

V. Recommendations Toward an Optimum Plan

As we have seen, the research program planned for the major tokamak facilities will produce impressive scientific progress in the next two years. This is a credit to the talented and dedicated scientists who work on these facilities. It is also due in large measure to past financial investments in hardware capabilities, which unfortunately will be curtailed in the future. In addition, the operating budgets for these facilities remain below what is necessary for them to be fully productive. In this environment, producing an *optimum* research plan, as requested in our charge, is a genuine challenge. The critical issue is the balance between experimental operation to fully exploit present capabilities and upgrades to improve on those capabilities. Having reviewed the combined program of all three facilities, we make two recommendations which, if implemented, will lead to an improved plan for their operation.

> **Recommendation #1.** DIII-D operating time in FY 1997 should be increased by ~50% (within their reference budget level) in order to increase the scientific output in all research areas, and to foster DIII-D's role as a major national collaborative research facility. In achieving this, the DIII-D program should consider reducing the downtime for and/or delaying the divertor upgrade installation. (Approved by the review panel by a vote of 8 in favor, 5 opposed, and 1 abstention.)

We found that more experiments in the DIII-D facility is a priority need for all research areas, including transport, MHD, and divertor physics. There has been a substantial investment in DIII-D hardware capabilities over the years; recent ones include new divertor diagnostics and a long-awaited profile control system that will soon be operational, representing a quantum leap in capability for physics studies. Because the DIII-D research program is conducted as a multi-institutional collaboration (about half the research staff are from outside the host institution), it is important to maximize the opportunity for collaborating scientists to conduct experiments. These considerations combine to place a high premium on operating time (along with the associated planning and analysis effort to make it scientifically productive). However, only 12 weeks of operating time is currently planned for DIII-D in FY 1997, the same as in FY 1996. In order to obtain the most scientific benefit from DIII-D, we think it is important to increase the scientific output in FY 1997, so we recommend this increase in operating time. We also recognize the radiative divertor upgrade as a high priority for divertor, boundary physics, and plasma-wall interaction research (although substantial progress can be made with the present configuration), as well as for high-performance core plasma studies. Its installation has already had to be delayed and split into two phases due to budget reductions, and ideally one would prefer to avoid further delays. However, installation of the first phase of this upgrade in FY 1997 requires a vent of approximately half a year, in which time the machine is obviously inoperable. Thus we face a conflict of priorities. Delaying the divertor installation to increase operating time is a painful tradeoff, if it must be made, but one that we consider warranted by the need to increase the rate of scientific progress next year.

> **Recommendation #2.** Additional resources (~$1M) should be applied to the Alcator C-Mod program to increase its near-term scientific output and to build up scientific capabilities needed for the long-term:
>
> – Diagnostic neutral beam and associated diagnostics.
> – Completion of the 8-MW ion cyclotron range-of-frequencies (ICRF) heating system.
> – Divertor cryopump.
>
> Assuming a fixed total budget for the major facilities, the resources should be obtained through equal reductions in the TFTR and DIII-D programs (~$0.5M each). (Approved by the review panel by a vote of 7 in favor, 2 favoring reductions in DIII-D only, 2 favoring reductions in TFTR only, and 3 abstentions.)

The development of Alcator C-Mod capabilities has been hampered by tight budgets for its entire operating life. We believe it is necessary to speed up the investment in this facility to ensure that it will be competitive in the long term, since it will be one of only two major U.S. tokamaks operating after 1998. The diagnostic neutral beam will support diagnostics to measure the current profile, ion temperature, rotation velocity, and fluctuations. Completing its basic auxiliary heating complement of 8 MW will enable Alcator C-Mod to operate near the beta (pressure) limit, and is cost-effective because it will make use of source capacity already installed. Both upgrades are critical for Alcator C-Mod's long-term advanced-tokamak program, and both involve collaborations with other institutions, an approach which we believe should be encouraged. The divertor cryopump is needed to improve particle control flexibility for the divertor physics program, currently the main emphasis on Alcator C-Mod. Besides these upgrades, the additional resources recommended will allow modest expansions in research staff and operating time, resulting in immediate increases in scientific output. We would prefer it if the additional resources for Alcator C-Mod could be made available without impacting other parts of the fusion program. However, under the assumption of a constrained total budget for the major facilities, there is no alternative but to offset the increase with reductions in DIII-D and TFTR. We recommend it be shared equally to avoid making an excessive impact on either one and to make clear that there is no adverse judgment against either one implied by this recommendation.

VI. Summary Conclusions

The major tokamaks and their scientific teams provide the U.S. program with a strong set of capabilities for addressing the major physics issues for fusion plasmas and improving the tokamak concept. This finding confirms the FEAC's favorable assessment of the facilities' capabilities and their potential to contribute in the restructured program.

The community has developed a research plan for the major tokamak facilities that will produce impressive scientific benefits over the next two years. The plan is well aligned with the new mission and goals of the restructured fusion energy sciences program recommended by FEAC. Budget increases for all three facilities will allow their programs to move forward in FY 1997, increasing their rate of scientific progress. With a shutdown deadline now established, the TFTR will forego all but a few critical upgrades and maximize operation to achieve a set of high-priority scientific objectives with deuterium-tritium plasmas. The DIII-D and Alcator C-Mod facilities will still fall well short of full utilization. Increasing the run time in DIII-D is recommended to increase the scientific output using its existing capabilities, even if scheduled upgrades must be further delayed. An increase in the Alcator C-Mod budget is recommended, at the expense of equal and modest reductions (~1%) in the other two facilities if necessary, to develop its capabilities for the long-term and increase its near-term scientific output.

Appendix A. Charge

Dr. Robert W. Conn, Chair
Fusion Energy Advisory Committee
School of Engineering
University of California, San Diego
9500 Gilman Drive
La Jolla, CA 92093-0403

Dear Dr. Conn:

This letter forwards two charges intended to follow up on specific recommendations made by your Committee in its Advisory Report on "A Restructured Fusion Energy Sciences Program." The report calls for expeditiously conducting two specific programmatic reviews to help the Department set the technical priorities of the restructured program:

- o A Major U.S. Facilities Review
- o An Alternative Concepts Review

The first review should be dealt with directly. As indicated by the enclosed charge, the second review is a little more involved and may require a longer time scale to fully address. I would like the committee to consider the fundamental investment strategy that we should use in funding alternative concepts. In the near term, however, we would like you to provide us with an assessment of one element within the catagory of alternative concepts, that of spherical tokamaks. Although the Fusion Energy Advisory Committee (FEAC) has suggested that the Alternative Concepts Review should also encompass inertial fusion energy, DOE is preparing a separate charge on that topic.

Please carry out the Facilities Review and the Alternative Concepts Review in parallel, using additional expertise outside of the FEAC's membership as necessary, so that the restructuring process may proceed. I would like to have your recommendations regarding facilities and, at least, the spherical tokamak aspects of the alternative concept review by mid-April.

The Department is most appreciative of the continued dedication shown by all FEAC members and your willingness to provide advice on important issues as we enter a period of unprecedented changes in the U.S. fusion science program. I will look forward to hearing the Committee's recommendations on these matters.

Sincerely,

Martha A. Krebs
Director
Office of Energy Research

Enclosures

Charge to the Fusion Energy Advisory Committee for a Major Fusion Facilities Review

In its report to DOE of January 27, 1996, the Fusion Energy Advisory Committee (FEAC) recommended that a major U.S. fusion facilities review be immediately carried out as part of making the transition to a Fusion Energy Sciences Program. The purpose of this review is to examine the progress, priorities, and potential near-term contributions of TFTR, DIII-D, and Alcator C-MOD (and other facilities as appropriate), and produce an optimum plan for obtaining the most scientific benefit from them. This optimzation should be within the context of the overall recommendations of the report on "A Restructured Fusion Energy Sciences Program" and should work within the funding level for these three facilities in the President's FY 1997 Budget Request.

The Department therefore requests the FEAC to organize and conduct such a review as expeditiously as possible, using whatever approach it deems most appropriate. In carrying out the review, the FEAC is encouraged to involve foreign participants in the review process.

There are specific points that the review should address:

- What are the highest priority near-term (~2 years) scientific objectives to be accomplished with these facilities to advance the goals of the U.S. Fusion Energy Sciences Program?

- What actions could be taken to more effectively use these facilities to address the objectives identified above? For example, changes in theory and modeling collaborations, in international collaborations, in enabling technology capabilities, in operating schedules, and in the allocation of resources among the facilities should be considered.

- In the case of TFTR, if the resources are available to permit operation of TFTR through FY 1997, what are the specific scientific objectives that would merit continuing operations through FY 1997 and into FY 1998? How would you measure progress toward such objectives in a review in mid FY 1997?

The FEAC's findings and recommendations in response to this charge should be delivered to the Director of Energy Research by mid-April.

Charge to the Fusion Energy Advisory Committee for an Alternative Concepts Review

In its report to DOE of January 27, 1996, the Fusion Energy Advisory Committee (FEAC) recommended that a review of Alternative Concepts be carried out as part of making the transition to a Fusion Energy Sciences Program. This review should fundamentally be directed at recommending an investment strategy for funding alternative concepts. What criteria, in addition to scientific excellence, should determine the effort devoted to the Alternative Concept Program (for example, similarity to or difference from the tokamak, power density, size, etc.)? Within the general guidelines of this recommendation, the Department requests the FEAC to organize and conduct such a review as expeditiously as possible, using whatever approach it deems most appropriate. Although FEAC recommended that inertial fusion energy (IFE) should be considered as part of the alternative concepts review, the Department recognizes the distinct characteristic of IFE and will request a review of IFE in a separate charge.

It is generally recognized that the various alternative concepts are at significantly different levels of development. Within this context, the review should address the following:

1. Review the present status of alternative concept development in light of the international fusion program. As part of this review, consider not only the prospects for alternative concepts as fusion power systems but also the scientific contributions of alternative concept research to the Fusion Energy Sciences Program and plasma science in general.

2. The review should produce an overall strategy for a U.S. alternative concepts development program including experiments, theory, modeling/computation and systems studies, which is well integrated into the international alternative concepts program. The U.S. plan and supporting documentation should include but not be limited to:

    - recommendations on how best to collaborate in alternative concepts where our international partners already have large experiments (e.g., the stellarator),

    - recommendations for encouraging new innovations in alternative concepts,

    - a methodology for assessing on a comparative basis the scientific progress of alternative concepts in their early stages of development, and

- a set of criteria for use in determining when an alternative concept is ready to undertake a "proof-of-principle" scale experiment. For this purpose, consider the Princeton Large Torus as the proof-of-principle experiment that validated the tokamak concept.

3. The spherical tokamak is recognized to be a scientifically advanced alternate. Based on the FEAC recommendations to enhance research on alternative concepts, the FY 1997 budget request contains proposed funding for the National Spherical Tokamak Experiment (NSTX) at Princeton. An experiment of this size and scope could be considered a "proof-of-principle" for this concept. There are several ongoing spherical tokamak programs and several new grant applications also under review. We are not asking you to review any specific proposals. Rather an assessment of the readiness of this concept to move to "proof-of-principle" experimentation would provide a useful example to be carried out early in the overall review process. This assessment should specifically address, in the international context, the present theoretical understanding and experimental data base of the spherical tokamak concept. In addition, the potential for such spherical tokamak research to resolve key physics and technology issues of importance to both the conventional tokamak and the spherical tokamak as a reactor in its own right should be considered.

The FEAC's findings and recommendations with regard to the spherical tokamak assessment should be delivered to the Director of Energy Research by mid-April. The overall review of alternative concepts should be delivered by mid-July.

Appendix B. Major Facilities Review Panel and Science Committee Members

Major Facilities Review Panel

Dr. George H. Neilson, Jr.* (Panel Chair)
Oak Ridge National Laboratory

Dr. Benjamin A. Carreras
Oak Ridge National Laboratory

Dr. Daniel A. D'Ippolito
Lodestar Research Corporation

Dr. Otto Gruber
Max-Planck-Institut fur Plasmaphysik, Germany

Dr. Mitsuru Kikuchi
Japan Atomic Energy Research Institute, Japan

Dr. Kevin McGuire
Princeton Plasma Physics Laboratory

Dr. Douglass E. Post
ITER Joint Central Team

Prof. James D. Callen* (Scicom Chair)
University of Wisconsin

Prof. Patrick H. Diamond*
University of California at San Diego

Prof. Kenneth W. Gentle
University of Texas at Austin

Dr. E. Bickford Hooper
Lawrence Livermore National Laboratory

Dr. Earl S. Marmar*
Massachusetts Institute of Technology

Dr. Cynthia Kieras Phillips*
Princeton Plasma Physics Laboratory

Dr. Tony S. Taylor*
General Atomics

* Member of Scientific Issues Subcommittee (SciCom)

FEAC Scientific Issues Subcommittee (SciCom)

Prof. James D. Callen (Chair)
University of Wisconsin

Prof. Patrick H. Diamond
University of California at San Diego

Prof. Farrokh Najmabadi
University of California at San Diego

Dr. William M. Nevins
Lawrence Livermore National Laboratory

Prof. Stewart C. Prager
University of Wisconsin

Dr. Dale Smith
Argonne National Laboratory

Dr. Tony S. Taylor
General Atomics

Dr. Michael C. Zarnstorff
Princeton Plasma Physics Laboratory

Prof. Gerald A. Navratil (Vice Chair)
Columbia University

Dr. Earl S. Marmar
Massachusetts Institute of Technology

Dr. George H. Neilson, Jr.
Oak Ridge National Laboratory

Dr. Cynthia Kieras Phillips
Princeton Plasma Physics Laboratory

Prof. Marshall N. Rosenbluth
ITER Joint Central Team

Dr. Emilia R. Solano
University of Texas at Austin

Dr. Kenneth L. Wilson
Sandia National Laboratories

Appendix C. Major Facilities Review Questions

1) Identify the highest-priority areas of scientific research for your program. For each area,
    - What are the scientific goals and how do these advance the new mission and scientific goals of the U.S. Fusion Energy Sciences Program (see FEAC report, Appendix A)?
    - What are the key scientific issues to be resolved?
    - What progress have you made to date and what is the current status of scientific understanding and predictive capability?

<u>Questions 2-5 should be answered assuming the reference budget levels provided by the Office of Fusion Energy Sciences for FY 1997 and FY 1998.</u>

2) For each research area identified in Question 1,
    - What further progress in scientific understanding and predictive capability will be made through FY 1998? Identify the key deliverables and any other means of measuring progress at intervals of a year or less. (TFTR: Specifically include deliverables and any other means of measuring progress through March, 1997).
    - Which of the planned facility upgrades in your answer to Question 3 are critical to progress in this area?
    - How much of the operating time in your answer to Question 4 will be allocated to this area in FY 1997 and FY 1998?

3) What are your planned facility upgrades (including diagnostics)? For each one,
    - What is the current status?
    - What is the planned availability?
    - What is the total cost and how much of it will be spent in FY 1997? in FY 1998?

4) What is your operating schedule through FY 1998 and how much operating time is planned in each year? How much contingency time is allowed for unforeseen developments?

5) To what degree will there be “full utilization” of your facility in FY 1997 and FY 1998? Where would you place your facility’s operation in that period on a scale of 0 to 100, where 0 denotes an idle facility and 100 denotes full utilization?

6) In order of priority, what actions could be taken by the Fusion Energy Sciences Program to more effectively utilize your facility and what would be the scientific benefit? If an action would require a change in budget allocation, indicate how much. Actions to consider include, but are not limited to, the following:
   - Addition of facility upgrades or acceleration of those already planned.
   - Addition of resources to increase or make better use of operating time.
   - Addition of staff to augment scientific or technical skills. To what extent could such needs be met through collaboration?
   - Support from other parts of the program, such as theory and modeling.
   - Improved coordination with other parts of the world program, e.g., joint studies with other facilities in the U.S. or abroad.

7) In terms of facility utilization and scientific progress, what would be the impact of a budget reduction of $1M? $2M? $5M?

8) How does your program help the U.S. Fusion Energy Sciences Program establish and maintain leadership in concept innovation?

9) How does your program advance the resolution of ITER's physics R & D needs, particularly those identified by ITER as urgent and high-priority?

10) How does your program advance the development of enabling technologies for fusion science? of low-activation materials?