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PPPL

**Laboratory
Program Development
Activities**

for Fiscal Year 1993

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Preface

In January, 1993, the Princeton Plasma Physics Laboratory (PPPL) obtained formal approval from the Department of Energy (DOE) for a program of discretionary research and development called Laboratory Program Development Activities (LPDA). Under the LPDA program, the Laboratory Director establishes a total funding amount (typically about 1.0% of total operating funds, not to exceed 2.0%) to be approved by DOE and then used at the Laboratory's discretion for research and development in magnetic fusion and related technologies. The primary purpose of the LPDA program is to provide support for Laboratory efforts relating to the early exploration and development of scientific concepts and techniques that are identified in the course of the Laboratory's normal technical work and which have the potential for contributing importantly to the Laboratory's future programs. For FY93, a total LPDA budget of \$1,250K was established.

The Laboratory divides its LPDA projects into three categories. First, approximately 40% of the funding (in FY93) was allocated to studies of possible new magnetic confinement devices and to advanced-tokamak theoretical studies in support of new confinement initiatives; projects in this category were carried out under active management by the Research Council, and the emphasis of the work evolved to meet changing Laboratory priorities. Second, approximately 30% of the funding was allocated to smaller, individual initiatives in magnetic fusion, which were selected on the basis of technical merit and institutional value from a much larger number of proposals submitted for review by the Program Committee. Third, approximately 30% of the funding was allocated to individual proposals which sought to apply plasma physics and fusion-related expertise to non-fusion applications; projects in this category were selected following review by the Program Subcommittee on Non-Fusion Research.

Most of the LPDA projects met their FY93 goals, as will be apparent from the following summary reports. However, progress on design studies of possible new confinement devices was inhibited in FY93 by a shortage of engineering resources, and the funds allocated to this category were not fully costed. However, the pace of design activity, both on a spherical tokamak experiment and on a medium-scale long-pulse tokamak device, has acceler-

ated recently, and considerable progress on these projects is expected in FY94. Completion of the planned workscope on some of the non-fusion projects was precluded by the relatively late award of funds in this area; the more promising of these activities are expected to be continued in FY94.

Advanced Tokamak Studies

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Advanced tokamak configurations make use of cross-sectional shaping and profile control to obtain confinement and beta exceeding contemporary scalings and bootstrap fraction approaching unity.

The current profile control would be provided by a combination of proven current-drive techniques, including neutral-beam current drive (NBCD), fast-wave current drive (FWCD), and lower-hybrid current drive (LHCD). The relatively low efficiencies of these methods of current drive is compensated largely by the high fraction of current being carried by the bootstrap effect. Thus the bootstrap current greatly leverages the external current drive to produce a configuration which has been shown by the ARIES studies [1] to scale to an attractive reactor.

Density and temperature profile control is obtained by controlling the boundary conditions and by affecting the local transport coefficients. The goal is to obtain self-consistent profiles in the sense that the largest gradients occur in the regions where the relative transport is predicted to be the lowest.

New Tokamak Operating Regimes

The most promising new operating regime for advanced tokamaks that have been identified is called the non-monotonic q configuration. The principal feature of this configuration is a region with significant negative magnetic shear, $dq/d\psi$, where ψ is a poloidal magnetic flux label and $q(\psi)$ is the safety factor. The shear is negative over the central region of the plasma and then returns to the conventional positive shear near the plasma edge. This is illustrated in Fig. 1 where the q -profile for a negative shear configuration is compared with that from a conventional configuration. The associated plasma current density in the negative shear case is hollow. Negative magnetic shear has several beneficial effects. Among these are complete stability to high- n ideal-MHD (magnetohydrodynamic) ballooning modes in the negative magnetic shear region [2,3]. This implies that ballooning-mode stability does not impose any constraint on the pressure gradient there. It permits strong peaking of the pressure profile, which can allow a large ratio of bootstrap to total

plasma current, I_{bs}/I_p , and also produce a bootstrap current profile with favorable shape for stability. The hollow current profile associated with the negative magnetic shear enables raising q_{axis} without simultaneously raising q_{edge} , thereby allowing the plasma current to remain high. Raising q_{axis} is known to be beneficial to both low and high- n instabilities [4,5]. Negative magnetic shear can also contribute to improved plasma confinement by acting to help suppress the primary destabilizing mechanisms for trapped-particle modes [6].

Several other potentially attractive advanced tokamak configurations were also examined during the course of this work, but do not look as attractive overall as the non-monotonic q mode. These include a second-stability configuration similar to the equilibrium used in the ARIES-II reactor study [1], but with a reversed current layer near the plasma boundary to help stabilize the $n=1$ kink mode. Second stability configurations with very high values of the axis safety factor, q_0 , have been studied, but have been found to have such low values of stable beta as to be impractical.

Ideal-MHD Analysis

Ideal-MHD stability analysis was carried out for the non-monotonic q configuration using the BALMSC [3] code for the high- n ideal-MHD ballooning modes and PEST2 [7] for the low- n ideal-MHD kink modes. The high- n ballooning modes are found to be stable for this configuration up to a β_N^* -value of 7.2 for the negative magnetic shear case and 5.1 for the conventional shear case. Above these values, high- n instability occurs in the region where the magnetic shear begins to rise rapidly near the plasma edge. The stability to $n=1, 2$, and 3 external kink modes with boundary conditions corresponding to a conducting shell at infinity and at $1.3a$, where a is the plasma minor radius, have been examined. This shell position of $1.3a$ was chosen to represent the stabilizing effects of the conducting structures and vacuum vessel. For no conducting shell, the highest stable β_N^* -values for the negative magnetic shear and conventional shear cases are 2.5 and 4.2, respectively. However, when we include the stabilizing effects of the conducting shell, the values increase to

Figure 1. The safety factor profiles for the negative magnetic shear and conventional shear cases at β_N^* values of 6.3 and 5.1, respectively.

7.2 and 6.4, respectively. Combining the results of the kink and ballooning stability, the β_N^* -limit is 7.2 for the negative magnetic shear case and 5.1 for the conventional shear case.

Transport Analysis

In order to address the effect of negative magnetic shear on confinement, a comprehensive kinetic toroidal eigenvalue code [8,9] was employed to investigate the linear stability of toroidal drift-type modes. The analysis includes dynamics associated with the most prominent microinstabilities: the ion temperature gradient modes (η_i modes) and the trapped-particle modes. For sufficiently large negative magnetic shear, the orbit-averaged magnetic curvature and gradient drifts change from destabilizing to stabilizing for the majority of the trapped particles, so that enhanced confinement may be expected. Although this effect of negative magnetic shear on trapped-particle instabilities is well known [6], until recently the means for creating such large negative magnetic shear in tokamaks was not established. The other dominant destabilizing mechanism for toroidal drift modes is the ion temperature gradient, parameterized by η_i . For typical tokamak parameters, suppression of this mechanism requires $\eta_i < \eta_{icrit}$, with η_{icrit} in the range from 1 to 2 (Ref. [10]). It is important to note that the density and temperature profiles which optimize the bootstrap current for the negative magnetic shear equilibrium reported here, correspond to $\eta_i < \eta_{icrit}$ in the same region of the plasma interior where $s < 0$. For the present discussion of plasma transport the definition of magnetic shear will be $s = (r/q)(dq/dr)$, where r is the (horizontal) flux surface minor radius. It is found that between the radii where s has a minimum and η_i has a minimum, there is a completely stable region for the toroidal drift-type instabilities. This corresponds to the region where $s < 0$ and $\eta_i < \eta_{icrit}$ are simultaneously satisfied. The linear growth rates for the toroidal drift-type instabilities thought to be responsible for anomalous transport for the negative magnetic shear and conventional shear cases are shown in Fig. 2.

Figure 2. The linear growth rates of the toroidal drift mode, for the negative magnetic shear and conventional shear cases, calculated in the electrostatic limit. Here, the perpendicular wavelength is chosen to approximately maximize the growth rate. Collisions and a carbon impurity are included in the calculation.

PIES Studies

To assess tearing-mode stability for the advanced tokamak configurations, we have used the three-dimensional PIES code [11] to calculate the saturated island widths in these configurations. Although there was some concern that the non-monotonic q profile would be unstable to tearing or double tearing modes, we find that this configuration is stable at $\beta = 0$. We also find that it is robust as beta is increased, particularly when compared to the other advanced tokamak configurations that we have looked at, such as the ARIES-I profiles.

Figure 3 shows a Poincaré plot for $\beta = 3.1\%$, corresponding to a beta on axis of about 15%. There is a $5/4$ island of modest width, but there is no evidence of the numerous high- n islands that sometimes appear at high beta for other profiles. These results are consistent with the experimental observations for reversed shear profiles in JET, where for a somewhat different q -profile, $4/3$ modes were observed.

Figure 3. Poincaré plots showing the tearing-mode instability for the non-monotonic q profile at $\beta = 3.1\%$

MH3D Studies

The MH3D code is a three-dimensional time-dependent resistive MHD code. This code is fully non-linear, and has all toroidal and finite-pressure effects included in it. It uses a potential-stream function formulation to allow accurate and efficient integration of the 3-D resistive MHD equations for tokamak parameters.

Up until now, this code has been used exclusively to study “internal” MHD instabilities, that is, those which do not perturb the plasma boundary. As part of the Advanced Tokamak Studies initiative, the capabilities of this code have been extended so that it can now study free-boundary instabilities and the interaction of these with nearby conducting walls and other conducting

structures. This capability is significant in its own right, and this code is now being used to study resistive wall stabilization of the $n=1$ kink mode for long times in the presence of plasma rotation.

Once this new free-boundary capability is fully debugged and exercised, it is planned to further extend the capabilities of this code to allow it to model feedback systems designed to actively stabilize the $n=1$ mode at high beta. This will be the thrust of our FY95 work. If the MH3D code shows this active feedback to be practical, it would be appropriate to develop an experimental proposal to test this result on one of the existing tokamaks such as PBX-M.

Summary

The Advanced Tokamak Studies program at PPPL during FY93 has investigated a number of different new modes for operating tokamaks using several different theoretical tools. The most promising new operational mode found during the year is called the non-monotonic q configuration. It is characterized by having negative magnetic shear in the plasma center, as would be consistent with a current profile which is peaked slightly off-axis. The favorable properties of this operating mode include an almost fully self-sustained current profile from the bootstrap effect, excellent stability to ballooning modes, and a possibility of enhanced confinement through suppression of toroidal drift-type instabilities. The external kink modes are the limiting instability and require additional stabilization through the presence of a conducting shell at approximately 1.3 times the minor radius. Work on the resistive MHD modes is underway and will continue, as will a study of active feedback control applied to tokamak instabilities.

References

- [1] R.W. Conn, F. Najmabadi, *et al.*, "The ARIES-I Reactor Studies Final Report," UCLA Report UCLA-PPG-1323 (1991).
- [2] A. Sykes and M.F. Turner, "A Stable Route to the High β_p Regime," in *Controlled Fusion and Plasma Physics* (Proc. 9th Europ. Conf., Oxford, United Kingdom, 1979) Vol. 1, Contributed Papers (UKAEA Culham Laboratory, 1979) 161.
- [3] J.M. Greene and M.S. Chance, "The Second Region of Stability Against Ballooning Modes," *Nuc. Fusion* **21** (1981) 453.

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- [4] M.J. Gerver, J. Kesner, and J.J. Ramos, "Access to the Second Stability Region in a High-Shear, Low-Aspect-Ratio Tokamak," *Phys. Fluids* **31** (1988) 2674.
 - [5] A. Sykes, J.A. Wesson, and S.J. Cox, "High- β Tokamaks," *Phys. Rev. Lett.* **39** (1977) 757.
 - [6] B.B. Kadomtsev and O.P. Pogutse, "Plasma Instability Due to Particle Trapping in a Toroidal Geometry," *J. Experimental and Theoretical Physics (USSR)* **51** (1966) 1734; English translation: *Soviet Physics JETP* **24** (1967) 1172.
 - [7] R.C. Grimm, R.L. Dewar, and J. Manickam, "Ideal MHD Stability Calculations in Axisymmetric Toroidal Coordinate Systems," *J. Comp. Phys.* **49** (1983) 94.
 - [8] G. Rewoldt, W.M. Tang, and M.S. Chance, "Electromagnetic Kinetic Toroidal Eigenmodes for General Magnetohydrodynamic Equilibria," *Phys. Fluids* **25** (1982) 480.
 - [9] G. Rewoldt, W.M. Tang, and R.J. Hastie, "Collisional Effects on Kinetic Electromagnetic Modes and Associated Quasilinear Transport," *Phys. Fluids* **30** (1987) 807.
 - [10] G. Rewoldt and W.M. Tang, "Toroidal Microinstability Studies of High-Temperature Tokamaks," *Phys. Fluids B* **2** (1990) 318.
 - [11] A. Reiman and H. Greenside, "Calculation of Three-Dimensional MHD Equilibria with Islands and Stochastic Regions," *Computer Phys. Commun.* **43** (1986) 157.

Princeton Spherical Tokamak Experiment

M. Ono, S.M. Kaye, and the PSTX Team

The Princeton Spherical Tokamak Experiment (PSTX) is intended to investigate the confinement and magnetohydrodynamic (MHD) physics of spherical tokamak plasmas ($R/a \leq 1.5$) at performance levels comparable to those of intermediate-scale machines at higher aspect ratio. To do this, the PSTX will operate in a previously unstudied region of parameter space at $R \approx 0.8$ m, $a \approx 0.55$ m, $I_p \leq 1$ MA, $T_e \approx 1$ keV, and $n_e > 10^{19}$ m⁻³. The very-low-aspect-ratio design affords several advantages, such as high beta within the first stability region and confinement and stability improvement owing to the greater access to the good magnetic curvature region. The proposed scale of the device is a logical step beyond the first-generation spherical tokamaks, such as the START at the Culham Laboratory, and is a predecessor to a potentially attractive compact-spherical-tokamak volume neutron source or advanced-fuel-cycle reactor.

Plasma Regimes in PSTX

Initial operation of PSTX will be with ohmic heating only. Extrapolating the START confinement results to the parameter regime of PSTX using either the ITER89-P or Lackner-Gottardi L-mode scaling (which fits the START results quite well), confinement times of 40 msec, beta values (defined using the vacuum toroidal magnetic field) of up to 4.5% (about one third the Troyon beta limit, also defined using the vacuum toroidal magnetic field), and electron temperatures of approximately 1 keV for $I_p = 1$ MA, $B_t = 0.5$ T, $\langle n_e \rangle = 3 \times 10^{19}$ m⁻³, and a self-consistent ohmic-heating power of 1.5 MW can be expected.

These performance projections are borne out in simulations of PSTX plasmas using the TSC code. Enhanced performance may be expected if the plasmas transition into the H-mode (high-confinement mode); the recent threshold scaling work by the ITER H-Mode Database Working Group implies a transition threshold power of approximately 400 kW for PSTX, well below the ohmic-heating power. For these parameters, the current flattop time of 300 msec (for 1 MA) is at least three times the current redistribution time,

giving the ability to study current-profile modification using noninductive currents, such as the bootstrap current in the near term with ohmic operation, and possibly radio-frequency (rf) current-drive in a later phase. Other physics issues that can be studied during ohmic operation include confinement and stability improvement, neoclassical physics including bootstrap current drive, and the use of the natural elongation and triangularity of low-aspect-ratio plasmas as a means of avoiding a demanding active plasma control system. Improved confinement at low aspect ratio will be studied directly; the PSTX is designed with the capability of operating over a range of aspect ratio from $R/a = 1.45$ to 2.5 .

After the initial operating period of ohmic-only plasmas, the device can be upgraded to include auxiliary heating/current-drive, advanced divertors, and long-pulse operation. Even modest auxiliary heating (about 4 MW) will allow attainment of the Troyon beta limit (13%) for H-mode plasmas; this low power requirement is easily satisfied through neutral beams or rf heating, and it will allow for exploration of possible magnetohydrodynamic stability improvements at low aspect ratio. The auxiliary heating would also produce high enough power densities so that, along with the long field-line path-length from plasma to neutralizer plate, meaningful divertor studies could be performed.

PSTX Design Approach

To satisfy next-generation spherical-tokamak experimental requirements, the PSTX should have sufficient flexibility for various plasma operation scenarios, as well as excellent access for plasma diagnostics. The device should be upgradable for auxiliary heating/current-drive, divertor studies, and long-pulse operation. Lastly, the device should be economical in terms of construction and operation. To satisfy these requirements, the present PSTX design takes advantage of unique low-aspect-ratio opportunities for structural efficiency. The device consists of only five main demountable components: (i) vacuum vessel, (ii) center-stack, (iii) outer toroidal-field coils, (iv) poloidal-field coils, and (v) support structures. These components can be fabricated independently off-site and assembled with minimum on-site effort. This demountable design also makes future upgrades feasible and less costly. Additional construction cost savings can be achieved by utilizing S-1 components and the existing toroidal field (TF), poloidal-field (PF), and ohmic-heating (OH) supplies which will be shared with PBX-M. The set of diagnostic and control hardware will utilize the available TFTR equipment for an additional cost reduction.

The PSTX Tokamak

A three-dimensional finite-element model of the PSTX tokamak is shown in Fig. 1. The main device components are:

- A vacuum chamber (approximately 11-feet wide by 10-feet high with 0.5-inch thick stainless-steel wall) composed of upper and lower dorns (the two halves of the S-1 vacuum chamber) and a middle cylindrical chamber with all-metal seals. The chamber, which is without any electrical dc-breaks, is being designed to withstand anticipated eddy-current forces due to worst-case disruptions. The continuous vacuum chamber also shields the ohmic coil leads from the voltage spikes induced during disruptions. Importantly, the vacuum chamber provides basic structural

support for the coil systems (TF, OH, and PF). The vacuum chamber includes twelve continuous poloidal limiters as plasma facing components. The chamber contains twenty-two large vertical flanges (14 to 16.5 inches in diameter), twelve 18-inch and twenty-four 8-inch horizontal flanges.

- A center stack, the most elaborate component of PSTX, is located in the core of the tokamak. It contains the 108-conductor inner TF bundle, the four-layered ohmic-heating solenoid, and the PF #1-A and 1-B Coils. The OH solenoid and PF #1-A coils are used

Figure 1. A three-dimensional finite-element model of the Princeton Spherical Tokamak Experiment (PSTX) tokamak.

in conjunction with other PF coils to provide up to two volt-seconds of ohmic-heating capability, which is calculated to be sufficient for 1-MA operation. These coils are encased in a 0.25-inch Inconel cylinder, which also serves as the inner vacuum boundary.

- Twelve outer TF coils (each containing 9 conductors) that connect to the inner TF bundle and complete the 108-turn TF-coil circuit.
- Poloidal-field coils #2, #3, and #4 (together with PF #1-B) used for plasma stability and shaping. These coils are from the S-1 device and they are left mounted on the vacuum vessel in the same positions as they were in S-1.
- Supporting structures, designed to support various static and dynamic forces. The support legs support the entire weight of the device with an adequate safety margin. The support structures between coils and vacuum vessel are designed to take various anticipated electromagnetic and thermal forces.

Preconceptual Engineering Studies

A Preconceptual engineering design study was carried out for the PSTX device components and auxiliary systems. A detailed PSTX component-fabrication cost and schedule was developed; this includes the vacuum chamber, center-stack, outer TF coils, PF coils, plasma-facing components, and support structures. The vacuum chamber design was analyzed using a finite element stress analysis code. A detailed stress analysis study has been initiated to model the response of the PSTX components to various static and dynamic forces, such as disruption and thermal stresses. The central tool for this study is a finite-element model of the PSTX tokamak structure, as shown in Fig. 1. Each component can be geometrically altered or enhanced independently by the interactive graphical preprocessor and postprocessor. The model can be exercised and modified to provide an optimum design of the vacuum vessel and the support structures. The eddy current analyses for the vacuum vessel structure for various moving disruption outputs from the TSC code have been carried out using an eddy current and force code.

Various auxiliary systems were examined including the water-cooling system, TF motor-generator supplies, PF and OH rectifier supplies, vacuum pumping system, control system, and data acquisition system. The Instrumentation and Control System, will use 500-point PLC-based hardware that will include all the necessary logic to interface to the area safety interlock

points, as well as all the standard machine control points. The Clock Synchronization System will take advantage of TFTR hardware that will no longer be in use. The set of diagnostic hardware will also benefit from available TFTR equipment. Data acquisition software presently in use by PBX-M will be in place for the initial operating period of PSTX. The site preparation requirements, including dismantling the Princeton Large Torus (PLT), have also been examined. Through these studies, the basic engineering soundness of PSTX has been confirmed and a relatively detailed cost and schedule has been developed.

Acknowledgments

We gratefully acknowledge the important participation of Y.K.M. Peng of the Oak Ridge National Laboratory in helping to develop both the motivation and specific design for PSTX.

Medium-Scale Long-Pulse Device Study

R.J. Goldston and R. Woolley

In the event that funding for the Tokamak Physics Experiment (TPX) is not available, due to financial constraints on the US Department of Energy (DOE), the requirement to accomplish the mission of TPX in some way will not disappear. To prepare at least a partial response to this eventuality, a small effort is being devoted to investigating the question of whether a smaller, copper coil device (MDEV) could meet some fraction of the TPX mission. In general, it has been found that the high-duty-factor aspects of the TPX mission could not be preserved, but some fraction of a long-pulse advanced tokamak mission could be retained, albeit at overall plasma parameters which would result in a much larger extrapolation from MDEV to a demonstration power reactor (DEMO) than from TPX. Of course MDEV would not provide the technological benefit of experience with superconducting coils provided by TPX.

Overview of MDEV

The device design under consideration would be close to one half the linear dimension of TPX, with a comparable (or possibly slightly higher) magnetic field. It would employ copper magnetic-field coils, cooled with pressurized, flowing liquid nitrogen. Heat dissipated in the coils would be removed from the cooling loop in an atmospheric-pressure liquid-nitrogen bath, which would boil off during a pulse. Indeed, the pulse length might ultimately be limited by the acceptable nitrogen boil off. Cost and access constraints would make provisions for remote maintenance impossible, which would significantly limit the deuterium duty factor in MDEV.

The physics parameters of such a device would be much more modest than those of TPX ($I_p R/a \sim 5$ MA versus 9 MA), making the extrapolation to DEMO more distant. The “density window” between fast electron delocalization at low density and collisional modification of the bootstrap current profile at high density is narrower in MDEV than in TPX, but the loss of operating window is at the less attractive low-density end.

MDEV Design Status

No detailed design, nor even preconceptual layout, has begun. Studies have concentrated on scaling the results from the SSAT-R (Steady-State Advanced Tokamak-Resistive) design, which was developed during the National Task Force effort, to the MDEV device size, in order to get a feeling for coil stresses and thermomechanical issues. In general, it appears that a continuously liquid-nitrogen-cooled device of this scale is not impossible, and its capital cost would be substantially lower than that of TPX. However, much more study is needed to understand such a device, which would use a relatively untested active cooling technology. The safety and environmental aspects of the use of large quantities of liquid nitrogen also need to be assessed.

The key feasibility question for MDEV is whether adequate access can be provided for heating and current drive, for a "slot" divertor, for diagnostics, and for maintenance, in such a small device. The MDEV would be located in the TFTR Hot Cell (not the main Test Cell as planned for TPX). Use of TFTR neutral beams for plasma heating would probably be precluded by inadequate access between the toroidal-field coils.

Collaborations Planning and Exploration

P.H. Rutherford and K.M. Young

The goal of future PPPL fusion collaborations will be to provide experienced research personnel, engineering support, and advanced auxiliary device hardware to several operating toroidal confinement facilities for the overall benefit of the US and world fusion effort. The collaborations will make use of special TFTR expertise, but will focus programmatically on those areas of advanced tokamak research which will be major components of the future Tokamak Physics Experiment (TPX) program. By implementing this program of collaborations in FY95-98, the Laboratory will retain the key technical expertise needed to operate, and obtain maximum programmatic benefit from, the TPX device scheduled to begin operations in FY2000.

In FY95, much of the Laboratory's experimental effort will still be devoted to analysis and interpretation of the TFTR data and to operation of PBX-M. Indeed, because of the importance of TFTR's deuterium-tritium (D-T) experiments to the overall fusion program, especially the International Thermonuclear Experimental Reactor (ITER), it is the Laboratory's plan that approximately 80% of the TFTR experimental team [as measured by full-time equivalents (FTEs)] will remain on TFTR through FY95. Nonetheless, it will be possible to initiate in FY95 a program of collaborations on other US and foreign facilities that will grow in FY96 and subsequent years to a level (in research FTEs and their direct support) corresponding approximately to the present TFTR experimental program.

In FY93, a small amount of Laboratory Program Development Activities (LPDA) funding was allocated for planning and exploratory discussions relating to possible future collaborations. Typically, the funds were used to support trips to facilities on which collaborations are to be proposed, and for the effort needed for the technical analysis of various possible hardware components of these collaborations.

JET

An initial meeting between a team from PPPL (D.M. Meade, K.M. McGuire, and K.M. Young) and the Joint European Torus (JET) management group,

including Drs. Keilhacker, Gibson, Thomas, Stott, and Jacquinet, was held at JET to investigate potential areas for collaboration after TFTR shuts down. The discussions were very successful in helping PPPL to define principal areas to be developed. In the area of diagnostics, PPPL is now preparing a study of fiber optics in the TFTR radiation environment, with the intention of developing an active spectrometer for ion temperature measurements on JET to be part of the TFTR contribution to the physics program in later years. It was also possible to identify a number of physics and instrumentation topics where the JET team would like to see PPPL support. The visit was used also to develop the details of the JET participation in the TFTR D-T program. A particularly good interaction from this meeting is JET's participation in the new alpha-scattering diagnostic on TFTR, whose success in interpretation will feed directly into the subsequent JET experiment. In areas other than diagnostics, possible collaborative activities were identified in the confinement physics and transport analysis (using TRANSP) of the JET D-T experiments, in radio-frequency heating and current drive experiments (lower-hybrid current drive and fast-wave current drive), in physics and neutral-beam operations and, with a nearer-term focus, in various areas of engineering and computer support.

Japan: LHD (NIFS) and JT-60U (JAERI)

The purpose of these visits by K.M. Young (in part with D.M. Meade and K.M. McGuire) and by D. Johnson was to discuss with the management and staffs of the Large Helical Device (LHD) at the National Institute for Fusion Science (NIFS), Nagoya, and of the JT-60U device at the Japan Atomic Energy Research Institute (JAERI) the possibilities of TFTR staff contributing to their programs. While the emphasis was on the period after FY94, some very successful arrangements for collaborative activity in FY93 were made with the JT-60U group. Concentrating on the diagnostics aspects of the visit, discussions were held mostly with Drs. Fujiwara, Fujita, and Hamada about PPPL's potential contributions to LHD. Clear possibilities exist for (i) developing their bolometer system, with funding in 1997, (ii) helping them with their neutral-particle analyzers (to be followed up by a visit to PPPL later this year), (iii) theoretical studies, and (iv) short-term consulting visits. At JAERI, it was immediately agreed that PPPL could (i) contribute to their planning of how to upgrade the JT-60U electron temperature measurements, (ii) help with the implementation of their Motional Stark Effect (MSE) system, and (iii) make suggestions to enhance their ability to do fluctuation measurements. In addition, D. Johnson went to JAERI and worked with their

staff on three possible ideas for the Thomson scattering system, one of which has now been chosen for implementation at the end of 1995. The main hardware components will be provided by Japanese industry, but the JT-60U group are eager for PPPL to continue supporting the optical design. The Electron Cyclotron Emission (ECE) systems were reviewed, and an improved method of density interpretation from the visible bremsstrahlung signal has been provided by H. Park. Regular consultation on the MSE has been provided, and more help will be needed soon to make their multichannel system work.

Tore Supra

Several areas of possible enhanced collaborations between PPPL and Tore Supra were discussed in a visit by K.M. McGuire in April. These include (i) transport studies using the TRANSP code, (ii) stability and ripple calculation by the PPPL theory group, (iii) limiter alignment techniques, both from the physics and engineering standpoint, and (iv) possible help with neutral-beam engineering and operation.

MIT (C-Mod)

Following meetings at the management level in March, at which tentative ideas were developed on possible equipment TFTR physicists could bring to the Alcator C-Mod device in FY95 and thereafter, B. Grek, A. Ramsey, G. Schilling, and K.M. Young visited the Massachusetts Institute of Technology (MIT) in May to discuss detailed possibilities with the appropriate C-Mod experimentalists. Among the MIT contacts were E. Marmor, J. Terry, B. Lipschultz, and R. Watterson. There were extensive discussions about a possible diagnostic neutral beam, which would be necessary if any of the active spectroscopy ideas which PPPL could pursue on C-Mod are to be implemented. In the areas of spectroscopy and Thomson scattering, there was considerable discussion of divertor diagnostics, which the MIT group see as being key to their operation. Initial discussions of possible areas of confinement physics collaboration were held (subsequently these have been developed somewhat further) which should lead to a well-integrated program. In addition the PPPL group had an opportunity to see the data-acquisition system in operation as C-Mod was just being brought up.

General Atomics (DIII-D)

As part of a series of discussions with General Atomics (GA) on possible collaborations on plasma control systems, R.E. Bell visited GA in February to

examine and discuss the digital plasma control system now in use on DIII-D, and to explore other possible collaborations on plasma control. The DIII-D tokamak had for many years relied on an analog system to control the plasma shape and position. However, DIII-D has recently developed a digital control system which is capable of handling far more complex control schemes than their analog system could. The implementation of control algorithms in software rather than hardware also gives a digital system greater flexibility. The DIII-D Advanced Plasma Control System was developed principally by J. Ferron. It is a digital system based on the Intel i860 microprocessor. The major innovation, on the part of GA, was to create a FIFO memory daughter board which serves as an interface between the digitizers and the microprocessor. A digitizer controller card was also created to supervise the transfers.

The digital system developed at GA would also satisfy the needs of PBX-M to improve its analog control scheme. The sharing of identical hardware would also make possible the development of mutual software packages. While the control algorithms on DIII-D and PBX-M would necessarily differ, the interface software could be shared. Unlike PPPL, the availability of software development personnel is limited at GA. Moreover, the PBX-M device, once it has acquired a similar digital system, would benefit from the DIII-D's experience with this system. Thus, the software development resources at PPPL would benefit GA in the development of a general user interface for mutual use.

Another area for possible collaboration between PBX-M and DIII-D might be on advanced plasma control algorithms. For example, PBX-M has been examining neural networks as a means of quickly calculating useful plasma parameters so that they may be used in a feedback loop.

GA has expressed an interest in collaborations in these areas since this trip. Discussions between PPPL/PBX-M and GA/DIII-D are continuing.

Divertor Simulator Studies

S.A. Cohen for the IDEAL Team

Heat and helium exhaust from tokamaks have long been recognized as major challenges to the design of a reliable and robust steady-state fusion reactor. To address these problems, the International Thermonuclear Experimental Reactor (ITER) Joint Central Team (JCT) issued a *Request for Task Proposal* that describes a plasma facility for simulating divertor conditions. This facility would test the physics of novel divertor concepts, as well as the response of prototype modules to long-term ITER-like plasma exposure. The research supported by the activity described in this report focused on finding a simulator design that could produce the plasma parameters and geometry necessary for testing divertor concepts and prototype modules in a time frame consistent with the ITER schedule. Achieving an accurate cost estimate and ensuring principles of conservative engineering were also high priority in this preconceptual design. The studies were carried out primarily by scientists from the Princeton Plasma Physics Laboratory and Grumman Corporation, with supporting studies by scientists and engineers from seven other US laboratories, ten private companies, and the Kurchatov Institute.

ITER Divertor Experiment and Laboratory

Present tokamaks are inadequate for ITER divertor testing because of low fluence, low power levels, small divertor dimensions, short-pulse lengths, low duty factor, poor access, low attained neutral pressure, and long turn-around time between tests. Single-component test facilities, such as electron-beams, though useful for heat-flux testing, cannot provide the necessary integrated environment, which mandates simultaneous exposure to energetic hydrogen atoms and hot plasma. The design point (ITER Divertor Experiment and Laboratory—IDEAL) upon which this effort concentrated was a 30-m long magnetized plasma device, heated by 10 MW of radio-frequency (rf) power in the ion cyclotron range of frequencies (ICRF) (see Fig. 1). Predicted parameters are listed in Table 1. These satisfy or exceed the requirements set by the Joint Central Team. It is particularly noteworthy that the plasma fluence, an essential element for the divertor technology testing program, attainable in IDEAL exceeds—by a factor greater than 300—the combined value obtainable in all the major tokamaks in the world.

The IDEAL is designed for continuous steady-state operation. The axial magnetic field of 2.5-3.5 tesla would be supplied by 30 one-meter-long superconducting magnets of 1-m nominal bore. The superconducting magnet design is based on nearly identical specifications from the MRI industry. Plasma shaping and stability are accomplished with normal-conductor quadrupole magnets, mounted on the exterior of the approximately 0.6-m diameter vacuum vessel. Rapid assembly and maintenance procedures are possible in this linear modular configuration. For example, the time to replace the divertor modules is less than one week, a factor of 25 faster than tokamaks.

Figure 1. ITER Divertor Experiment and Laboratory (IDEAL) assembly—the 30-meter linear machine.

Table 1. ITER Divertor Experiment and Laboratory (IDEAL) Parameters.

Parameter	IDEAL	JCT Spec*
B (T)	3.5	2.5
$L_{ }$ (m)	10	5
$A_{\perp}$ (cm ²)	100	100
Plasma Shape (cm × cm)	5 × 20	—
Divertor Module (m)	0.4 × 2.5	0.2 × 2.0
$n_e(0)$ (10 ²⁰ m ⁻³)	2	1
$T_e(0)$ (eV)	150	100
$T_i(0)$ (eV)	1000	300
$T_e(d)$ (eV)	1-100	1-100
$q_{ }(0)$ (MW/m ²)	6000	1000
Ion Species	D	D
Pulse Duration (sec)	>1000	>400
Duty Factor (%)	35	>20
Heating Power (MW)		
ICRF	10	—
ECH	0.2	—
OH	1.0	—

* These specifications are based on discussions with the ITER Director and represent slight revisions from those originally issued.

Plasma parameters would be controlled by means of a versatile ICRF heating system, a central gas box for fueling, and pumps placed in the divertor regions at both ends of IDEAL. The ICRF system can simultaneously supply power at four different frequencies, allowing both ion heating, via the fundamental ion cyclotron resonance, and electron heating, via Landau damping. The radio-frequency saddle-coil antenna design is efficient, requiring four-times less power throughput and voltage standoff capabilities than presently attained on other plasma devices. Density control via ICRF coupling is also assured because of the proper choice of wavelength possible with $m=1$ antennae. Excellent on-axis heating ($q_{||} = 670 \text{ MW/m}^2$ per megawatt of input power) is another benefit of this antenna design. Operation at densities over $2 \times 10^{20} \text{ m}^{-3}$ is possible using fast wave excitation. Initial hydrogen heating experiments will be superseded by hydrogen/deuterium mixtures, and finally pure deuterium plasmas, after the rf power supplies are converted to 20-40 MHz operation.

Electron temperatures up to approximately 150 eV have been predicted by fluid codes and 100 eV by kinetic codes. Ion temperatures can exceed 1 keV. The central plasma density would be variable between 1 and $20 \times 10^{20} \text{ m}^{-3}$. Divertor region densities would be 10-100 times higher. The experimental physics basis for IDEAL is strongly supported by data from ICRF-heated mirror machines.

Schedule, Siting, and Cost

The schedule for construction and operation of IDEAL must meet the ITER Engineering Design Activity requirements. Engineering design lasts 6 months; procurement, concurrently started, extends for 12 months; and construction, beginning with site preparation, lasts for 30 months.

An 18-month physics assessment period would test ultrahigh recycling, gas-target, and radiative-divertor concepts. In pure hydrogen and mixed hydrogen/deuterium plasmas, power flows would exceed 1 GW/m^2 for 10 sec, at a duty factor of 5%. Sheath effects, helium exhaust, and cross-field plasma transport would also be studied. Five divertor changeovers will be made. Two one-year technology testing periods would ensue, with duty factor being raised to 35%. Pure deuterium plasmas would be used and pulse lengths at full power would exceed 1000 sec. Erosion/redeposition, thermal fatigue, and loss-of-flow-accident studies would be carried out. The IDEAL would provide, within three years of its completed construction, a plasma fluence equal to about ten years of ITER operation at its nominal 10% duty factor. The operating plan for IDEAL is illustrated in Fig. 2.

PPPL is a suitable site for IDEAL because of its capability to handle beryllium components and extensive site credits (*ca* \$15M). These include ICRF power supplies, magnet power supplies, liquid helium refrigerators, operation hall, diagnostic equipment, and trained personnel.

The total cost for IDEAL at PPPL is estimated to be \$32M, including a 25% contingency. (A 25-m device, heated with 2 MW of ICRF, could be constructed for about \$24M.) Operating expenses are about \$10M per year.

A presentation on IDEAL was made to the US ITER Steering Committee (ISCUS), based on the single 30-m design point, which has now been described in detail in PPPL TM 402 (Ref. [1]). The ISCUS stated that both the costs and the physics performance were credible and that existing (tokamak and electron-beam) facilities cannot adequately address all the relevant physics and technology issues for the ITER divertor. At a subsequent IDEAL review at the ITER San Diego Co-Center, the ITER Director reaffirmed his strong support for a divertor simulator with the full capabilities described in TM-402. Various aspects of the IDEAL preconceptual design have been described in several conference presentations [2-6].

Figure 2. Plasma fluence to IDEAL divertor plates during the physics and technology phases in terms of ITER burn years at either 1.7 GW (open shapes) or 3.0 GW (solid shapes). Circles represent high-recycling divertor (HRD) plates. Squares represent gas-target divertor (GTD) tubes. Triangles represent the Joint Central Team's choice for technology testing.

References:

- [1] S.A. Cohen, R. Majeski, R. Motley, S. Mendelsohn, B. Genzlinger, R.B. Hooper, *et al.*, "IDEAL Pre-Conceptual Design Report," Princeton University Plasma Physics Laboratory Report TM-402 (1993).
- [2] R.W. Motley, R. Majeski, S.A. Cohen, M. Diesso, and J.R. Wilson, "ICRF Coil for the IDEAL Plasma," *AIP Conference Proceedings APS 10th Topical Conference on RF Power in Plasmas*, American Institute of Physics, Boston, April 1993.

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- [3] S.A. Cohen, M. Diesso, R. Majeski, R. Motley, D. Post, L. Zakharov, S. Mendelsohn, M. Phillips, and R. Gentzlinger, "The Experimental Program of the IDEAL Device," IEEE Meeting, 15th Symposium on *Fusion Engineering*, Hyannis, 1993, to be published.
 - [4] R.C. Gentzlinger, S. Mendelsohn, B. Abel, I. Birnbaum, U. Christensen, S. Kalsi, J. Mueller, M. Phillips, J. Swinton, D. Weissenburger, S. Cohen, E. Fredd, R. Majeski, R. Motley, and R. Walls, "Ideal Pre-Conceptual Design Development," IEEE Meeting, 15th Symposium on *Fusion Engineering*, Hyannis, 1993, to be published.
 - [5] M.W. Phillips, R. Gentzlinger, J. Swinton, D. Weissenburger, S.A. Cohen, R. Majeski, and R. Motley, "Stability issues of the IDEAL Conceptual Design Study," IEEE Meeting, 15th Symposium on *Fusion Engineering*, Hyannis, 1993, to be published.
 - [6] S.A. Cohen, B.J. Braams, M. Diesso, H. Fishman, R. Majeski, A. Pigarov, V. Pozharov, R. Motley, and K. Werley, "Fluid Modeling of the IDEAL Device," 20th European Physical Society Meeting on *Plasma Physics and Controlled Fusion*, Lisbon, 1993, to be published.

Gyrofluid Simulation

G.W. Hammett, M.A. Beer, W. Dorland and S.A. Smith

The goal of this program is to develop realistic nonlinear three-dimensional simulations of tokamak plasma turbulence in order (i) to understand anomalous transport in present experiments, (ii) to increase the confidence of projections to future fusion reactors, and (iii) to aid the search for further enhanced-confinement regimes. This goal is coming closer to reality because of the dramatic growth of massively parallel computing and significant advances in plasma theory. "Gyrofluid," or "gyro-Landau-fluid," equations provide a way to reduce the dimensionality of the gyrokinetic equations to a small set of fluid equations, while retaining models for approximating important kinetic effects such as wave-particle resonances (Landau damping) and gyro-orbit averaging.

Code Development

The gyrofluid approach is relatively new [1]. Efforts have been made to generalize the equations to include more toroidal effects and to test them against established gyrokinetic results while, at the same time, developing a nonlinear toroidal gyrofluid code and carrying out initial simulations. Highlights of technical progress in the past year include:

- Successful benchmarking of nonlinear gyrofluid and gyrokinetic codes in typical parameter regimes, including the identification of some special cases where the gyrofluid approach needs many moments (and would be inefficient) for accuracy [2,3].
- Extension of the toroidal gyrofluid equations to higher moments (for more accuracy near marginal stability) and to include neo-classical trapped-particle effects, important in controlling poloidal rotation.
- Generalization of the nonlinear gyrofluid turbulence code to a flux-tube in toroidal geometry, using a field-line coordinate system proposed by S. Cowley for efficient representation of nonlinear tokamak turbulence problems.

Applications

At present, the code models toroidal ion-temperature-gradient-driven turbulence, which is probably dominant in the core of many tokamaks. Nonadiabatic electron dynamics need to be added to destabilize the drift-wave branch, which is important in some experimental regimes. More detailed parameter scans with higher resolution need to be done, but initial nonlinear studies have found two important physics results, illustrated in Figs. 1 and 2, which may be summarized as follows [4]:

- An important nonlinear mechanism is the feedback between sheared flows generated by the turbulence and the stabilizing influence of these flows on the turbulence. (This effect has been noted in H-mode studies and edge-turbulence simulations by others, but our work was the first to show that it is important for core turbulence as well. Previous ion-temperature-gradient simulations had used a variant of an adiabatic electron response which allowed the electrons to strongly short-out the radial electric field associated with $E \times B$ flows.) It appears crucial to properly treat all of the flow generation and damping mechanisms (including neoclassical toroidal effects). If the sheared flow is artificially suppressed, then elongated streamers form and the heat transport increases without bounds (Fig. 1).

Figure 1. Three-dimensional nonlinear gyrofluid simulations of toroidal ion-temperature gradient turbulence for $\eta_i = 4$, $s = 1.5$, $L_n/R = 0.4$, $T_i = T_e$, and $q = 2.4$. In the figure, χ_i versus time when poloidal flows are suppressed (dashed line) or allowed to self-consistently evolve (solid line).

- The nonlinear spectrum is peaked at significantly longer wavelength ($k_\theta \rho_i = 0.15$) than the fastest growing linear instability ($k_\theta \rho_i = 0.4$), producing a spectrum (Fig. 2) with qualitative features similar to the beam-emission-spectrometry (BES) measure-

ments. More work needs to be done to understand what controls the scaling of the peak $k_\theta \rho_i$ in our simulations. (BES measurements on TFTR indicate $k_\theta \rho_i$ goes down towards the edge, and the dimensionless scaling experiments indicate Bohm transport, suggesting that the dominant k_θ should scale independently of ρ_i .) Nevertheless, it is encouraging to see that nonlinear effects can produce long-wavelength features similar to the experiments.

Figure 2. Time-averaged fluctuation spectrum versus $k_\theta \rho_i$ showing that the spectrum is downshifted from the fastest growing linear mode. The same case as in Fig. 1 (with self-consistent poloidal flows).

References

- [1] W. Dorland and G.W. Hammett, "Gyrofluid Turbulence Models with Kinetic Effects," *Phys. Fluids B* **5** (1993) 812.
- [2] W. Dorland, "Gyrofluid Models of Plasma Turbulence", Ph.D. Thesis, Princeton University, 1993.
- [3] Invited talk at the 1993 American Physical Society Division of Plasma Physics meeting. Submitted for publication as S.E. Parker, W. Dorland, R.A. Santoro, M.A. Beer, Q.P. Liu, W.W. Lee, G.W. Hammett, and C.Z. Cheng, "Comparisons of Gyrofluid and Gyrokinetic Simulations of Tokamak Turbulence," *Phys. Fluids B*.
- [4] Invited talk at the 1993 Sherwood International Fusion Theory Conference. Published as G.W. Hammett, M.A. Beer, W. Dorland, S.C. Cowley, and S.A. Smith, "Developments in the Gyrofluid Approach to Tokamak Turbulence, Simulations," *Plasma Phys. Control. Fusion* **35** (1993) 973.

Feedback Kink Study

**M.S. Chance, C.Z. Cheng, S.C. Jardin, J.L. Johnson,
J.Manickam, and E.J. Valeo**

Theoretical magnetohydrodynamic (MHD) studies of the external kink mode in advanced tokamaks with a nearby conducting shell generally show that the $n=1$ mode is the most dangerous, the higher modes generally setting in at higher beta values, or when a closed shell is further from the plasma or a partial wall is less enclosing. This is convenient since any proposed active feedback scheme could then be designed with relatively simple electromechanical structures. In order for the active feedback to be effective, the mode must be slowed to tolerable time scales by the use of passive stabilizers. This is done by positioning them such that they would completely stabilize the kink modes if the plates were perfectly conducting; then, since the actual plates would have finite resistivity, the modes will be slowed down to roughly the order of the L/R time of the plates, which is designed to be long enough to accommodate the active feedback. Theoretical models are being constructed to simulate and assess the effect of these plates. PBX-M's passive plates are also already providing an experimental data base. Although the theory so far neglects the resistivity of the plates, it can closely simulate the real geometry, so that the requisite plates for any particular application can be designed.

Active feedback requires knowing the physical characteristics of the mode for properly shaping the feedback coils, and also designing pickup loops outside the plasma to trigger the feedback circuits with the correct phase. To this end, the requisite theory has been developed to calculate and display the eddy currents in a conducting shell due to a simulation of a growing kink mode. Since these currents are also exactly those needed to suppress the mode, by keeping the shell at a constant distance away from the plasma, it can be ascertained from the current distribution where one should optimally distribute the feedback forces. An example shown in Fig. 1, taken from a simulation of a proposed Tokamak Physics Experiment (TPX) discharge, shows that the feedback must be localized mostly at the unfavorable curvature side of the plasma to be effective. This is unfortunately also where one needs to gain access to the plasma discharge for beam and wave heating, current drive,

diagnostics, etc. The passive plates and the feedback mechanism must therefore be carefully designed.

The pick-up (Mirnov) loops simulation, also theoretically calculated, may provide a more practical and realistic feedback scenario since, unlike the passive plates, the Mirnov loops do not provide any stabilizing forces of their own.

Two avenues for active feedback are being pursued concurrently; namely, magnetic feedback coils and radio-frequency (rf) ponderomotive forces.

Magnetic Feedback Coils

This is the traditional and most direct method, in which the appropriate coils can be designed using theoretical and experimental (i.e., PBX-M) results. Such coils are being designed for a possible PBX-M upgrade. It is of course mandatory that any feedback scheme be tested for proof-of-principle and effectiveness before installation on large, costly devices. On the theoretical front, a scheme has been proposed for incorporating the simulation into a modified NOVA code. This would include the incorporation of the inhomogeneous forcing terms from the feedback coils. These terms would manifest themselves as modifications of the vacuum boundary conditions in the code.

Radio-Frequency Ponderomotive Force

There is some uncertainty about the viability of using rf ponderomotive forces to stabilize the kink, even though there is strong evidence that it works effectively on the interchange modes in mirror machines. Nevertheless, strong supporting theoretical calculations and also the possibility of performing a

Figure 1. The simulated eddy current pattern in a closed conducting shell due to an external kink perturbation in TPX. The shell is unfolded to a two-dimensional l_ϕ versus l_θ plane over one toroidal and poloidal period in which l_ϕ and l_θ are normalized to the local major and minor circumferences, respectively. The plot shows the helical structure of the currents strongly localized in the outer-major-radius side of the torus.

convincing experimental test on PBX-M motivate PPPL's involvement in a study in which SAIC is playing a major advocate role. The existing relevant theoretical papers are being reviewed and estimates carried out of the power needed, versus that available from the ion-Bernstein-wave system of PBX-M. It was concluded that substantial savings of power would be realized with phase-modulated (as opposed to continuous) beams, thereby bringing in line the available and required power. As a prelude to the actual stabilization of the kink mode, an axisymmetric experiment is proposed which will examine whether rf ponderomotive forces are able to suppress a vertical axisymmetric instability. This is being formulated, with the rf modification of the equilibrium simulated by incorporation of the relevant terms into the TSC code.

Stellarator Studies

J.L. Johnson, D.A. Monticello, F.W. Perkins,
and A.H. Reiman

During the next decade, the international fusion research community will build and operate facilities that will decide whether the tokamak (TPX) or stellarator [W7-X, Large Helical Device (LHD)] will be the superior steady-state magnetic fusion reactor candidate. PPPL's stellarator studies project is intended to bring PPPL's unique computational capabilities to bear on three-dimensional magnetohydrodynamic (MHD) equilibrium and stability areas and to increase the Laboratory's interactions with stellarator programs in other parts of the world. In addition, codes developed at PPPL have important applications to tokamak research, particularly in assessing the consequences of magnetic-field errors. Most of the effort to date has been with the PIES code—a three-dimensional equilibrium code that can treat configurations with magnetic islands. In contrast, the standard VMEC code assumes nested toroidal flux surfaces. A program to examine stellarator stability with the KSTEP program (in which three-dimensional equilibria are averaged over the toroidal angle and PEST-like analyses are carried out) is continuing.

Equilibrium

Our equilibrium studies have been in collaboration with A. Salas (Madrid) on using the PIES code for TJ-II heliac equilibrium studies and with P. Merkel (Garching) on PIES code calculations of W7-X heliac equilibria with magnetic islands, where configurations with $\langle \beta \rangle = 3.3\%$ have been found. Merkel converted the PIES code to a form where it can handle free-boundary equi-

Figure 1. The solid curves are from PIES code. The dots are for the VMEC code. Plots of $1/2\pi$ (rotational transform) versus Ψ (toroidal flux) for various values of $\langle \beta \rangle$.

libria and it is intended to start utilizing this feature. Our major effort has been to extend a comparison of PIES and VMEC calculations, which was done earlier for a particular ATF (Advanced Toroidal Facility) configuration at zero beta and at and $\langle\beta\rangle = 1\%$, to higher values of β . Good agreement was achieved for $\langle\beta\rangle = 2\%$, with reasonable agreement at $\langle\beta\rangle = 3\%$, as can be seen in Fig 1. The shapes of the magnetic surfaces agree well for $\langle\beta\rangle \leq 2\%$. At $\langle\beta\rangle = 3\%$, the PIES code calculations show magnetic islands that cover approximately 5% of the plasma cross section [Figs. 2(top) and 2(bottom)]. A new free-boundary version of the VMEC code was recently acquired. It is intended to use the PIES and VMEC codes to carry through investigations of two ATF configurations (one with good magnetic surfaces and one where islands have been introduced intentionally with error fields) that have been investigated experimentally. These codes should be useful for a stellarator reactor study that is being started by J. Lyon and the ARIES Team.

Figure 2. Plots of magnetic surfaces for $\langle\beta\rangle = 3\%$ from the (top) PIES code and (bottom) the VMEC code.

Stability

A few years ago, the KSTEP code was extended by our Kyoto colleagues to investigate the linear stability properties of an equilibrium configuration produced by the VMEC code. A finite-aspect-ratio Green's function has been incorporated into the code, improving its effectiveness for free-boundary stability considerations. Application of this code indicates that the ATF configuration is unstable with respect to a free-boundary kink mode if $\langle \beta \rangle \geq 2\%$. The Oak Ridge National Laboratory group believes that this mode can be stabilized by modifying the externally imposed quadrupole field. It is planned to use the VMEC and KSTEP codes to investigate the appropriate changes. Our KSTEP code collaborations with Japan form a part of an international effort to benchmark and compare a variety of stellarator stability codes; a test case has been selected for benchmarking. Good agreement has been achieved for a particular LHD configuration with the results from KSTEP (run by Nakamura at Kyoto) and TWIST (run by Drozdov in Moscow), and it is expected that Cooper (Lausanne) and Anderson (Livermore) will do the calculations with the TERPSICHORE code, Schwab (Garching) will use the CAS-3D code, and Garcia (Madrid) will try the same calculations with an Oak Ridge National Laboratory code.

High-Field Magnet Studies

P. Bonanos

The National Science Foundation (NSF) awarded a grant to Princeton University for construction of a high-field magnet at the Princeton Plasma Physics Laboratory (PPPL). The proposal requesting NSF support was submitted from the Physics and Electrical Engineering Departments, with support from the PPPL Engineering Department. The effort described here is on high-field magnet concepts which would utilize PPPL power supplies. The magnet would be a general-purpose facility, primarily intended for condensed-matter physics research, but other uses would be considered. Although Laboratory Program Development Activities (LPDA) funds were allocated in FY93, the effort to date has utilized only the NSF funds.

Magnet Design

The magnet is a small-bore, pulsed, cylindrical solenoid providing central fields of 60 tesla or more. The design shown in Fig. 1 is a six-coil magnet. This subdivision provides spaces for liquid nitrogen flow between coils and permits cooldown after a pulse in times of the order of one hour. This also allows placement of structural reinforcement periodically, within the winding, so that the innermost layer of any coil is not too distant from a strain-limiting structural support tube.

Power for the outer five coils is provided by the PPPL C-Site motor generator (MG) sets. This is a 12-generator supply rated at 200 MW, 400 MJ, with a maximum current per generator of 22.3 kA

Figure 1. Cross section of the six-coil magnet with internal support cylinders. The magnet coils are represented by the rectangles with the crosses. (93E0624)

and a voltage per generator of 750 volts. Twelve pairs of leads are required to connect the coil to the generators. The innermost coil is powered by a rectifier supply with an open-circuit voltage of 3 kV, a current rating of 22 kA, and a voltage of 1.1 kV at full current. The outer coil is powered first and, as the current approaches its maximum value, the small inner coil is then pulsed with a 0.1-second flat-top. The outer coils may also be operated separately, with the inner coil removed, to provide more than 40 tesla in a 13-cm bore.

The conductor materials proposed to be used include the new high-conductivity copper-nickel-beryllium alloy for most of the coils and the conventional chromium-zirconium copper for the outer coil, where stresses are lower.

Schedule

This is a three year project, begun in the summer of 1992. The magnet is proposed to be located in the Princeton Large Torus (PLT) area at PPPL's C-Site using the available generator connections.

Analysis of Helically Wound Solenoids

G.V. Sheffield, R.A. Ellis, III, and L.M. Lontai

Finite element analysis (FEA) is generally used to determine the performance limits of electromagnets. For a typical pancake-wound solenoid, it is only necessary to model a small azimuthal section. Detailed local models are used to analyze details such as layer-to-layer transitions.

Helically wound solenoids pose special problems for finite element analysis. In the azimuthal direction, the full extent of the solenoid must be represented. The vertical position of a conductor varies with angle and, since adjacent radial layers are wound in opposite directions, interpolation elements are generally required to connect grid points at the boundaries between layers.

Data Generator for NASTRAN

A data generator has been developed to prepare a NASTRAN model of a helical solenoid, using a small set of parameters as input. These parameters include the conductor and insulation dimensions, number of turns, number of radial layers, and angular mesh increment. A true representation of a helically wound bar of copper, wrapped with insulation, is generated. Spline interpolation elements are used to connect grid points at the boundaries between layers. Model size can be reduced by combining a conductor turn and its surrounding insulation elements into elements with orthotropic properties (see Fig. 1). The data generator allows the user to specify which turns are to be mod-

PPPL#93X0273

Figure 1. Reduction of grid points through use of orthotropic properties.

eled with orthotropic properties. A load generator, which uses the background field, conductor current, and model geometry as inputs, has been written. Forces and temperatures are output in NASTRAN format.

Application to High-Field Magnet

The data generators have been tested and are presently being used to perform a detailed analysis of the proposed design for the High-Field Magnet (HFM). The portion of the High-Field Magnetic to be analyzed is a six-layer, thirty-eight-turn solenoid. It is one of the most highly stressed regions. Because of vertical symmetry, only nineteen turns need be analyzed. An additional data generator creates a finite-element representation of the tabs that connect the layers. The coil is divided into eight-degree angular increments. Orthotropic properties are used away from the areas of interest in order to reduce model size.

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Several test cases have been run on a model representing a small part of the solenoid (see Fig. 2). Two out of the six layers, and three out of the nineteen turns, are modeled. Two of the turns are represented by orthotropic properties. Connection of the tabs to each other is modeled with multi-point constraints.

The results of the NASTRAN code runs are postprocessed using the TAURUS code, or the MSC/XL code, on a local workstation. Displacement plots and stress contours are available.

Figure 2. Finite-element geometry.

X-Ray Lithography with Tokamak Radiation

**S. Suckewer, C.H. Skinner, D. DiCicco, H. Kugel (PPPL),
and J. Maldonado (IBM)**

The primary motivation of investigating X-ray lithography with tokamak radiation is the possibility of building a relatively inexpensive device which will provide a high flux of X-rays, uniformly covering large areas of wafers. The experiments reported here form a first stage: the exposure of photoresists to the X-ray flux from the PBX-M tokamak in order to experimentally determine the exposure level needed. The results are relevant to two possible lithographic techniques: proximity (contact) lithography and projection lithography. In proximity lithography, a mask is positioned in front of the resist, and the shadow cast on the resist forms the desired exposure pattern that is then chemically developed. This technique is presently used on synchrotrons using X-rays in the 20-Å region. An alternative technique, presently under development, is projection lithography in which synthetic multilayer mirrors are used to project an image of the mask pattern onto the resist. For this technique, X-rays in the 130-Å region are used, since it is in this region that the mirrors are most reflective. Measurements were made on the X-ray flux from PBX-M in both the 20-Å and 190-Å regions.

Experimental Arrangement

A vacuum chamber was designed, fabricated, and mounted on the PBX-M tokamak. The vacuum chamber was compact (8 inches by 4 inches diameter) and included a pump-out port, a flange to connect it to a gate valve on PBX-M, and a remotely controlled shutter to control the exposure of the resist. Samples of silicon wafers, coated with photoresist, were positioned in the chamber and were approximately 60 cm from the PBX-M plasma. Both polymethyl methacrylate acid (PMMA) and polybutene-1 sulfone (PBS) photoresists were used. The experiments were run in parallel with the regular PBX-M experimental program, and the resists were exposed in experimental conditions ranging from ohmic plasmas to plasmas with radio-frequency and high-power neutral-beam heating. It is noted that the conditions were far from optimized for high X-ray flux. In particular, strenuous efforts are made

through boronization to reduce the plasma impurities in PBX-M to a minimum—this also minimizes the line radiation available for X-ray exposure. Also, the vacuum chamber was mounted radially to the plasma, whereas a tangential mounting would receive a larger X-ray flux. To compensate for these limitations, the resist was exposed to multiple plasma discharges.

Up to four X-ray filters were positioned in front of the four-resist samples to limit the wavelength range of the X-ray flux. These included foils of Lexan, carbon, aluminum and copper, which have absorption edges in the extreme ultraviolet and X-ray regions. A transmission electron microscope (TEM) grid was placed in contact with the resist to expose a high contrast pattern in the resist. A separate arrangement was used to mimic projection lithography. An existing molybdenum/silicon multilayer mirror was positioned to reflect the incident X-rays through 90 degrees. The mirror reflectivity was measured to be 20% at 190 Å with a passband of 20 Å [a much higher reflectivity (approximately 60%) would be available at 135 Å]. An aluminum filter was used to filter out ultraviolet light. After exposure, the resist was developed in 2:1 MEK:IPA and examined visually. The discharge conditions—total radiated power, hard X-ray flux, radio-frequency and neutral-beam-injection heating power—was also logged for the shots used to expose the resist.

Experimental Results

The results for proximity (contact) lithography can be expressed in terms of the number of discharges needed to obtain a good exposure of the resist. A 1700-Å-thick copper filter was used to limit the flux to X-rays in the 15- to 30-Å region. In a series of experiments, resists were exposed to ohmic discharges (total input power was approximately 0.5 MW) and to discharges with high-power neutral-beam heating (3 MW). In summary, exposure to 10-54 ohmic discharges gave a fair-to-very-good exposure of PBS resist. Discharges with high-power neutral injection were ten times brighter, exposing the resist in 1-4 high-power discharges. The projection lithography results at 190 Å were that 10-44 discharges gave a fair-to-good exposure.

In conclusion, satisfactory resist exposure for both contact and projection lithography was obtained by accumulating the flux from 1 to 44 PBX-M discharges (depending on experimental conditions). It is noted that the impurity level and radiated power in these discharges were low. Accordingly, it is expected that in a tokamak optimized for lithography (seeded with impurities radiating in the desired wavelength range), it will be possible to expose resist in only one shot of a ohmic-powered discharge.

Magnetospheric Plasma Circulation

R.B. White, H. Okuda, and W. Park

The transfer of mass, momentum, and energy into the magnetosphere is driven by the solar wind across the magnetopause boundary layer, and the self-consistent internal dynamics of a plasma determine the convection pattern in the magnetosphere. The solar-wind plasma and its energy are eventually dissipated into the ionosphere in the form of field-aligned current, energetic particles, and terrestrial radiation during a magnetospheric substorm. The triggering mechanism and the subsequent development of a substorm remains one of the most challenging problems in the physics of the magnetosphere. The problem includes both macroscopic and microscopic plasma physics that determine global circulation of a plasma within the magnetosphere and its coupling to the ionosphere via field-aligned current.

Magnetospheric Substorms

For the first year of this effort, the focus has been on the magnetospheric substorm, and in particular on some of the mysteries associated with substorms, such as the impulsive acceleration of charged particles up to hundreds of keV, or even a few MeV, energies. During the work, a collaboration was initiated with Professor W. Heikkila of the Center for Space Sciences at the University of Texas at Dallas. A joint proposal has been submitted to National Aeronautics and Space Administration (NASA) seeking future funding on the magnetospheric substorms.

The starting point of the work is the recognition that the cross-tail current is filamentary during the growth phase of a substorm. Auroral arcs are connected to individual filaments, and hence one must look for instabilities of current filaments in the magnetotail to understand the triggering mechanism for a substorm. It is assumed that an outward meander of the current deep in the magnetotail will create an inductive electric field, which in turn drives the motion of charged particles along the geomagnetic field. This is the well-known substorm current diversion. In order to model the particle acceleration to a few hundred keV, or a few MeV, as observed in the magnetosphere and at the upper atmosphere during a magnetospheric substorm, self-

consistent global magnetohydrodynamic (MHD) simulations in two and three dimensions must be carried out first in order to determine the magnetic configuration of the tail and the associated inductive electric field [1]. The earlier version of our MHD code for the tail dynamics is being modified to employ proper boundary conditions [2]. Since the MHD code does not properly handle particle acceleration due to a field-aligned electric field, a particle code must be used at some point in conjunction with an MHD code in order to determine the energization of the particles in the geomagnetic tail during a substorm. While no results have been obtained from the two- and three-dimensional MHD simulations yet, there are a number of space observations in which particle acceleration to a few hundred keV, or a few MeV, energies have been reported [3]. From these observations, one can estimate the magnitude of an electric field along the geomagnetic field necessary for the observed acceleration.

Shown in Fig. 1 is a sketch of the simulation model for particle acceleration in the geomagnetic tail during a substorm. Two- and three-dimensional MHD simulations will be performed using the model shown in Fig. 1(b) where a global change of the tail configuration will be studied self-consistently. The system will be driven by a dawn-dusk electric field generated by the solar wind flowing across the geomagnetic field. In order to model the particle acceleration, a model simulation will be performed first in which a given electric field lies along a magnetic field, as shown in Fig. 1(a). The electric field may represent an induction field, associated with the magnetic field variation during the growth phase of a substorm. The model simulation will be followed by self-consistent simulations in two and three dimensions.

Consider a one-dimensional model along a magnetic field shown in Fig. 1(a). The model represents a low-density ($n = 0.05\text{-}0.1/\text{cm}^3$) and high-temperature ($T_e = 0.1\text{-}1$ keV and $T_i = 1\text{-}10$ keV) tail plasma, subject to a field-aligned electric field. Assuming the particles are accelerated to a few hundred keV or a few MeV energies during

Figure 1. Sketch of the geomagnetic tail and the numerical simulation models. A one-dimensional particle simulation model shown in (a) will be first used to study particle acceleration along the tail magnetic field in the presence of a parallel electric field. Two- and three-dimensional MHD simulation models shown in (b) will be used to study self-consistent formation of the magnetotail and particle acceleration.

the time of about 1-10 seconds [3], one can estimate that the parallel electric field is in the range $E_{\parallel} = (0.1-1.0)$ mV per meter. The distance that a particle travels along the magnetic field during that time may be a few thousand or a few tens of thousand kilometers. Since the particles cannot be freely accelerated in a plasma due to plasma instabilities, it is interesting to study the acceleration process by means of particle simulations in order to determine the energy distribution for high-energy particles.

In the following, preliminary results of one-dimensional numerical simulations will be presented in the presence of a parallel electric field. A system length of 1024 grid points (1024 Debye lengths) corresponding to a length of about 1000 km will be used. The Debye length is about 1 km in this region of the magnetosphere. The real proton-electron mass ratio will be used in the simulations presented here ($m_i/m_e = 1837$). The electric field strength in the simulations is held at 5 mV per meter during the course of the simulation.

Shown in Figs. 2(a) and (b) are the average drift speeds of the electrons and ions with time. At the early phase of the simulation, electrons and ions are accelerated freely in opposite directions due to the electric field. As soon as the relative drift speed exceeds the electron thermal speed, then the electron-ion two-stream (Buneman) instability is triggered, causing the growth of the wave electric field. The electric field in turn traps the electrons, slowing them down from the free acceleration and causing the saturation of the instability. Since the plasma is subject to a dc electric field, however, electron acceleration resumes after the saturation until the second phase of the two-stream instability is triggered when the electron drift speed exceeds the electron thermal speed and the same process repeats.

Particle distributions shown in Fig. 2(c) for the electrons and in Fig. 2(d) for the ions indicate that the electrons can be separated into two populations: a beam accelerated by the electric field and a plateau caused by the two-stream instability. Both the

Figure 2. Results of a one-dimensional particle simulation using the model shown in Fig. 1(a). Shown in the figure are: (a) average electron drift speed with time, (b) average ion drift speed with time, (c) electron velocity distribution, and (d) ion velocity distribution at $\omega_{pe} t = 5000$.

beam energy and the thermal energy of the plateau are of the order of a few hundred keV at this time. Ions are accelerated both in the direction of and opposite to the direction of the electric field. The acceleration in the direction opposite to the electric field is caused by the wave electric field associated with the two-stream instability. These simulation results support the space observation of the inverse velocity dispersion [3]. Details will be reported soon [4].

References

- [1] W.J. Heikkila, "The Reasons for Magnetospheric Substorms and Solar Flares," *Solar Phys.* **88** (1983) 329.
- [2] K. Min, H. Okuda, and T. Sato, "Numerical Studies on Magnetotail Formation and Driven Reconnection," *J. Geophys. Res.* **90** (1985) 4035.
- [3] D. Sarafopoulos and E. Sarris, "Inverse Velocity Dispersion of Energetic Particles Bursts and Driven Reconnection," *Planet. Space Sci.* **36** (1988) 1181.
- [4] H. Okuda and W.J. Heikkila, "Particle Acceleration and the Inverse Velocity Dispersion in the Magnetotail," in preparation.

Projection Lithography with X-Ray Laser

**S. Suckewer, S. Cowley, L. Meixler,
C. Skinner, and E. Valeo**

This initiative seeks to demonstrate the feasibility of developing a discharge-pumped X-ray laser (DPXL) for X-ray projection lithography. The project had a late start, since the Laboratory Program Development Activities (LPDA) funds were allocated only in the second half of the fiscal year. The LPDA funds are for plasma calculations and for a demonstration of the conceptual feasibility of the idea, in order to provide a sound basis for preparation of a Cooperative Research and Development Agreement (CRADA) proposal. (The Laboratory's partner would be Maxwell Laboratories, which would provide significant support for the project, both technical and financial.)

At a series of meetings, the personnel to be involved in the project identified two key technical issues related to the development of a DPXL. The first issue addressed was the very low plasma impedance, which creates a difficult problem of energy transfer from the fast coaxial line to the X-ray laser target. Relatively simple plasma modeling indicates that, in order to avoid impedance mismatch between the plasma and the fast-pumping coaxial line, the X-ray laser plasma target should be several centimeters long and less than one millimeter in diameter. The possibility of decreasing the impedance of a low-inductance and high-current power supply was also considered. It was decided to pursue the second approach (high-current power supply) in collaboration with the technical staff from Maxwell Laboratories and the Naval Research Laboratory (NRL).

The second issue which was analyzed and will require special attention is a very fast termination current after the plasma temperature reaches its maximum. Difficulties here arose from the large energy stored inductively. Two methods were considered in more detail: (a) a very fast current switch, which diverts current from target plasma to parallel circuit, and (b) change of impedance of the load (target plasma) in order to reflect energy from the target area after the first circle. It was decided that the combination of these two methods should be pursued as the best solution to this problem.

In order to investigate these two crucial problems of the DPXL concept, it was decided to build a prototype of a simplified coaxial line and target chamber, in order to measure power deposition to the plasma as a function of plasma temperature and density. The same target chamber will be used for testing an NRL power supply (modified by Maxwell Laboratories). Design of the fast coaxial current line and target chamber has started; both will be built at PPPL. Construction of an experimental system is expected to begin early in December, 1993, with the experimental program scheduled to begin in January, 1994.

Appendix

Fiscal Year 1993 Financial Summary

(Amounts in Thousand Dollars)

	<u>CC/WP</u>	<u>Allocated</u>	<u>Costed</u>
Advanced Tokamak Studies	1999/ADVT	200.0	245.0
Princeton Spherical Tokamak Experiment	1999/PSTX	150.0	69.7
Medium-Scale Long-Pulse Device Study	1999/MDEV	200.0	43.9
Collaborations Planning and Exploration	1999/COLL	60.0	36.8
Divertor Simulator Studies	2999/DVSM	140.0	131.0
Gyrofluid Simulation	3999/GYFL	60.0	60.4
Feedback Kink Study	3999/FDBK	50.0	50.7
Stellarator Studies	3999/STEL	60.0	63.2
High-Field Magnet Studies	7999/HFMG	35.0	0.0
Analysis of Helically Wound Solenoids	7999/MAGN	35.0	36.4
X-Ray Lithography with Tokamak Radiation	7999/XLTR	55.0	50.8
Magnetospheric Plasma Circulation	7999/GPCM	100.0	108.6
Projection Lithography with X-Ray Laser	7999/DPXL	<u>58.0</u>	<u>50.7</u>
Total		1203.0	947.2

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