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SUMMARY

The Fusion Program of Oak Ridge National Laboratory (ORNL), a major part of the national fusion program, encompasses nearly all areas of magnetic fusion research. Collaboration among staff from ORNL, Martin Marietta Energy Systems, Inc., private industry, the academic community, and other fusion laboratories is directed toward the development of fusion as an energy source for the future. The achievements resulting from this collaboration are documented in this report, which is issued as the Fusion Energy Division's annual progress report for 1985 and also contains information from components of the Fusion Program that are external to the division (about 15% of the program effort).

The areas addressed by the Fusion Program include the following:

- experimental and theoretical research on magnetic confinement concepts,
- engineering and physics of existing and planned devices,
- development and testing of diagnostic tools and techniques,
- assembly and distribution to the fusion community of databases on atomic physics and radiation effects,
- development and testing of the technologies needed to heat and fuel fusion plasmas,
- development and testing of superconducting magnets to contain fusion plasmas, and
- development and testing of materials for fusion devices.

The integration of these activities into a unified program, with a structure that actively fosters interaction and collaboration among program components, is a major factor in the successes documented in this report. Highlights from program activities follow.

Toroidal Confinement Experiments

The Advanced Toroidal Facility (ATF) torsatron was the major focus of work in the area of toroidal confinement experiments during 1985. Studies have documented more clearly the flexibility of the device, which will allow exploration of a range of magnetic configurations, and have elucidated features that relate to control of plasma density and impurities. Construction of the device progressed well; completion is set for December 1986. Advances were made in techniques for producing nonaxisymmetric components that meet exacting tolerances. The ATF experimental program was more clearly defined, and extensive preparations for device operation were well under way at year's end.

Advanced projects are being carried out in two areas. Stellarator reactor studies are focusing on ATF-like configurations with high beta at even lower aspect ratios. In the tokamak area, the Spherical Torus Experiment (STX) was developed to the proposal stage and favorably reviewed by the Department of Energy (DOE), but no funds are available for construction. A modest level of work continues on important technical issues of the STX in the hope that the prospects will improve.

Confinement studies were carried out to establish physics requirements and options for compact ignition tokamak (CIT) experiments. The CIT is the focus of a national effort to define the next large fusion device.

Active collaborations continued in several areas. Involvement in research on the Tokamak Fusion Test Reactor (TFTR) at Princeton Plasma Physics Laboratory (PPPL)

reached its highest level during 1985. The collaboration between ORNL and the Junta de Energia Nuclear of Spain resulted in a proposal to build a flexible heliac, TJ-II. The proposal received first-round approval from the European Atomic Energy Community (EURATOM). A novel technique, developed at ORNL, for aligning torsatron coils was applied by a U.S.-U.S.S.R. team on the URAGAN-3 device at the Kharkov Physico-Technical Institute in the U.S.S.R. Work also proceeded toward pump limiter tests on TEXTOR under a collaboration involving the United States, the Federal Republic of Germany, and France and on Tore Supra under a U.S.-France collaboration.

The ORNL bumpy torus program was terminated by DOE following the FY 1986 and 1987 budget reductions. The innovative ELMO Bumpy Square (EBS) design developed in the program will be fully documented and then archived to await a more auspicious funding situation.

Atomic Physics and Diagnostics Development

Activities in the Atomic Physics and Plasma Diagnostics Development Program, which is carried out by the ORNL Physics Division, are divided among atomic collisions research, data compilation, and development of plasma diagnostics.

Two major experimental developments during 1985 enhanced the atomic collisions research effort: the implementation of an ion source that produces highly charged ions from metals, and the installation of a new charge analyzer in the electron-ion crossed-beams experiment, which increases the maximum charge state for ionization cross-section measurements from 6 to 15. The result has been expansion of the database and key new information on the roles of indirect mechanisms and metastable states in ionization.

The Controlled Fusion Atomic Data Center continued to provide bibliographies and compilations of atomic data to the fusion community. Major effort was devoted to preparing a revised and expanded series of recommended data for fusion research.

The plasma diagnostics program concentrated on developing advanced diagnostics for fusion experiments, relying mainly on optical and laser technology. A system that uses Thomson scattering of a pulsed laser beam to measure the density and velocity of alpha particles was developed. A test stand was built for the development of a far-infrared (FIR) laser interferometer system for ATF, and development and implementation of an FIR multichannel polarimeter system for TFTR continued.

Plasma Theory

The development and optimization of advanced confinement systems such as ATF, TJ-II, EBS, and STX have benefited from advances made in theoretical work at ORNL in nonlinear magnetohydrodynamic theory and in the transport theory of nonaxisymmetric systems.

The plasma theory effort concentrates on toroidal confinement theory and is directed toward understanding, improving, and optimizing the confinement of plasmas in fusion devices. To achieve these goals, new theories are developed and validated with experimental results. The approach is to identify critical problems and then to integrate the plasma physics disciplines needed to solve them. Collaboration, both within the theory effort and

with ORNL experimental groups, university groups, and the international fusion physics community, is emphasized.

Key results during 1985 include establishing the physics basis for STX, defining the operating regime for the CIT, and developing fast and effective ways to evaluate transport coefficients for three-dimensional configurations. Other significant results include optimizing the equilibrium of the flexible heliac, demonstrating that the addition of a toroidal field to the Heliotron-E experiment will allow stable access to the second stability regime, and developing the concept of using rf waves to scatter particles into or out of a resonant zone.

Plasma-Materials Interactions

The Plasma-Materials Interactions (PMI) Program addresses issues related to the interaction of hot plasmas with surrounding material surfaces. These issues include characterization of the plasma edge, heat and particle removal, and plasma-surface phenomena.

The PMI group is preparing plasma edge diagnostics for ATF to study plasma density and temperature, particle and heat fluxes, impurity production and transport, and the configuration of the edge plasma.

Studies of heat and particle removal involve international collaboration. With Japan and the Federal Republic of Germany, the PMI group has carried out initial H_α measurements on TEXTOR and has investigated several concepts for a helium removal experiment. In collaboration with France, several concepts for a pump limiter to be used on Tore Supra were studied.

Studies of plasma-surface phenomena in 1985 concentrated on graphites, wall carbonization, erosion and redeposition, and postmortem analysis of limiters. Fundamental characteristics of graphites suitable for fusion technology were studied and documented, and an outgassing experiment to evaluate selected materials was begun. Wall conditioning studies were concentrated on the production and characterization of carbon films. Sputtering measurements of several fusion-relevant materials were made. Erosion/redeposition measurements were performed in collaboration with groups from the Joint European Torus (JET) and TEXTOR, and postmortem analysis was carried out on the limiter from the Heliotron-E experiment in Japan. Other initiatives included studies of surface effects on the performance of rf components.

Plasma Technology

The Plasma Technology Section concentrates on developing and testing methods of heating and fueling fusion plasmas. These methods currently include pellet fueling and rf heating.

The development of pellet injectors continues with vigor. Numerous applications are in progress or planned for TFTR, Doublet III-D (DIII-D) at GA Technologies, Tore Supra, ATF, JET, Alcator-C at the Massachusetts Institute of Technology (MIT), and JT-60 in Japan.

Future applications of rf heating in plasma devices will require high power at high frequencies with excellent coupling efficiency to carry power from the source to the

plasma. A high-power rf feedthrough was developed and is in use on TEXTOR. Additional feedthroughs will be used on TFTR, Tore Supra, and ATF. A short-pulse antenna for ion cyclotron heating has been installed on DIII-D and will be tested to establish the physics characteristics of the antenna-plasma interaction.

The Radio-Frequency Test Facility (RFTF) has operated with its first steady-state plasma. This facility is an important component of the national rf technology development activity, for which ORNL is the lead laboratory.

Superconducting Magnet Development

Since 1977, superconducting magnet development in the ORNL Fusion Program has been focused on the Large Coil Task (LCT), an international effort that involves the United States, EURATOM, Japan, and Switzerland in the construction and operation of the International Fusion Superconducting Magnet Test Facility in Oak Ridge. Experimental operation of this six-coil array of superconducting magnets signaled the achievement of a major milestone in the LCT. At year's end, the facility was being cooled to the required cryogenic operating temperatures, and preparations were being made for a program of single-coil and three-coil tests, to be followed by tests on the full six-coil array. It appears that the goals of the LCT test program can be achieved. Development of advanced superconductor concepts continued at a modest level, and magnet technology work was carried out in support of the RFTF and the ATF.

Fusion Engineering Design Center

At the Fusion Engineering Design Center, representatives from industry work with staff members from the national laboratories involved in fusion research to provide conceptual designs and cost estimates for future fusion reactor projects.

The Design Center took a lead role in assessing the engineering feasibility of a very compact tokamak experiment with copper coils. The assessment concluded with an endorsement of the feasibility of this class of device, and the Design Center began studies of the Ignitor concept, developed by Bruno Coppi of MIT, and arrived at a design configuration. Many features of this configuration have been incorporated into the national baseline conceptual design for the CIT.

The Design Center continued to participate in the mirror fusion program by contributing to the Minimars design effort, a two-year program to develop and describe an attractive tandem mirror reactor concept. In addition, the Design Center continued to lead the engineering activities for the International Tokamak Reactor (INTOR) program and provided the primary point of U.S. interaction with the Next European Torus team in Europe and with the Fusion Engineering Reactor (FER) team in Japan.

Additional work included a study of advanced commercial tokamaks and development of a cost accounting system that is applicable to all magnetic fusion reactor design studies.

Materials Research and Development

The Materials Research and Development Program investigates alloys for fusion reactor structural applications, special-purpose materials such as ceramics, and materials for superconducting magnets.

Development of austenitic stainless steels is carried out in collaboration with Japan. Spectral tailoring experiments in the Oak Ridge Research Reactor and the High-Flux Isotope Reactor are being used to explore the irradiation responses of these steels in regimes appropriate to near-term reactors such as the FER. Other experiments explore higher damage levels to determine maximum useful lifetime and to provide information on which to base further development. Exploratory studies of manganese-stabilized austenitic stainless steels, which are possible low-activation structural materials, continued. Radiation effects in vanadium alloys were explored; these alloys appear to be subject to the same forms of radiation damage as other alloy systems. Research on ferritic steels has continued to explore the response of these alloys to neutron damage. Attempts to develop low-activation versions of ferritic steels by substituting tungsten or vanadium for molybdenum and niobium produced promising initial results.

Corrosion studies addressed the compatibility of materials used for structure, coolant, and breeders and the mechanisms of corrosion and mass transfer. Emphasis shifted to studies of low-activation alloys, with continuing research on standard alloys aimed at developing corrosion-resistant materials and possible inhibitive techniques for liquid-metal environments.

The Special-Purpose Materials Program for development of insulators and stabilizers for superconducting magnets focused on providing facilities at the National Low-Temperature Neutron Irradiation Facility for research on the effects of radiation on magnet materials. This facility will provide temperature and irradiation conditions similar to those at the magnets of a fusion reactor.

Neutron Transport

The Neutron Transport Program carries out experiments on neutron and photon transport in fusion materials, analysis of experimental data, and computer modeling of transport. Experiments conducted at ORNL and on TFTR at PPPL provided data that verified analytic models, which will be used in the design and analysis of future fusion experiments. Work is also performed in the areas of nuclear data evaluation and cross-section processing. The Radiation Shielding Information Center supplies a broad range of services to the fusion community.

The range of achievements described in this report clearly illustrates the diversity of these activities and the strengths of the ORNL Fusion Program.

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Chapter 1

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Toroidal Confinement Experiments

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1. TOROIDAL CONFINEMENT EXPERIMENTS

SUMMARY OF ACTIVITIES

The Advanced Toroidal Facility (ATF) torsatron was the major focus of work in the area of toroidal confinement experiments during 1985. Studies have documented more clearly the flexibility that ATF will provide in investigating a range of magnetic configurations and have elucidated features that relate to control of plasma density and impurities. The construction project progressed well, with completion set for late 1986. The experimental program was more clearly defined, and extensive preparations were well under way at year's end. These efforts are described in Sect. 1.1.

In the area of advanced projects, discussed in Sect. 1.2, activities were carried out on two fronts. Stellarator reactor studies focused on ATF-like configurations that retain the potential for high beta at even lower aspect ratios, and the spherical torus concept was developed to the proposal stage and favorably reviewed by the Office of Fusion Energy.

Confinement studies were carried out to establish physics requirements and options for compact ignition tokamak (CIT) experiments, as an aid for design selection and optimization. Details of these and other advanced studies are given in Sect. 1.3.

Active collaborations continued in several areas. This was the peak year for ORNL involvement in research on the Tokamak Fusion Test Reactor at Princeton Plasma Physics Laboratory (Sect. 1.4). The collaboration of ORNL and the Junta de Energia Nuclear (JEN) of Spain on the design of a TJ-II experiment for JEN resulted in a proposal presented to the European Atomic Energy Community (EURATOM) for Phase I review (Sect. 1.5). A novel technique for aligning torsatron coils was applied by a combined U.S.-U.S.S.R. team on the URAGAN-3 device at the Kharkov Physico-Technical Institute (Sect. 1.6). Work also proceeded toward pump limiter tests on TEXTOR under a collaboration involving the United States, the Federal Republic of Germany, and France (Sect. 1.7) and on Tore Supra under a U.S.-France collaboration (Sect. 1.8).

Abstracts are presented in Sect. 1.9 for work resulting from experiments on ELMO Bumpy Torus-Scale and in Sect. 1.10 for work in other areas.

1.1 THE ATF PROGRAM

The Advanced Toroidal Facility (ATF) program has four major elements: continuing physics studies of the ATF magnetic configuration, the construction project itself, sharper definition of the early experimental program, and preparation of the operational and experimental systems to implement the experimental program. These elements are detailed in the sections that follow.

1.1.1 Physics Studies

B. A. Carreras, R. H. Fowler, J. H. Harris, J. F. Lyon, P. K. Mioduszewski, and J. A. Rome

Physics studies of the ATF in 1985 were aimed at better documentation of the magnetic configuration in preparation for experiments. Two elements were stressed: the flexibility to study a wide range of magnetic configurations and the control of plasma density and impurities for steady-state operation.

1.1.1.1 Range of magnetic configurations

The flexibility that allows modification and control of the magnetic configuration derives from the three degrees of freedom in the poloidal field (PF) coil system: (1) the poloidal flux, which controls the net plasma current (which is normally zero); (2) the dipole moment, which permits shifts of the magnetic axis inward or outward and variation of the vacuum magnetic well depth; and (3) the quadrupole moment, which permits poloidal shaping of the magnetic surfaces and changes in the central rotational transform $\iota(0)$.

Figure 1.1 shows the $\phi = 0^\circ$ magnetic flux surfaces for a range of plasma shifts ($R_{\text{axis}} = 2.05$ to 2.15 m) and plasma shaping ($I_{\text{MID}}/I_b = -0.27$ to $+0.27$). The case in the center of the figure is the standard ATF configuration. The plasma ranges from basically triangular shapes to elliptical shapes to shapes with a bifurcated magnetic axis [$\iota(0) = 0$].

Figure 1.2 shows some of the variations in the magnetic configuration parameters that can be obtained by changing the current in the middle set of vertical field (VF) coils for the unshifted configurations ($R_{\text{axis}} = 2.1$ m) in the central column of Fig. 1.1 and by shifting the magnetic axis for the unshaped configurations ($I_{\text{MID}}/I_b = 0$) in the central row of Fig. 1.1. Fairly large variations in $\iota(0)$, edge transform $\iota(a)$, and magnetic well or hill are possible, but the average plasma radius is approximately constant. Figure 1.3 shows the radial profiles of ι and V' ($= \oint d\ell/B$, where the prime indicates differentiation with respect to the flux function ψ) for the extremes of the I_{MID} and Δ_v ranges in Fig. 1.2.

Another feature of the flexibility of ATF, illustrated in Fig. 1.4, is the variation of $\iota(a)$ with beta and with the quadrupole moment of the VF coil system for zero-current equilibria. Increasing the quadrupole moment decreases the transform but increases the "beaniness" of the plasma. Through a combination of beta and shaping, the plasma can attain high beta while maintaining a transform very close to the optimum vacuum transform. This method of achieving high beta improves both the equilibrium and the stability of the plasma. The equilibrium improves because of a reduction in the variation of $\oint d\ell/B$ in a field period, which reduces the shift of the magnetic axis with beta. Although the shift is reduced, there is no reduction of the magnetic well, because the shaping deepens the well.

Fig. 1.1. Vacuum magnetic flux surfaces at the $\phi = 0^\circ$ location for different values of magnetic axis radius and mid-VF coil current normalized to the HF coil current (I_{MID}/I_h).

Fig. 1.2. Variation of vacuum configuration properties vs (a) I_{MID}/I_h for centered configurations and (b) axis shift Δ_v for $I_{MID} = 0$.

1.1.1.2 Particle and impurity control

The control of particles and impurities is a critical issue for steady-state operation of stellarators. Modeling of the plasma edge has shown that heat and particle flux patterns are very complicated in these configurations; computer calculations of magnetic field lines and thermal ion orbits, as well as experimental observations, indicate that the fluxes will not be distributed uniformly over the plasma surface but will be concentrated in certain areas (the "divertor traces" seen in Heliotron-E). This is illustrated for ATF in Fig. 1.5, which shows field lines and 1-keV protons for launch points 2 cm outside the last closed flux surface. These areas of concentrated flux are natural locations for heat and particle removal.

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Fig. 1.3. Radial profiles of ϵ and $V/V(0)$ for configurations at the extremes of the I_{MID}/I_0 and Δ_v variations in Fig. 1.2.

Fig. 1.4. Paths of constant $\alpha(0)$ for zero-net current equilibria. These paths can be followed to high beta by increasing I_{MID}/I_h as beta increases.

Fig. 1.5. Helical strips of (a) field lines and (b) 1-keV H^+ orbits passing through a surface 5 cm outside the last closed flux surface.

An experiment designed to gather information that will guide the design process for pump limiters and divertors was performed on Heliotron-E. A movable pump limiter module was installed in a horizontal midplane port. Inside the last closed flux surface, the pressure buildup in the pump limiter was comparable to that in tokamaks; outside the last closed surface, the pressure was low because the limiter did not match the complicated flux pattern. Details of this experiment are given in Chap. 4.

1.1.2 Construction Project

T. C. Jernigan, project manager

P. B. Thompson, engineering manager

L. R. Baylor, R. D. Benson, C. Bridgman, W. D. Cain, K. K. Chipley, M. J. Cole, D. L. Coppenger, G. R. Dyer, P. H. Edmonds, W. M. Fletcher, J. W. Forseman, R. D. Foskett, R. H. Fowler, W. A. Gabbard, D. H. Gray, E. L. Halstead, J. H. Harris, D. I. Hatch, G. H. Henkel, R. L. Johnson, G. H. Jones, L. M. Jordan, V. D. Latham, J. F. Lyon, L. A. Massengill, J. A. Mayhall, J. F. Monday, R. N. Morris, G. H. Neilson, B. E. Nelson, J. A. Rome, M. J. Saltmarsh, W. D. Shipley, F. W. Snodgrass, F. Sorrells, C. K. Thomas, L. M. Vinyard, J. E. Warwick, J. M. Whatley, J. A. White, J. C. Whitson, D. E. Williamson, and J. L. Yarber

Construction on ATF continued during 1985. Highlights from the year include completion of detailed design, delivery of the VF coils and the castings for the structural shell, and production starts on the helical field (HF) coils and vacuum vessel.

As shown in Fig. 1.6, ATF is a torsatron device consisting of two HF coils, three sets of VF coils, a vacuum vessel, and a toroidal structural shell to support the magnetic loads. The total estimated cost for ATF construction is \$19.9 million. Completion of construction is scheduled for December 1986.

1.1.2.1 HF coils

Construction of the helical coils is proceeding well after the delayed start due to the extended prototype development process. All of the stainless steel castings for the structural tees used in the HF coil segments have been delivered by Esco Company. CBI Services, Inc., has begun delivery of the structural tees with the required features machined in them and the machined and formed copper turns. Final delivery is scheduled for September 1986. Assembly of the segments at ORNL is proceeding well.

1.1.2.2 VF coils

Princeton Plasma Physics Laboratory (PPPL) has completed and delivered all three sets of VF coils. The lower sets are now in place below ATF awaiting final installation.

Fig. 1.6. Artist's sketch of ATF, showing main components.

1.1.2.3 Vacuum vessel

Pittsburgh-Des Moines Corporation is fabricating the vacuum vessel from 6-mm stainless steel. The welding form has been completed, and the stainless steel skin has been welded from the outside. The remaining steps are to remove the welding form from the inside, complete the welds from the inside, and attach the vacuum flanges. Delivery is expected in June 1986.

1.1.2.4 Structural shell

Westinghouse Nuclear Components Division is building the structural shell. It is made up from 24 castings with mating flanges machined in each casting. Numerous problems in the production of the castings have been overcome by Abex Corporation. The standard HK stainless steel alloy required many modifications to obtain the necessary strength, ductility, and magnetic permeability. Afterward, a large number of tries was needed to obtain the required dimensions of the casting. All the castings have been delivered for final machining, and Westinghouse has begun assembly of the shell. Delivery of the lower half of the shell is scheduled for March 1986.

1.1.2.5 ATF final assembly

The sequence of final assembly for ATF, shown in Fig. 1.7, begins with the delivery of the lower half of the structural shell in March 1986. Next, the lower 12 segments of the HF coils will be installed and aligned on the lower shell. In July 1986, the vacuum vessel will be installed, followed by the upper 12 segments of the helical coils and the upper structural shell. Finally, the bus work, cooling system, and VF coils will be installed.

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Fig. 1.7. Assembly sequence for major components of ATF. (a) Upper half of structural shell, (b) intermediate connecting panels, (c) upper HF coil segments, (d) one-piece vacuum vessel, and (e) lower HF coil segments in lower half of structural shell.

1.1.3 Experimental Program

P. H. Edmonds, J. F. Lyon, and G. H. Neilson

The experimental program on ATF will focus on the physics of the torsatron as a toroidal confinement configuration with potential advantages over the tokamak. Among these advantages are a vacuum magnetic field possessing a "built-in" rotational transform and optimum MHD stability properties, good particle and energy confinement, and inherently disruption-free, steady-state operation. The planned experimental program is designed to take advantage of these properties as early as possible, although full exploitation will require several years.

The plans for the first two operating years, FY 1987 and FY 1988, are shown in Figs. 1.8 and 1.9. Important goals in the first year are to implement all three heating systems and most of the essential diagnostics and to begin exploration of the high-beta mode of operation using up to 3 MW of neutral beam injection into target plasmas created with second harmonic electron cyclotron heating (ECH) at a nominal magnetic field of 1 T. Careful attention will be given to areas especially crucial to successful stellarator operation, such as quality of magnetic surfaces and control of the plasma-material interface. The surface quality will be addressed by applying techniques for measuring field errors, aligning coils, and probing surfaces that are being developed in collaboration with the URAGAN-3 group at Kharkov Physico-Technical Institute in the U.S.S.R.¹ and with Auburn University. Control of the plasma-material interface will be investigated at first using instrumented graphite limiters, together with probes, cameras, and H_α monitors to diagnose the plasma edge region.

In the second year, the diagnostic set will be complete enough to permit detailed studies of confinement, stability, and impurity behavior. In particular, a heavy-ion beam

Fig. 1.8. ATF experimental schedule for FY 1987.

Fig. 1.9. ATF experimental schedule for FY 1988.

probe (HIBP) for measuring radial profiles of the electrostatic potential will be operational. This measurement is crucial to the understanding of confinement in stellarators and is being carried out in collaboration with Rensselaer Polytechnic Institute (RPI). Power will be increased in all three heating systems, also, and these capabilities will permit studies of the low-collisionality operating mode using high-power neutral beam injection into target plasmas created with fundamental ECH at a magnetic field of ~ 2 T. Studies of heating in the ion cyclotron range of frequencies (ICRF), plasma-materials interactions, and particle control will proceed in parallel, complementing the main program.

In later years, the role of ICRF heating will be expanded as the power is raised to the ~ 2 -MW level and a number of different launch configurations are tested. Optimized pump limiters will be combined with the intrinsic magnetic topology in the torus edge region in order to study divertor-like configurations. Confinement and stability studies begun earlier will continue, and helical-axis configurations will be investigated. Experiments will increasingly be aimed at providing the foundations for the eventual goal of high-beta, steady-state operation of ATF.

1.1.4 Experimental Preparations

R. A. Brown, G. R. Dyer, P. H. Edmonds, R. D. Foskett, W. A. Gabbard,
J. C. Glowienka, G. H. Neilson, M. J. Saltmarsh, W. L. Wright,
and R. B. Wysor

The preparation of operational and experimental systems for ATF began in late 1984, following completion of the Impurity Study Experiment (ISX-B) and ELMO Bumpy Torus-Scale (EBT-S) programs. The ISX-B tokamak and much of its supporting equipment were removed. A number of parallel tasks were begun to modify existing systems and

build new ones to support the ATF experimental program described in Sect. 1.1.3. Major facility improvements include an upgrade of the HF power supply (HFPS), implementation of a status and control system based on programmable logic controllers (PLCs), and installation of an enclosed, elevated control room. Other areas in which substantial progress was made in 1985 include the heating systems and several of the diagnostics. Some of this work is described in more detail in the following subsections.

1.1.4.1 Helical field power supply

D. D. Bates, A. Y. Broverman, P. H. Edmonds, R. E. Hill, G. H. Neilson, L. E. Nickles, J. A. White, R. E. Wright, and W. L. Wright

The HFPS was formerly used as the toroidal field supply on the Oak Ridge Tokamak (ORMAK) and ISX. An upgrade was necessitated by the higher load impedance and longer pulse lengths of ATF. Being replaced with higher capacity components are the rectifier assemblies, the rectifier transformers, the primary 13.8-kV feeder system and switchgear, and the 161- to 13.8-kV substation serving this facility. In 1985, the eight rectifier assemblies were stripped and the old transformers removed. The contract for eight 5-MVA rectifier transformers was placed with Brown Boveri Company of Bland, Virginia, and delivery was scheduled for early 1986. Most of the rectifier components were procured. When completed, the upgraded supply will satisfy both the pulsed and the steady-state requirements of ATF.

1.1.4.2 Status and control system

W. R. DeVan, B. C. Duggins, R. D. Foskett, D. H. Gray, J. T. Patton, and J. N. Sumner

The status and control system will be used for monitoring and controlling the state of facility subsystems (e.g., valves, switches, pumps, and pressure and temperature readouts). Programmable logic controllers were selected for this purpose, replacing hard-wired relay logic, because they offer a flexible ladder network and operator interface, documentation, and a reduction in cabling at relatively low cost. A system architecture using four Gould-Modicon Model 584 CPUs was defined. Purchase orders for the CPUs and input-output modules needed for initial operation were placed. Much of this equipment is being obtained as surplus from the Gas Centrifuge Enrichment Program, resulting in substantial cost savings for ATF.

1.1.4.3 Control room

A. M. Alban, R. D. Foskett, and J. N. Sumner

The ATF control room will improve both the operating environment for personnel and equipment and cable management. The control room floor is elevated ≈ 2 m (7 ft) above the main experiment floor, providing ample and accessible space for cable trays and electronic cabinets, which will reduce control room clutter. The structure was procured from and installed by CID Associates of Oakmont, Pennsylvania.

1.1.4.4 Electron cyclotron heating

D. D. Bates, T. S. Bigelow, H. D. Kimrey, T. L. White, and W. L. Wright

The ATF ECH heating system consists of a high-voltage gyrotron power supply, the gyrotron oscillator tube, the gyrotron superconducting magnet system, the quasi-optical waveguide system for transmitting the ECH power to ATF, the ECH cooling systems, and the ECH system instrumentation, controls, and interlocks. The high-voltage gyrotron power supply was acquired from the EBT-S program and can produce 90 kV at 10 A with a separate, floating, well-regulated supply at 40 kV and 100 mA.

The gyrotron oscillator² is a high-power, steady-state microwave tube that can produce 200 kW cw at 53.2 GHz in the TE_{02}^0 circular electric mode. The frequency was chosen to match the maximum electron cyclotron frequency on axis in ATF; however, most of the operation of ATF will be at half-field, so that second harmonic ($\omega = 2\omega_{ce}$) ECH will play a major role in producing target plasmas for subsequent heating with neutral beam injection.

The ECH waveguide system, which is under construction, is shown schematically in Fig. 1.10. The system has several unique features, including an all-vacuum waveguide transmission with an rf gate valve for vacuum isolation rather than a barrier window, TE_{02}^0 mode-controlled transmission using purely quasi-optical bends rather than corrugated bends, cw power handling, and a very broad bandwidth that allows gyrotrons at widely separated frequencies to use the same transmission system.

Initially, the launcher will be a simple open-waveguide structure radiating in the TE_{02}^0 mode due to budget constraints. Later, the launcher will be upgraded to a focusing Vlasov³

Fig. 1.10. ECH system for ATF.

system that can direct the ECH power to a small spot in a controlled polarization to optimize single-pass absorption near the "saddle point" in the ATF magnetic geometry. This system is shown in Fig. 1.11.

1.1.4.5 Neutral beam injection

P. H. Edmonds, L. A. Massengill, M. M. Menon, R. N. Morris, J. A. Rome, J. E. Warwick, and R. E. Wright

Plasma heating in ATF will be carried out predominantly by means of neutral beam injection. The injectors, originally used in ISX-B, are being modified for ATF.⁴ These modifications include (1) magnetic shielding of the ion source and the neutralizer, (2) a new front end for mechanical adaptation to the ATF port, and (3) an adjustable beam-defining aperture to minimize beam interception on the ATF vacuum wall. Each injector, as shown in Fig. 1.12, is equipped with a duoPIGatron ion source that can deliver 100 A of 40-keV ion beam with a divergence profile of $1.1^\circ \pm 0.1^\circ$ (HWHM) and an atomic species content of $(80 \pm 5)\%$. The pulse duration of the beam is 0.3 s. Three such injectors will be used in ATF; together, they can provide up to 4.5 MW of neutral power.

Tangential injection was chosen in order to minimize orbit losses. Figure 1.13 shows the view looking into the plasma through the injection port in a tangential direction. Although the injection port is quite large (≈ 1 m), because of the complex structure of the vacuum wall the maximum permitted beam diameter is considerably smaller. Moreover, small changes in injection angle around the tangential direction can significantly affect the vacuum wall loading relative to the beam power deposited in the plasma. The injection angle was optimized using Monte Carlo techniques, as described in ref. 5. The regions of the vacuum vessel subjected to direct interception by the beam are shown in Fig. 1.14 for the optimum direction; Fig. 1.15 shows the useful part of the beam power deposited in the plasma. A beam-defining aperture, located 350 cm downstream, shields the vacuum wall from significant beam loading. Moreover, to take into account practical features such as non-Gaussian intensity distribution and imperfect focusing of the beamlets, the aperture size is adjustable over a range of 20–25 cm.

The effect of the ion source characteristics on the power deposited to the plasma was also examined. It showed that, for existing ion sources, changing the focal length in a range of 350–450 cm is of little benefit. However, improving the beamlet divergence has a significant effect on the power deposited in the plasma.

Transport simulations show that unbalanced injection could drive substantial currents (tens of kiloamperes), thereby altering the rotational transform profile enough to degrade confinement. However, calculations of beam thermalization indicate that fast-ion orbit losses will be small for either co- or counter-tangential injection. Since there is no penalty for counter-injection, plans are to install the first two beam lines in opposing directions.

Fig. 1.11. Focusing Vlasov launcher system.

Fig. 1.12. ATF neutral beam injection system.

Fig. 1.13. View into ATF through the injection port along a tangential direction.

1.1.4.6 Ion cyclotron rf heating

F. W. Baity, W. E. Bryan, and K. W. Prince

A conceptual design for a fast-wave ICRF heating antenna for ATF has been completed. The antenna is a compact loop design developed at ORNL in the rf technology program (see Sect. 5.2). The entire structure mounts on one of the large "Tennessee" flanges of ATF and is movable with respect to the plasma edge. Three views of the antenna are shown in Figs. 1.16–1.18. The face of the antenna was shaped to match the contour of the edge of the plasma in ATF, and the current strap is oriented to launch primarily the fast-wave polarization.

Fig. 1.14. Regions of ATF vacuum vessel subjected to direct beam interception.

Fig. 1.15. Loci of beam particle ionizations by plasma.

Fig. 1.16. Antenna structure mounted on ATF (plan view).

The resonant double loop design of this launcher incorporates vacuum variable capacitors that provide tuning and impedance matching to the 50- Ω transmission line from the transmitters. This antenna is designed to operate at 200 kW in the range from 15 to 30 MHz, which covers a number of fast-wave modes in ATF, depending on the magnetic field in the torus, as shown in Fig. 1.19.

The Faraday shield consists of two layers of copper-plated stainless steel tubing with graphite coating on the plasma side. This design has been shown to be very transparent to rf fields and can easily be water cooled for steady-state/long-pulse operation by the addition of manifolds at the sides of the shield. This design has been used for the compact loop antenna on the Doublet II-D experiment at GA Technologies.

1.1.4.7 Thomson scattering

P. S. Hays, R. R. Kindsfather, R. N. Morris, M. Murakami, S. L. Painter,
D. A. Rasmussen, C. E. Thomas, and E. Tolnas

The Thomson scattering design for ATF is a modification of the Scatpak II system used on ISX-B, which could provide a two-dimensional (2-D) profile of the electron temperature and density by shot-to-shot translation of both the multiple spectrometer viewing

Fig. 1.17. Antenna structure mounted on ATF (elevation view).

Fig. 1.18. Antenna structure mounted on ATF (end view).

Fig. 1.19. Fast-wave modes available in ATF.

system and the vertically directed ruby laser beam. Much of the equipment from the ISX-B system can be reused on ATF. The modifications planned either are required in order to deal with problems specific to the ATF device and plasma conditions or are improvements of the ISX-B system.

The Thomson scattering diagnostic for ATF is shown in Fig. 1.20. The ruby laser beam enters the vacuum vessel through one of the square vertical diagnostic ports. The beam can be translated radially (between plasma discharges) in order to obtain scattering over the full width of the plasma. The scattered light is viewed through one of the horizontal "Tennessee" ports. The vertical and horizontal port centerlines are offset toroidally by half a field period (15°). The vertical ports are centered on the $\theta = 0^\circ$ plane. In order to maximize our view of the scattered light, rotating and translating mirrors relay the light to an $f/9$, fixed-focal-length lens, as shown in Fig. 1.21. The laser translates along the dotted line shown in the figure. The mirrors rotate and translate in order to maintain a constant

Fig. 1.28. Thomson scattering system outline, showing the lasers, ATF, collection optics, and data acquisition electronics.

distance between the laser beam object plane and the lens. The lens images the scattered light onto each of 15 fiber bundles, which relay the light to 15 spectrometers. The spectrometer-dispersed light is collected by an array of photomultiplier tubes (PMTs), and the signals are stored in the computer for between-shot analysis. The spectrometer arrangements for three different temperature ranges that were used on ISX-B will be reused on ATF.

The view of the plasma obtainable with this light collection system is shown in Fig. 1.22. Areas marked with an X fill 100% of the $f/9$ cone of the lens (i.e., no vignetting). The numbers in the rest of the area show the percentage of vignetting for those points. The view of the top of the plasma is obstructed by the vacuum vessel trough, which holds the HF coil windings. The view of the lower part is limited by the slanted edge of the horizontal port. For each laser pulse, the vertical profile between these limits can be obtained. The laser path and the relay mirrors can be remotely adjusted between plasma discharges. In this way it is possible to obtain a 2-D profile that includes a radial profile, all points of which are measured by the same central spectrometer. The ability to obtain a radial profile from a single spectrometer retains the option of density calibration against the density profile obtained from the FIR laser interferometer.

Fig. 1.21. The collection optics system. A rotating mirror and a rotating and translating mirror are used to relay light to a lens in a fixed position with a fixed focal length.

The complex vacuum vessel geometry of ATF, coupled with the obtuse viewing angle, does not allow for the installation of a full viewing dump like that used on ISX-B. To compensate, other methods are used to reduce the effects of stray ruby laser light. On ISX-B, long vertical vacuum extensions with multiple light baffles were used to reduce stray light scattering. In addition, filters were inserted between the lens and the fiber optics to selectively attenuate the ruby light. For ATF, we intend to reuse the vacuum extension and baffle design and to double the thickness of the ruby filters. The ruby filters allow measurement of only the red side of the Thomson-scattered spectrum, so cesium-doped gallium arsenide photocathode PMTs are required for light detection. Additional design concerns specific to ATF include stray magnetic fields larger than those in ISX-B and problems associated with the microwaves used for both plasma startup and discharge cleaning of the vacuum vessel.

For ATF, significant improvements have been made to the Scatpak II design. These improvements include moving the spectrometer and PMT arrays and associated electronics out of the machine enclosure area. This will allow access to them during machine operation. The laser input window and the beam dump designs have been modified to provide

Fig. 1.22. View of the plasma. Most of the plasma can be viewed without vignetting. All of the light in an $f/9$ cone reaches the lens from points indicated with X's. The numbers that identify the other points represent the percentage of an $f/9$ cone that reaches the lens.

for easier alignment and more reliable operation. The collection optics system has been designed to provide f -number matching of the fiber optics to the spectrometer input slit. This should result in a higher spectrometer throughput. A more extensive alignment target has been designed in order to allow a nearly complete mapping of the image plane to the two-dimensionally viewed region of the plasma.

One additional area in which an improvement over the ISX-B system is expected involves calibration of the diagnostic. In addition to the conventional method employed in ISX-B, the feasibility of a novel calibration scheme utilizing Rayleigh scattered light from a tunable dye laser is being evaluated. During Rayleigh scattering calibration the dye laser beam can be substituted for the ruby beam. The dye laser output can be tuned over the full wavelength range of the spectrometer. Tuning the output of the dye laser and measuring the Rayleigh scattered light (when the vacuum vessel is backfilled with N_2 or H_2) with

the spectrometer allows a complete characterization of the scattering geometry and the spectrometer spectral response and output channel shapes. Translation of the beam allows this calibration for each spatial location. This method also accounts for long-term changes in the spectral and transmission properties of the vacuum window caused by sputtering of material during plasma operation. The complete calibration can be done without a vacuum opening or disassembly of any of the viewing optics components.

A detailed estimate of the number of photons that will be detected by the different spectrometer channels has been made. It was found that the ATF system should perform essentially as well as it did on ISX-B for similar plasma conditions. A comparison of the estimate to data obtained by Scatpak II during ISX-B operation shows agreement within a factor of two.

The Thomson scattering system installation will begin in mid-April 1987. We expect to obtain preliminary data by August 1987.

1.1.4.8 Heavy ion beam probe

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An important benchmark of ATF performance will be the correlation of confinement with ambipolar fields. Modeling studies have shown that the poloidal $E \times B$ rotation due to ambipolar electric fields of modest strength ($\phi \sim T$) can profoundly affect collisionless orbit losses. Experimentally, electric fields are obtained from spatial derivatives of electrostatic plasma potential. The only direct method for obtaining spatially resolved measurements of ϕ is through the use of an HIBP. The ATF project is procuring an HIBP through a subcontract with RPI. Previous HIBP installations [on ISX-B, EBT, the Tandem Mirror Experiment (TMX) at Lawrence Livermore National Laboratory, and, most recently, the Texas Tokamak (TEXT) at the University of Texas in Austin] were essentially two-dimensional in that the incoming primary beam trajectories and the exiting secondary ion trajectories would lie in or near the same vertical plane. In most cases, potential profiles were obtained along a curved line usually approximating a plasma radius. In the case of EBT, 2-D contour plots that mapped the potential over the entire plasma cross section were obtained. In contrast to all previous installations, the ATF system will be three-dimensional (3-D); the incoming and outgoing ion trajectories will not lie in the same plane, and the plasma potential mapping will be over curved 3-D surfaces with significant poloidal and toroidal extent. The reasons for this difference are the complex magnetic field topology of ATF and the limited diagnostic access. The system will incorporate as much of the ISX-B HIBP system as possible, the main part of which is a 160-keV accelerator. Primary beam injection will be vertically upward through a lower vertical port with the beam accelerator located on the mezzanine. Detected secondaries will pass through an adjacent horizontal port. Recent work has centered on optimizing injection parameters. The effort is now focusing on optimizing the detector location so as to maximize the size of the viewed plasma. Two detectors may be used to increase the field of view. Preliminary studies have indicated that increasing the beam energy by about a factor of two will permit further increases in viewing volume. Installation is planned for the summer of 1987.

1.2 ADVANCED PROJECTS

1.2.1 ATF-II and Stellarator Reactor Studies

B. A. Carreras, J. H. Harris, J. F. Lyon, J. A. Rome, J. Sheffield, and I. N. Sviatoslavsky

Studies of an ATF-based reactor and of an eventual successor to ATF, the ATF-II, focused on magnetic configurations that retain the high-beta potential of ATF at even lower plasma aspect ratio $R/\bar{a}$. Since a given average plasma radius $\bar{a}$ is required for adequate confinement, lower values of $R/\bar{a}$ lead to a more compact and economically attractive reactor.⁶ The most developed stellarator reactor cases, the Heliotron-H⁷ and the Wendelstein ASRA6C,⁸ have $R \approx 20$ m and $\bar{a} \approx 1.6$ – 1.8 m, or $R/\bar{a} \approx 12$. In contrast, tokamak reactor cases typically have $R/\bar{a} \leq 5$. The ATF torsatron with $R/\bar{a} \approx 7$ is a step in the right direction, but further improvements can be made.

Figures 1.23 and 1.24 compare a low-aspect-ratio configuration ($R/\bar{a} = 4.9$) with the standard ATF case. The new configuration, designated ATF', is similar to the standard ATF in that it is an unmodulated ($\phi = \ell\theta/m$), $\ell = 2$ torsatron wound on a torus with a circular cross section, but it differs in that it has a lower coil aspect ratio ($R/a_c = 3.24$ vs 4.42) and fewer field periods (9 vs 12). The flux surfaces, $\psi(r)$, and $V'(r) = \oint d\ell/B$ are similar for the two configurations, but the ATF' configuration has higher shear inside the $\psi = 1/2$ surface, and access to the second stability region in the new configuration is obtained by the increase of shear with beta at the $\psi = 1/2$ surface, rather than by deepening the magnetic well, as in ATF. In fact, the $\psi = 1/2$ surface remains outside the magnetic well as beta increases. The Mercier criterion indicates stability, except for $r/\bar{a} = 0.85$ – 0.9 , at $\beta(0) = 13\%$.

An important parameter for the design of a device fuel with deuterium and tritium (D-T) is R/Δ , where Δ is the minimum distance between the plasma edge (here taken to be the $\psi = 1$ surface) and the center of the helical coil. Here Δ is the sum of the scrapeoff layer thickness t_{so} , the first-wall thickness t_{fw} , the blanket thickness t_{bl} , the shield thickness t_{sh} , the dewar thickness t_d , the coil case thickness t_{cc} , and the conductor half-width t_{hw} . Since Δ is approximately constant (because t_{cc} varies as $\sqrt{R}$ for a given current density j) and is dominated by the shield thickness t_{sh} required to protect the superconducting helical coil from nuclear heating, a lower R/Δ leads to a more compact reactor; R/Δ is 8.4 for ATF' and 9.6 for ATF. Since R/a is considerably smaller for ATF', the ratio of plasma size to Δ is more favorable for ATF' than for ATF ($\bar{a}/\Delta = 1.72$ for ATF' vs 1.46 for ATF).

The values of $\bar{a}$ and Δ also depend on the cross section of the HF coil. For the cases shown in Figs. 1.23 and 1.24, the ratio of poloidal extent to radial extent $K = 2$. A filamentary coil comparison case has $R/\bar{a} = 3.96$, $R/\Delta = 8.73$, and $\bar{a}/\Delta = 2.2$, so there may be room for further improvement, especially since no attempt has been made to optimize the PF coil geometry.

Achieving a compact reactor requires minimizing Δ . For $t_{so} = 0.2$ m, $t_{fw} = 0.01$ m, $t_{bl} = 0$ (where Δ is minimum), $t_d = 0.09$ m, and $t_{cc} = 0.03$ m, $\Delta = 0.33$ m. The conductor half-width is given by $t_{hw} = (\pi B_0 R / 2 \mu_0 M K j)^{1/2}$, and the shield thickness $t_{sh} = 0.66 + 0.083 \ln(\Gamma_n/p_d)$ for steel and $0.49 + 0.058 \ln(\Gamma_n/p_d)$ for tungsten, where Γ_n is the neutron

Fig. 1.23. Vacuum flux surfaces for the $m = 12$ ATF (a,b) and the $m = 9$ ATF' (c,d). The ATF has more axis shift and more triangularity than ATF' because of its lower aspect ratio.

wall flux (in megawatts per square meter) and p_d is the nuclear power density at the superconductor edge (in milliwatts per cubic centimeter). For the reactor scaling study, we used $B_0 = 5$ T, $K = 2$, $j = 5$ kA/cm², $\Gamma_n = 2.4$ MW/m², and $p_d = 0.1$ mW/cm³, with a tungsten shield under the helical coils on the small major radius side. For these assumptions, the distance from the plasma edge to the conductor edge is $x = 1.0$ m, which yields $R = 10.75$ m and $\bar{a} = 2.19$ m for an ATF'-based reactor and $R = 12$ m and $\bar{a} = 1.83$ m for an ATF-based reactor. These values for R are roughly half those used in more conventional stellarator reactor designs.

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Fig. 1.24. Comparison of (a) rotational transform profiles and (b) $V = \int R/B$ for ATF and ATF'. The ATF has more central shear and a deeper vacuum magnetic well than ATF', but the well does not extend out as far in t .

The shield thickness of 0.67 m assumed here is relatively conservative, since $t_{sh} = 0.53$ m is needed if it is assumed that the cryoplant capacity needed to cope with nuclear heating in the coils is 2% of the net electrical output. For the lower value, the insulators receive 3×10^{10} rad of damage after 30 full-power years, and the magnet must be annealed every 8 full-power years. For the higher value ($t_{sh} = 0.67$ m), the insulator damage is reduced to 3×10^9 rad, the cryoplant capacity is 0.2% of the electrical output, and the magnet need not be annealed.

For the ATF geometry, the coil shield takes up $\approx 36\%$ of the first-wall perimeter, so only $\approx 60\%$ of the first wall area is available for breeding tritium. To obtain a net breeding ratio of 1.05, the local ratio must be 1.75. This can be obtained in a helium-cooled LiPb blanket with a beryllium multiplier that has a ferritic steel reflector behind it to enhance multiplication and add shielding. Such a blanket would produce a local energy multiplication of 1.55 and an overall multiplication of 1.26.

If the shield can be omitted, as it could in an ignition experiment with copper magnets, then an even smaller device results. For $\bar{a} = 1$ m, $B_0 = 5$ T, and $x = 0.23$ m, an unshielded device would have $R = 4.9$ m, $t_{hw} = 0.35$ m, and $j = 1.4$ kA/cm², with $16.6B_0^2$ megawatts needed for water cooling and a 137-s pulse obtained at 5 T for liquid nitrogen cooling of the copper coils. If the plasma size is reduced to $\bar{a} = 0.8$ m, then $R = 3.9$ m, $t_{hw} = 0.23$ m, and $j = 2.57$ kA/cm², with $19.7B_0^2$ megawatts needed for water cooling and a 40-s pulse obtained at 5 T for liquid nitrogen cooling.

The ignition experiment with copper coils is designated ATF-II. Its mission would be to demonstrate steady-state H operation at reactor-level parameters and short-pulse D-T operation with $Q \geq 10$ in a later upgrade. The parameters envisioned are $R = 4\text{--}5$ m, $\bar{a} = 0.8\text{--}1$ m, $m = 9$, $\epsilon(0) \approx 0.25$, and $\epsilon(a) \approx 0.95$. For the present design studies, the copper coils are assumed to be jointed and either cooled continuously with water or cooled to ≈ 70 K with liquid nitrogen before a shot. The pulse length for D-T operation with liquid nitrogen cooling would depend on the thermal inertia and is estimated to be $\approx 40\text{--}20$ s for $B_0 = 5\text{--}7$ T. The operating field for steady-state H operation with water cooling would depend on the available power and cooling capacity.

1.2.2 Spherical Torus Experiment

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The work begun last year⁹ on the Spherical Torus Experiment (STX) has continued, and a point design has been analyzed in some detail. Theoretical work has confirmed the extrapolation of Troyon scaling¹⁰ for beta to aspect ratios as low as 1.5. This has increased our confidence in the potential for high beta values in STX. Theoretical calculations also indicate that a low-aspect-ratio tokamak will have excellent stability against tearing modes. A complete report of the physics and engineering work that led to our proposal to the Department of Energy (DOE) is contained in ref. 11.

The proposal to build STX was submitted to DOE in August 1985 and reviewed in September. The recommendation of the review committee was favorable, but approval for construction has not been obtained. We have been encouraged by DOE to continue the physics and engineering studies.

Recent work has focused on a recommendation by the review committee to increase the ohmic flux and refine the power supply requirements. Work is under way to modify a free boundary equilibrium code (EFIT) obtained from GA Technologies. This code will be used to optimize the expanded boundary divertor and to study the range of plasma shapes that can be achieved in STX.

1.3 ADVANCED STUDIES

N. A. Uckan, W. A. Houlberg, and J. Sheffield

1.3.1 Overview

A comprehensive confinement study, using both zero-dimensional (0-D) (analytic) and one-and-one-half-dimensional (1½-D) transport models, has been carried out to establish the physics requirement and options for compact ignition tokamak (CIT) experiments having typical parameters in the ranges $B_0 \sim 8\text{--}13$ T, $I \sim 8\text{--}12$ MA, $a \sim 0.4\text{--}0.5$ m, $\kappa = b/a \sim 1.6\text{--}2.0$, $R_0 \sim 1.1\text{--}1.5$ m. The global analysis¹² used in this study complements calculations made with the predictive 1½-D transport code WHIST¹³ and expands the code's POPCON (Plasma OPeration CONtour) analysis capability. It allows not only rapid assessment of ignition requirements and plasma operating regimes of a particular device, but also formulation of nearly universal contour plots (in density-temperature space) for classes of devices with equivalent performance. The latter feature allows rapid determination of the "figure-of-merit" parameter(s) needed to achieve a certain performance, thus providing a useful guidance for design selection and/or optimization.

Contour plots generated are used to assess the requirements for driven (high- Q) and ignited operation, the prospect of achieving ignition with ohmic heating alone, the optimal path to ignition, the size of the ignition operating window, the range of alpha particle physics that can be addressed, the role of density and beta limits, the sensitivity to variations in confinement models and plasma shape, and the degree to which confinement depends on alpha power. The confinement scalings considered range from an optimistic model based on ohmically heated plasmas to pessimistic models (L-mode scalings) that include confinement degradation by both auxiliary and alpha heating. Favorable H-mode confinement seen in tightly closed divertor geometries is also considered. For a wide range of confinement models, the "figure-of-merit" parameter aB_0^2/q_* (or, equivalently, B_0I) is found to give a good indication of relative performance of the devices. Direct comparisons of the global and 1½-D WHIST transport calculations show good agreement over the entire range of confinement models and device parameters of CIT.¹⁴

For conservative density ($\langle n_{20} \rangle_{\max} \approx 1.5B_0/q_*R_0$) and beta ($\beta_{\text{crit}} \approx 3I/aB_0\%$) limits and assuming confinement degradation with alpha power, the results of the study¹⁴ show that a class of CIT devices with $aB_0^2/q_* \sim 25\text{--}30$ ($B_0I \sim 100\text{--}120$), $\kappa \sim 1.8$, and $A = R_0/a \sim 2.8 \pm 0.2$ (corresponding to a set of device parameters $a \sim 0.4\text{--}0.5$ m, $R_0 \sim 1.1\text{--}1.4$ m, $B_0 \sim 10\text{--}11$ T, $I \sim 10\text{--}11$ MA) should be ignitable with a finite ignition window (ignition margin $M \geq 1$) under modestly conservative L-mode confinement scalings (such as Kaye-Goldston). A small ignition window restricts the alpha particle physics

studies.¹⁵ Under more pessimistic scalings, or more stringent operational limits on $\langle n \rangle_{\max}$ and β_{crit} , $Q \sim 10$ operating space should be accessible with modest auxiliary power ($P_{\text{aux}} \sim 10\text{--}15$ MW). However, under more optimistic confinement scalings (i.e., H-mode, or no degradation due to alpha power, or sufficiently relaxed operational limits) a large ignition margin ($M \geq 1.5$) can be realized, which opens up a broader range of alpha particle physics studies.

Figure 1.25 shows the maximum attainable ignition margin vs aB_0^2/q_0 for some of the confinement scalings in a typical tokamak geometry. Plasma performance contours of a representative CIT with $aB_0^2/q_0 \approx 25$ ($B \sim 10$ T, $I \sim 10$ MA) are shown in Fig. 1.26 for two confinement models: (a) a favorable ohmic scaling (optimistic model) and (b) a typical L-mode scaling (pessimistic model). Performance projections under various confinement models are summarized in Fig. 1.27. Although the projections vary widely, devices with $aB_0^2/q_0 \sim 25\text{--}30$ (with $B_0 I \sim 100\text{--}120$) should achieve at least $Q \sim 5\text{--}10$ under many of the more pessimistic scaling laws.

1.3.2 Abstracts

1.3.2.1 "A Simple Procedure for Establishing Ignition Conditions in Tokamaks"¹²

N. A. Uckan and J. Sheffield

A method of establishing ignition conditions and plasma operating regimes over large regions of parameter space (R_0/a , b/a , aB_0^2/q , etc.) under various physics assumptions

Fig. 1.25. Maximum attainable ignition margin vs aB_0^2/q_0 for (a) ohmic-like (optimistic) and (b) auxiliary (L- and H-mode) scalings.

Fig. 1.26. Steady-state plasma performance contours for a device with $aB_0^2/q_e \sim 25$ ($B \sim 10$ T, $I \sim 10$ MA). Shown are auxiliary power and Q contours and density ($n_{20} = n_{max} = 1.5B/q_e R_0$) and beta limits. Confinement models: (a) ohmic scaling (optimistic model) and (b) L-mode scaling (pessimistic model).

(χ_e , χ_i , q_ψ , β_{crit} , n_{crit} , etc.) using a simple global model is presented. Contour plots of ignition, supplementary power, and plasma heating and operating windows are generated. These are then used to analyze the potential physics design space, operating regimes, and plasma performance characteristics of small ($R_0 \sim 1-2$ m), high-field ($B_0 \sim 8-13$ T) tokamak ignition experiments.

1.3.2.2 "Physics Evaluation of Compact Tokamak Ignition Experiments"¹⁴

N. A. Uckan, W. A. Houlberg, and J. Sheffield

At present, several approaches for compact, high-field tokamak ignition experiments are being considered. A comprehensive method for analyzing the potential physics operating regimes and plasma performance characteristics of such ignition experiments with 0-D (analytic) and 1½-D (WHIST) transport models is presented. The results from both calculations are in agreement and show that there are regimes in parameter space in which a class of small ($R_0 \sim 1-2$ m), high-field ($B_0 \sim 8-13$ T) tokamaks with $aB_0^2/q_e \sim 25 \pm 5$ and $\kappa = b/a \sim 1.6-2.0$ appears ignitable for a reasonable range of transport assumptions. Considering both the density and beta limits, an evaluation of the performance is presented for various forms of χ_e and χ_i , including degradation at high power and sawtooth activity. The prospects of ohmic ignition are also examined.

Fig. 1.27. Confinement projections for a typical CIT with $aB_0^2/q_e \sim 25$ ($B \sim 10$ T, $I \sim 10$ MA).

1.3.2.3 "Physics Requirements for an Attractive Magnetic Fusion Reactor"⁶

J. Sheffield

The plasma parameters beta β and thermal diffusivity χ_E required for a se^{16} sustained D-T reactor plasma may be written in terms of the geometry and technological requirements of the reactor. While there are, separately, constraints on each parameter, it is instructive to combine them in a single parameter $\langle \beta \rangle / \chi_E$ to characterize the reactor region. In a separate study of a generic toroidal magnetic fusion reactor it has been found that, for a fusion reactor to be competitive with alternative sources of electricity, it must

have a fusion island weight of not more than half that of the STARFIRE and MARS reference reactors. A simple model, based upon this result, is used to determine the region of minimum $\langle\beta\rangle/\chi_E$ and the self-consistent physics and technology requirements for a reactor as a function of its plasma geometry. The requirements are compared with experimental achievements and theoretical predictions for the parameters $\langle\beta\rangle$, χ_E , and $\langle\beta\rangle/\chi_E$, for a variety of magnetic configurations—tokamak, stellarator, reversed-field pinch, bumpy torus, tandem mirror, and field-reversed theta pinch. It is shown that all these configurations are, "classically," capable of reaching the attractive reactor region and that good progress is being made experimentally toward that goal.

1.3.2.4 "Noninductive Current Drive in Tokamaks"¹⁶

N. A. Uckan

This report provides a brief survey of possible noninductive current drive mechanisms for tokamaks. It was originally prepared as part of the lecture notes for fusion courses taught at The University of Tennessee by the author. The primary application of driving currents noninductively is the possible continuous (steady-state) operation of a tokamak reactor through the indefinite sustainment of the poloidal magnetic field. Thus, the amount of power required to maintain the current is of crucial importance. In these notes the discussions are restricted to those schemes that are studied the most (experimentally and/or theoretically) and to those that are likely to be reactor contenders. Various current drive mechanisms may be grouped into four classes: (1) injection of energetic particle beams; (2) launching of rf waves; (3) hybrid schemes, which are combinations of various rf schemes (rf plus beams, rf and/or beam plus ohmic heating, etc.); and (4) other schemes, some of which are specific to reactor plasma conditions requiring the presence of alpha particle or intense synchrotron radiation. Particle injection schemes include current drive by neutral beams and relativistic electron beams. The rf schemes include current drive by the lower hybrid (LH) waves, the electron waves, the waves in the ion cyclotron range of frequencies, etc. Only a few of these approaches, however, have been tested experimentally, with the broadest database available for LH waves. Included in this report are (1) efficiency criteria for current drive, (2) current drive by neutral beam injection, (3) LH current drive, (4) electron cyclotron current drive, (5) current drive by ion cyclotron waves—minority species heating, and (6) current drive by other schemes (such as hybrids and low-frequency waves).

1.4 THE ORNL-PPPL COLLABORATION ON TFTR

J. D. Bell, C. E. Bush, R. J. Colchin, J. L. Dunlap, L. C. Emerson, A. C. England, W. A. Gabbard, D. L. Hillis, S. Hiroe, H. C. Howe, D. P. Hutchinson, R. R. Kindsfather, R. A. Langley, C. H. Ma, M. Murakami, V. K. Paré, D. A. Rasmussen, R. K. Richards, J. E. Simpkins, S. D. Scott, C. E. Thomas, and J. B. Wilgen

The ORNL staff continued to be active in the wide range of research activities on the Tokamak Fusion Test Reactor (TFTR) cited in the previous annual report.¹⁷ Their contributions have been a significant factor in the TFTR program at PPPL.

The level of effort in support of TFTR is illustrated in Fig. 1.28. Off-site assignment to Princeton continues to be the primary means of implementing this collaboration. Of the 22 ORNL staff members who were involved in 1985, 15 participated under that arrangement, for a total of 8.5 work-years of support from assignments. Short visits to PPPL (less than 30 days) and work done on-site at ORNL contributed an additional 3 work-years. The level of effort will continue its gradual decline as emphasis at ORNL shifts to preparation for physics operation of ATF. The formal collaboration with PPPL ends at the end of the current fiscal year.

Fig. 1.28. Level of effort involved in support for TFTR.

1.5 THE ORNL-JEN COLLABORATION ON TJ-II

J. Botija, J. L. Cantrell, B. A. Carreras, K. K. Chipley, J. H. Harris, T. C. Hender, S. P. Hirshman, T. C. Jernigan, A. Lopez, V. E. Lynch, J. F. Lyon, R. Martin, R. N. Morris, B. E. Nelson, A. Pueblas, J. A. Rome, and E. R. Solano

The Junta de Energia Nuclear (JEN) and ORNL have been collaborating for the past two years on the design of the TJ-II stellarator experiment under a grant from the Joint U.S.-Spain Committee on Cooperation in Science and Technology. Construction of TJ-II¹⁸ itself will be funded by various elements of the Spanish National Fusion Program and by EURATOM (since Spain joined the European Community on January 1, 1986). The TJ-II

device will be the largest research facility in the Spanish fusion program, with operation planned for 1989.

As discussed in the previous annual report,¹⁹ in the first year the class of stellarator-type devices was selected for study and several magnetic configurations were analyzed, leading to two candidates for TJ-II. This year a specific coil configuration, the flexible heliac, was chosen for TJ-II; physics and engineering trade-off studies were done for different design options; the expected behavior of TJ-II was analyzed; and a proposal for EURATOM Phase I review was completed.

1.5.1 Selection of the Configuration

The first year's studies narrowed the field of TJ-II candidate configurations to a low-aspect-ratio torsatron and a flexible heliac. Low aspect ratio ($R/\bar{a}$) is desirable because it gives a large plasma (average radius $\bar{a}$) for a given device size (major radius R).

The low-aspect-ratio torsatron ($R/\bar{a} = 5$) had two helical windings ($\ell = 2$) with nine field periods on a torus with a triangular cross section. This concept was rejected for TJ-II for a number of reasons. First, the torsatron is too similar to the low-aspect-ratio ATF torsatron ($R/\bar{a} = 7$). Second, the torsatron requires high-power plasma heating early in the program because of the already well-developed world torsatron program. Third, the flexible heliac concept is of world-wide interest, offers unique opportunities for significant contributions to stellarator research, and is not being studied in a large-scale experiment elsewhere.

1.5.2 Properties of the Configuration

Figure 1.29 shows the flexible heliac concept²⁰ chosen for TJ-II. The four-field-period ($m = 4$) system has 32 circular TF coils linked by two central conductors (a circular conductor and an $\ell = 1$ winding). The centers of the TF coils follow a helical path around the central circular coil in phase with the helical path of the $\ell = 1$ winding. This coil configuration produces bean-shaped magnetic flux surfaces that rotate around the central circular coil in the same manner as the TF coils, as shown in Fig. 1.30. The TJ-II parameters are a major radius R_0 of 1.5 m, a TF coil radius of 0.4 m, a swing radius for the TF coils of 0.28 m, and a swing radius for the $\ell = 1$ winding of 0.07 m. The two VF coils required to center the plasma are located at $R = 2.25$ m and $Z = \pm 0.56$ m. The coil currents (≤ 230 kA in the TF coils, ≤ 100 kA in the VF coils, ≤ 300 kA in the central circular conductor, and ≤ 300 kA in the $\ell = 1$ winding) are chosen to give control over plasma size, shape, rotational transform, and to some extent shear over a wide range of values at $B = 1$ T (pulse length 0.5 s).

The $\ell = 1$ winding gives the device its most characteristic property: flexibility, the capability to attain a wide range of magnetic configurations. In Fig. 1.31, lines of constant rotational transform per field period (t_0/m) are shown for part of the achievable range of hardcore currents (I_{cc} = circular coil current, I_{hc} = helical coil current); the value of t_0/m extends over a wide range, from 0.2 to 0.8. The magnetic surfaces corresponding to several of the t_0/m values are shown in Fig. 1.32. High central conductor currents (particularly I_{hc}) push the plasma off the central conductor structure. Thus, following contours of constant t_0 (Fig. 1.31), the larger plasmas occur at the higher currents, with average radii

Fig. 1.29. The coil configuration of the TJ-II Flexible Heliac.

of up to 0.25 m being typical. Above some radius, however, larger plasmas are less bean-shaped and thus have shallower magnetic wells. The magnetic well depth is also a function of the ϵ_0 value. At low ϵ_0 the wells are very shallow (typically 0.5%), whereas at high ϵ_0 the wells are much deeper (typically 6%).

The usual ϵ profile is very flat, with most of the shear at the plasma edge. Some control over the shear can be obtained by providing the $\ell = 1$ coil with two independent current feeds such that the centroid of the helical current distribution can be radially shifted. By powering only the inner turns ($r_{cc} = 5.25$ cm) or the outer turns ($r_{cc} = 8.75$ cm) of the $\ell = 1$ winding, the shear can be changed from markedly positive to negative. This double control over the ϵ profile allows the exploration of an extremely wide range of configurations by changing only the coil currents without any mechanical modifications to the device.

Fig. 1.30. Flux surfaces at toroidal angles $\phi = 0^\circ$, 11.25° , and 22.5° for circular TF coils and at $\phi = 22.5^\circ$ for square TF coils.

Fig. 1.31. Contours of ϵ_0/m showing the wide range achievable by varying the relative currents in the central conductors.

Fig. 1.32. Flux surfaces for various values of τ_0/m .

1.5.3 Trade-Off Studies

Physics and engineering trade-off studies for different design options addressed the proximity of the TJ-II plasma to the coils, the sensitivity of the magnetic flux surfaces to various field errors, and the geometry of the vacuum vessel. Figure 1.30 illustrates the proximity of the TJ-II flux surfaces to the $\ell = 1$ winding and to the TF coils. These distances are shown in Fig. 1.33 for $\epsilon_0 = 0.96, 1.52$, and 2.12 , assuming a 3.5-cm half-thickness for the $\ell = 1$ winding (square cross section and 10 kA/cm^2 assumed) and an 8-cm radial thickness for the TF coil (5 kA/cm^2 assumed). These distances can be increased by using thinner TF coils, noncircular TF coils (Fig. 1.30), higher current density in the central windings, or more poloidal elongation of the $\ell = 1$ winding cross section. As Fig. 1.33 shows, for a reasonable plasma-to-conductor spacing, the limiting factor is the distance to the $\ell = 1$ winding. For the same value of transform, increasing the poloidal extent of the $\ell = 1$ winding requires more current (95, 135, and 215 kA at $\epsilon_0 = 1.46$ for radial winding thicknesses of 7, 4, and 2.5 cm, respectively), so the winding cross-sectional area increases. For square or D-shaped TF coils, the total central coil currents remain unchanged, but a larger fraction is needed in the circular coil.

Fig. 1.33. Separation between the edge of the TJ-II coils and a flux surface vs the average radius of the flux surface for three values of ϵ_0 .

Since the TJ-II magnetic configuration has high rotational transform and low shear, it is sensitive to small magnetic field errors arising from coil misalignments, especially those that break the basic 4-fold symmetry arising from the toroidal periodicity. The most dangerous resonant perturbations are those with low toroidal mode numbers n for various poloidal mode numbers m . For ϵ values between 1 and 2.7, the most dangerous non-symmetry-breaking (4-fold symmetric) perturbations (in order of decreasing importance) have m/n values of 4/2, 4/3, 8/3, 4/4, 8/5, and 12/5. Typically, for symmetry-preserving distortions such as "squaring" of the central conductors, deformations of 1–3 cm can produce serious deterioration of the magnetic surfaces. For symmetry-breaking perturbations (e.g., a horizontal shift or tilt of the central conductors with respect to the TF coils), the most dangerous errors (in order of decreasing importance) for ϵ between 1 and 2.7 are those with m/n values of 1/1, 2/1, 3/2, 5/2, 4/3, 5/3, 7/3, 8/3, 5/4, 7/4, 9/4, 6/5, 7/5, 8/5, 9/5, 11/5, 12/5, 13/5, and 14/5. When the rotational transform falls at these resonances, coil displacements of 1–3 mm significantly perturb the flux surfaces. In all cases, however, a relatively small change in the central coil currents moves the magnetic configuration away from the resonant ϵ value and restores flux surface quality.

The TJ-II coil configuration has a significant impact on the vacuum vessel geometry and on the access for plasma diagnostics and heating. One design option was a large cylindrical vacuum vessel containing all but the VF coils. Another option was a vacuum vessel interior to the TF coils; the total vacuum wall area was reduced, but device assembly would be more difficult. For both design options, sufficient access exists for quasi-tangential neutral beam injection. The interior vacuum vessel option, as shown in Fig. 1.34, was chosen for TJ-II. Although it looks complicated, it is made up of simple

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Fig. 1.34. Modular interior vacuum vessel design for TJ-II.

components that can be easily fabricated. Increasing the number of TF coils per period has little effect on flux surface quality but increases the depth of the magnetic well and decreases the access distance between coils (well depths of 1.3%, 2.4%, 3.3%, and 3.8% and maximum intercoil distances of 50, 36, 27, and 21 cm are obtained for 6, 8, 10, and 12 coils per period, respectively).

1.5.4 Expected Behavior

Analysis of the expected behavior for TJ-II focused on studies of MHD behavior and plasma confinement.

Preliminary studies have been made of 3-D MHD equilibria in the TJ-II device. Studies for other heliacs indicated the importance of fine-scale control over ϵ and shear to avoid low-order resonances where the beta limit is very low. Equilibrium studies for TJ-II with both the NEAR code and the BETA code have shown that good equilibria exist up to at least $\beta_0 = 8\%$, and it appears likely that this value can be extended by further refinements to the numerics and to the ϵ ranges studied. So far only the Mercier stability of these equilibria has been studied; in the helically symmetric limit, studies indicate very favorable stability properties for heliacs. The Mercier results from the BETA code for the TJ-II equilibria studied show stability in all cases.

Plasma confinement studies have concentrated on the confinement of collisionless ions by following energetic ion orbits and on the calculation of the particle and energy cross-field diffusion coefficients using a drift kinetic equation approach. For both studies, the magnetic field ripple near the magnetic axis was varied by toroidally modulating the TF coil currents, and the role of the ambipolar electric field was examined. It is found that confinement improvements due to modest radial electric fields (corresponding to $e\phi/kT \sim 1$) are much more important than those obtained by minimizing the magnetic field ripple. Figure 1.35 shows the reduction, by more than a factor of 10, in collisionless 0.5-keV H^+ orbit losses as the magnitude of the potential ϕ is increased to $e\phi \sim kT$. Calculations of the neoclassical heat conduction coefficients for TJ-II indicate an energy confinement time of 5–15 ms, more than sufficient for the physics studies planned.

1.6 THE U.S.-U.S.S.R. COLLABORATION ON URAGAN-3

R. J. Colchin, S. I. Fedotov, J. H. Harris, D. K. Lee, F. I. Ozherel'ev, O. S. Pavlichenko, D. P. Pogozhev, J. A. Rome, J. D. Treffert, and V. M. Zalkind

If the major axes of the HF and VF coils of a torsatron plasma confinement device are not aligned to within $\sim 1\%$ of the device major radius, the resulting error field can break up the closed magnetic flux surfaces and reduce the effective plasma confinement volume. A novel technique for aligning torsatron coils was tested for the first time on the URAGAN-3 device²¹ at the Kharkov Physico-Technical Institute in the Soviet Union as part of a U.S.-U.S.S.R. fusion research exchange. The concept²² was originated at ORNL during the ATF design studies and involves reprogramming the currents in HF and VF coil pairs so as to produce null points in one or more of the field components near the major axis of each coil set. By making measurements at a number of different vertical

Fig. 1.35. Decrease in 0.5-keV collisionless proton losses with radial potential ϕ . Cases A, B, and C refer to the standard, modulated TF, and overmodulated TF cases for TJ-II.

locations, it is possible to determine the location and tilt of the axis of each coil pair tested. The coil sets can then be shifted so that all their axes are parallel. This alignment of all of the coil dipole moments eliminates the most likely source of large-scale field errors that could break magnetic surfaces having low-order rational values (e.g., $1/2$) of rotational transform.

While the concept was considered attractive for use on ATF, a number of uncertainties needed to be resolved in order to make it a practical technique. The coil currents in the HF and VF coil sets can be set up in a number of different configurations, and it was not known which of these would produce the most reliable measurements. The magnitude of the magnetic field components to be measured is directly proportional to the coil currents used; if reduced coil currents are used to facilitate steady-state operation and to avoid coil force problems (the currents are configured differently from those used in normal plasma operation, for which the coils and support structure are designed), the field near the null can become quite small (≤ 1 G), which presents practical measurement difficulties, such as interference from the earth's magnetic field.

URAGAN-3 was an appropriate choice for a trial experiment because flux surface measurements had shown that the plasma volume was reduced from its design value by flux surface breakup caused by field errors. Flux surface calculations at ORNL showed that effects like these could be produced by relative shifts of ≥ 5 mm between the HF and VF coil sets.

In preparation for the exchange, the Kharkov team constructed a precise magnetic probe positioning device with three degrees of freedom. The probe could be moved 3 m vertically along the centerline of the torus. The magnetic field sensor signal and position were recorded automatically with a computer data acquisition system. The necessary preparations to reconfigure the coil currents in a variety of ways were also made.

At ORNL, elements of existing computer codes for nonaxisymmetric field calculations were combined to generate the variety of magnetic field maps needed to interpret the measurements during the limited duration (three weeks) of the exchange visit to Kharkov. The completed program can produce the maps needed to align the coils of URAGAN-3, ATF, or any other torsatron configuration with a simple change of input parameters.

Experimenters from ORNL and the University of Wisconsin spent three weeks working with the Kharkov team to make the actual magnetic measurements. The experiment went quite smoothly because sufficiently accurate measurements could indeed be obtained using currents low enough for steady-state operation, which greatly facilitated data-taking. It was found that the axis of the VF coils could best be determined by measuring the location of the radial field null obtained when the top and bottom coils are fed with opposing currents and that the axis of the helical windings can best be determined from the periodic azimuthal structure of the radial field nulls produced when two helical windings are fed with opposing currents. An additional check of the system as a whole can be obtained by examining the spatial structure of the vertical component of the field that results when the HF and VF coils are fed with currents of opposite sign, with the relative magnitudes chosen so as to produce a null in the net vertical field at a convenient vertical location.

The results obtained during the exchange showed that the VF coils were horizontally shifted ~ 10 mm with respect to the HF coils. A rough mechanical measurement of the coil positions supported this conclusion, and the coils have been realigned. Experiments to assess the resulting improvements to the magnetic topology of URAGAN-3 are now under way.

The successful demonstration of this alignment technique will reduce the uncertainties in preparing to align the coils of the larger ATF device. The combinations of coil currents and field component measurements that produce good results are now known, so that the design of the measurement apparatus can be simplified and the data analysis streamlined.

1.7 THE U.S.-FRG-JAPAN COLLABORATION ON ALT-II

P. H. Edmonds, D. L. Hillis, P. Mioduszewski, and T. Uekan

The Advanced Limiter Test II (ALT-II) is a collaborative effort between the United States, the Federal Republic of Germany, and Japan. The goal is to study the operation of a full toroidal pump limiter for heat and particle removal in future high-power, steady-state fusion devices. Staff members from ORNL contribute to project and engineering

activities and participate in the experimental program. In particular, contributions were made to the identification of project deliverables and appropriate milestones and to the testing of the limiter manipulator mechanism, including the successful solution to its high-temperature galling problem. Contributions to the physics program include installation and testing of H_α detectors on TEXTOR and further evaluation of the diagnostics requirements for a helium removal experiment. Two H_α detector systems were designed and built at ORNL, tested on an ECH plasma test stand, and then taken to Jülich and installed on TEXTOR. They performed well and provided a database on signal-to-noise levels on TEXTOR necessary for the design of H_α arrays for ALT-II.

Several concepts have been investigated and the associated diagnostics identified to implement a helium removal experiment on ALT-II. Simple modeling of the TEXTOR plasma, combined with the capabilities of an available diagnostic lithium beam, indicates that spectroscopic techniques would provide a measurement of the helium removal efficiency at the plasma edge, but not in the plasma core. When, at a later date, the TEXTOR neutral beams are available, the helium removal efficiency in the plasma core can be determined with nuclear reaction techniques or with charge-exchange-enhanced spectroscopy.

1.8 THE U.S.-FRANCE COLLABORATION ON TORE SUPRA

C. C. Klepper, P. Mioduszewski, and T. Uckan

The pump limiter program on Tore Supra is a collaborative effort between the United States and France. Participation of ORNL staff involves the development of an experimental plan, conceptual design of a pump limiter system for heat and particle removal, testing and implementation of suitable pump limiter diagnostics, operation of the pump limiters, and coordination of the overall task. Engineering and fabrication of the pump limiter modules are the responsibility of the Sandia National Laboratories at Albuquerque.

In the past year, simple models have been developed and have been valuable in computing the specifications for a preliminary design of a Tore Supra pump limiter. Such computations show that a single pump limiter is sufficient for density control but insufficient for accommodating the large particle source resulting from the pellet injection planned for Tore Supra. A preliminary design featuring six pump limiters has been proposed to handle the desired particle exhaust without exposing limiter surfaces to heat fluxes higher than those that can be tolerated with active cooling.

1.9 ABSTRACTS OF WORK ON THE EBT EXPERIMENT

W. H. Casson, J. C. Glowienka, J. R. Goyer, O. E. Hankins, G. R. Haste, D. L. Hillis, S. Hiroe, R. K. Richards, L. Solensten, and J. B. Wilgen

The ELMO Bumpy Torus (EBT) research program was phased down in 1984; activity during 1985 consisted of the theoretical work described in Chap. 3 and the documentation of experiments carried out on the EBT device. The graduate students who had been conducting research on EBT continued work on their theses; two of them completed this effort

and were awarded their doctorates. Abstracts of these theses and of other documentation follow.

1.9.1 *Far-Infrared Scattering on ELMO Bumpy Torus and Impurity Study Experiments*²³

W. H. Casson

A far-infrared (FIR) laser scattering diagnostic was installed on the EBT experiment to measure density waves and fluctuations. The system was tested and calibrated using a TPX plastic Bragg acoustic cell and was shown to be fully functional. Only low signal levels were observed during most plasma conditions including fast- and slow-wave ion cyclotron heating (ICH) experiments. It was demonstrated that the level of fluctuations in EBT-S was much lower than in EBT-I. Also, penetration of ICH-driven waves into the plasma core was evidenced. Pulsed experiments provided evidence of an instability, probably related to the rapid increase in ECH power, that had a remarkably short lifetime.

Unresolved forward scattering of FIR radiation was accomplished on ISX-B by spectrum analyzing the output signals from the FIR interferometer system. MHD instabilities and fluctuation levels were observed, in agreement with other plasma diagnostics.

1.9.2 *An Experimental Investigation of the Properties of the Hot Electron Rings in the ELMO Bumpy Torus Device Using Hard X-ray Diagnostic Techniques*²⁴

O. E. Hankins

A thorough understanding of the hot electron rings in EBT is essential to the bumpy torus concept of plasma production and confinement. The hot electrons are produced via electron cyclotron resonant heating (ECRH) using a 28-GHz cw gyrotron or several 18-GHz klystrons. Measurements of the electron temperature, density, and geometry of the hot and warm components of a hot electron ring were performed for a number of experimental conditions using hard X-ray measurement techniques, diamagnetic flux pickup loops, and synchrotron radiation measurement techniques. An axial array of NaI detectors was used to observe X-ray bremsstrahlung energy spectra resulting from collisions between ring electrons and core plasma ions and electrons. Typical hot electron component temperatures and densities for 28-GHz microwave power operation are 350 to 450 keV and 0.8×10^{11} to $2 \times 10^{11} \text{ cm}^{-3}$, respectively. Typical hot electron component temperatures and densities for 18-GHz microwave power operation are 110 to 150 keV and 0.5×10^{11} to $1 \times 10^{11} \text{ cm}^{-3}$, respectively. Warm component temperatures and densities for 18-GHz and 28-GHz microwave power operation are 20 to 40 keV and 1×10^{11} to $6 \times 10^{11} \text{ cm}^{-3}$, respectively. Ring length is typically 6 to 10 cm. Ring thickness and radial location were measured by inserting a skimmer probe and by dropping small metal balls through the hot electron ring. The results show that the ring thickness is about 6 cm for 28-GHz

operation and about 4 cm for 18-GHz operation and that the ring does not follow the second harmonic resonance location but instead resides radially outside the resonance. The total stored energy of a single hot electron ring measured by diamagnetic flux pickup loops approaches 40 J and is in good agreement with that deduced from hard X-ray measurements. Using the experimental measurements of ring properties and comparing them with theoretical calculations of the critical ring stored energy needed to form a magnetic well or reverse $\int d\ell/B$, it has been determined that the hot electron rings in EBT do not possess sufficient stored energy to create a magnetic well and stabilize the core plasma against flute and simple interchange modes.

1.9.3 "ELMO Bumpy Torus Programme"²⁵

EBT Group

The EBT experiment was designed to test the idea of stabilizing a multiple mirror plasma by the use of energetic electron rings, while eliminating end losses by closing the device into a circular toroid. The experiment did provide a quiescent mode of operation; however, there were major radial losses of particles and energy. These were attributed to effects associated with the uniformly distributed toroidal curvature; an improved device, the ELMO Bumpy Square (EBS), was proposed to eliminate this problem.

1.9.4 "A Comparative Measurement of the Neutral Density and Particle Confinement Time in EBT-S"²⁶

J. C. Glowienka and R. K. Richards

The neutral density and particle confinement time in EBT-S have been determined by two different techniques. These involve a spectroscopic measurement of molecular and atomic hydrogen emissions and a time-decay measurement of a fast-ion population using a diagnostic neutral beam. The results from both diagnostics exhibit identical trends for either estimate, although the absolute values differ by a factor of 2 to 3. The observed variations with fill gas pressure and microwave power from either technique are consistent with measurements of electron density and temperature. In this paper, the measurement techniques are discussed, and the results are compared in the context of consistency with independently observed plasma behavior.

1.10 ABSTRACTS OF OTHER WORK

The abstracts in this section describe analyses of experiments performed on the ISX-B experiment, the Divertor and Injection Tokamak Experiment (DITE), the Texas Tokamak (TEXT), and the Heliotron-E device; discuss the use of charge-exchange spectroscopy as a diagnostic tool in fusion plasmas; and summarize the results of the ORNL tokamak program from 1972 to 1984.

1.10.1 "Spectroscopic Studies of Impurity Fluxes and Concentrations in the ISX-B Beryllium Limiter Experiment"²⁷

R. C. Isler, K. Behringer, E. Källne, P. D. Morgan, N. J. Peacock, P. H. Edmonds, P. K. Mioduszewski, E. A. Lazarus, P. J. Lomas, C. H. Ma, M. Murakami, and R. V. Neidigh

An experiment to test beryllium as a limiter material has been performed in the ISX-B tokamak. One facet of the studies centered on characterizing impurity influxes and concentrations. The radiation from both low-Z (Be, C, N, O) and intermediate-Z (Ti, Cr) contaminants was measured at the limiter surface and at a wall location 90° away from the limiter to assess the relative contributions of metallic impurities from the two different sources. The effects of limiter melting, with concomitant reduction of both low-Z and intermediate-Z elements, were also documented. The analysis, including atomic rate coefficients, for interpreting spectral radiation in terms of production rates is discussed in detail.

1.10.2 "Relative Calibration of Grazing Incidence Spectrometers from Charge-Exchange Excitation of Hydrogenic Ions"²⁸

R. C. Isler and R. A. Langley

Charge transfer from hydrogen atoms having energies below 35 keV to fully ionized low-Z impurities takes place predominantly in two excited levels. It is shown that the subsequent radiative cascades from C^{5+} , N^{6+} , and O^{7+} can be used to obtain the relative response of grazing incidence spectrometer systems in the region between 102 and 1164 Å. In principle, the technique can be extended to 19 Å. The method is useful in plasma devices employing neutral beams for heating or for diagnostic probing. The absolute sensitivity of a system is then established by normalizing to only one branching ratio measurement from a pair of lines having accurately known transition probabilities. Results from the ISX-B tokamak are presented.

1.10.3 "Control of Gaseous Impurities by Pump Limiters"²⁹

R. C. Isler, P. K. Mioduszewski, and E. C. Crume, Jr.

Nitrogen has been puffed into the ISX-B tokamak as a test gas for evaluating the effect of pump limiters on impurity control. The gas is introduced either through a nozzle at the limiter tip or from a location at the wall of the vacuum chamber. The interior nitrogen content is monitored by charge-exchange excitation of N VII lines and depends only slightly on the location of the N_2 source or on whether the pumps are activated or not. The edge radiation from N IV is also influenced little by activating the pumps when the source is at the limiter tip, but it is strongly influenced when the source is at the wall location. The spectral data suggest that the reservoir of nitrogen in the scrapeoff layer is efficiently pumped while the nitrogen content of the main plasma is not affected. The implication is that, in the quasi-steady interval of a discharge, radial diffusion is very slow compared to parallel transport in the vicinity of the limiter tip.

1.10.4 "Impurity Accumulation in ISX-B During Counter-Injection—Are Alternative Hypotheses Valid?"³⁰

R. C. Isler, P. D. Morgan, and N. J. Peacock

Neutral-beam-heated experiments in the ISX-B tokamak have been interpreted as showing impurity accumulation with counter-injection but not with co-injection. It has recently been suggested that there is no difference of transport in ISX-B during these operations and that the experimental results are artifacts of a recombining plasma caused by greater influxes of impurities during counter-injection. It is shown here that this interpretation does not realistically take account of atomic processes and impurity ionization balances. In addition, several new pieces of data are presented that support the previous conclusions of the ISX-B group. These include observations of accumulation of fully stripped light ions by charge-exchange spectroscopy, for which detailed analysis of the atomic physics is not required, and the measurement of almost identical production rates and influxes of limiter material for both senses of injection.

1.10.5 "Rotation Scalings and Momentum Confinement in Neutral Beam Injected ISX-B Plasmas"³¹

R. C. Isler, A. J. Wootton, L. E. Murray, R. A. Langley, J. D. Bell, C. E. Bush, A. Carnevali, P. H. Edmonds, D. P. Hutchinson, R. R. Kindsfather, E. A. Lazarus, C. H. Ma, J. K. Munro, M. Murakami, G. H. Neilson, S. D. Scott, and C. E. Thomas

Scalings of the central rotation in nongettered, co-injected ISX-B discharges have been measured as a function of beam power, electron density, and plasma current. Extensive studies are made possible by exploiting charge-exchange excitation (CXE) of O VIII lines to measure Doppler shifts. The rotation velocity, $v_\phi(0)$, tends to saturate at $1.0\text{--}1.2 \times 10^7$ cm/s when $P_b \approx 0.5$ MW, showing little further increase up to the maximum input of 2 MW. It is independent of n_e and I_p . Momentum confinement times in quasi-steady plasmas are 10–16 ms for $\bar{n}_e = 4.5 \times 10^{13}$ cm⁻³. Counter-injection discharges always disrupt, but $v_\phi(0)$ prior to this event is the same as for co-injection plasmas. The addition of a third beam line, permitting injection of up to 2 MW of *balanced* neutral beam power, has allowed comparisons of the energy and particle confinement in rotating and nonrotating plasmas having the same total neutral beam input. In those cases where impurity buildup can be avoided, it is found that the ISX-B empirical scaling of energy confinement time is reproduced with balanced injection. Thus, the unfavorable $P_b^{-2/3}$ dependence of τ_E^0 is not the result of rotation. Studies of impurity behavior under differing injection conditions have been extended to include fully stripped low-Z ions. The results are consistent with previous investigations of metallic elements that revealed strong dependences on the sense (co- vs counter-) of injection. Potentials calculated from momentum balance, using measured rotation profiles and typical plasma density and temperature profiles, are in qualitative agreement with those measured directly for various combinations of co- and counter-injection.

1.10.6 "Wave Absorption at the Second Harmonic of the Electron-Cyclotron Frequency in a Tokamak Plasma"³²

F. S. McDermott,^{*} G. Bekefi,^{*} A. C. England, S. E. Attenberger, D. B. Batchelor,[†] P. H. Edmonds, R. C. Goldfinger,[‡] R. R. Kindsfather, E. A. Lazarus, M. Murakami, G. H. Neilson, J. B. Wilgen, and A. J. Wootton

Wave absorption in the ISX-B tokamak [Phys. Rev. Lett. **44**, 647 (1980)] at the second harmonic ($\omega = 2\omega_{ce}$) of the electron-cyclotron frequency is reported. Measurements of the absorption of a wave polarized in the extraordinary mode and propagating perpendicular to the toroidal magnetic field are in agreement with computations.

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[‡]Computing and Telecommunications Division, Martin Marietta Energy Systems, Inc.

1.10.7 "Experimental Results on Tokamak Beta Limits from ISX-B"³³

G. H. Neilson, A. J. Wootton, J. D. Bell, C. E. Bush, A. Carnevali, J. L. Dunlap, P. H. Edmonds, A. C. England, W. L. Gardner, H. C. Howe, D. P. Hutchinson, R. R. Kindsfather, R. C. Isler, R. A. Langley, E. A. Lazarus, C. H. Ma, M. Murakami, V. K. Paré, S. D. Scott, C. E. Thomas, R. M. Wieland, J. B. Wilgen, and W. R. Wing

It is observed that the expression $\beta = A\langle\beta\rangle q_\psi/\kappa = 18\text{--}22\%$ defines one of the operational boundaries for ISX-B and corresponds to empirical beta limit scalings obtained in other tokamaks. Although this expression closely agrees with simple expressions for ideal-MHD instability thresholds, it is not obvious that it constitutes a real tokamak beta limit. Confinement degradation is observed when β is well below this operating boundary, and there is neither experimental evidence nor a satisfactory theoretical model connecting this phenomenon with ideal-MHD activity.

1.10.8 "Measurements of Periodic Ripple Transport in the ISX-B Tokamak"³⁴

S. D. Scott, J. F. Lyon, J. K. Munro, D. J. Sigmar,^{*} S. C. Bates, J. D. Bell, C. E. Bush, A. Carnevali, J. L. Dunlap, P. H. Edmonds, W. L. Gardner, H. C. Howe, D. P. Hutchinson, R. C. Isler, R. R. Kindsfather, E. A. Lazarus, C. H. Ma, M. Murakami, L. E. Murray, G. H. Neilson, V. K. Paré, P. A. Staats,[†] C. E. Thomas, R. M. Wieland, W. R. Wing, and A. J. Wootton

The effect of periodic toroidal field (TF) ripple on ion confinement has been studied in the ISX-B tokamak by comparing neutral-beam-heated plasma performance with 9 and 18 TF coils. Three ripple physics issues were treated by these experiments: (1) enhanced ion thermal conductivity, (2) enhanced loss of energetic ions, and (3) ripple damping of beam-induced toroidal plasma rotation, which may affect the plasma losses. Under a wide

variety of plasma conditions, ripple reduced the central-ion temperature by a factor of approximately two (600 eV $\rightarrow$ 300 eV). Ion temperature was found to be nearly independent of applied neutral-beam power in the large ripple configuration (9 TF coils). These results are shown to be in reasonable agreement with theoretical models of ripple transport. Charge-exchange measurements of the fast-neutral flux indicated no loss of fast passing ions due to ripple, but a large depletion of the fast ions trapped in local ripple wells, as expected theoretically. The central toroidal rotation velocity was reduced by a factor of six by ripple, yielding a momentum confinement time substantially less (factor of about seven) than that expected from standard theoretical expressions for ripple-enhanced ion viscosity.

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1.10.9 "Charge Exchange Observations and Analysis in the DITE Tokamak"³⁵

B. P. Duval, N. C. Hawkes, S. J. Fielding, R. C. Isler, and N. J. Peacock

Measurements of the enhancement of line intensities from oxygen and carbon ions during beam injection of hydrogen atoms into the DITE tokamak have been made for neutral energies of 25 keV and beam powers up to 2 MW. Observations of the spectrum from O encompass the Lyman series in the XUV region (1s- np , $n < 9$) and $\Delta n = 1$ transitions in the VUV and visible regions. The different selection rules for these transitions allow a differentiation between C/X and electron impact excitation of the different (n , l) states. Using theoretical cross-sections and a cascade model for C/X recombination into hydrogenic ions, the concentrations of impurity nuclei (e.g., O^{8+}) are determined from the $\Delta n = 1$ VUV lines. This analysis is unsuitable for the Lyman series which lack the C/X signature. Their intensity is determined primarily by the ionization balance, and the excitation from ground state hydrogenic ion. l state mixing is considered. No indication of statistical repopulation of the sublevels is observed for $n < 7$.

1.10.10 "Long-Time Impurity Confinement as a Precursor to Disruptive Instabilities in Ohmically-Heated Plasmas"³⁶

R. C. Isler, W. L. Rowan, and W. L. Hodge

The confinement times of impurities in ohmically heated tokamaks are shown to lengthen considerably as the density-limit boundary is approached and exceeded either by the increase of n_e itself or by the lowering of B_T or I_p . Observations made on the Texas tokamak indicate that a change in the balance of diffusion and inward convection mechanisms governing impurity transport precedes the onset of partial or major disruptions. These studies also explain the reason for the wide differences of impurity behavior previously observed among various tokamaks.

1.10.11 "Studies of Currentless, High-Beta Plasma in the Heliotron E Device"³⁷

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A currentless plasma with a volume-averaged beta value of 2% has been produced with neutral beam heating. Target plasmas were created by second harmonic resonance heating with electron cyclotron waves (150–350 kW and 53.2 GHz) at a magnetic field strength of 0.94 T. Neutral beam injection (23–30 keV and 1.3–2.6 MW) was used to heat the plasma further. MHD stable and unstable high-beta plasmas were observed. The Q-mode plasmas were produced with the help of intense neutral gas puffing. Properties of the MHD activity and confinement of high-beta plasma are discussed and compared with theoretical studies.

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1.10.12 "Charge-Exchange Spectroscopy for Plasma Diagnostics"³⁸

R. C. Isler

The use of charge-exchange spectroscopy as a diagnostic tool in fusion plasmas is discussed. This technique has been employed to measure the concentrations of fully stripped ions, to perform in situ calibrations of grazing incidence spectrometers, and to determine ion temperatures and rotation velocities from the Doppler effect. The influence of collisions and external fields on the effective emission cross sections of charge-exchange excited spectral lines is assessed.

1.10.13 "Results of the ORMAK, ISX-A and ISX-B Programmes"³⁹

ORMAK-ISX Group

Results obtained with the ORMAK, ISX-A, and ISX-B tokamaks at Oak Ridge National Laboratory (1972–1984) are summarized. Topics considered are plasma heating, confinement, MHD studies, impurity effects, pellet fueling, and limiters and surface physics.

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Chapter 2

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Atomic Physics and Plasma Diagnostics Development

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2. ATOMIC PHYSICS AND PLASMA DIAGNOSTICS DEVELOPMENT

SUMMARY OF ACTIVITIES

The activities within the Atomic Physics and Diagnostics Development Program are divided among atomic collisions research, data compilation, and plasma diagnostics development. The majority of this research is supported through the ORNL Fusion Program but is carried out within the Physics Division.

The atomic collisions research focuses on inelastic processes that are important for determining the energy balance and impurity transport in high-temperature plasmas and for diagnostic measurements. This activity is not directed toward a particular fusion device but concentrates rather on developing a better basic understanding of collision processes involving highly ionized atoms at the relatively low kinetic energies that are characteristic of magnetic fusion plasmas. Close coordination of experimental and theoretical programs guides the selection of key experiments and provides both benchmarks and challenges for theory. Two major recent experimental developments have enhanced our ability to address fusion-related atomic collision processes. First, the capability of our electron cyclotron resonance (ECR) multicharged ion source to produce highly charged ions from metals, such as iron, has been developed and implemented for crossed-beams experiments. Second, the successful implementation of a new charge analyzer in the electron-ion crossed-beams experiment increases the maximum charge state for ionization cross-section measurements from 6 to 15, allowing full advantage to be taken of the charge-state capabilities of the ECR source. Measurements and theoretical calculations of cross sections for electron-impact ionization of multicharged iron and magnesium-like ions have expanded the existing database and have also produced key new information on the roles of indirect mechanisms and metastable states in ionization. Significant parallel development of theoretical methods for treating ions with complex electronic structures has been crucial to this effort, as well as to calculations of the dielectronic recombination process. New experimental initiatives include crossed-beams measurements of electron-impact dissociation of water-based ions and studies of electron ejection during the neutralization of multicharged ions at metal surfaces, both of which have applications in divertors and impurity control.

The atomic physics group also operates the Controlled Fusion Atomic Data Center, which, with a network of consultants, searches and categorizes publications on atomic collisions and prepares and distributes bibliographies and compilations of recommended data to the fusion research community. The major effort during the last several years has been the preparation of a revised and expanded series of recommended data for fusion research.

The plasma diagnostics program concentrates on the development of advanced diagnostics for existing and future magnetic fusion experiments, relying primarily on optical and laser technology. The current emphasis is on the application of infrared lasers to the diagnostics of ignited multicore, nonuniform plasmas. A technique being developed uses Thomson

scattering of a pulsed carbon dioxide laser beam to measure the density and velocity distribution of fusion-product alpha particles. A detailed feasibility study has focused on the influence of toroidal magnetic field on the scattered power spectrum and on methods to enhance the signal-to-background ratio by judicious choice of scattering geometry. The conceptual designs of the long-pulse laser and the heterodyne receiver have been completed. A test stand has also been constructed in our laboratory for development of an advanced multichannel far-infrared (FIR) laser interferometer system for the Advanced Toroidal Facility (ATF), and substantial improvements have been made in laser output power and frequency stability. Support is also continuing on the development and implementation of an FIR multichannel polarimeter system for the Tokamak Fusion Test Reactor (TFTR), both at Princeton Plasma Physics Laboratory and on a test stand in our laboratory. Recent advances include development of a method for antireflection coating of optical elements and development of an optical isolator and specialized signal-processing electronic circuitry. Emphasis in this program continues to be placed on computer-based remote handling systems for all the laser diagnostic systems, which will become more important with each new generation of fusion experiments.

2.1 EXPERIMENTAL ATOMIC COLLISIONS

2.1.1 The ECR Multicharged Ion Source

F. W. Meyer and J. W. Hale

During the past year, performance of the electron cyclotron resonance (ECR) multicharged ion source has been optimized. The first stage of the ion source was redesigned to operate at 10.6 GHz in the ECR mode. Microwave power to the first stage is obtained from the main 10.6-GHz supply, providing power to the second stage by means of a low-reflectivity variable power divider. This improved first stage has been found to significantly enhance the stability of very high charge-state ion beams and of ion beams of the lighter elements. In order to compensate for the lower magnetic field in the first stage relative to earlier overdense operation, an auxiliary coil was added to the middle second-stage coil.

A third axial field coil supply was installed to maximize axial field tuning capability. In this new configuration, the currents in all three axial field coils can be varied independently over a sufficiently wide range to find the optimum axial magnetic field profile.

The capabilities of the ECR source were extended to metallic ion production with a solid sample feed technique. A metallic vapor is created from a thin foil sample heated by insertion into the second-stage plasma edge. The foil is clamped to the end of a lever arm that can be rotated about a pivot. Rotation of the sample into the plasma edge is achieved via a mechanical linkage to a linear motion feedthrough located on the microwave feedthrough flange. A 0.2-mm-thick sample of type 302 stainless steel sheet was used in this manner to obtain beams of iron, chromium, and nickel ions. Charge states of iron up to 15+ were produced with sufficient beam intensity for use in crossed-beams experiments.

Table 2.1 summarizes ion currents now attainable with the ECR source for beams produced from three representative permanent gases and for iron beams produced by the solid sample feed technique.

Table 2.1. Representative ORNL ECR source beam currents
Oxygen beam currents for 10-kV source voltage; argon and krypton beam currents for 12-kV source voltage (15- by 15-mm charge analyzer slits)

Charge state	Beam current (μA)			
	O	Ar	Kr	Fe
+1	105	110		10
+2	95	120		<i>a</i>
+3	80	90		20
+4	<i>a</i>	75		<i>a</i>
+5	83	<i>a</i>	20	23
+6	50	65	25	25
+7	2.5	73	26	<i>a</i>
+8	0.1	105	27	<i>a</i>
+9		45	33	20
+10		<i>a</i>	31	10
+11		3.0	33	5
+12		0.7	40	<i>a</i>
+13			23	2
+14			21	<i>a</i>
+15			10	0.5
+16				
+17			1.0	
+18			<i>a</i>	
+19			0.25	

^aIndicates m/q degeneracy with contaminant beam.

2.1.2 Electron-Impact Ionization of Multicharged Iron Ions

D. C. Gregory, F. W. Meyer, A. Müller, and P. Defrance^{*}

Electron-impact ionization cross-section measurements have been made for a number of multicharged iron ions as part of our continuing work in support of the Fusion Program. In addition to its astrophysical significance, iron is expected to be a contaminant in any laboratory plasma contained in a stainless steel vessel, so it is one of the critical elements for which atomic collision cross sections are needed in order to model plasmas and to interpret diagnostics. The only data available for ionization of iron ions were crossed-beams measurements for Fe^+ (ref. 1) and Fe^{2+} (measured at ORNL²) and rate coefficient measurements³ for Fe^{8+} – Fe^{10+} (which are considered to have rather large absolute uncertainties). The development of the new ECR ion source and recent improvements in our

experimental apparatus have made it possible to extend ionization measurements to higher charge states.

The experimental apparatus is shown in Fig. 2.1. After entering the ultrahigh-vacuum chamber, the incident ion beam is electrostatically analyzed in the charge purifier. The beam then passes through the interaction region, where it is crossed at 90° by a magnetically confined electron beam of variable energy. The ions are separated by charge state in the post-collision analyzing magnet, and the ionized signal ions are counted in a channeltron detector, while the parent ion beam current is measured in one of two movable cups. The apparatus is capable of resolving parent/product ion charge ratios in the range 5/6 to 15/16.

In the present study, absolute cross sections for electron-impact ionization of Fe^{5+} , Fe^{6+} , and Fe^{9+} were measured as a function of electron energy. Fe^{5+} and Fe^{6+} cross sections are about twice those predicted for direct ionization by distorted-wave theory.⁴ Concurrent work on indirect ionization theory for these ions⁵ attributes the difference to the process of electron-impact excitation of the target ions to autoionizing levels, and good agreement is found between theory and experiment when these excitation-autoionization contributions are included. The outstanding feature of the Fe^{9+} cross-section curve is that the ionization onset is observed 50 eV below the energy expected for the ground state ion, indicating a large metastable component in the target ion beam coming from the ECR ion source. The presence of a large number of highly excited metastables in Fe^{9+} is therefore predicted in many plasmas containing iron contaminants.

Fig. 2.1. The ORNL electron-ion crossed-beams apparatus.

The rate coefficients for ionization of all iron ions for which crossed-beams data exist are summarized in Fig. 2.2. It is clear that the earlier rate coefficient measurements for Fe^{9+} (ref. 3) were correct in slope but a factor of three low in magnitude. We expect to extend our studies of iron ions to even higher charge states and to include other metal ions of interest to the fusion community.

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2.1.3 Summary of "Electron-Impact Ionization of Li-Like Multicharged Ions"⁶

D. H. Crandall, R. A. Phaneuf, D. C. Gregory, A. M. Howald, D. W. Mueller,
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Measurements of electron-impact ionization of ions are of general interest because of the uncertain state of basic collision theory as applied to ionization and because of the important role of ionization in laboratory and astrophysical plasmas. In recent years, we have provided accurate experimental data from crossed-beams experiments along a number of isoelectronic sequences in order to test theoretical approaches in a consistent manner

Fig. 2.2. Rate coefficients for electron-impact ionization of iron ions. The curves are calculated from the cross-section measurements discussed in the text, and the two points are rate measurements for Fe^{9+} .

and to establish a database of results for ionization of multicharged ions.^{7,8} The present results for B^{2+} and O^{5+} complete a series of experiments on ionization of Li-like ions for the first five ions of the isoelectronic sequence. Because of the absence of metastable ions in the beams, simple electronic structure, and prominence of emission lines from plasmas, the lithium sequence is a prime candidate for extensive study by experiment and theory. These measurements also provide a reasonable basis for extrapolation of the predicted magnitude of indirect ionization effects to higher members of this sequence.

The first measurements⁹ on ionization of the Li-like ions C^{3+} , N^{4+} , and O^{5+} provided a mild surprise in that inner-shell excitation followed by autoionization produced a measurable contribution to the ionization cross section. Subsequent calculations¹⁰ showed that almost all of the inner-shell excited states of these ions would decay by autoionization rather than by radiative stabilization and that the calculated inner-shell excitation cross sections were roughly appropriate for the observed structure in the ionization cross sections. However, trends in the experimental data indicated that the relative importance of excitation-autoionization increased more rapidly with ionic charge than predicted by theory.¹¹ In addition to the basic questions raised about collision physics, this discrepancy implied significant uncertainty for even the most refined predictions of ionization and excitation rates for other, more highly charged ions of this sequence (e.g., Fe^{23+} , which is important in analyzing high-temperature plasmas of interest in fusion and astrophysics).

The present studies were undertaken in an attempt to clarify the scaling of excitation-autoionization effects in multiply charged Li-like ions. In the six years since the publication of the original data, a number of improved calculations of both excitation and direct ionization have been published, as have experimental data¹² for Be^+ . We have added measurements for ionization of B^{2+} (previously unmeasured) and have remeasured the O^{5+} system, which was significantly larger than theory in the previous analysis. Sophisticated distorted-wave calculations¹³ for direct ionization have been taken as a basis for extraction of the excitation cross sections from the total ionization measurements for each member of the isoelectronic sequence from Be^+ through O^{5+} .

The experimental technique and apparatus have been described.¹⁴ The B^{2+} measurements used the ORNL PIG ion source, while the latest O^{5+} results were obtained with the new ORNL ECR ion source. In both cases, the experiments were pursued with the highest level of precision in the energy regions where the excitation threshold occurs. Absolute uncertainty in the overall magnitude of the cross-section curve was about $\pm 8\%$ of the peak cross section at good confidence level, while the relative uncertainty in the excitation threshold region is considerably less. The B^{2+} measurements are in reasonably good agreement with the distorted-wave calculations of Younger,¹³ both in shape and in magnitude (experimental results are consistently 10% higher than theory). The O^{5+} measurements are in excellent agreement with the distorted-wave results up to about 400 eV, where the cross section suddenly increases by about 7%. Independent measurements¹⁵ corroborate the existence of this feature, although the physical process causing it is still unidentified.

The extracted total excitation cross sections (obtained by subtracting Younger's direct ionization prediction—scaled where appropriate—from the total measurements) for the Li-like ions studied to date are presented in Table 2.2. The ratios of these deduced excitation cross sections to six-state close-coupling calculations of total excitation are also included in the table, and the agreement is seen to be excellent for all members of the

Table 2.2. Comparison of excitation cross sections for Li-like ions

Ion	Cross section (10^{-19} cm^2)		Ratio of experiment and theory
	Six-state close-coupling theory	Presently deduced experiment	
Be^+	9.3	20 ± 8	2.1
B^{2+}	4.1	4.0 ± 1.0	1.0
C^{3+}	2.24	2.3 ± 0.7	1.0
N^{4+}	1.27	1.6 ± 0.4	1.3
O^{5+}	0.74	0.9 ± 0.3	1.2

isoelectronic sequence except Be^+ . The reason for this agreement, as opposed to the marked disagreement of previous analyses, is twofold. First, an error in the distorted-wave ionization calculations used as a basis for obtaining the excitation cross sections has recently been corrected, producing much more accurate estimates of the direct ionization component for each ion. Second, the new O^{5+} results indicate a smaller indirect ionization component than the previous measurements. It is felt that the new data are more accurate and precise than the previous results due to the improved beam intensity and stability obtained with the new ion source. The only remaining discrepancy between theory and experiment for excitation in this isoelectronic sequence is found in the lowest charge-state ion of the sequence. Additional comparisons between theory and experiment for selected highly charged Li-like ions will be necessary, however, to ensure the accuracy of scaled cross sections many charge states further along the isoelectronic sequence.

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2.1.4 Summary of "Electron-Impact Ionization of Mg-Like Ions"¹⁶

A. M. Howald, D. C. Gregory, F. W. Meyer, R. A. Phaneuf, A. Müller, N. Djuric, and G. H. Dunn

The implementation of the ECR multicharged ion source provided an opportunity to extend our systematic studies of collision processes along ionic sequences. The first intersecting-beams experiments involved cross-section measurements for electron-impact ionization of three ions in the magnesium isoelectronic sequence: S^{4+} , Cl^{5+} , and Ar^{6+} . These experiments were performed in collaboration with the Joint Institute for Laboratory Astrophysics (JILA) group led by G. H. Dunn, which previously made measurements for Al^+ , the first member of the sequence.

The electron-ion crossed-beams apparatus and experimental technique have been described in detail in previous reports.¹⁴ The experimental total single-ionization cross-section measurements are compared in Fig. 2.3 to scaled Coulomb-Born calculations⁴ for direct ionization and to the predictions of the Lotz semiempirical formula.¹⁷ The changes

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Fig. 2.3. Experimental cross sections for electron-impact single ionization of S^{4+} , Cl^{3+} , and Ar^{6+} . Dashed curves are Lotz semiempirical predictions for direct ionization, and the solid curve for Ar^{6+} is the distorted-wave calculation of Younger. For S^{4+} and Cl^{3+} , the solid curves are scaled from that for Ar^{6+} by the inverse squares of their respective ionization potentials.

in slope of the cross sections at electron energies near 170, 190, and 220 eV for S^{4+} , Cl^{5+} , and Ar^{6+} , respectively, are signatures of the onset of $2p$ - nl inner-shell excitation, followed by autoionization. This indirect excitation-autoionization (EA) mechanism contributes a relatively larger fraction to the total ionization cross section as the ionic charge increases along the sequence. Detailed distorted-wave theoretical calculations of this process, performed for these ions by Pindzola, Griffin, and Bottcher, are summarized in Sect. 2.2.3 of this report.

A number of interesting features emerge from a detailed analysis of the data. First, the experimental ionization onsets are lower than the ionization potentials of the $2p^6 3s^2$ ground state, indicating that a significant fraction of the incident Mg-like ions are in metastable levels of the $2p^6 3s 3p$ configuration. Second, the single-ionization cross sections show contributions due to direct $2p$ inner-shell ionization. The $2p$ vacancy state is predicted to decay almost completely by autoionization, in which case $2p$ ionization would contribute to double rather than single ionization. This was perplexing, until it was realized that removal of a $2p$ electron from a metastable $2p^6 3s 3p$ Mg-like ion produces a Na-like ion in the $2p^5 3s 3p$ configuration, in which many of the levels are also metastable, with lifetimes against both radiative and Auger decay of tens of microseconds or longer.¹⁸ Such times are longer than flight times in the apparatus. This realization also has allowed us to resolve some outstanding issues relative to earlier experiments on Na-like ions, which were characterized by large autoionizing backgrounds. The apparently missing contribution of the $2p$ - $3p$ EA mechanism to single ionization of Na-like Al^{2+} (ref. 19) may be attributed to metastability of the $2p^5 3s 3p$ levels.

Low-lying metastable levels clearly play important roles in the state dynamics of many high-temperature plasmas. Such levels may well account for unresolved discrepancies between predictions and recent measurements on Ne-like ions in X-ray laser plasmas.²⁰

2.1.5 Electron-Impact Dissociation of H_3O^+

P. A. Schulz, D. C. Gregory, F. W. Meyer, and R. A. Phaneuf

Electron-impact dissociative recombination ($e^- + AB^+ \rightarrow A + B$) has long been recognized as an important process at electron energies of a few electron volts and below in astrophysical plasmas and in other areas of research such as excimer laser development. The competing process of electron-impact dissociation ($e^- + AB^+ \rightarrow A^+ + B + e^-$), however, generally has a larger cross section at energies above 5 eV and has received relatively little attention. The present study of dissociation of H_3O^+ was undertaken to determine the absolute cross sections for the various dissociation channels and to determine, if possible, the mechanisms leading to dissociation. The cross sections for dissociation of H_3O^+ due to collisions with background gas have also been determined.

The apparatus used in this experiment is exactly the same as that described in Sect. 2.1.4. The postcollision magnetic charge analyzer can also serve as a mass analyzer, so no modifications were necessary. H_3O^+ (and D_3O^+) ions were produced in the first stage of the ORNL ECR ion source and then allowed to drift through the second stage of the ion source before being accelerated. For this application, the second stage was used only as a drift section to transport the H_3O^+ ions to the extraction region (i.e., no

microwave power was applied). Since the transit time through the second stage (on the order of a millisecond) was long compared to the excited vibrational-state lifetimes of the ions, the target ion beam was probably quite "cold" in the collision region. This important feature of the ECR ion source may make it an unexpectedly valuable source for experiments involving molecular ions.

Cross sections for the process of dissociation of the 5-kV H_3O^+ ions in collision with residual gas in the target region were estimated from the background count rate in the signal detector. The cross sections for the possible dissociation products are $3 \times 10^{-18} \text{ cm}^2$ for O^+ , $4 \times 10^{-17} \text{ cm}^2$ for OH^+ , and $7 \times 10^{-16} \text{ cm}^2$ for D_2O^+ . The exact composition of the residual gas is not known, but it is most likely a combination of N_2 , H_2 , and CO . These effective cross sections are based on a number of assumptions, but are almost certainly accurate within a factor of four. The relative magnitudes of the cross sections are more accurate, and it is somewhat surprising that roughly an order of magnitude separates the relative probabilities of dissociation into each of the channels.

Electron-impact dissociation of H_3O^+ yields O^+ , OH^+ , and H_2O^+ products. Since the apparatus could not adequately separate H_2O^+ from H_3O^+ , the production of D_2O^+ from target D_3O^+ ions was studied in its place. This reaction is assumed to have the same cross section as that producing H_2O^+ .

The cross sections for electron-impact dissociation of H_3O^+ to produce O^+ as a function of energy are shown in Fig. 2.4. The threshold is observed near 18 eV, which is consistent with the estimated onset of this dissociation channel from the ground vibrational

Fig. 2.4. Cross sections for electron-impact dissociation of H_3O^+ to O^+ .

state of H_3O^+ . The cross section rises sharply at the threshold, then continues to rise gradually to a peak of $8 \times 10^{-18} \text{ cm}^2$ at about 150 eV. The gradual rise is characteristic of an ionization process, and we conclude that this dissociation channel is fed predominantly by ionization of H_3O^+ . The cross-section curve for production of OH^+ from H_3O^+ is shown in Fig. 2.5. The threshold is between 14 and 17 eV, and, in contrast to O^+ , the maximum cross section of $45 \times 10^{-18} \text{ cm}^2$ is observed at threshold. This energy dependence (maximum cross section at threshold) is characteristic of excitation of ions, and we conclude that the dominant channel for production of OH^+ is through excitation of H_3O^+ . The cross section for production of D_2O^+ from D_3O^+ is not shown here. Due to the high rate of D_2O^+ production through collisions of the target ions with residual gas, high background count rates made the accumulation of signal statistics difficult at low energies and made normal diagnostic measurements even more difficult. The peak cross section for this dissociation channel is about $70 \times 10^{-18} \text{ cm}^2$, but we were unable to extend measurements to low energies with sufficient accuracy to determine whether the dissociation is produced through excitation or ionization of the parent D_3O^+ ion.

This initial study of electron-impact dissociation of molecular ions has opened a new field of research for our laboratory. Future studies will include ionization or dissociation of ions that are of interest in laser development or astrophysical studies.

Fig. 2.5. Cross sections for electron-impact dissociation of H_3O^+ to OH^+ .

2.1.6 Ion-Atom Merged-Beams Experiment

C. C. Havener, R. A. Phaneuf, and P. A. Schulz

Our objective is to measure the total electron capture cross section for collisions of multicharged ions with hydrogen (or deuterium) in the energy range of 1–500 eV/nucleon. At such low energies, theoretical models remain essentially untested. In some multiply ionized systems, an orbiting mechanism is predicted to give very large electron capture cross sections with a $1/v$ velocity dependence.

Details of the apparatus can be found elsewhere.²¹ Successful merging of 5- to 7-keV beams of D^0 with 30- to 40-keV beams of N^{3+} in a 76-cm-long ultrahigh-vacuum merge path has been carried out, yielding center-of-mass energies of 4–70 eV/nucleon. A D^- beam from a duoplasmatron ion source is photodetached using a 1.06- μm Nd:YAG laser, producing a highly collimated beam of D^0 . A N^{3+} beam was produced by the ORNL PIG ion source. Significant improvements in beam line optics, vacuum, and noise reduction have led to the operating parameters presented in Table 2.3. The secondary electron emission coefficient γ needed to calibrate the neutral Faraday cup can be experimentally determined in situ to a few percent. Signals in the D^+ detector (channel electron multiplier) resulting from beam-beam interaction are measured above the background from beam-gas interactions by using a standard beam modulation scheme to chop both the multicharged ion and neutral beams and to gate dual counters to record both signal-plus-background and background events. The form factor is a measure of the two-dimensional beam-beam overlap and is measured at four positions along the merge path by two mechanical knife-edge scanners and two commercial rotating wire scanners. The two rotating wire scanners provide a real-time two-dimensional profile of the beams that has proved essential in tuning the D^0 and N^{3+} beams to achieve a good overlap throughout the merge path.

Table 2.3. Typical operating properties for ion-atom merged-beams experiment

D^- beam	$E = 4.00 \text{ keV}$ $I(D^-) = 1.5 \text{ } \mu\text{A}$, $I(D^0) = 13 \text{ nA}$ $\gamma = 0.59 \pm 0.02$
N^{3+} beam	$E = 36.0 \text{ keV}$ $I(N^{3+}) = 1.0 \text{ } \mu\text{A}$
Center-of-mass energy	35.9 eV/nucleon
Merge path pressure	
Base	$2 \times 10^{-10} \text{ torr}$
With beams	$7 \times 10^{-10} \text{ torr}$
D^+ background rate due to D^0 stripping	3 kHz
Noise from N^{3+} beam	50 Hz
Beam-beam signal	120 Hz
Average two-dimensional form factor	2.5 cm^{-2}
Time for s.d. of 10% on signal	3 min

Extensive diagnostics have been performed on the beam-beam signal observed in the $N^{3+} + H$ system. Horizontal and vertical scans of the signal indicate that all of the signal is being collected. The signal was observed to be independent of the beam chopping frequency above 500 Hz, indicating that beam-produced gas pressure modulation effects were being correctly subtracted from the background. However, since the N^{3+} beam was obtained from the ORNL PIG ion source, measurements of the beam-beam overlap, signal, and diagnostics on the signal were not possible before the characteristics of the N^{3+} beam changed. This instability and the presence of an unknown admixture of metastables in the N^{3+} beam made meaningful measurements of a cross section impossible.

The merged-beams experiment is presently being installed on the new ORNL ECR ion source. This source has produced ion beams significantly more intense and stable than the ORNL PIG ion source and in considerably higher charge states. This will allow measurements of more interesting systems, such as O^{5+} and C^{3+} , which are Li-like and contain no metastables to complicate comparisons between theory and experiment.

2.1.7 Multicharged Ion-Surface Scattering at Grazing Incidence

F. W. Meyer, C. C. Havener, W. Heiland, S. Overbury, and D. Zehner

In an experiment designed to investigate the feasibility of studying ion-surface interactions using the ORNL ECR multicharged ion source, we have measured the energy distributions of secondary electrons produced during collisions of H-like and He-like light ions with a gold surface at grazing incidence. The study of multicharged ion-surface interactions is of considerable current interest, in large part due to the significant neutralization potential energy that the multicharged projectiles carry into the collision. Experiments studying the reflected ion charge state distributions and the secondary electron emission coefficients for collisions with tungsten surfaces have been reported.^{22,23} The present experiment is the first, to our knowledge, to investigate the energy distributions of secondary electrons produced during multicharged ion-surface collisions.

Beams of N^{5+} , N^{6+} , O^{6+} , and O^{7+} were produced by the ORNL ECR ion source at 10-kV source potential. After charge selection and beam collimation, the ions were made incident at 5° on the (110) surface of a gold single crystal mounted in an ultrahigh-vacuum chamber equipped with a four-grid low-energy electron diffraction (LEED) system. The condition of the surface of the gold crystal was determined by the use of electron-induced Auger spectroscopy and the observation of LEED patterns. In addition, the energy distributions of the multicharged-ion-induced secondary electrons were obtained by using the LEED grids in the retarding field mode and numerically differentiating the transmitted secondary electron current vs retarding voltage curves.

Figures 2.6 and 2.7 show the secondary electron spectra observed for N^{6+} and O^{7+} incident projectiles. Both spectra show two peaks superimposed on a "kinetic emission" background having a monotonic energy dependence. The higher energy peaks at ≈ 350 eV and ≈ 480 eV for the N^{6+} and O^{7+} induced spectra, respectively, are believed to be associated with Auger decay of the K-shell vacancy in the H-like projectile via electrons in highly excited states. These high-lying states are populated during the collision by resonance neutralization from the valence band of the solid. Lower energy peaks at ≈ 250 eV

Fig. 2.6. N^{6+} incident on gold at 5° . Top: transmitted secondary electron current vs retarding voltage; bottom: secondary electron energy distribution (bar indicates "instrumental" width introduced by numerical differentiation).

occur at the same energy in both spectra and are believed to be associated with Auger decay of a target-atom inner-shell vacancy created during the collision.

Secondary electron spectra were obtained for the case of a "clean" surface that was etched by a 10-keV Ar^{1+} beam and subsequently annealed and for the case of a "dirty" surface that had just been exposed to atmosphere. The "clean" surface was characterized as having less than 20% of a monolayer of carbon surface contamination. The spectra acquired under the two surface conditions were indistinguishable.

Fig. 2.7. O^{7+} incident on gold at 5° . Top: transmitted secondary electron current vs retarding voltage; bottom: secondary electron energy distribution (bar indicates "instrumental" width introduced by numerical differentiation).

Secondary electron spectra were also measured for N^{5+} and O^{6+} incident projectiles, which have ground-state configurations with filled K-shells. Although the latter spectra were dominated by the "kinetic emission" background, the same peaks observed for the H-like incident ions could be discerned, albeit greatly reduced in size. These spectra indicate that a small fraction of the He-like ions extracted from the ECR source are created in $1s2s$ metastable states. Since these metastable ions necessarily have K-shell vacancies, the secondary electrons they produce during surface collisions manifest the same Auger peaks observed for the H-like ions.

Further work is planned to determine the extent to which ion-induced secondary electron spectroscopy may be used as a quantitative tool for studying the metastable content of ion beams, as well as to study in greater detail the mechanisms underlying ion neutralization and secondary electron emission during multicharged ion-surface interactions.

2.2 ATOMIC COLLISION THEORY

2.2.1 Summary of "Ionization Processes in Transition Metal Ions"⁵

M. S. Pindzola, D. C. Griffin, and C. Bottcher

In the last decade, experiment and theory have shown¹¹ that indirect resonance processes may make substantial contributions to the electron-impact ionization of transition metal ions in low stages of ionization. Contributions to the electron-impact single-ionization cross section can be made by the following processes:

and

where A represents an arbitrary ion with charge q . The first process is called direct ionization, the second excitation-autoionization, and the third resonant-recombination double autoionization. Contributions to the electron-impact double-ionization cross section can also be made by the following processes:

and

The first process is called ionization-autoionization and the second excitation-double autoionization.

Though other processes are possible, these five are of most interest in fusion plasmas. We are engaged in a survey of these processes for all stages of ionization of the stainless steel impurities present in such plasmas, notably iron. Our approach is to make an initial survey using configuration average approximations; that is, if the initial state belongs to the complex $3p^5 3d^2$, we average over all L-S terms in this complex. If some case shows interesting features or anomalies, or if the average approximation is dubious for any reason, we carry out calculations with full allowance for intermediate coupling in the final state. Recently we have developed the ability to include configuration interactions as well, such as the mixing of $3p^5 3d^2$ with $3p^3 3d^4$. For systems with several d -electrons, considerable optimization of algorithms is needed to make these calculations feasible at all. A single case may still take several hours to run on a Cray-XMP.

We began with low stages of ionization, as described in ref. 24, and have now reached Fe^{13+} in the iron sequence. The single-ionization cross section is plotted as a function of electron impact energy in Fig. 2.8. The enhancement due to the indirect process [Eq. (2.2)] is about a factor of three over a wide range.

Much interesting physics is contained in these cross sections. We cite two examples. In low stages of ionization, double ionization can be comparable with single due to the process described by Eq. (2.3). This is the case²⁴ in Fe^{2+} . In higher stages, low-lying metastable states (living longer than 1 μs) can dominate ionization in both plasmas and beam experiments, which appears to be the case in Fe^{9+} . Metastable states are discussed further in Sect. 2.2.3 and in the experimental sections (Sects. 2.1.2 and 2.1.4) of this chapter.

2.2.2 Summary of "Dielectronic Recombination in External Electric Fields"²⁵

D. C. Griffin, M. S. Pindzola, and C. Bottcher

Dielectronic recombination (DR) can be thought of as a two-step process. In the first step, a free electron collisionally excites an N -electron ion and is simultaneously captured into a doubly excited, autoionizing state of the $(N + 1)$ -electron ion:

Fig. 2.8. Electron-impact single-ionization cross section of Fe^{13+} . Thresholds for various excitation-ionization processes are indicated by arrows.

It can then autoionize back to the initial state i or to an excited state k of the N -electron ion:

$$X_j^*(Z, N+1) \rightarrow X_k^*(Z, N) + e^- . \quad (2.7)$$

However, it can also emit a photon and decay to an excited state f of the $(N+1)$ -electron ion:

$$X_j^*(Z, N+1) \rightarrow X_f^*(Z, N+1) + h\nu . \quad (2.8)$$

If this state is below the first ionization limit of the $(N+1)$ -electron ion, it will be stable, and the DR process will be complete.

We have considered DR transitions associated with $\Delta n = 0$ excitations. For such transitions within high-temperature plasmas, recombination through the multitude of doubly excited Rydberg states with very high principal quantum numbers will normally dominate the total DR process. Therefore, the cross section will be very sensitive to the

existence of external electric fields, which surely are present in plasmas and electron-ion beam experiments. First of all, such fields can ionize electrons in high Rydberg states and thereby decrease the DR rate. Second, electric fields redistribute the angular momentum among the doubly excited Rydberg states, which tends to open up more recombination channels and enhance the DR rate.

In an earlier study,^{26,27} we examined the systematics of field mixing in dielectronic recombination in the lithium and sodium isoelectronic sequences by employing the configuration average and linear Stark approximations. This provided physical insight into field-mixing effects and enabled us to determine the maximum field enhancement of DR cross sections as a function of ionization stage. More recently, we have performed extensive distorted-wave calculations of dielectronic recombination as a function of electric field strength. The intermediate-coupled, field-mixed eigenvectors used to represent the doubly excited Rydberg states were determined by diagonalizing a Hamiltonian that includes the spin-orbit, internal electrostatic, and Stark matrix elements. These calculations then properly take into account the mixing between individual doubly excited states with the same value of n but different values of l due to the presence of an external electric field. However, as in earlier studies, we do not include mixing between Rydberg states with different values of n , and we ignore the effect that the electric field may have on the incoming continuum electron.

Studies are complete on the alkali-like ions extensively investigated by means of the merged-beams experiment at ORNL.²⁸ A typical comparison of theory and experiment is shown in Fig. 2.9. The data are fitted by the calculation corresponding to the mean field in the collision region of the apparatus (25 V/cm).

These calculations are very expensive, and, unfortunately, they have not so far suggested any simple model for the variation of cross section with field, especially at very low field values (<10 V/cm).

2.2.3 Summary of "Role of Metastable States of Complex Ions in Crossed-Beam Experiments"²⁹

M. S. Pindzola, D. C. Griffin, and C. Bottcher

We have studied the contributions of excitation-autoionization to the electron-impact ionization of several multiply charged ions in the magnesium isoelectronic sequence. The largest contributions are due to $2p \rightarrow 3p$ monopole and $2p \rightarrow 3d$ dipole excitations, which are of current interest in short-wavelength laser research.³⁰ The recent experimental crossed-beams measurements of Howald et al.¹⁶ for S^{4+} , Cl^{5+} , and Ar^{6+} show a clean separation between sizable excitation-autoionization features and the background direct ionization process at incident electron energies around twice the ionization threshold. These measurements present a clear challenge to theoretical interpretation and calculation.

Electron-impact ionization of an atomic ion labeled A can follow two paths:

Fig. 2.9. Theoretical values of the DR rate coefficients for S^{2+} (as measured in the electron-ion merged-beams experiment) as a function of electron energy for five different electric fields F in the interaction region. $F = 0$ V/cm; --- $F = 5$ V/cm; - - - $F = 25$ V/cm; - · - $F = 125$ V/cm; — $F = 625$ V/cm. The calculations include all resonances up to $n_{\text{max}} = 63$. The circles are the data from the merged-beams experiment.

or

The first process is direct ionization, and the second is excitation-autoionization. It is assumed that these processes are independent and that radiative stabilization of the autoionization states is negligible for low stages of ionization along the magnesium isoelectronic sequence. The latter assumption was tested by explicit calculation of branching ratios for several of the doubly excited configurations considered in this paper.

Most of the calculations were performed using the average configuration distorted-wave approximation. The energy spread within configurations was then established by a procedure we call the average statistical model (ASM), in which the total cross section is divided among the levels of the final complex according to the statistical weights of the various terms. For S^{4+} in the threshold energy region, we compared the ASM with a detailed intermediate-coupled level-to-level calculation. Except for some refined details, the ASM is acceptable.

Results for S^{4+} are shown in Fig. 2.10 and compared with the experimental crossed-beams measurements of Howald et al.¹⁶ over a wide energy range. Although the overall agreement is quite good, three discrepancies were noted. (1) The measured ionization threshold is too low. (2) The experimental excitation-autoionization features are spread over a wider energy interval than predicted by theory. (3) Experiment and theory diverge increasingly above 240 eV. The same discrepancies were repeated in Cl^{5+} and Ar^{6+} .

The puzzle was resolved by repeating the calculations for ions initially in the metastable $2p^63s3p^3P$ state, which live longer than the transit time through the apparatus. These results are compared with the same data¹⁶ in Fig. 2.11; the discrepancies are seen to be resolved. The data are consistent with between 50% and 100% metastables in the beam for all the ions studied.

We believe that low-lying metastable states will dominate the measured cross sections for many highly ionized species. They may also dominate the physics of the same species

Fig. 2.10. Electron-impact ionization of S^{4+} . Solid curve: total cross section from the $2p^63s^2$ ground configuration in the average statistical model; dashed curve: direct cross section only; solid circles: experimental measurements. The discrepancies between theory and experiment are discussed in the text.

Fig. 2.11. Electron-impact ionization of S^{4+} . Solid curve: total cross section from the $2p^6 3s 3p$ excited configuration in the average statistical model; dashed curve: direct cross section only; solid circles: experimental measurements. The discrepancies between theory and experiment (Fig. 2.10) have been resolved.

in plasmas, including X-ray laser plasmas, where they may help to resolve the long-standing disagreement between observations and computer simulations.

2.2.4 Summary of "Towards Numerical Solution of the Threshold Ionization Problem"³¹

C. Bottcher

We have continued to study the Peterkop model of low-energy ionization, based on the Schrödinger equation

$$\left[-\frac{1}{2} \left(\frac{\partial^2}{\partial r_1^2} + \frac{\partial^2}{\partial r_2^2} \right) + \left(\frac{1}{r_1 + r_2} - \frac{1}{r_1} - \frac{1}{r_2} \right) \right] \psi = \left(E + \frac{1}{2} \right) \psi. \quad (2.11)$$

The calculation of ionization cross sections is hindered by a lack of understanding of the asymptotic solutions of Eq. (2.11). We have made progress in two directions.

The first direction is obtaining asymptotic solutions of Eq. (2.11) with uniform error on the hyperspherical arc $0 \leq \alpha \leq \pi/4$, where $(r_1, r_2) = (\rho \cos \alpha, \rho \sin \alpha)$. These solutions are obtained by joining exact semiclassical wave functions valid in $\alpha_1 \leq \alpha \leq \pi/4$ to a "near-axis expansion" valid in $0 \leq \alpha \leq \alpha_2$, where $\alpha_2 > \alpha_1$.

The exact semiclassical wave function has the form

$$\psi(\xi, \eta) = [P(\xi, \eta)\omega(\xi, \eta)]^{-1/2} \exp \left[-i \int_{\xi_0}^{\xi} ds P(\xi, \eta) \right], \quad (2.12)$$

where (ξ, η) refer to a bundle of classical trajectories emanating from a single "condensation point" (ρ_0, α_0) . The initial condition, apart from energy, is η , and the distance along the trajectory is ξ ; two adjacent trajectories η and $\eta + \Delta\eta$ are separated by a distance $\omega(\xi, \eta)\Delta\eta$ at later times. Thus, ω is the cross section of a tube of flux. As usual, P is the momentum along ξ . When the trajectories have been computed, (ξ, η) have to be mapped numerically onto (ρ, α) .

The near-axis expansion is obtained by making the ansatz (for $r_1 \gg r_2$)

$$\psi(r_1, r_2) = r_2^{1/2} H_1^{\pm} [(8r_2)^{1/2}] \exp \left[ikr_1 + \sum_{\lambda=2}^{\infty} \frac{f_{\lambda}(r_2)}{r_1^{\lambda}} \right] \quad (2.13)$$

in Eq. (2.11), where $H_n^{\pm}$ are Hankel functions and $\kappa^2 = 2E$. The functions $f_{\lambda}(x)$ satisfy a hierarchy of nonlinear ordinary differential equations that are readily integrated numerically.

To compare Eqs. (2.12) and (2.13), we calculate $\Delta S = \text{Re } S(\rho, \alpha) - \kappa\rho$ both ways, where $\psi = \exp(iS)$. Typical results are shown in Fig. 2.12. The two expressions have a

Fig. 2.12. $\Delta S (= \text{Re } S - \kappa\rho)$ vs α for $\kappa = 1$, $\rho = 90\alpha$. SC is the exact semiclassical result, Eq. (2.12), and NA+, NA- are the solutions defined by Eq. (2.13).

common interval of validity around 30° (this number depends on ρ) and so may be patched together to span the whole range of α . The arbitrary constant, which may be added to one or the other expression for ΔS , is independent of ρ , so that we have a uniform solution of Eq. (2.11), valid in this example for all $\rho \geq 40\alpha_0$. Our hope is that this solution can be extended to smaller values of ρ by numerical integration.

The second direction is the application of Eq. (2.12) to obtain exact threshold laws numerically. The ionization probability for producing two particles of energies ϵ and $E - \epsilon$ can be written, for small E , as

$$P(E, \epsilon) = CE^{1+\alpha} f(\epsilon/E) . \quad (2.14)$$

Analytic treatments³² valid near $\epsilon = E/2$ suggest that $\alpha \approx 0.1269$ and $f \approx 1$. The function P/E turns out to be simply proportional to ω in Eq. (2.12). Our calculations reproduce $\alpha = 0.127 \pm 0.001$, but show deviations of f from unity, as shown in Fig. 2.13. Indeed, f has a bona fide singularity, albeit a weak one: $f \sim \epsilon^{-1/4}$ near $\epsilon = 0$. These results agree fairly well with purely classical calculations³³ of f , except for the singularity; either behavior of f is consistent with experimental data. The singularity is, of course, a semiclassical pathology, which emphasizes the need for a fully quantal solution of the problem.

Fig. 2.13. The function $f(x)$ defined in Eq. (2.14), calculated from Eq. (2.12). The error bar indicates residual numerical uncertainty.

2.3 CONTROLLED FUSION ATOMIC DATA CENTER

C. F. Barnett, H. B. Gilbody, D. C. Gregory, P. M. Griffin, C. C. Havener, A. M. Howald, H. T. Hunter, R. K. Janev, M. I. Kirkpatrick, C. R. Mahon, E. W. McDaniel, F. W. Meyer, T. J. Morgan, R. A. Phaneuf, M. S. Pindzola, and E. W. Thomas

The Controlled Fusion Atomic Data Center (CFADC) was established at ORNL in 1959. Its primary task is to collect, review, evaluate, and recommend numerical atomic collision data of interest in controlled thermonuclear fusion research. The CFADC operates with the equivalent of two full-time staff members, as well as a number of expert consultants from the research community, upon which it depends heavily. The major activities of the data center are threefold:

- production of up-to-date bibliographies of fusion-related publications on atomic collision processes in computer files for online searching and publication,
- preparation and publication of compilations of recommended atomic collision data for fusion research, and
- processing of user requests for specific data or information about relevant atomic processes.

The data center actively participates in cooperative agreements with the Atomic Data Center for Fusion at the Institute for Plasma Physics (IPP) of Nagoya University, Japan, the Atomic and Nuclear Data Center of the Japan Atomic Energy Research Institute (JAERI), and the Atomic and Molecular Data Unit of the International Atomic Energy Agency (IAEA) in Vienna. Our staff continues to participate in the IAEA International Atomic and Molecular Data Center Network and to advise the IAEA data center on its activities. Our bibliographic computer tapes are sent to these centers every three months and form the basis for the quarterly IAEA *International Bulletin on Atomic and Molecular Data for Fusion*.

Our data center also maintains a close cooperation with the Atomic and Molecular Processes Information Center at JILA in Boulder, Colorado, and with the National Bureau of Standards Data Center on Atomic Spectral Lines and Transition Probabilities in Washington, D.C.

During this reporting period, a major effort has been directed toward the preparation of an updated and expanded version of the compilation of recommended tabular and graphical data titled *Atomic Data for Controlled Fusion Research* and published in 1977 as ORNL-5206 and ORNL-5207 (commonly referred to as the "Redbooks"). The new multiple-volume series is named *Atomic Data for Fusion* and will consist initially of five volumes (ORNL-6086 through ORNL-6090). These volumes, with their (expected) publication dates, are as follows:

- Vol. 1: Collisions of H, H₂, He, and Li Atoms and Ions with Atoms and Molecules (10/86),
- Vol. 2: Collisions of Electrons with Atoms and Molecules (3/87),
- Vol. 3: Particle Interactions with Surfaces (published 2/85),
- Vol. 4: Spectroscopic Data for Iron (published 2/85),
- Vol. 5: Collisions of Carbon and Oxygen Ions with Electrons, H, H₂, and He (5/86).

All tabular and graphical data for the new series are computer-generated in publication-ready format. To make the recommended data more compatible with the needs of fusion plasma modelers, a computer code has been implemented to calculate Maxwellian rate coefficients from the cross sections and to produce Chebyshev polynomial fits to both the cross sections and rate coefficients. This will facilitate our plan to eventually place these recommended collision data in user-accessible files on the National Magnetic Fusion Energy Computing Center (NMFEC) network.

2.4 ADVANCED PLASMA DIAGNOSTICS DEVELOPMENT

2.4.1 Alpha Particle Diagnostics

D. P. Hutchinson, K. L. Vander Sluis, R. K. Richards, and C. H. M^a

In an ignited D-T fusion reactor, the 3.5-MeV fusion-product alpha particles will provide the energy required to sustain ignition. Techniques for monitoring these products as they thermalize and give up their energy to the plasma will be crucial to the development and operation of a successful reactor. During the past year, we have concentrated our efforts on development of the necessary technology to perform laser Thomson scattering on alpha particles in a burning plasma. With this technique, information about the velocity distribution of these products is obtained from the Doppler shift of the laser radiation, which is scattered at small angles.

One feature that is essential to the success of this diagnostic is an optical-heterodyne scheme for detection of the scattered radiation. A test stand has been built to analyze and optimize the noise figure of the multichannel heterodyne receiver for the experiment. The primary components of this facility are a 5-W waveguide, single-mode CO₂ laser; a 500 K blackbody source; and the associated electronics to heterodyne-detect the blackbody signal from the detector. The test stand will also be used to measure the absorption coefficient of a CO₂ hot cell. This cell will be used in the receiver optics assembly to attenuate any stray laser light entering the detection system. A low-power sequence-band CO₂ laser has also been developed and tested for use as a local oscillator in this system.

The resonator design chosen for the high-power pulsed CO₂ laser will restrict operation to a single fundamental mode, an essential feature for the small scattering angles required by the experiment. The pulsed laser source for the scattering experiment will be a 100-J Lumonics Model 602 laser. The laser cavity is being modified to include a 1-m-long, low-pressure, transverse discharge cell and unstable resonator optics. The low-pressure cell, containing a He-CO₂-N₂ laser mixture, will be fired by a thyatron approximately 90 μ s before the main discharge of the pulsed laser is initiated. The population of the upper laser level in the low-pressure cell will reach a maximum after the 90- μ s delay, resulting in a slower rise time and longer duration of the 100-J pulse. The unstable resonator optics configuration will restrict operation of the laser to a single fundamental mode. The intracavity low-pressure cell and unstable resonator should produce a pulse length of approximately 1 μ s with a beam divergence compatible with the requirements of the scattering experiment.

We have defined an experiment on a nonburning plasma that we believe will demonstrate the feasibility of the alpha-particle scattering diagnostic. In our original feasibility study,³⁴ we predicted that the scattered signal from the alpha particles would be three to five times the signal from background plasma electrons over a particular range of frequency shifts from line center (5 to 20 GHz). Therefore, we plan to perform a small-angle scattering experiment from electrons in a nonburning plasma. If we are able to measure the scattered signal from the background electrons, then we may proceed to further development with some confidence that the diagnostic system will have the sensitivity to measure the alpha particle density and velocity distribution in an ignited plasma. In addition, the detection of scattered radiation from the electrons at larger frequency shifts (40 to 60 GHz) would permit a study of the high-energy tail of the electron distribution in a region inaccessible to standard Thomson scattering diagnostics.

2.4.2 Multichannel Interferometry

W. H. Casson, C. H. Ma, R. K. Richards, H. T. Hunter, D. P. Hutchinson,
K. L. Vander Sluis, and C. A. Bennett

The interferometer system from the Impurity Study Experiment (ISX) was moved from the Fusion Energy Division's facility at the Y-12 plant to the development laboratory at the X-10 site for modifications necessary for installation on the Advanced Toroidal Facility (ATF) in late 1986. Observation of laser stability and noise indicated that changes to the basic laser design were necessary in order to improve resolution and reliability. The interferometer system consists of two independent laser systems operated at a constant frequency difference of about 1 MHz. Each laser system comprises a CO₂ pump laser and a far-infrared (FIR), optically pumped laser cavity. The output wavelength can be chosen from a very large number of known laser lines by changing the molecular gas and the pump frequency. An analysis of the required spatial resolution and refractive effects on ATF dictates a shorter wavelength than that used on ISX. The most attractive candidate is the 119- μ m line of methyl alcohol, which is one of the strongest emission lines in the far infrared. For accurate Abel inversion of chordal data from the highly elliptical cross section of the ATF plasma, a 15-channel system is necessary. Operation of this number of channels, each with an individual system of optics, would require a prohibitive amount of alignment work and design complexity. The alternative selected is a single optical beam, highly elliptical in cross section, that will be focused onto a detector array after passing through the plasma. Each element of the detector array will consist of a Schottky diode mounted in a corner cube reflector. The total number of optical elements can thus be reduced from about 225 to around 30.

Modifications made to the CO₂ pump laser include elimination of the internal Brewster windows, redesign of the electrodes, and addition of a slightly different set of cavity optics. The Brewster windows, which were mounted between the electrical discharge tube and the internal optical elements, were a known source of beam distortion and thermal noise. They were eliminated by installing the optics within the vacuum of the discharge but physically separated from the discharge by a reasonable distance. Distortion of the discharge itself was reduced by designing electrodes that are symmetric about the

discharge axis. The pump laser was further improved by replacing the plane grating, used for selection of the CO₂ emission line, with a spherical curved grating and replacing the curved output coupler with a plane coupler. This effectively decreases the power density on the grating surface, optimizes use of the discharge volume, and places the output beam waist at the output coupler. Another improvement was made in the pump source by replacing the ZnSe lens used to focus the beam into the FIR cavity with a curved reflector, thereby further reducing beam distortion.

Most laser control functions are automated by a complex feedback arrangement overseen by a small computer control system. The CO₂ laser will be tuned for optimum power and frequency by an acousto-optic cell monitoring device. The FIR output will be monitored for output power and for the frequency difference between the two laser systems. The operator will have a continuously updated readout of all laser functions and will have override control at all times. The resulting improvements in the interferometer will make it possible for a single operator to start up and operate the laser system and to monitor the data output and analysis.

2.4.3 Far-Infrared Source Development

D. P. Hutchinson, J.-N. Juang, and C. H. Ma

The availability of a suitable directly excited laser source of FIR radiation would be of significant benefit in plasma diagnostics, in terms of both simplicity of operation and cost. We have begun the development of a directly excited HCN laser that we intend to use for studying density fluctuations in the ATF device currently under construction in the Fusion Energy Division. In the past, HCN lasers were very large and required long downtimes for maintenance. Recent work by several experimenters in Japan and the U.S.S.R. has pointed to the possibility of the construction of a reliable, high-power, rf-driven HCN laser operating at a wavelength of 337 μm . We have undertaken the construction of a small rf-driven HCN laser to evaluate its performance as a source of FIR power for plasma diagnostic applications. Our initial tests have been very encouraging, and we are proceeding with the design of an HCN laser with an integral rf power oscillator.

2.4.4 Calibration of the FIR Polarimeter on TFTR

C. H. Ma, D. P. Hutchinson, K. L. Vander Sluis, D. K. Mansfield,* H. Park,* and L. C. Johnson*

We have been engaged in the development of a multichord FIR polarimeter for the Tokamak Fusion Test Reactor (TFTR) at Princeton Plasma Physics Laboratory (PPPL). During the past year, simultaneous measurements of electron density and Faraday rotation in TFTR plasmas have been achieved. Density measurements have been made successfully on all five channels of the FIR interferometer/polarimeter system. However, due to limited financial resources and time, a decision was made by PPPL to delay the multichord measurements of the Faraday rotation. The system has been routinely employed to study ohmically heated and neutral-beam-heated plasmas, including plasmas into which multiple solid

hydrogen pellets were injected. It is believed that this is the first successful simultaneous measurement of electron density and Faraday rotation in multiple pellet-injected plasmas. Apparatus for expansion to ten-channel operation has been designed, constructed, and tested. The installation of the ten-channel system on TFTR is under way.

A schematic diagram of the interferometer/polarimeter system on TFTR is shown in Fig. 2.14. The system consists of a Michelson-type interferometer made from two cw 119- μm methanol lasers, optically pumped by a CO_2 laser. The beat frequency of the FIR lasers is usually adjusted to approximately 1 MHz by tuning the cavity length. The probing and reference beams are divided into ten channels (only five channels were in operation by April 1985) by metallic meshes. The probing beams go through the plasma from bottom to top and are returned by the retro-reflectors. All optical components are mounted on a diagnostic support structure totally isolated from the TFTR machine and supported on seismic isolators. The FIR laser beams are initially linearly polarized. A wire-grid polarizer is used as an analyzer to provide the Faraday rotation signal. The analyzer is adjusted to pass the outgoing probing beam and reflect any rotated component of the return beam. This component is mixed with a portion of the reference beam and is directed onto the

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Fig. 2.14. Schematic of the FIR interferometer/polarimeter system on TFTR.

polarimeter detector. Schottky diodes are used for all detectors. The output of the polarimeter detector is filtered, amplified, and fed into an envelope detection circuit. If the ellipticity of the polarization of the return beam is negligible, the output of the detection circuit is proportional to $\sin(\theta_F)$, where θ_F is the Faraday rotation angle of the polarization vector, and is proportional to the line integral of electron density times the poloidal magnetic field along the double path of the probing beam. Part of the beam from the reference laser is mixed first in the reference detector with a portion of the source laser, which is split off before passage through the plasma, and the remainder is guided to the signal detector to mix with part of the return beam. The outputs of both detectors are also filtered, amplified, and fed into a digital phase detection circuit to extract the phase shift produced by the electron density. Due to limited resources, only the outputs of the phase detectors were digitized for computer storage and processing as of April 1985. The output of the envelope detection circuit is displayed on oscilloscopes for photographic recording (only one channel).

The interferometer/polarimeter system was used on TFTR for more than four months. Figure 2.15 shows the time-resolved traces of line electron density and rotation of a typical neutral-beam-heated plasma discharge. During this discharge, three solid hydrogen pellets were injected into the plasma at intervals of approximately 0.5 s. The abrupt changes of density and Faraday rotation occur during a period of $\approx 400 \mu\text{s}$.

An algorithm for the solution of the wave propagation equation has been developed and used to determine the polarization evolution on the Poincaré sphere. Computer codes have been used to calculate the ellipticity ϵ and the rotation angle of the vibrational ellipse ϕ of the polarization.

The significant parameters of TFTR are as follows: major radius $R = 265 \text{ cm}$, minor radius $a = 85 \text{ cm}$, central electron density $n_0 = 10^{14} \text{ cm}^{-3}$, plasma current $I_p = 2.5 \text{ MA}$, toroidal field $B_t = 5.2 \text{ T}$, and wavelength $\lambda = 119 \mu\text{m}$. In this case, the maximum ϕ is approximately 15° with a maximum ϵ of 0.045 for the double path of the beam. The output signals of the interferometer detector, V_i , and of the polarimeter detector, V_p , can be expressed by

$$V_i = V_{i0} \cos(\phi) [1 + \epsilon^2 \tan^2(\phi)]^{1/2} \cos(\Delta\omega t + \psi + \psi_i)$$

and

$$V_p = V_{p0} \sin(\phi) [1 + \epsilon^2 \cot^2(\phi)]^{1/2} \cos(\Delta\omega t + \psi + \psi_p) ,$$

where V_{i0} and V_{p0} are the calibration constants for the interferometer and polarimeter, respectively; ψ is the phase shift due to electron density; and ψ_i and ψ_p are given by

$$\psi_i = \tan^{-1}[\epsilon \tan(\phi)]$$

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Fig. 2.15. Time variation of (a) line electron density and (b) Faraday rotation measured by the multichord FIR in erferometer/polarimeter system on TFTR. The abrupt changes of the line density and Faraday rotation are caused by the injection of solid hydrogen pellets.

and

$$\psi_p = \tan^{-1}[\epsilon \cot(\phi)] .$$

It can be seen from these equations that both the amplitude and the phase of the signals depend on the ellipticity of the polarization, which cannot be measured easily due to limitations on the experimental techniques. The error in density measurement due to neglecting the ellipticity is very small (0.024%), and the error in the Faraday rotation measurement is approximately 1.7%.

During the period of operation, sinusoidal oscillation of the Faraday rotation signal due to cross-coupling of the laser beams was observed. A modified theory for the polarimeter that includes the cross-coupling was developed. A comparison between the theory and the experiments gave excellent agreement. Experiments for calibration of the polarimeter were also carried out to determine the effects on the sensitivity of the system and the rotation angle due to the cross-coupled beams.

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Chapter 3

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Plasma Theory

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R. A. Dory

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3. PLASMA THEORY

SUMMARY OF ACTIVITIES

Fundamental contributions have been made to the theoretical physics of plasma confinement devices, general plasma areas, and related fields.

The nonlinear analysis and turbulence theory developed for studying confinement physics has applications as well in solar plasma and reactor confinement physics. Correlations have been established between the two-point spectral theory and the results of both simple and more sophisticated simulation codes.

A comprehensive theory has been developed for the complicated physical and mathematical problem of neoclassical (collisional) transport in the fully three-dimensional (3-D) magnetic structures of stellarator confinement devices. This work has built upon previous pioneering work that led to compact and efficient 3-D equilibrium analysis codes and to similarly compact and efficient representation of the velocity-space behavior of long mean-free-path plasmas.

Procedures have been developed to analyze the "full wave" rf field problem, eliminating the common geometrical optics approximation—the validity of which is debatable at best for present plasma confinement experiments. This will form the basis for eventual detailed analyses leading to the energy deposition profiles required to determine energy flows in the Advanced Toroidal Facility (ATF) and other devices.

The stability theory brought to broad maturity by development of nonlinear turbulence models has been supplemented by a "neoclassical MHD" model, which begins to incorporate the kinetics of long mean-free-path behavior. This comes into play when confinement devices enter the temperature regimes of burning fusion plasmas. Similarly, kinetic phenomena from alpha particle fusion "ash" have been incorporated into both stability and transport analyses.

Strong support has been provided for the U.S. effort to define and design a fusion ignition device, the Compact Ignition Tokamak (CIT), at a size consistent with credible funding requirements. Essential analysis of the fusion burn dynamics has been carried out, including determination of the tolerance of the preconceptual design to effects that degrade confinement. A novel operating scenario has been suggested, using ramped fields and currents to establish a hot plasma discharge within the pulse time limits defined by cost factors.

Theoretical contributions were an essential ingredient in the development of several innovative plasma confinement concepts:

- An ELMO Bumpy Square device was proposed, as a reconfiguration of the present ELMO Bumpy Torus. Unfortunately, the current funding climate makes it unlikely that it will be constructed.
- The Spherical Torus Experiment (STX) proposal remains a possible candidate for a future DOE initiative. Among many novel factors recommending it is the recently completed analysis of tearing modes in very low aspect ratio tokamaks.

- The TJ-II flexible heliac, a joint conceptual design by ORNL and the Spanish Junta for Nuclear Energy (JEN), has been proposed by JEN for construction in Madrid and has received initial approval by EURATOM for proceeding to a full engineering design.
- Attractive low-aspect-ratio configurations have been developed both for the reactor phase of the stellarator program and as candidates for a next step in the ORNL program, ATF-II.
- For the Heliotron-E device at Kyoto, a new approach has been proposed to access the "second stability regime," where plasma effects provide improved stability. In this case, increased shear is provided by toroidal field coils to accomplish the stabilization.

Broad contributions were made to the development of new theoretical models and the understanding of confinement processes, fluctuation phenomena, effects of ambipolar potentials, bootstrap currents, particle drifts, plasma edge phenomena (scrapeoff layer, limiters, divertors, ergodic field line regions), plasma fueling, injected pellet ablation, rf antennas, and energy deposition. Guidance was provided for engineering understanding of the rich geometrical aspects caused by the helical toroidal structures of the ATF device and of potential successors up to the fusion demonstration reactor stage.

In preparation for the increased load from the International Fusion Superconducting Magnet Test Facility and the ATF stellarator device, an integrated computing network was defined and implemented in initial form. It makes use of the improved performance per unit cost (hardware and maintenance) available within the VAXcluster concept. The capability is centered on an 8600 central processor with clustered worker units organized around a hierarchical storage "disk farm."

This past year's work is summarized in more detail in the sections on fundamental areas of research, Sects. 3.1-3.3, and on the integrating and experiment interpretation efforts, Sects. 3.4-3.6. Computing support activities are summarized in Sect. 3.7.

3.1 MHD EQUILIBRIUM, STABILITY, AND TURBULENCE

The magnetohydrodynamic (MHD) equilibrium, stability, and turbulence effort is characterized by the parallel use of analytic and numerical fluid models to study the behavior of magnetically confined plasmas. Equilibria are determined, and their linear and nonlinear stability properties are analyzed. Calculations also are made of nonlinearly evolved fluctuation levels and their consequences for particle and heat transport. Research is carried out on two strongly interconnected levels. First, fundamental physics research is performed, generally using simplified geometry, to unveil the basic mechanisms that underlie plasma behavior. This work establishes a sound basis for further theoretical development. Theoretical results are then applied to specific devices, providing basic physics understanding and tools for configuration optimization and for plasma modeling.

In the area of equilibrium theory, calculational techniques were improved to provide more efficient computation and a better physics understanding. A new two-dimensional

(2-D) averaging method approach to the three-dimensional (3-D) equilibrium problem permits a fast reconstruction of the magnetic surfaces from experimental magnetic measurements. This technique is especially well suited to the Advanced Toroidal Facility (ATF). A better differentiation scheme for the 3-D equilibrium spectral codes gives fast and accurate results with relatively coarse radial grids. Applications of equilibrium theory were mainly oriented toward configuration optimization. The equilibrium properties of tokamaks with very small aspect ratio were investigated thoroughly, with particular application to the Spherical Torus Experiment (STX). The flexible heliac equilibrium was optimized for application to the design of the TJ-II device. In addition, further studies were made of the effect of the added toroidal field on the Heliotron-E equilibrium.

In the area of linear stability theory, work continued to derive a set of MHD equations valid in the plateau regime that would include the "neoclassical" instabilities of Shaing and Callen. Alpha particle effects were added to the ballooning stability theory for tokamaks. Linear stability theory was applied to the analysis of several very diverse configurations. We continued to map the stability boundaries of ATF, taking into account more of the degrees of freedom of the device. Stable equilibria with beta as large as 41% were found for a low-aspect-ratio ($R/a = 1.5$) tokamak. For the design parameters of the STX device, a beta of nearly 20% should be accessible. The addition of a toroidal field to the Heliotron-E device was shown to provide a stable access path to the second stability regime. The stabilization is due to an increase of the shear with beta, not to a magnetic well effect. The Mercier stability of the flexible heliac was studied. Low-aspect-ratio configurations were found to be stable for beta as large as 5%, and significant additional optimization is possible.

In the area of nonlinear stability theory and turbulence, the combination of analytical and computational work led to several significant new results. A detailed study of g -mode turbulence saturation phenomena showed that the saturation mechanism is probably not that predicted by the simple mixing length model because the level of fluctuations and associated losses found was higher than that predicted by that model. The effects of low aspect ratio on the nonlinear evolution of the $m = 1$ tearing mode were analyzed. Depending on the q profile and aspect ratio, either a saturation of the island or an incomplete reconnection takes place. These results show that the sawtooth oscillation phenomenon is more complicated than the Kadomtsev model predicts. The numerical results have many of the features observed in experiments. Also, it was shown that at very low aspect ratio, the saturated magnetic islands corresponding to the $m > 1$ tearing modes are smaller than predicted by the large-aspect-ratio theory. This implies that the stable operating regime for low-aspect-ratio tokamaks is broader than thought earlier.

These physics studies require continual development and evolution of efficient codes and techniques. In addition to those already discussed, a new stellarator averaging method code for free boundary equilibria was developed and tested in the past year. In order to take advantage of the increased computing capability at the National Magnetic Fusion Energy Computing Center (NMFEC)—in particular, the Cray-II four-processor computer—we have worked to adapt several of our large codes to a multitasking environment. As an example, we have operated the nonlinear FAR code, which solves the full MHD equations for a tokamak configuration, in a multitasking version on the Cray-II.

The following abstracts of work performed in this effort are organized by device type, rather than by area, because many of them report on work covering more than one area.

3.1.1 Stellarators—Abstracts

3.1.1.1 "Finite Beta Magnetic Surfaces in a Fixed Toroidal Plane for Stellarator Configurations"¹

B. A. Carreras

The stellarator expansion reduces the equilibrium calculation to the solution of a Grad-Shafranov type equation in a 2-D geometry which allows for fast equilibrium calculations. However, the subsequent reconstruction of the 3-D magnetic surfaces either is limited to the inner surfaces or requires a cumbersome 3-D mapping. Here, the problem of reconstruction of the magnetic surfaces is reformulated. A Grad-Shafranov equation is derived for each $\phi = \text{const}$ plane and is solved using real-space boundary conditions. The solution of the equation gives directly the magnetic surfaces in real space for the toroidal plane considered. This method provides an efficient approach to the reconstruction of finite-beta magnetic configurations from experiment. Free boundary equilibrium calculations based on this method will be presented.

3.1.1.2 "Resistive Ballooning Modes in Helical Axis Stellarators"²

W. A. Cooper and M. C. Depassier

The helical nature of the magnetic axis induces a curvature component that drives resistive ballooning modes that scale as the resistivity to the third power in a helical-axis stellarator.

3.1.1.3 "Resistive Ballooning Modes in Helical Axis Stellarators with Longitudinal Mass Flow"³

M. C. Depassier and W. A. Cooper

The equilibrium equation for systems with helical symmetry and longitudinal mass flow is derived for cases with temperature constant on a flux surface. The incompressible resistive ballooning mode equation for systems with a coordinate of symmetry and rigid longitudinal flow is also derived. A reduced ballooning equation is obtained by an expansion in $\epsilon = ah \ll 1$, where a is the minor radius and h is the helical pitch. The magnetic field line curvature is a quantity of order ϵ^2 . Thus, ignoring the effects of toroidal curvature and treating the system as helically symmetric is valid for $\epsilon^2 \gg \epsilon_t$, where ϵ_t is the inverse toroidal aspect ratio. An analytic solution of the ballooning equation is obtained in the electrostatic approximation and invoking the two-length-scale expansion for a model equilibrium with circular flux surfaces and a magnetic axis with dimensions comparable to the minor radius. In the small longitudinal velocity limit, the static mode width and growth rate, scaling as one-third the resistivity, are recovered. However, for small growth rates

and large longitudinal velocity, the interaction of the magnetic curvature and the centrifugal force with the pressure and mass density gradients that lie on the surface, respectively, determine the width of the mode. In this limit, the growth rates are driven by the interaction of the magnetic curvature and centrifugal force components due to the helical motion of the magnetic axis with the radial pressure and mass density gradients, respectively. These modes scale linearly with the resistivity at a reduced level.

3.1.1.4 "Helically Symmetric Equilibria"⁴

M. C. Depassier, W. A. Cooper, and S. P. Hirshman

The 3-D MHD energy principle is expanded to leading order in N^{-1} , where $N \gg 1$ is the number of toroidal field periods. By transforming to a rotating coordinate system, a set of reduced, helically symmetric force balance equations is obtained for a finite transform, ι/N , and a finite pitch, $h = N/R_0$, per field period. The resulting constraints on the moment amplitudes of the inverse coordinates are described. The numerical solutions for equilibria with zero net longitudinal current are in excellent agreement with those obtained from a 2-D helically symmetric version of the energy principle for $N \geq 500$.

3.1.1.5 "Studies of the Plasma Edge Region in Stellarators Using a Field Line/Orbit Code"⁵

R. H. Fowler and J. A. Rome

The region between the last closed flux surface and the vacuum vessel is examined by following field lines and particle orbits. Realistic multifilament models are used to compute the vacuum magnetic field. The geometrically correct vacuum vessel is used to determine wall loadings. Applications are made to the optimal positioning of a pumped limiter in Heliotron-E and the ion cyclotron resonant frequency (ICRF) antenna for ATF. It is found that losses occur in distinct wall stripes. The connection length to a pumped limiter is on the order of the connection length, $2\pi Rq$. Charged particle losses can differ significantly from field line losses due to mirror effects. Electric fields could be significant outside the plasma but are difficult to incorporate into the model.

3.1.1.6 "3-D Nonlinear MHD Calculations With Drift and Thermal Force Effects"⁶

L. Garcia, H. R. Hicks, B. A. Carreras, L. A. Charlton, and J. A. Holmes

A computer code (KITE) that solves a reduced set of MHD equations with diamagnetic and thermal force effects included has been constructed. It can use two different time integration schemes. A mostly-explicit time integration scheme is shown to be efficient for the nonlinear phase of tearing mode turbulence where numerical stability demands a small timestep. However, for linear calculations, or nonlinear ones in which the level of turbulence is low, a mostly-implicit approach is seen to be more efficient. The two numerical schemes yield the same solutions.

Nonlinear MHD calculations in which the solutions are represented by a finite Fourier series allow one to study the dependence of the nonlinear solution on its permitted

harmonic content. It is seen that for the 3-D, nonlinear, tearing mode disruption problem, the refinement of the solution by the addition of more modes leads to further destabilization. This contrasts to the rippling mode problem, in which the addition of modes is stabilizing. For the latter, a finite number of modes yield a converged solution that is a saturated state. This contrasting behavior reflects the basic physical mechanism of the nonlinear interaction in each case and not the particular numerical scheme used for the calculations.

3.1.1.7 "Torsatron Equilibrium and Stability Studies"⁷

T. C. Hender, B. A. Carreras, L. A. Charlton, L. Garcia, H. R. Hicks, J. A. Holmes, and V. E. Lynch

Equilibrium and stability results are presented for the ATF device. The results of 3-D equilibrium calculations and free boundary average method calculations are shown to be in good agreement with previous fixed boundary average method results. These favorable comparisons serve as a valuable validation of the simple and computationally efficient fixed boundary average method. Stability calculations for the free boundary average method equilibria are also in good agreement with fixed boundary calculations, showing instability only when the plasma is shifted inward with an applied vertical field.

3.1.1.8 "ORMEC: A Three-Dimensional MHD Spectral Inverse Equilibrium Code"⁸

S. P. Hirshman and J. T. Hogan

The Oak Ridge Moments Equilibrium Code (ORMEC) is an efficient computer code that has been developed to calculate 3-D MHD equilibria using the inverse spectral method. The fixed boundary formulation, which is based on a variational principle for the spectral coefficients (moments) of the cylindrical coordinates R and Z , is described and compared with the finite difference code BETA developed by Bauer, Betancourt, and Garabedian. Calculations for Heliotron, Wendelstein VIIA, and ATF configurations are performed to establish the accuracy and mesh convergence properties for the spectral method.

3.1.1.9 "A Spectral Method for Obtaining Three-Dimensional Magnetohydrodynamic Equilibria"⁹

S. P. Hirshman and D. K. Lee

A description is given of a new code, MOMCON (spectral moments with constraints), that obtains 3-D ideal MHD equilibria in a fixed toroidal domain using a Fourier expansion for the inverse coordinates (R, Z) representing nested magnetic surfaces. A set of nonlinear coupled ordinary differential equations for the spectral coefficients of (R, Z) is solved using an accelerated steepest descent method. A stream function λ is introduced to improve the mode convergence properties of the Fourier series for R and Z . Constraint equations relating the $m \geq 2$ moments of R and Z are solved to define a unique poloidal angle.

3.1.1.10 "Interactive Analysis of the Output from a System of Codes"¹⁰

V. E. Lynch, H. R. Hicks, and B. A. Carreras

A postprocessor has been developed to interactively analyze the results of numerical calculations. This is accomplished directly on the computer using self-identifying data files and an interactive analysis tree. This method will replace the hard-copy output, graph paper, and hand calculators presently required to plot or compare results in ways not anticipated at the time of the calculation. It also facilitates comparing results of independent calculations.

Our system of codes includes numerous MHD equilibrium, stability, nonlinear evolution, and diagnostic codes. Many of these calculations are sufficiently expensive to perform that they should not be rerun merely to obtain a few plots which were not originally requested. Since all of these codes produce different output files and are continuously evolving, we require the codes to write their significant data into a machine-readable output file with a table at the beginning of the file. This table contains some identifying information about the calculation, the number of arrays, and, for each array, its variable name, location in the file, number of dimensions, type of dimension (time coordinate, radial coordinate, mode number, etc.), and the minimum and maximum value for that dimension. The file is created as an absolute file, rather than text or binary, so that once the identifying table is read, the locations of the needed arrays can be directly read.

3.1.1.11 "Stellarator Expansion Methods of MHD Equilibrium and Stability Calculations"¹¹

V. E. Lynch, B. A. Carreras, L. A. Charlton, T. C. Hender, L. Garcia, H. R. Hicks, and J. A. Holmes

Two methods for performing stellarator expansion, or average method, MHD calculations are described. The first method includes the calculation of vacuum, equilibrium, and stability using the Greene and Johnson stellarator expansion in which the equilibrium is reduced to a 2-D problem by averaging over the geometric toroidal angle in real-space coordinates. In the second method, the average is performed in a system of vacuum magnetic coordinates. Both methods are implemented to utilize realistic vacuum field information, making them applicable to configuration studies and machine design as well as to basic research. Illustrative examples are presented to detail the sensitivities of the calculations to physical parameters, to show numerical convergence, and to show the comparison of these methods with each other and with other methods.

3.1.1.12 "Nonlinear Resistive g Mode and Electron Heat Conductivity in Torsatron/Heliotron Plasmas"¹²

K. C. Shaing and B. A. Carreras

An electron heat conduction coefficient χ_e induced by the nonlinear saturation of the resistive g mode in a stellarator is calculated. It agrees reasonably well with the anomalous χ_e values inferred from Heliotron-E discharges.

3.1.2 Tokamaks and Reversed-Field Pinches—Abstracts

3.1.2.1 *Toroidal Field Effects on the Stability of Heliotron E*¹³

B. A. Carreras, L. Garcia, and V. E. Lynch

The addition of a small toroidal field to the Heliotron-E configuration improves the stability of the $n = 1$ mode and increases the value of the stability beta critical. Total stabilization of this mode can be achieved with added toroidal fields between 5% and 15% of the total field. In this situation, the plasma can have direct access to the second stability regime. For the Heliotron-E configuration, the self-stabilization effect is due to the shear, not due to the magnetic well. The toroidal field threshold value for stability depends strongly on the pressure profile and the plasma radius.

3.1.2.2 "Resistive MHD Numerical Calculations in Toroidal Geometry"¹⁴

L. A. Charlton, B. A. Carreras, J. A. Holmes, H. R. Hicks, and V. E. Lynch

A computer code, named FAR, has been constructed which solves the full MHD equations in toroidal geometry. No ordering assumptions are used. The code is applicable to toroidal devices including tokamaks, stellarators, and reversed-field pinches (RFPs). Both linear and nonlinear terms may be included. For linear calculations, a fully implicit numerical technique is used which allows eigenvalues and eigenfunctions to be found in a few steps and also allows eigenfunctions to be studied which are not the most unstable. Nonlinear terms are added implicitly which requires a decrease in step size after the nonlinear regime is reached.

3.1.2.3 "Tokamak M.H.D. Activity: Calculations Using a Full Set of Nonlinear Resistive M.H.D Equations"¹⁵

L. A. Charlton (in collaboration with B. A. Carreras, T. C. Hender, H. R. Hicks, J. A. Holmes, and V. E. Lynch)

Calculations have been carried out, using a full set of MHD equations in toroidal geometry, to understand the effects of toroidicity and beta upon the nonlinear evolution of resistive modes and the consequent interpretation of MHD activity. Attention has been focused upon two distinct phenomena: $m = 1$ evolution and the tokamak disruption. Specifically, for cylindrical calculations done at moderate effective aspect ratio $A \sim 3$ and low beta, the experimentally observed "shrinking emitter" behavior is reproduced numerically when $0.9 \leq q_0 \leq 1.0$: the region surrounding the original magnetic axis is pushed away by the initial growth of the $m = 1$ island. The area around the original magnetic axis then returns to near its initial location. This behavior is not observed in a reduced equations model which yields the traditional Kadomtsev-type reconnection of the $m = 1$. As q_0 is lowered to $q_0 \leq 0.8$ or as the aspect ratio is made large ($A \sim 10$) the evolution obtained from the full set of equations reverts to the complete Kadomtsev-type reconnection. For smaller aspect ratios, $A \leq 2.5$, the $m = 1$ magnetic island is observed to

saturate to a helical steady-state solution. For fully toroidal calculations and moderate aspect ratio ($A \sim 4$) the shrinking emitter behavior is also seen. The persistence of the $m = 1$ island during this motion is also consistent with the experimental behavior described by Dubois et al. The second question addressed involves the effects of aspect ratio upon the major disruption as modeled using the multiple helicity interaction of tearing modes. Destabilization of the (2/1) mode due to toroidal couplings to satellite $n = 1$ components competes with stabilization due to geometric effects (as is also the case with reduced MHD equations). This geometric stabilization is greater for scans having fixed safety factor profiles (hence A -dependent current profiles) than for scans of fixed current profile. Multiple helicity calculations, which include the evolution of the important 2/1 and 3/2 modes, will also be presented.

3.1.2.4 "Nonlinear M.H.D. Calculations in Toroidal Geometry"¹⁶

L. A. Charlton, B. A. Carreras, H. R. Hicks, J. A. Holmes, and V. E. Lynch

A resistive MHD model derived for the large aspect ratio has been successful in understanding low-frequency MHD activity in tokamaks. Experimental results indicate, however, that the internal kink mode could play a role at high injection power. Since this mode is excluded by the ordering assumptions used for the large-aspect-ratio method, work was started to construct a computer code using a formalism which does not utilize these ordering assumptions. For this purpose, incompressible equations were developed which are similar to those of Azumi et al. Linear calculations from the resulting computer code FAR were presented earlier where it was found that the $n = 1$ results using the new formalism were identical to those obtained with large-aspect-ratio ordering at high beta ($\beta_0 > 15\%$, where β_0 is the value of beta at the magnetic axis) and at low beta ($\beta_0 < 1\%$). At intermediate values of beta, the FAR approach shows that the most unstable mode is the internal kink which is excluded by the large-aspect-ratio formalism. These calculations have now been extended to full nonlinear evolution. Under investigation are the nonlinear evolution above the internal kink threshold and also the low-beta reconnection behavior responsible for the "sawtoothing" phenomena in tokamaks. For the latter, it has been found in cylindrical calculations that the nonlinear tearing mode responsible for sawtoothing at larger aspect ratio saturates at small aspect ratio. Toroidal studies of this behavior are under way. Results of these studies will be presented.

3.1.2.5 "Numerical Calculations Using the Full MHD Equations in Toroidal Geometry"¹⁷

L. A. Charlton, J. A. Holmes, H. R. Hicks, V. E. Lynch, and B. A. Carreras

A computer code has been constructed that solves the full MHD equations in toroidal geometry. The code is applicable to toroidal devices, including tokamaks, stellarators, and RFPs. A fully implicit numerical technique is used that allows linear eigenvalues and eigenfunctions to be found in a very few computational steps. Although the present work describes the solution of the linearized equations, generalization of the numerical method to the solution of the nonlinear problem is straightforward. Use of the code is illustrated by calculating the $n = 1$ instability for a tokamak configuration. The results show the structural changes in the eigenfunctions as the plasma pressure is increased.

3.1.2.6 "Ballooning Stability Boundaries with Plasma Compression"¹⁸

W. A. Cooper, D. J. Sigmar, and S. P. Hirshman

The resistive compressible ballooning growth-rate-level curves and the ideal compressible ballooning stability boundary are determined in the parameter space of shear vs pressure gradient for realistic tokamak equilibria.

3.1.2.7 "Numerical Studies of Resistive Pressure Driven Turbulence"¹⁹

L. Garcia, B. A. Carreras, and P. H. Diamond

Resistive-pressure-driven turbulence has been proposed as an explanation of energy confinement degradation at the edge of the Heliotron-E stellarator. Here, nonlinear, multiple-helicity calculations using a resistive interchange model based on averaged equations obtained from the stellarator expansion are presented. In order to incorporate the effect of heating, the average pressure gradient is held constant during the calculations. Both electrostatic and electromagnetic models have been investigated.

The saturation level, fluctuation spectra, and the relative importance of various nonlinear mechanisms will be discussed. Comparison with the results of analytical calculations will be presented.

3.1.2.8 "Nonlinear Multiple-Helicity Calculations of Impurity-Gradient-Driven Turbulence"²⁰

L. Garcia, B. A. Carreras, and V. E. Lynch

The resistivity-gradient-driven turbulence is a good candidate to explain tokamak edge fluctuations for moderate values of parallel electron thermal conductivity. This model can be extended to higher values of electron thermal conductivity when the effect of impurities is included. Rutherford has shown that impurity-driven rippling modes could play an important role in the explanation of anomalous cross-field thermal transport, since they are not stabilized to any great extent by parallel electron thermal conduction.

A model including resistivity as well as Z_{eff} evolution has been developed. For normal tokamak profiles, the resistivity gradient is positive. When the Z_{eff} effects are included, it is clear that Z_{eff} profiles with positive gradient will increase the destabilization while negative gradients will have decreased the effect. That is translated not only with modification of linear growth rates, but also with change in the nonlinear saturation levels of the fluctuations and diffusion coefficient.

We have studied numerically two different cases using Macrotor parameters, one with the gradient length scale of Z_{eff} $L_Z = 0.40a$ and the other one with $L_Z = -0.65a$. The gradient length scale of the Spitzer resistivity is $0.32a$ in both cases. The mean square radial velocity at saturation changes (in normalized units) from 900 to 100, while it was 400 for the case with constant Z_{eff} profile. The results of these calculations confirm the basic turbulent saturation mechanism proposed in our previous work and agree with the analytical estimates based on such dynamical mechanism.

3.1.2.9 "Theory of Resistivity-Gradient-Driven Turbulence"²¹

L. Garcia, P. H. Diamond, B. A. Carreras, and J. D. Callen

A theory of the nonlinear evolution and saturation of resistivity-driven turbulence, which evolves from linear rippling instabilities, is presented. The nonlinear saturation mechanism is identified both analytically and numerically. Saturation occurs when the turbulent diffusion of the resistivity is large enough so that dissipation due to parallel electron thermal conduction balances the nonlinearly modified resistivity gradient driving term. The levels of potential, resistivity, and density fluctuations at saturation are calculated. A combination of computational modeling and analytic treatment is used in this investigation.

3.1.2.10 "Resistive MHD Stability of Low Aspect Ratio Tokamaks"²²

T. C. Hender, J. A. Holmes, B. A. Carreras, and L. A. Charlton

The linear and nonlinear resistive MHD stability of low-aspect-ratio tokamaks is studied using the FAR code. Extensive scans of linear stability for $m = 2$ and $m = 3$ modes have been made. For aspect ratio scans in which the q profile is maintained constant, it is found that decreasing the aspect ratio has a marked stabilizing effect, while for scans in which the current profile is held constant, decreasing the aspect ratio has only a modest stabilizing effect. Nonlinear calculations will be presented to show the effects of aspect ratio on nonlinear stability and disruptions. The effects on linear stability of elliptic and triangular shaping have also been studied. It is found that shaping has only a modest effect on stability. Results on the resistive stability of equilibria optimized to give the maximum stable beta to external kinks and ballooning modes will also be presented.

3.1.2.11 "The Reliability of Calculations with the RSF Algorithm"²³

H. R. Hicks, B. A. Carreras, and J. A. Holmes

Eastwood and Hopcraft have investigated the numerical stability properties of the initial-value scheme used in our RSF code. Their analysis relates numerical instabilities to physical conditions such as the magnitude and orientation of flow velocity. Such approximate numerical stability analyses do not, however, provide conclusive and general stability rules, particularly for a nonlinear scheme. Our approach is to take a more empirical point of view. An adaptive timestep control routine analyzes certain amplitudes for very short wavelength components. When these are observed to grow, the timestep is automatically reduced. In practice, this avoids all but the fastest-growing instabilities. This approach has the additional virtue of arriving at a timestep that is near the maximum feasible value, thus providing the greatest efficiency for this scheme.

We use two techniques to verify the numerical stability of the results obtained using this successive timestep algorithm. The first is to compare the results of up to four different numerical schemes. The second is to rerun a calculation with the same scheme but with a different timestep. These tests have been applied systematically in all the physics

applications we have made, thus excluding any contamination of results due to numerical instabilities.

We address the questions raised by Eastwood and Hopcraft. While we agree with many of their conclusions, we have some significantly different interpretations. We concur with them on numerous points, including the desirability of a better understanding of these numerical schemes with the goal of improved domains of numerical stability. We agree that schemes exist which in certain situations are computationally more efficient than RSF. In fact, we have developed and used such schemes since the RSF algorithm was first developed in 1978.

3.1.2.12 "External Inductance of an Axisymmetric Plasma"²⁴

S. P. Hirshman and G. H. Neilson*

The external inductance of an axisymmetric toroidal plasma with an arbitrary aspect ratio and cross section is obtained using a Green's function method. By varying an equivalent skin current density over the plasma surface, while keeping the total toroidal current fixed, the plasma boundary is made to coincide with a magnetic surface. Numerical computations of the self-inductance and mutual inductance as functions of aspect ratio and elongation are fitted to simple analytic formulas. The effect of distributed plasma current on the volt-seconds required to reach a prescribed net current is considered.

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3.1.2.13 "Reversed Field Pinch Calculations Using a Reduced Set of Equations"²⁵

J. A. Holmes and B. A. Carreras

The study of the turbulent dynamo effect in the RFP configuration using the resistive MHD model involves the excitation and interaction of a large number of plasma modes. Assuming a full set of MHD equations and realistic parameters for present RFP devices, it is found that numerical calculations exceed the available resources of the present computers in terms of both storage and time requirements. In order to facilitate the study of resistive MHD turbulence in the RFP configuration, we have adopted a reduced set of equations due to Strauss, which are derived in the limits of low-beta, force-free plasmas with the dynamical fluctuations ordered to be small and a slow time scale ordering which eliminates the fast compressional Alfvén waves. These equations require the time advancement of only two dynamical quantities, a magnetic flux function and a vorticity component. This reduction leads to increased computer speed and decreased storage compared with the full equations.

Computational results using these equations are compared with those of previous calculations, for both the RFP and tokamak configurations, using a full set of MHD equations. In particular, we observe the nonlinear mode stabilization mechanism due to the dynamical couplings which was reported previously. In addition, we present some initial results in the use of these equations to study resistive MHD turbulence.

3.1.2.14 "Small Aspect Ratio Tokamak Equilibrium and Stability"²⁶

J. A. Holmes, B. A. Carreras, L. A. Charlton, R. A. Dory, and T. C. Hender

One of the primary factors motivating the study of small-aspect-ratio tokamaks is the prospect of achieving high-beta MHD stable equilibria which is suggested by the extrapolation of empirical scaling formulae. Troyon's expression for the maximum beta equilibrium ideally stable to long-wavelength free boundary modes, ballooning modes, and Mercier criterion on axis was derived from the results of theoretical studies of the Joint European Torus (JET) and the International Tokamak Reactor (INTOR). Experimentally, the highest beta values for several tokamaks have been found to be near the limit given by this scaling law, but neither experimental nor theoretical application has been made at small aspect ratios ($A < 2$).

In this work the critical beta scaling is extended to small aspect ratio ($A \approx 1.5$) through numerical calculations of equilibrium and stability to free boundary, long-wavelength ($n = 1-3$) ideal modes as well as ideal ballooning modes. Systematic studies of these stability criteria as functions of plasma cross-sectional shape and equilibrium profile variations reveal the existence of high-beta ($\langle \beta \rangle \approx 20\%$), MHD stable tokamak equilibria having conventional cross-sectional shape.

3.1.2.15 "Resistive Tearing Modes in a Poloidal Divertor Tokamak"²⁷

E. Uchimoto,* J. D. Callen, and L. Garcia

A new 3-D resistive MHD initial value code is being developed to study the linear and nonlinear evolution of resistive tearing modes in a poloidal divertor tokamak configuration including the plasma outside the divertor separatrix. This work was prompted by unusual features observed in Wisconsin Tokapole II discharges, such as attainment of a low q value (≈ 0.5) and the absence of current-terminating major disruptions. The code is spectral in the toroidal direction and finite-differenced in the two Cartesian directions of a constant toroidal angle plane. Linear simulations of the $m = 2, n = 1$ tearing mode in a divertorless tokamak are shown to be consistent with the standard analytic theory. Nonlinear simulations demonstrate the saturation of the 2/1 island. Both linear and nonlinear results and their implications for the Tokapole II will be presented. In addition, a possibility of accelerating the convergence to the nonlinear saturated state by using a steepest descent method will be considered.

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3.1.2.16 "Investigation of Saturated Tearing Modes in Tokamaks Using the Resistive Chodura-Schlüter Method"²⁸

E. Uchimoto,* J. D. Callen, D. A. Skinner,* L. Garcia, and B. A. Carreras

Investigation is under way to find saturated magnetic islands in a tokamak. Our primary objective is to find such a state in a poloidal-divertor-configuration tokamak (e.g.,

the Wisconsin Tokapole II) to assess the effect of the divertor separatrix and noncircularity. To this end, a new 3-D computer code has been written to numerically advance the resistive MHF (magneto-hydro-friction) equations. This code differs from the standard Chodura-Schlüter code in that a small resistivity is introduced to allow rapid magnetic field tearing and in that all the dependent variables are Fourier-analyzed in the toroidal direction to facilitate an efficient mode representation. In order to handle the divertor separatrix, however, a finite difference representation is used in two Cartesian components on a poloidal slice. The code is currently being tested in a simpler geometry without the divertor rings.

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3.1.3 EBS and EBT Devices—Abstracts

3.1.3.1 "First-Order Displacement of a Magnetic Field Line in an ELMO Bumpy Torus"²⁹

D. K. Lee, C. L. Hedrick, and J. D. Callen

A perturbation procedure is employed to derive a formula for the field-line displacement in a magnetic configuration consisting of a dominant (unperturbed) field with closed field lines and a small additional (perturbation) field that causes the perturbed field lines to fail to close upon themselves. An expression for the spatial displacement of the perturbed field line per circuit, obtained in a first-order perturbation approximation in the Clebsch representation, is given in terms of an action integral representing a longitudinal adiabatic invariant. Numerical results are also presented to compare the first-order approximation with the exact relation.

3.1.3.2 Charge Uncovering Effects on Flute Instabilities in Hot Electron Plasmas³⁰

D. A. Spong

Recent measurements and concurrent theoretical equilibrium models of the ELMO Bumpy Torus (EBT) edge plasma region [as described by E. F. Jaeger et al. in *Magnetic Well Depth in EBT and Sensitivity to Hot Electron Ring Geometry*, ORNL/TM-9185 (1984)] have indicated that the hot electron ring beta β_{hot} at the C-T transition may not always be sufficient to produce the local minimum in the magnetic field thought to be necessary for MHD stability. This has led to the examination of other mechanisms that could account for the observed stability of the T-mode. In this report, an effect known as charge uncovering, which depends not on the value of β_{hot} but rather on the ratio $n_{\text{hot}}/n_{\text{core}}$, is studied.

3.2 TRANSPORT KINETICS: KINETIC THEORY

The transport kinetics effort integrates topics that cross the boundaries of the traditional subcategories of theoretical plasma physics: equilibrium, orbits, stability, heating, and transport. A major focus is the interaction of transport and heating—particularly formation and control of the electrostatic potential. An immediate goal is to provide theories and codes that permit direct comparison with near-term experiments; however, effort is also directed toward providing the background and insights necessary for understanding and predicting the behavior of longer-term experiments.

The self-consistent electric field plays an essential role in plasma confinement for nonaxisymmetric devices, and analysis of its role was extended in several areas. First, a new study of the electric field scaling predicted by neoclassical and nonclassical processes was undertaken. For plasma confinement geometries with large aspect ratios and symmetric magnetic fields satisfying $B(\theta, \zeta) = B(-\theta, -\zeta)$ (which includes many stellarators and heliacs), it was shown that the time-reversal symmetry of the collisionless orbits is sufficient to ensure that the local diffusion coefficient is independent of the sign of the radial electric field. The consequences of this property for neutral beam injection and for generic root-jumping models for stellarators were also considered.

The effects of asymmetric electric potentials on transport in nonsymmetric devices were analyzed further. The analytic expression for the poloidal asymmetry of the electric potential in EBT was extended to apply to the ELMO Bumpy Square (EBS). The theory employs a three-fluid model for the electrons (cool, warm, hot) with the warm component assumed to dominate potential formation (as suggested by EBT experimental data). Additionally, a code was developed to calculate the self-consistent, poloidally varying electric potential in the low-collision-frequency regime in a nonsymmetric, toroidally confined plasma.

A numerical code, DKES, has been developed for computing transport coefficients in nonsymmetric, 3-D plasma devices. Starting from a variational principle derived from the drift Fokker-Planck equation, accurate numerical solutions for the thermodynamic fluxes as functions of the collision frequency and the radial electric field may be computed for arbitrarily complicated magnetic field geometries. Effects resulting from the variation of the longitudinal adiabatic invariant along an orbit, due to particle transitions from helically trapped to toroidally trapped orbits, are included in the analysis. Such transitional particle dynamics are difficult to include correctly in analytic calculations, and yet they may account for a sizable portion of the transport in configurations with a finite number of field periods N . Benchmarking of the code has been partially completed, using both 2-D analytic results and 3-D Fokker-Planck results. Numerical evaluations of the fluxes and viscosity coefficients for arbitrary-aspect-ratio tokamaks in the banana regime are in excellent agreement with the 2-D analytic predictions. The 3-D Fokker-Planck results were obtained by Hitchon and Mynick for a model magnetic field strength $B = B_0[1 - \epsilon_t \cos \theta - \epsilon_b \cos(2\theta - N\zeta)]$, where $N \rightarrow \infty$.

Some new and interesting results for the transport coefficients have already been obtained as N is varied parametrically in the Hitchon-Mynick model field. Helical resonances satisfying $|m - N/\ell| \ll m$ (which for two poloidal modes, $m = 2$, requires

$N/\iota \approx 2$) are found to measurably enhance the transport fluxes in the plateau and Pfirsch-Schlüter regimes. Good agreement between the code and analytic results is obtained. The scaling of the transport coefficients with the radial electric field is found to depend on N . In the low-collision-frequency (superbanana) regime, the diffusion coefficient $D \sim (\nu_h \Omega_E)^p$ where $p = N/(N + 1)$. Thus, the variation from $p = 1$, for $N \rightarrow \infty$, to $p = 0.5$, for $N = 1$, spans the range of values expected from pure collisionless detrapping effects to collisional boundary layer effects.

A code called MAGC1 has been developed to efficiently compute the Fourier spectrum of an arbitrary function of $|\mathbf{B}|$, $f(|\mathbf{B}|)$, associated with a real, finite-pressure, 3-D equilibrium consisting of nested magnetic surfaces. By coupling MAGC1 with DKES, the transport analysis of realistic finite-beta plasma configurations (in contrast to model fields) is now feasible.

The fluid moment method has been used to extend the range of validity of several analytic calculations for transport coefficients in nonsymmetric systems. Neoclassical fluxes in the plateau regime for nonsymmetric toroidal plasmas have been calculated, retaining the full harmonic decomposition for the magnetic field. It is found that the helical modulation of $|\mathbf{B}|$ can contribute significantly to the fluxes when the resonance condition $N/\iota = m$ is satisfied for transiting particles. Here, N and m are toroidal and poloidal mode numbers appearing in the $|\mathbf{B}|$ spectrum. This resonance is also observed with the 3-D Fokker-Planck code, DKES. The analytically calculated enhancement over the $N \rightarrow \infty$ results is in close agreement with the numerical results. A tractable analytic form for the circulating particle distribution in the banana regime for nonsymmetric toroidal plasmas has been calculated. The parallel viscosity-driven fluxes, including the Ware flux and the bootstrap current, and the banana regime plasma resistivity can be computed from this distribution function. A code has been written to calculate the bootstrap current for an ATF stellarator configuration. A modification of the fluid moment method has proved useful in identifying the physical mechanism for inducing currents by rf waves. The fluid theory indicates that rf currents are generated by the absorption of both momentum and energy flux from the waves. The analytic results obtained are in good agreement with numerical results from Fokker-Planck codes.

Monte Carlo simulations have continued to be employed to estimate energy and particle confinement times in complex magnetic configurations and to benchmark emerging Fokker-Planck transport codes. Bounce-averaged Monte Carlo simulations for cool electrons and ions in EBT and EBS have been carried out. When poloidally asymmetric electrostatic potentials of the type observed in EBT are incorporated into the model, cool electron and ion lifetimes comparable to those observed experimentally (a few tenths of a millisecond) are obtained. Similar calculations carried out for EBS show increases of two orders of magnitude in the lifetimes. These increases are due to the improved magnetics of EBS, both directly and indirectly through the resulting decreased poloidal asymmetry in the electric potential.

Application of the fluid moment method to derivation and analysis of resistive MHD-like equations for tokamaks continued. New pressure-gradient-driven resistive instabilities in the banana-plateau and banana regimes were discovered and studied. Finally, in collaboration with colleagues from other institutions, further kinetic theory and turbulence theory calculations on tearing modes and kinetic Alfvén waves were made.

3.2.1 Ambipolar Potential—Abstracts

3.2.1.1 "Bifurcation Phenomena and the Radial Electric Field in a Nonaxisymmetric Plasma"³¹

D. E. Hastings

The radial electric field in a nonaxisymmetric plasma is known to be a multivalued function of the plasma state. It is shown that the theorem of minimum entropy production rate cannot be used to distinguish between different electric field solutions. The possibility of bifurcation phenomena in the coupled electric field, density, and temperature equations is examined. It is shown that the functional dependence of the plasma source can strongly influence the type of bifurcation that can occur.

3.2.1.2 "Additive Noise and the Radial Electric Field in Systems with Convective and Diffusive Losses"³²

D. E. Hastings and S. P. Auerbach*

The charge balance equation in systems with convective and diffusive losses is a nonlinear differential equation that is known to possess multiple solutions. A functional of the potential, which can be identified as akin to a generalized entropy production rate, is defined. It is shown that the addition of a small amount of noise in the system will pick the solution that globally minimizes this functional.

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3.2.1.3 "The Poloidal Potential in Low-Collisionality Regime in a Nonaxisymmetric Torus"³³

D. E. Hastings and J. S. Tolliver

The poloidal potential is calculated numerically in the low-collisionality regime for nonaxisymmetric tori such as stellarators and bumpy tori. It is found that even fairly deep into the superbanana regime, the poloidal potential retains the simple azimuthal dependence of the plateau regime.

3.2.1.4 "Ambipolarons: Solitary Wave Solutions for the Radial Electric Field in a Plasma"³⁴

D. E. Hastings, R. D. Hazeltine,* and P. J. Morrison*

The ambipolar radial electric field in a nonaxisymmetric plasma can be described by a nonlinear diffusion equation. This equation is shown to possess solitary wave solutions. A model nonlinear diffusion equation with a cubic nonlinearity is studied. An explicit analytic form for the solitary wave is found. It is shown that the solitary wave solutions are

linearly stable against all but translational perturbations. Collisions of these solitary waves are studied and three possible final states are found: two diverging waves, two stationary waves, or two converging waves leading to annihilation of the solitary waves.

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3.2.1.5 "Anisotropy of the Electrostatic Potential in EBT and EBS"³⁵

C. L. Hedrick

Here we develop an analytic expression for the experimentally observed poloidal asymmetry of the electrostatic potential, ϕ , in EBT and apply this expression to EBS. The theory is motivated by the experimental observations that: (1) there are at least three disparate populations of electrons in EBT (bulk electrons, "warm" electrons, and the hot electron rings) and (2) ϕ scales with the "warm" temperature. A multiple-fluid theory gives a hierarchy of equations—the lowest order equations being those of magnetic equilibria and radial transport. The new feature is the poloidal variation of macroscopic quantities—particularly those associated with the warm electrons. The poloidal variation in ϕ can be obtained from the warm electron transport equations provided one uses the dominant behavior of the other "fluids" to relate the asymmetry in ϕ to the asymmetry in the warm density via quasi-neutrality.

If one takes the isotropic limit for the warm electrons, one finds that the asymmetry in ϕ is proportional to the asymmetry in $\oint d\ell/B$. In EBT, the warm group was observed to be anisotropic. However, applying the more complicated anisotropic expression shows that in EBT the asymmetry in ϕ is relatively independent of anisotropy (because toroidal effects are uniformly distributed) and either expression is in accord with experimental observation.

For EBS the toroidal effects are concentrated in the corners and anisotropy is important. For EBS, the expression predicts a two order of magnitude reduction in the asymmetry of the potential—one order of magnitude occurs because of overall improvement of particle orbits ($\oint d\ell/B$) and the other order of magnitude occurs because trapped particle orbits are perfectly symmetric. This reduction in ϕ asymmetry is important because, as discussed in Sects. 3.2.2.9 and 3.2.2.10, it virtually eliminates convective losses for the bulk electrons, which can be an order of magnitude greater than diffusion/conduction losses.

3.2.1.6 "Consequences of Time Reversal Symmetry for the Electric Field Scaling of Transport in Stellarators"³⁶

S. P. Hirshman, K. C. Shaing, and W. I. van Rij

Most three-dimensional fusion plasma confinement geometries possess symmetric magnetic fields satisfying $B(\theta, \zeta) = B(-\theta, -\zeta)$, where θ and ζ are poloidal and toroidal flux coordinate angles, respectively. For such configurations with large aspect ratios, the time-reversal symmetry of the collisionless orbits is sufficient to prove that the local diffusion

coefficient, $D(E_r)$, is independent of the sign of the radial electric field, E_r . Consequences of this property for neutral beam injection and for generic root-jumping models for stellarators are discussed.

3.2.1.7 "The Poloidal Potential in the Low-Collisionality Regime in a Nonaxisymmetric Torus"³⁷

J. S. Tolliver and D. E. Hastings

The poloidal potential in the low-collisionality regime for nonaxisymmetric tori such as stellarators and bumpy tori is examined. Solutions in this regime are not accessible analytically because the orbits are very complicated and collisions are too weak to negate orbit details. We derive the appropriate kinetic equation and boundary conditions for the distribution function in the superbanana regime and present numerical solutions. Although in an axisymmetric tokamak the poloidal potential develops a complicated azimuthal dependence and decreases with collisionality, we show that, in a stellarator or bumpy torus, the poloidal potential retains the simple azimuthal dependence of the plateau regime even fairly deep into the superbanana regime and is nearly independent of collisionality. Thus the poloidal potential in a stellarator is as important in the banana regime as in the plateau regime.

3.2.2 Transport Coefficients—Abstracts

3.2.2.1 "Finite-Aspect-Ratio Tokamak Viscosities in the Deep Banana Regime"³⁸

E. C. Crume, Jr., K. C. Shaing, S. P. Hirshman, C. O. Beasley, Jr., and W. I. van Rij

Neoclassical parallel viscosities in the tokamak deep-banana regime have been calculated for arbitrary aspect ratios from numerical solutions of the linearized drift kinetic equation. These viscosities can be used to construct the cross-field neoclassical transport coefficients. The numerical solutions for the 2-D tokamak confinement geometry were obtained as a special case of a 3-D confinement geometry technique described by Hirshman et al. (see Sect. 3.2.2.2). The parallel viscosity results are compared with those given by Hirshman and Sigmar. Connection is also made with recent calculations of finite-aspect-ratio transport coefficients by Chang and Hinton and by Bolton and Ware.

3.2.2.2 "Numerical Computations of Transport Coefficients for Nonsymmetric Plasmas"³⁹

S. P. Hirshman (in collaboration with C. O. Beasley, Jr., E. C. Crume, Jr., C. Handy,* K. C. Shaing, and W. I. van Rij)

The linearized drift kinetic equation is solved numerically to obtain the Onsager transport matrix for a finite-beta plasma confined in a 3-D toroidal domain. Local transport coefficients relating the cross-field fluxes to the thermodynamic forces are computed as

continuous functions of the collision frequency. In particular, in the low-collision-frequency regime ($\nu < \omega_b$), the transport resulting from nonconservation of the longitudinal adiabatic invariant J (due to helical-trapped particle to toroidal-trapped particle transitions) is obtained. The boundary layer in velocity space resulting from these transition particles can be accurately treated using a velocity-space representation for the distribution function developed previously. The use of magnetic flux coordinates renders the present flux calculations valid for arbitrary plasma equilibria (with nonzero rotational transform), thus dispensing with the need for magnetic field models and small transform-per-field-period approximations characterizing analytic calculations. A transformation of the kinetic equation reduces the disparity in time scales associated with collisionless particle orbits and yields an accelerated numerical algorithm to obtain steady-state fluxes and viscosity coefficients. Significant improvements in precision and computational effort are found in comparison with Monte Carlo methods. The effect of a self-consistent radial electric field in the $\nu < \omega_D$ regime is studied. Applications to ATF and heliac configurations are given.

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3.2.2.3 "Plasma Transport Coefficients for Nonsymmetric Toroidal Confinement Systems"⁴⁰

S. P. Hirshman, K. C. Shaing, W. I. van Rij, C. O. Beasley, Jr., and E. C. Crume, Jr.

A variational principle is developed for computing accurate values of local plasma transport coefficients in nonsymmetric toroidal confinement configurations. Numerical solutions of the linearized drift Fokker-Planck equation are used to obtain the thermodynamic fluxes as functions of collision frequency and the radial electric field. Effects resulting from the variation of the longitudinal adiabatic invariant J along an orbit (due to particle transitions from helically trapped to toroidally trapped orbits) are treated. The velocity-space distribution resulting from trapped, circulating, and transition particle orbits is well represented by a Legendre polynomial expansion in the pitch angle coordinate. The computational effort is significantly reduced from that required with Monte Carlo methods through use of an efficient treatment of the disparity between the time scales of collisionless and collisional particle dynamics. Numerical computations for a stellarator configuration are presented.

3.2.2.4 "Neoclassical Transport Fluxes in the Plateau Regime in Nonaxisymmetric Toroidal Plasmas"⁴¹

K. C. Shaing, S. P. Hirshman, and J. D. Callen

The neoclassical fluxes in the plateau regime in an arbitrary large-aspect-ratio, non-axisymmetric toroidal system are calculated based on the moment equation approach. The results are applicable for any nonaxisymmetric system. Two specific examples, one for a stellarator and the other one for a rippled tokamak, are considered. It is found that for most stellarators, the helical magnetic field contribution to the particle and heat fluxes is

as important as the toroidicity contributions and should not be neglected. Further, the helical magnetic field contribution to the parallel viscosity is capable of reversing the directions of the bootstrap current and Ware pinch flux, while in a rippled tokamak, the neoclassical particle and heat fluxes are the same as those in an axisymmetric tokamak after the radial electric field and parallel flow velocity are determined without the external sources.

3.2.2.5 "Neoclassical Transport Theory in the Banana-Plateau Regime in Nonaxisymmetric Toroidal Plasmas"⁴²

K. C. Shaing, S. P. Hirshman, and J. D. Callen

Since current stellarators are usually operated in the banana-plateau regime and there are few problems which have not been clarified in the collisional ripple plateau regime in a rippled tokamak, the neoclassical transport theory in the banana-plateau regime in a nonaxisymmetric torus deserves further studies. We employ the moment equation approach to the problem without considering the external sources but including the heat flux effect. In contrast to the usual calculations in a stellarator, where an average over the helical ripple is performed so that a stellarator is approximated by an axisymmetric tokamak, the moment equation approach keeps the full nonaxisymmetric effects. This allows us to study the combined effects of the helical field and the toroidicity and of the radial electric field. Also, since the particle flux has been self-consistently separated according to the flux-friction relations, we clarify the "nonaxisymmetric" contribution to the particle flux in the collisional ripple plateau regime in a rippled tokamak. The particle flux in a rippled tokamak is similar to that of an axisymmetric tokamak after the radial electric field is determined. However, in a stellarator, the helical field also contributes to the particle flux. The details depend on the relative magnitude of ϵ_h , ϵ_t , and the ℓ and m numbers. A simplified form of the parallel viscosity in the banana regime will be presented. This allows us to calculate the bootstrap current *quantitatively* in any nonaxisymmetric system.

The dynamical behavior of the electric field in an arbitrary nonaxisymmetric system is also considered. In general, the electric field cannot be determined separately from the flow velocity. Two coupled time-dependent equations for any two of the four variables U_t , U_p , $U_{\parallel}$, and Φ' , where U_t , U_p , and $U_{\parallel}$ are toroidal, poloidal, and parallel flow velocities, must be solved to determine the radial electric field Φ' .

3.2.2.6 "Reply to Comments of Mynick and Hitchon"⁴³

K. C. Shaing, J. A. Rome, and R. H. Fowler

There appears to be an inconsistency between the time-scale ordering assumptions made in our work and those of Mynick and Hitchon. Even if we accept their definition of τ_t and the summary of their results as stated in their comment, the theory of Mynick and Hitchon can be compared conveniently with our results only in the two limiting cases ($p = 0$ and $p = 1$) of our parameter p , the retrapping probability for the toroidally trapped particles. Goldston and Towner have calculated a parameter equivalent to p for two cases. One depends on the collisional boundary layer effect, and the other depends on the finite

orbit width and the magnetic configuration. Mynick and Hitchon employed only the latter model in their calculations. We suspect that an interpretation like ours will prove to be necessary for $p \neq 0$, as is most likely to happen in a real situation.

We further note that the transport processes discussed in our letter come from the ripple-trapped (or helically trapped) portion of the particle orbits, independent of the value of p ; in contrast, Mynick and Hitchon state that for $p = 0$ the transport is caused by the ripple-trapped part of the orbits, while for $p = 1$ it is caused by the toroidally trapped part.

Finally, the functional dependence of the contribution of the ripple-trapped particles to the diffusion coefficients is determined by the ∇B drifts and ripple well depths; it should be the same for particles trapped at $\theta = 0$ and at $\theta = \pi$, even if the well depths are different, since the ∇B drift is proportional to $\sin \theta$. We therefore find it difficult to understand physically how the scalings of the ripple transport can be different at these two points. The scaling we give should be valid at both $\theta = 0$ and $\theta = \pi$.

3.2.2.7 "Viscous Effects in a Collisional Tokamak Plasma with Strong Rotation"⁴⁴

W. M. Stacey, Jr., and D. J. Sigmar

The full viscosity tensor for an axisymmetric toroidal plasma in the collisional regime (with strong rotation) is calculated, including gyroviscosity and $O(\epsilon)$ poloidal variations over the flux surface. It is shown that the resulting viscous force is of sufficient magnitude to account for the radial transfer of toroidal momentum that must be inferred in order to explain the rotation measurements in tokamak experiments. The consequences of a viscous force of this form and magnitude on particle transport and on the evolution of toroidal and poloidal rotation velocities are discussed.

3.2.2.8 "Rotation and Impurity Transport in a Tokamak Plasma with Directed Neutral-Beam Injection"⁴⁵

W. M. Stacey, Jr., A. W. Bailey,* D. J. Sigmar, and K. C. Shaing

The authors have extended their previous collisional-regime theory for rotation and impurity transport in a tokamak plasma with strong, directed neutral beam injection and strong rotation ($v_\phi \approx v_{th}$) plateau regime. The paper gives a summary of a kinetic theory derivation of the parallel viscous force in the strong rotation ordering and a self-consistent formalism for calculating ion and impurity rotation velocities and radial transport fluxes, as well as the radial electric field and the poloidal variation of the impurity density upon which the former strongly depend. Calculations for model problems representative of the Impurity Study Experiment (ISX-B) and the Princeton Large Torus (PLT) are presented. The predicted impurity transport exhibits features that are in agreement with experimental observations.

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3.2.2.9 "Bounce-Averaged Monte Carlo Energy and Pitch Angle Scattering Operators"⁴⁶

J. S. Tolliver

Beginning with a linearized Coulomb collision operator, the derivation of bounce-average energy and pitch angle scattering terms that have been suitably discretized for Monte Carlo use is described. The derivation assumes that the electrostatic potential is constant along a magnetic field line. The collision operators are applied to electron transport in EBT.

3.2.2.10 "Monte Carlo Calculations in Elmo Bumpy Square"⁴⁷

J. S. Tolliver and C. L. Hedrick

We have generalized the ORNL EBT bounce-averaged Monte Carlo code to work for the EBS geometry. The neoclassical diffusion coefficients in EBS are expected to be about an order of magnitude smaller than in EBT because of better centering of the collisionless orbits in EBS. Furthermore, the asymmetry in the electric field is expected to be much smaller in EBS than in EBT, thus nearly eliminating the convective flow of low-energy particles out of the confinement region along potential contour lines. We have used the Monte Carlo code to estimate particle lifetimes in EBS and compared these with estimated lifetimes in EBT. Because the EBS confinement is so much better than EBT, the Monte Carlo code requires large amounts of computer time in order to obtain good statistics on the loss rates. Preliminary indications are that the global confinement time of the cool population in an EBS with symmetric potential will be about two orders of magnitude better than in EBT with asymmetric potential, in keeping with the expected improvement based on analytic neoclassical diffusion coefficients.

3.2.3. Extensions to MHD—Abstracts

3.2.3.1 "Neoclassical MHD Equations for Tokamaks"⁴⁸

J. D. Callen and K. C. Shaing

The moment equation approach to neoclassical-type processes is used to derive the flows, currents, and resistive MHD-like equations for studying equilibria and instabilities in axisymmetric tokamak plasmas. The resultant "neoclassical MHD" equations differ from the usual reduced equations of resistive MHD primarily by the addition of the important (for $\nu \sim 1$) viscous relaxation effects within a magnetic flux surface. The primary effects of the parallel, poloidal viscous relaxation are: (1) rapid ($\sim \nu_{ii}$) damping of the poloidal ion flow; (2) addition of the bootstrap current contribution to Ohm's law; and (3) an enhanced (by B^2/B_θ^2) polarization drift-type term due to parallel flow inertia, which causes the equations to depend only on the poloidal magnetic field B_θ . Gyroviscosity (or diamagnetic viscosity) effects are included to properly treat the diamagnetic flow effects. A nonlinear version of the neoclassical MHD equations has been derived and shown to satisfy an energy conservation equation with dissipation arising from Joule heating, bootstrap current effects, classical and neoclassical diffusion, and viscous heating.

3.2.3.2 "Neoclassical MHD Instabilities and Transport in Tokamaks"⁴⁹

J. D. Callen and K. C. Shaing

The moment equation approach to neoclassical-type processes is used to derive the perturbed flows, currents, and resistive MHD-like equations for studying instabilities in tokamak plasmas. These "neoclassical MHD" equations are valid for arbitrary collisionality ν_e and low- to medium-mode-number modes for which the electrons are fluidlike. The main differences from the regular resistive MHD equations are: (1) the effective polarization current has ρ_m/B^2 replaced by ρ_m/B_θ^2 , where B_θ is the poloidal magnetic field; (2) Pfirsch-Schlüter currents cancel the geodesic curvature terms; (3) the flux-surface-averaged Ohm's law contains a (fluctuating) bootstrap current contribution due to the parallel viscous damping [at rate $\mu_e \sim \sqrt{\epsilon} \nu_e (1 + \nu_{*e})$] of pressure-gradient-driven poloidal flows. Possible "neoclassical MHD" instabilities include pressure-gradient-driven, tearing, and rippling type modes. The pressure-gradient-driven modes have growth rates

$$\gamma_\mu \approx n^{2/3} S_\theta^{1/3} \beta_\theta^{2/3} \times (\mu_e/\nu_e)^{1/3} / \tau_{A\theta} = [4(m_e/m_i) \mu_e \omega_e^2]^{1/3}$$

in an $\mathbf{E} \times \mathbf{B}$ toroidal rest frame ($\omega = \omega_E + i\gamma_\mu$) and radial extent

$$\delta_\mu \sim n^{-1/3} S_\theta^{-1/3} [\beta_\theta (\mu_e/\nu_e)^2]^{1/6}.$$

For $\gamma_\mu < \omega_{*e}$, diamagnetic drift frequency effects reduce the growth rate and radial extent by factors of $(\gamma_\mu/\omega_{*e})^2$ and $(\gamma_\mu/\omega_{*e})^{1/2}$, respectively.

To estimate the anomalous transport induced by the pressure-gradient-driven modes, we follow the procedures developed by Carreras and Diamond for resistive ballooning instabilities. We find for the anomalous electron heat diffusivity

$$\chi_e \sim n_M^{1/2} (n/\mu_0) (\mu_e \beta_\theta / \nu_e)^2 \sqrt{m_i/m_e} / (1 + n_M^{2/3} \omega_{*e}^2 / \gamma_\mu^2)^{3/4},$$

where n_M is a maximum mode number for which the electrons are fluidlike. In the outer confinement region the ω_{*e} correction becomes negligible and $\mu_e/\nu_e \sim \sqrt{\epsilon}/\nu_{*e}$. Then, utilizing an ohmic power balance to determine the scaling of T_e , we find $\beta_\theta \sim (m_e/m_i)^{1/6} \sim 0.3$ and $\tau_E \sim n_e^{13/14} a^{4/7} R_0^{13/7} g^{5/7} B_p^{1/7}$, which is close to the TFTR empirical scaling law $\tau_E \sim n_e a R_0^2 q^{0.8}$. For $\nu_e < 1$ the estimated χ_e is similar to but slightly (by $1/\sqrt{\epsilon}$) larger than the Carreras-Diamond result. We will also discuss the anomalous particle transport, toroidal momentum damping, and possible modifications of the neoclassical pinch and bootstrap current effects due to neoclassical MHD turbulence in tokamak plasmas.

3.2.3.3 "A Pressure-Gradient-Driven Tokamak 'Resistive Magnetohydrodynamic' Instability in the Banana-Plateau Collisionality Regime"⁵⁰

J. D. Callen and K. C. Shaing

The moment equation approach to neoclassical processes is used to derive the linearized electrostatic perturbed flows, currents, and resistive MHD-like equations for a tokamak plasma. The new features of the resultant "neoclassical magnetohydrodynamics," which requires a multiple-length-scale analysis for the parallel eigenfunction but is valid in the experimentally relevant banana-plateau regime of collisionality, are (1) a global Ohm's law that includes a fluctuating bootstrap current resulting from "parallel" electron viscous damping (at rate μ_e) of the poloidal flow due to the perturbed radial pressure gradient; (2) reduction of the curvature effects to their flux surface average because Pfirsch-Schlüter currents cancel out the lowest-order geodesic curvature effects; (3) an increased polarization drift contribution with B^{-2} replaced by B_θ^{-2} , where B_θ is the poloidal magnetic field component. An electrostatic eigenmode equation is determined from $\nabla \cdot \mathbf{J} = 0$. For the unstable fluid-like eigenmodes, the new viscous damping effects dominate (by $\epsilon^{-3/2}$) over the curvature effects, but the growth rates still scale roughly like resistive- g or resistive-ballooning modes, $\gamma_{\mu\tau_A} \sim n^{2/3} S_N^{1/3} \beta_T^{2/3} (\mu_e/\nu_e)^{1/3}$. Diamagnetic drift frequency corrections to these new modes are also discussed.

3.2.3.4 "Kinetic Approach to a Pressure-Gradient-Driven Tokamak Resistive Instability in the Banana Regime"⁵¹

K. C. Shaing and J. D. Callen

A new resistive, low-frequency ($\omega \leq \omega_{ba} \sim \nu_a$, where ω is the frequency of the mode and ω_{ba} and ν_a are the bounce frequency and collision frequency for species a) instability is found in a tokamak geometry. The mode is driven unstable by the parallel viscous (bootstrap) current induced by the radial pressure gradient. It has a growth rate $\gamma \sim (\omega_e \nu_i)^{1/3}$ for $\nu_i \geq \omega_e$, where ω_e is the electron diamagnetic drift frequency and ν_i is the ion-ion collision frequency. The frequency dependence of the parallel viscosity is also derived and utilized to show that drift wave branches of these modes are stable, but that this new mode remains unstable when the diamagnetic drift effects are included.

3.2.4 Microstability and Turbulence—Abstracts

3.2.4.1 "Turbulent Stabilization of the Tearing Mode in Tokamak Plasmas"⁵²

E. Esarey,* J. P. Freidberg,* K. Molvig,* C. O. Beasley, Jr., and W. I. van Rij

The effect of turbulent electron diffusion from stochastic electron orbits on the stability of low-beta fluctuations is considered through the use of the normal stochastic approximation. A set of coupled, self-adjoint equations is derived for the fluctuation potentials Φ and A_1 . Solutions to this set of equations describe both unstable finite-beta drift waves

when analyzed for high- m modes and the tearing mode when analyzed for low- m modes. For the tearing mode, it is shown that stability is obtained for sufficiently large values of the diffusion coefficient. Provided $D_e \sim 1/n$, this implies that a density threshold must be surpassed before the tearing mode is observed. Physically, turbulent electron diffusion prohibits the formation of a perturbed current within a finite region about the rational surface. At higher densities, the inclusion of a finite electrostatic potential, Φ , gives an additional stabilizing term to the dispersion relation, which physically represents ion inertial effects. This ion inertial effect implies that, in the absence of diffusion, the tearing mode is stabilized for ion betas above some critical value.

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3.2.4.2 "The Effect of Runaway Electrons on the Tearing Mode"⁵³

E. Esarey,* K. Molvig,* C. O. Beasley, Jr., W. I. van Rij, and T. K. Chu†

The $m = 2$ tearing mode is analyzed using a kinetic theory. The analysis is carried out using two different models for the equilibrium plasma current. In the first model a drifted Maxwellian electron distribution describes the equilibrium current, while in the second model the current is described by an electron beam on the tail of a Maxwellian distribution. The second model represents a current composed primarily of runaway electrons. In both models the equilibrium current profile is held constant. It is shown, both analytically and numerically, that in certain parameter regimes the $m = 2$ tearing mode is stabilized when the runaway electrons carry the total current. In the presence of runaway electrons, the real frequency of the tearing mode is shifted from the electron drift frequency, ω_{*e} , to slightly above ω_{*e} . The shift in mode frequency above ω_{*e} indicates that more of the available energy must be utilized to sustain the plasma particle oscillations, which stabilizes the tearing mode. A comparison to experimental observations is also presented.

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3.2.4.3 "Spectral Properties of the Kinetic Alfvén Wave in Cylindrical Tokamak Geometry"⁵⁴

W. I. van Rij, G. Vahala,* and D. J. Sigmar

It is shown, for a tokamak-like cylindrical plasma and small poloidal and toroidal mode numbers, that the full gyrokinetic shear Alfvén spectrum is discrete when $k_{\parallel} \neq 0$. Calculations of the global eigenmodes of the MHD limit, and their first-order finite Larmor radius (FLR) corrections, are presented. First-order FLR theory is found to be a good approximation to full FLR theory, even for modes with high radial node number. In

the presence of a positive density gradient, such as that which may occur with pellet fueling of the plasma, the existence of global MHD eigenmodes above the Alfvén continuum is confirmed.

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3.3 RF HEATING AND CURRENT DRIVE

The rf heating and current drive effort is aimed at developing fundamental theory as well as computational modeling capability for plasma wave generation, propagation, and absorption and plasma response in all fusion-relevant frequency regimes (Alfvén, ion cyclotron, lower hybrid, and electron cyclotron). Application of the models to detailed comparisons with experimental measurements provides insight into the experiments and feedback for further refinements in the theory.

Over the past year, intensive effort has gone into full-wave ion cyclotron resonance heating (ICRH) calculations in various magnetic geometries. For this purpose, a number of 2-D finite difference codes were developed and applied. One code was developed to treat axially invariant plasmas and used to study wave propagation and absorption in straight tokamak geometry. The possibility of attaining pure fundamental ICRH was investigated. A study was made of the structure of resonance surfaces in a cold plasma model. Results from the code also were used to assist the Plasma Technology Section in the design of antennas for Doublet IIID and Tore Supra. Another 2-D finite difference code was developed to treat helically symmetric plasmas. This code is being used to study the possibility of pure fundamental ICRH in stellarators, to study the structure of cold plasma resonances in torsatron geometry, and to optimize antenna designs for ATF. A 2-D finite difference code also was developed to study axisymmetric mirror plasmas. Full-wave 1-D calculations were carried out for warm plasmas to study the details of mode conversion and cyclotron damping in slab geometry. These calculations represent an initial step toward including finite temperature effects in the 2-D codes. Also, Fourier transform methods in flux coordinates were employed to investigate ion cyclotron wave propagation in helically symmetric stellarator geometry.

Technical progress in the electron cyclotron range of frequencies over the past year has included further improvement to the RAYS geometrical code for application to non-axisymmetric plasmas such as ATF. Detailed ray tracing calculations of fundamental and second harmonic electron cyclotron heating (ECH) in ATF were made, including a preliminary study of the effect of wall reflections. Among the collaborative efforts with universities, we especially note the collaboration with New York University on investigations of strongly damped cyclotron waves in nearly parallel stratified plasmas, using a nonlocal kinetic, full-wave model, and the collaboration with the University of Wisconsin on studies of a coherent nonlinear mechanism for hot electron generation at the second harmonic.

The possibility of exploiting a single particle resonance as a means of influencing the poloidal variation of the electric field or for impurity removal in ATF (and perhaps in tokamaks) was identified. The mass dependence of the energy at which this resonance

occurs suggests that the process would be most effective for impurities. It might be possible to scatter particles either into or out of such a resonant zone with rf waves and thereby selectively influence the loss rate of certain species. Since this resonance occurs at low energy, low power requirements would be anticipated—a particularly attractive feature.

3.3.1 Abstract of "Strong Damping of Electron Cyclotron Waves in Nearly Parallel Stratified Plasmas"⁵⁵

D. B. Batchelor, A. Fruchtman,^{*} K. S. Riedel,^{*} and H. Weitzner

In mirror-like devices, where the magnetic gradient is along the direction of the magnetic field (parallel stratification), the right circularly polarized component of the E field is not suppressed. The warm plasma dispersion relation predicts essentially total absorption of the extraordinary mode. However, because of the strong damping and the rapid variation of k near resonance, geometric optics is not necessarily valid. Adapting a calculation done by Weitzner for ICRH, we introduce three small parameters, $\epsilon_1 = v_{th}/c$, $\epsilon_2 = c/\omega L$, $\epsilon_3 = (\omega_{pe}/\omega)^2$, and perform a boundary layer analysis in the resonance region. For many parameter values, we prove the extraordinary mode is completely absorbed. The integrodifferential equations are solved numerically.

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3.3.2 Abstract of "Electron Cyclotron Resonance Heating in Parallel Stratified Plasmas"⁵⁶

D. B. Batchelor, A. Fruchtman,^{*} and H. Weitzner

We study the interaction of electrons with electromagnetic waves when the equilibrium magnetic field is approximately parallel stratified near the cyclotron resonance surface. The Vlasov-Maxwell equations are solved in a boundary layer around the resonance surface since the cold plasma theory is invalid there. The form of the equations in the boundary layer depends on the relative magnitudes of some characteristic parameters such as the ratio of the free-space wavelength to the equilibrium scale length, the ratio of the thermal velocity to the velocity of light, and the ratio of the plasma frequency to the wave frequency. We concentrate on two cases. In one case, when the electron density is large in some sense, the waves are totally absorbed. In a second case, when the electron density is small, the Budden tunneling effect is present. The asymptotic solutions can be found analytically. However, in order to find the energy deposition profile, we have to solve numerically integrodifferential equations, which result from the fact that the currents are of a nonlocal form. Numerical and analytical results will be presented for both cases.

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3.3.3 Abstract of "Nonlinear Wave-Particle Interactions at Cyclotron Harmonics with Collisional Decorrelation"⁵⁷

M. D. Carter, D. B. Batchelor, and J. D. Callen

When particle orbits become trapped very near a cyclotron resonance, the quasi-linear model of weakly perturbed, uncorrelated passages through resonance breaks down and nonlinear effects become important if the wave-particle autocorrelation time is less than the collisional decorrelation time. Analytic formulae are derived describing the nonlinear behavior including relativistic detuning and perturbed bounce motion. The results include large energy excursions at bounce-resonant values of the turning point, and the connection to quasi-linear behavior is made. It is also demonstrated that perpendicularly propagating waves at the second harmonic electron cyclotron frequency can strongly interact with very cold (<1 -eV) electrons.

Coulomb collisional decorrelation has been modeled for electrons trapped near the second harmonic resonance using a Monte Carlo code. Significant enhancement in the heating over the quasi-linear model is found for certain ranges of collisionality and autocorrelation time.

3.3.4 Abstract of "Fokker-Planck Heating Coefficients Utilizing Nonlinear Wave-Particle Interactions at Cyclotron Harmonics"⁵⁸

M. D. Carter, J. D. Callen, and D. B. Batchelor

For electrons trapped in a magnetic mirror and subjected to a nearly monochromatic wave at a harmonic of the cyclotron frequency, nonlinear particle orbits become important when decorrelation timescales become longer than the timescale for nonlinear effects. The nonlinear orbit can be described by an effective potential and two constants of motion. The resulting periodic energy excursions may be decorrelated by Coulomb collisions, drifts through the nonlinear interaction region, low-frequency waves (near the excursion frequency $\sim$ MHz), and other mechanisms. Diffusive heating still occurs for these orbits but not in the manner of small kicks per passage through resonance as considered by quasi-linear theory. Fokker-Planck coefficients are generated as a function of pitch angle and energy by utilizing the nonlinear orbits and the above decorrelation mechanisms. Interactions of very cold electrons with ECH waves at the second cyclotron harmonic are also considered with regard to breakdown of neutral gases for plasma startup.

3.3.5 Abstract of "Collisional Effects on Coherent Nonlinear Wave-Particle Interactions at Cyclotron Harmonics"⁵⁹

M. D. Carter, J. D. Callen, D. B. Batchelor, and R. C. Goldfinger

When particle orbits are trapped very near a cyclotron harmonic resonance, the quasi-linear concept of weakly perturbed, uncorrelated passages through resonance breaks down, and nonlinear effects become important. In numerical as well as analytic studies, it

is demonstrated that relativistic detuning of the resonance can be important for electrons even at low initial energies (~ 20 eV) and that coupling to perturbed parallel motion can lead to strong interactions for values of the turning point where the wave frequency differs from a harmonic multiple of the bounce-averaged gyrofrequency by an integral multiple of the bounce frequency. The resultant motion is described by large periodic energy excursions for which small-angle Coulomb collisions or other randomization processes are required to realize net heating. Analytic formulas are derived describing the energy excursion behavior and heating in a mildly relativistic limit. Also, a Monte Carlo numerical model of the collisional effects on the orbits has been employed to study electron heating at the second harmonic cyclotron resonance and to test the analytic results. In certain regimes of collisionality, a strong enhancement over quasi-linear heating has been found.

3.3.6 Abstract of "ICRF Wave Equations in Stellarator Configuration"⁶⁰

W. A. Cooper

We reduce the wave propagation equation in the ICRH frequency regime to the evaluation of the $\nabla\rho$ and $\mathbf{B} \times \nabla\rho$ components in a (ρ, θ, ϕ) flux coordinate system because the parallel wave electric field is small. This electric field is expressed as $\mathbf{E} = V\nabla\rho + U\mathbf{B} \times \nabla\rho$. For a straight stellarator configuration, Fourier analysis in the ignorable coordinate ϕ is simple. Fourier analysis in the poloidal angle θ leads to a truncated set of ordinary differential equations (ODEs), with jump conditions due to an applied antenna current in the poloidal direction on a given flux surface. The boundary conditions are determined by invoking regularity at the origin and the vanishing of U at the conducting wall.

3.3.7 Abstract of "ICRH Wave Propagation Equations in Stellarator Configurations"⁶¹

W. A. Cooper

The ICRH wave propagation equations are expanded in a (ρ, θ, ϕ) flux coordinate system with the wave electric field $\mathbf{E}$ expressed as $\mathbf{E} = V\nabla \ln \rho + U(\mathbf{B} \times \nabla \ln \rho)$. Because the mod- B surfaces in a 3-D stellarator are hyperboloidal in shape, we model the device as a helically symmetric system (with ϕ the coordinate of symmetry). A reduced set of ICRH wave equations is derived by an expansion in $\epsilon = ah \ll 1$, where a is the minor radius and h is the helical pitch. The coefficients of these equations are simplified further by considering a model MHD equilibrium with concentric elliptical flux surfaces.

3.3.8 Abstract of "ICRH Wave Propagation Equations in Stellarator Geometry"⁶²

W. A. Cooper

The ICRH wave propagation equations are expressed in a (ρ, θ, ϕ) flux coordinate system, where ρ is the radial variable that identifies the flux surfaces and θ and ϕ are

angle-like variables of period 2π . In the ICRH frequency regime, the component of the wave electric field along the equilibrium magnetic field lines is small. This allows us to reduce the problem to the evaluation of the $\nabla\rho$ and $\mathbf{B} \times \nabla\rho$ components of the original vector equation. We express the wave electric field as $\mathbf{E} = V \nabla \ln \rho + U(\mathbf{B} \times \nabla \ln \rho)$, where we have neglected the small parallel electric field. It is convenient to define the vector $\mathbf{G} = i\omega\sqrt{g}\mathbf{B}_1 \times (\mathbf{B} \times \nabla\rho) + \sqrt{g}\mathbf{E} \times [\nabla \times (\mathbf{B} \times \nabla\rho)]$, where $\mathbf{B}_1$ is the wave magnetic field, ω is the wave frequency, and $\sqrt{g}$ is the Jacobian of the transformation from Cartesian to flux coordinates. The $\mathbf{B} \times \nabla\rho$ component of the wave equation yields an expression for the radial derivative of $W = \rho\mathbf{G} \cdot \nabla\rho$. An expansion of $\mathbf{G} \cdot \nabla\rho$ yields an equation for the radial derivative of U . The $\nabla\rho$ component of the ICRH wave equation yields an expression for V .

The mod- B surfaces in 3-D stellarator configurations are hyperbolas. We can model this geometry with a helically symmetric stellarator because such systems have similarly shaped mod- B surfaces. Fourier analysis in the ignorable coordinate ϕ then becomes trivial. At the magnetic axis, the radial derivative of U and W can be expressed as linear combinations of $E_0 \sin \alpha$ and $E_0 \cos \alpha$. E_0 is the wave electric field magnitude on axis and α is the angle between the midplane and the direction of this field as it crosses the axis. Fourier analysis in the poloidal angle yields a set of coupled ODEs from which we can construct an algorithm using finite differencing in ρ that employs an iterative shooting technique with E_0 as the complex eigenvalue. A converged solution is obtained when the resulting value of U vanishes at a conducting wall boundary.

3.3.9 Abstract of *Electron Cyclotron Heating Calculations in ATF*⁶³

R. C. Goldfinger and D. B. Batchelor

The RAYS geometrical optics code has been used to calculate electron cyclotron wave propagation and heating in the ATF device under construction at ORNL. The intent of this work is to predict the outcome of various heating scenarios and to give guidance in designing an optimum heating system. Particular attention is paid to the effects of wave polarization and antenna location. We investigate first and second harmonic cyclotron heating with the parameters predicted for steady-state ATF operation. We also simulate the effect of wall reflections by calculating a uniform, isotropic flux of power radiating from the wall. These results, combined with the first-pass calculations, give a qualitative picture of the heat deposition profiles. From these results we identify the compromises that represent the optimum heating strategies for the ATF model considered here. Our basic conclusions are that second harmonic heating with the extraordinary mode (X-mode) gives the best result, with fundamental ordinary mode (O-mode) being slightly less efficient. Assuming the antenna location is restricted to the low magnetic field side, the antenna should be placed at $\phi = 0^\circ$ (the toroidal angle where the helical coils are at the sides) for fundamental heating and at $\phi = 15^\circ$ (where the helical coils are at the top and bottom) for second harmonic heating. These recommendations come directly from the ray tracing results as well as from a theoretical identification of the relevant factors affecting the heating.

3.3.10 Abstract of "Propagation and Absorption of Electron Cyclotron Waves in ATF"⁶⁴

R. C. Goldfinger and D. B. Batchelor

The ray tracing code, RAYS, has been adapted to trace rays in the electron cyclotron frequency range for the ATF stellarator device to be built at ORNL. Several new features have been added to the code. These include calculation of radial heating profiles, $P(\psi)$ (ψ = toroidal magnetic flux), and flexibility to be used in general 3-D toroidal geometry for either analytically defined or numerically defined plasma equilibria. Rays are propagated in a beam simulating anticipated antenna design using the usual geometrical optics approximations with cold plasma theory. Wave absorption due to the fundamental electron cyclotron resonance is calculated along the trajectories. An approximation to a radial heating profile is derived for each beam by calculating a histogram of fractional power absorption as a function of toroidal magnetic flux.

The main objective in the study is to determine how to maximize the first-pass absorption of the beam and to have this heating occur as near as possible to the magnetic axis. It turns out that the complicated topology of the mod- B resonance surfaces is the primary factor in determining the wave absorption. To achieve optimal heating, the cyclotron resonance must be at the saddle point of the magnetic field and as close as possible to the magnetic axis. This requirement means that considerations such as the natural symmetry of the system, the separation of the saddle point and magnetic axis, and the gradient perpendicular to the ray trajectories all help point to a preferred launch location for any given equilibrium.

Only single-pass propagation of ordinary mode rays is calculated; neglecting wall reflections in this model is a reasonable approximation due to the strong absorption. Each ray tracing calculation takes a conical bundle of eight rays to simulate a beam which has a FWHM $\sim 3^\circ$. The plasma model and magnetic field can be defined either analytically or numerically. Some preliminary runs have been done using a 3-D cubic spline interpolation of numerical data, which is the output from finite-beta equilibrium codes. However, the majority of work to date has been with an analytically defined model. In this model the Biot-Savart law is integrated to give the vacuum field due to filamentary current sources representative of ATF. The plasma density is parametrized by the toroidal magnetic flux, ψ , obtained from an analytic approximation to computed flux surfaces for the vacuum case. The assumed plasma and wave parameters are those expected for typical ATF operation: $f_{\text{wave}} = 53.2$ GHz, $T_e = 500$ eV, $n_e = 10^{13}$ cm $^{-3}$, $B_0 = 1.9$ T.

The complicated topology of the stellarator geometry causes the ray tracing results to be highly sensitive to coil currents, antenna location, and finite-beta effects. For the specific case of ATF parameters, we show numerous examples of ray tracing results for various launch locations and magnetic field levels. We present curves of net absorption as a function of magnetic field level since the location of the cyclotron resonance is crucial and operation at various coil currents is anticipated. Also presented are plots of resonance surface and flux contours. Experience with these runs allows us to derive a qualitative picture of generic ECH in stellarators. An extended discussion of these qualitative effects is presented with the resultant implications for design of ECH systems.

3.3.11 Abstract of "Fast Wave Heating in Tokamak and Stellarator Magnetic Fields"⁶⁵

E. F. Jaeger, D. B. Batchelor, and H. Weitzner

It has been suggested that fast wave plasma heating at the fundamental cyclotron frequency of the bulk ions may be enhanced in stellarator geometry. We intend to address this question by solving Maxwell's equations for the ICRF wave fields in a helically symmetric stellarator magnetic field. Preliminary calculations for a straight tokamak with and without poloidal magnetic fields show no resonant heating at the fundamental cyclotron resonance. However, heating at the two-ion hybrid resonance is observed. Energy deposition profiles and antenna loading impedance are calculated in the two-ion hybrid case.

3.3.12 "ICRF Wave Propagation and Absorption in Tokamak and Mirror Magnetic Fields—A Full-Wave Calculation"⁶⁶

E. F. Jaeger, D. B. Batchelor, H. Weitzner, and J. H. Whealton*

Global solutions for the ICRF wave fields in a straight tokamak with rotational transform and a poloidally symmetric mirror are calculated in the cold plasma limit. The component of the wave electric field parallel to $\mathbf{B}$ is assumed zero. Symmetry in each problem allows Fourier decomposition in one ignorable coordinate, and the remaining set of two coupled, 2-D partial differential equations is solved by finite differencing. Energy absorption and antenna impedance are calculated using a simple collisional absorption model.

When large gradients in $|B|$ along $\mathbf{B}$ are present in either geometry, ICRF heating at the fundamental ion cyclotron resonance is observed. For the mirror, such gradients are always present. But for the tokamak, the rotational transform must be large enough that $\mathbf{B} \cdot \nabla B \geq O(1)$. For smaller transforms more typical of real tokamaks, only heating at the two-ion hybrid resonance is observed. This suggests that direct resonant absorption at the fundamental ion cyclotron resonance may be possible in stellarators where $\mathbf{B} \cdot \nabla B \sim O(1)$ naturally.

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3.4 CONFINEMENT OPTIMIZATION AND IMPROVEMENT

The confinement optimization and improvement effort integrates physics results from MHD equilibrium and stability, transport, and heating analyses in order to devise ways of improving confinement in toroidal systems. The three main approaches followed are optimization of magnetic fields to improve equilibrium and stability properties, optimization and control of the electrostatic potential in order to improve transport properties, and improvement of particle and impurity control at the plasma edge. The objectives of this effort are not limited solely to finding improved configurations or regimes of improved confinement.

It is also very important to find ways to experimentally vary the basic parameters that are thought to be crucial for plasma confinement, in order to test applicability of the optimization concepts.

Two attractive new low-aspect-ratio stellarator configurations have been devised. One of these is a new, lower-aspect-ratio ($A_c = 3$) version of ATF. It has the same rotational transform profile as ATF but is stabilized at high beta by shear rather than by deepening of the magnetic well. Another, quite different, low-aspect-ratio torsatron results if the coils are wound on a torus of triangular cross section.

An effort to produce stellarators with larger separations between the plasma and the coils was started. The larger separation is desirable because it will allow reactors with smaller major radii. Configurations with a small plasma in a large coil set have been improved by making small changes in the coil winding law (the method of Cary and Hanson) to heal the broken flux surfaces at the plasma edge. Methods used at the Max Planck Institute were employed to move the minor radius of the coil outward while maintaining the same outer flux surface of the plasma.

Initial studies to characterize the edge region of ATF were performed to assist in the design of efficient particle and impurity control schemes. A region about a centimeter in width, with a very long field line connection length, was found just outside the last closed flux surface. A single pump limiter located there may provide adequate pumping capacity for the experiment. In this regard, experiments with the ISX-B pump limiter were performed on Heliotron-E. Theoretical analysis of the damage on the limiter head is consistent with the predicted natural divertor topology on Heliotron and ATF. Outside the ergodic separatrix on ATF, the field lines are mostly directed poloidally. Thus, the problem of protecting a leading edge on a particle collector, such as a pump limiter, may be avoided.

Extensive studies were performed on the ultralow-aspect-ratio, paramagnetic, STX tokamak design. Monte Carlo transport calculations showed that transport in STX could be better than in a conventional tokamak, even though there would be more trapped particles in STX. This follows from the partial omnigenicity of the flux surfaces and mod- B contours. In addition, neutral beam deposition and thermalization calculations were used to verify the efficiency of the major proposed heating method on STX.

Studies of the flexible heliac concept continued. Extensive transport calculations on two possible models for the TJ-II flexible heliac were made using both Monte Carlo simulations (for collisionless orbits) and the DKES code. The significant role of the radial electric field in healing the loss cone was demonstrated. Comparisons with simpler model fields show that the effect of reducing the corner-trapped population (by increasing the aspect ratio) has much less impact on transport than does controlling the radial electric field. The importance of choosing ϵ to be related to the golden mean (the most irrational number) was recognized.

Theoretical understanding of the complex geometry of stellarators has led to continued advances in construction procedures for ATF. Three forming fixtures for the helical field (HF) coils were machined at Y-12 using our new and innovative computerized setup and control techniques. The coil joint positions were held to 10-mil tolerances. The same techniques are being used to adjust the ATF vacuum vessel so that it fits the HF coils.

3.4.1 Abstract of "Physics Studies for the TJ-II Flexible Helic" ⁶⁷

T. C. Hender, J. L. Cantrell,^{*} B. A. Carreras, H. R. Hicks, V. E. Lynch, J. F. Lyon,[†]
and A. Lopez-Fraguas[‡]

The Spanish TJ-II facility will be a four-field-period ($M = 4$) flexible heliac with a 1.5-m major radius. The $\ell = 1$ winding added to the central conductor structure in the flexible heliac design permits the rotational transform (ι) and shear to be independently varied over a wide range. Values $0.1 < \iota_0/M < 0.8$ may be achieved with plasmas of up to 25-cm average radius. At fixed ι_0/M the shear and magnetic well depth may be varied by altering the mix of $\ell = 1$ and central conductor currents. Good equilibria have been found up to $\beta_0 = 9\%$ at $\iota_0/M = 0.36$. For other ι_0/M farther from low-order resonances it is expected that higher β_0 equilibria exist. Although the magnetic configuration is sensitive to symmetry-breaking field errors for ι_0/M at low-order resonances, a small change in central conductor currents restores flux surface quality. Adequate plasma-wall separation and neutral beam injection access are achieved in TJ-II with circular toroidal field (TF) coils.

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3.4.2 Abstract of "Configuration Studies for the TJ-II Device" ⁶⁸

A. Lopez^{*}
for the ORNL Design Team

Configuration scans and design calculations for the TJ-II device are presented. TJ-II is a collaborative experiment which is to be built in Madrid, Spain. The physical constraints for TJ-II are a major radius of 1.5 m and a 1-T magnetic field. A wide range of potential configurations have been investigated. The most promising configuration is the flexible heliac, which is a heliac with an $\ell = 1$ winding added to the central conductor structure. This additional $\ell = 1$ winding permits the independent variation of ι and shear over a wide range. Other benefits of this winding include a deepening of the magnetic well and a reduction of the total current on the central conductors. Design calculations including field error studies will be presented for an optimized flexible heliac.

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3.4.3 Abstract of "ATF Neutral Beam Injection Optimization"⁶⁹

R. N. Morris and J. A. Rome

The initial phase of operation of the ATF at Oak Ridge will require neutral beam heating with expected parameters of 1.5 MW of 40-keV hydrogen for a period of 300 ms. Calculations to date using the Oak Ridge stellarator beam deposition code, STABIT, have indicated approximately 70% of the neutral beam energy can be deposited within the plasma even though beam aperturing and careful beam aiming are required. STABIT has been modified to include the complex geometry and to output the Monte Carlo birth points in flux coordinates. Results for both Gaussian and experimental beam profiles will be presented along with wall strikes, temperature rise, and sputtering to the vacuum vessel.

3.4.4 Abstract of "Optimization of Neutral Beam Injection into ATF"⁷⁰

J. A. Rome and R. N. Morris

For the initial phase of operation of the ATF at Oak Ridge, heating will be provided by the three ISX-B neutral injection systems. Each of these beams can provide 1.5 MW of 40-keV hydrogen for 300 ms.

Although ATF has very large ports (1 m) for tangential injection, the extremely complex geometry of the vacuum vessel intrudes upon the beam path. Using the present injectors, about 15–20% of the injected beam would hit the wall at locations which are difficult to shield.

The Oak Ridge stellarator beam deposition code, STABIT, has been extensively modified to fully account for the source geometry (grid shape, beam divergence, focal length), injection geometry (source location and aiming), and vacuum vessel geometry. It gives the ion locations in flux coordinates so that it is efficient to integrate the orbits during thermalization.

There are four options that can be used to improve the deposition geometry. The beam line can be moved and aimed. The helical coils interfere with most positions, so very little improvement can be achieved. The focal length of the source can be lengthened. On ISX-B, the focal point was at the small entrance port. On ATF, better performance is achieved by focusing the beam inside the port, at the place where the helical coil intrudes. The beam divergence of the ISX-B sources is 1.3° . If it is reduced to 1.0° , it yields a dramatically improved performance. Finally, a shield can be placed just outside the entrance port. The shield can be designed to intercept the trajectories of all particles which would hit the insides of the HF coils. Although this still leads to a reduction in the power delivered to the plasma, the blocked particles can be handled outside the vacuum vessel without generating any plasma impurities.

Engineering studies must determine the most cost-effective of these solutions.

3.4.5 Abstract of "Confinement Studies in a Helical Magnetic Axis Configuration Including Electric Potential Effects"⁷¹

E. R. Solano,* J. A. Rome, and J. F. Lyon*

Using Boozer's guiding center equation we have obtained the orbits for the flexible heliac TJ-II. The magnetic field on a flux surface in this configuration can be described by $B = B_0(1 + \epsilon_t \cos \theta + \epsilon_h \cos(4\phi - \theta) + \epsilon_c \cos 4\phi + \epsilon_r \cos 32\phi)$, where ϵ_t is the toroidal modulation, ϵ_h the helical one, ϵ_r corresponds to the local toroidal field ripple, and ϵ_c describes the "corner" associated with the $1/R$ variation in the four field periods. The two new terms, ϵ_c and ϵ_r , cause most particle trapping in this geometry. With no radial electric field, 30% of the ions are lost (cross the last flux surface) within a collision time. The introduction of an electric potential produces an important improvement in the confinement. Values of $e\phi/T$ as low as 0.4 reduce the losses to practically zero.

*Toroidal Confinement Physics Section.

3.5 ALPHA PARTICLE PHYSICS

The alpha particle physics effort has expanded to encompass such diverse areas as the analysis of instabilities (both kinetic MHD and purely kinetic modes), alpha transport modeling (2-D, i.e., in energy and radius), thermal instability investigations (including the effects of tritium pellets), and theoretical support of important alpha diagnostics (e.g., calculation of the scattering function for CO₂ laser Thomson scattering on the alphas).

Calculations of alpha particle effects on ballooning modes have evolved during the past year toward more realistic models of fusion-generated alphas. Comprehensive calculations of growth rates and marginal stability boundaries have been carried out using a slowing-down model for the alpha energy distribution. The ballooning mode calculation has been successfully coupled to finite-beta equilibrium codes and to improved analytic equilibrium models. Improved stability over the large-aspect-ratio equilibrium model is found. A technique for treating an isotropic alpha pitch angle distribution has been developed; previously, only the deeply trapped limit had been considered. Preliminary results indicate enhanced stability compared with the deeply trapped limit results.

3.5.1 Abstract of "Theoretical Studies of Alpha Effects"⁷²

D. J. Sigmar

Using the total power balance of an ignited tokamak plasma as a guideline, a range of alpha-driven effects is surveyed regarding their impact on achieving and maintaining fusion burn. Specific examples of MHD and kinetic modes and multispecies transport dynamics

are discussed, including the possible interaction of these categories of effects. The ensuing dynamics is nonlinear, and (using present knowledge) reliable prediction of a steady-state operating point is difficult and uncertain.

3.5.2 Abstract of "Effects of an Alpha Particle Slowing-Down Distribution on Tokamak Ballooning Modes"⁷³

D. A. Spong, D. E. Hastings, D. J. Sigmar, and W. A. Cooper

Energetic trapped alpha particles interact through their precessional drift with high-mode-number ballooning modes in tokamaks. Due to the energy dependence of the precessional drift, the consequences of this interaction depend on the form of the alpha distribution function. Results are compared here for two forms of alpha distribution: a Maxwellian and a slowing-down distribution. The latter has a significantly greater influence on ballooning stability because of the larger fraction of particles in the energy range of the drift resonance. Also, due to the dependence on the critical velocity ($\propto T_e^{1/2}$), the stability boundaries and growth rates now are strong functions of the background electron temperature T_e . Parameters typical of tokamak breakeven experiments such as the Tokamak Fusion Test Reactor (TFTR) are considered.

3.5.3 Abstract of "Effects of Isotropic Alpha Distributions on Ballooning Modes in Finite Beta Tokamaks"⁷⁴

D. A. Spong, D. E. Hastings, D. J. Sigmar, and W. A. Cooper

A method for analyzing the coupled kinetic equations describing hot species effects on ballooning modes will be discussed when the hot distribution is uniform in pitch angle. Such a model is especially relevant to the case of fusion-generated alphas, which are born isotropically and form a slowing-down distribution with mean energy determined by the background plasma electron temperature. A second feature of increased realism in our model is the option to use a numerical treatment of the equilibrium parameters appearing in the ballooning equation. Work is under way on coupling a steepest-descent moments equilibrium code to the stability calculation. This incorporates higher harmonics in the curvature drive, kinetic, and bending energy terms than are present in the simple shifted circle model and allows treatment of high-beta equilibria where the alpha precessional drift can be reversed.

3.5.4 Abstract of "Alpha Particle Destabilization of Ballooning Modes in Tokamaks for Various Energy Distributions"⁷⁵

D. A. Spong, D. J. Sigmar, D. E. Hastings, W. A. Cooper, and K. T. Tsang^{*}

The effect of alpha particles on tokamak ballooning modes has been examined in the deeply trapped limit for a large-aspect-ratio tokamak equilibrium. Both a Maxwellian and a slowing-down distribution for the alphas have been considered. For parameters where the

average hot species precessional drift frequency is in the range of the ballooning mode frequency, such modes can be significantly destabilized. This results in a decreased critical background plasma beta for ballooning instability and a loss of access between first and second stability regions. We examine the regimes of tokamak operation where this can occur and compare the effects of the two different alpha energy distributions in general and for TFTR-like plasmas.

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3.5.5 Abstract of "Effects of Trapped Alpha Particles on Ballooning Modes in Tokamaks"⁷⁶

D. A. Spong, D. J. Sigmar, W. A. Cooper, D. E. Hastings, and K. T. Tsang*

The effects of a trapped, precessing alpha particle population on ballooning modes are examined for a large-aspect-ratio, shifted circular flux surface tokamak equilibrium. The alphas are modeled in the deeply trapped limit and with a Maxwellian distribution in energy. The resulting kinetic ballooning equation is solved numerically, and the dependence of the eigenvalues and stability boundaries on shear, background pressure gradient, and ratio of hot to background temperatures and densities is investigated. In the low-frequency regime ($\omega \ll \omega_{dH}$ = alpha precessional drift frequency), the alpha component has a stabilizing influence, while in the intermediate-frequency range ($\omega \lesssim \omega_{dH}$) the alphas destabilize ballooning modes through interaction with the trapped particle precessional drift resonance. Parameter ranges that should be typical of alpha production in near-term tokamak devices such as TFTR are considered.

*Science Applications International Corporation, Boulder, Colo.

3.5.6 Abstract of "Alpha-Driven Fast Magnetosonic Wave Heating in Tokamak Plasmas"⁷⁷

W. R. Sutton III,* D. J. Sigmar, and G. H. Miley*

An alpha-driven fast magnetosonic wave instability is investigated in tokamak plasmas for propagation transverse to the external magnetic field at frequencies several times the alpha gyrorate. A 2-D differential quasi-linear diffusion equation is derived in $v_{\perp}$ - $v_{\parallel}$ geometry. The quasi-linear diffusion coefficients in the small parameter $k_{\parallel}/k_{\perp}$ are expanded and the problem is reduced to one dimension by integrating out the $v_{\parallel}$ dependence. Reactor-relevant information is obtained using data from the one-dimensional (1-D) formulation in a one-and-one-half-dimensional (1½-D) tokamak transport code. Contour plots of the alpha threshold fraction are used to identify the instability regions in the n_e - T_i plane. Alpha/background electron fractions as low as 10^{-6} to 10^{-4} may trigger the instability. For a typical reactor-size tokamak, an enhancement of the fraction of the alpha

energy transferred to ions by as much as 1.5 can occur for $T_i = T_e$ at 7 keV. Still, due to the rapid equilibration of electron and ion temperatures, a <1-2% increase in fusion power occurs overall.

*University of Illinois Fusion Studies Laboratory, Urbana.

3.5.7 Abstract of "Effects of Alpha Particles on the Scattering Function in CO₂ Laser Scattering"⁷⁸

L. Vahala,* G. Vahala,* and D. J. Sigmar

It is shown that the alpha-particle contribution to the scattered power can be dominant in the coherent scattering of a CO₂ laser in a Maxwellian plasma. For $T_i = T_e = 10$ keV, the optimal forward-scattering angle is around 1.0°, with detection of the electron density fluctuation wavenumbers $k_{\perp} \gg k_{\parallel}$ (relative to the toroidal magnetic field). A strong resonance occurs at the lower hybrid frequency. Because of the strong dependence of the scattered signal on the alpha-particle temperature and the alpha distribution function, it seems feasible that CO₂ laser scattering, using heterodyne techniques, could give detailed local information on fusion alphas.

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3.6 PLASMA MODELING AND EXPERIMENT INTERPRETATION

Fluid transport simulation codes represent the widest range of contact between plasma theory and experimentally measured quantities. Because of this broad contact, these codes are used both to interpret experimental results in terms of theoretical models and to provide guidance for new experiments. It is also important to note that the nature of experiments is such as to require, in addition to plasma physics, substantial efforts in the areas of atomic and molecular physics and plasma-surface interactions.

In order to define the operating regime for the Compact Ignition Tokamak (CIT), parametric studies were performed with the WHIST code. Ignition margins with various confinement models were examined in order to make recommendations on size and field strength. A startup sequence with simultaneous ramping of the toroidal field and plasma current was identified as the most promising means of maximizing the flattop phase when the total pulse length is limited by heating of the normal TF coils. Additional WHIST calculations were used to support the claim that skin current formation could be avoided even at high current ramp rates (≈ 3 MA/s). Experiments on Alcator-C and TFTR that could be used to verify the concept were identified. Sawtooth activity was found to have a profound influence on the performance of a CIT, including the fusion rate, energy confinement, and poloidal magnetic flux consumption (volt-seconds required in the primary of the

transformer). The validity of extending existing models for sawtooth activity into the ignition regime remains an issue that will continue to be addressed in these studies. Fast alpha particle thermalization times, densities, and pressures were evaluated. Even though a high density leads to short classical thermalization times, it also leads to a higher fusion rate. Then the most critical parameters for alpha-induced kinetic and MHD effects (the fast density and beta relative to the thermal density) and beta are only a function of temperature. Temperatures in the range of 10–20 keV should allow investigation of potentially significant alpha-induced kinetic and MHD effects, whether the plasma is ignited or driven.

Small stellarator/torsatron reactors were also analyzed. Feasibility studies of high- Q or ignited steady-state systems with 300–500 MW of fusion power were performed and show promise of attractive reactors. Maintenance of an electrostatic potential of $e\phi/kT \approx 2-4$ appears to be a critical issue, and it is desirable to press toward lower-aspect-ratio coil designs.

Useful and interesting new results have come out of pellet modeling studies. The neutral shielding model for pellet ablation has been upgraded to include a model for an ionized shield extended along the magnetic field lines and a model for fast alpha and beam ion effects in both the neutral shield thickness and surface evaporation rate. At the 2nd International Pellet Fueling Workshop, all assessments of pellet ablation and the resulting technology requirements used earlier versions of the computer models developed at ORNL, and much interest was expressed in obtaining newer versions of the models as they develop. In addition to numerous groups within the United States, this user community includes the Japan Atomic Energy Research Institute (JAERI), for applications to the Japanese JT-60 experiment; JET, for collaborative experiments with ORNL on this EURATOM project; and Garching, for studies on ASDEX and Wendelstein.

Data on pellet injection in the ohmically heated Alcator-C, TFTR, Doublet III (DIII), Poloidal Divertor Experiment (PDX), and ISX-B were collected for extensive benchmarking of the basic electron-dominated portion of the ablation model in collaboration with the pellet injector development group at ORNL. Initial results show excellent agreement with penetration depths and line-integrated density increases where multichord data are available. An attempt is being made to expand the database to include experiments outside the United States. To complement this effort, data on pellet injection in neutral-beam-heated plasmas were assembled for TFTR, PDX, DIII, and ISX-B for benchmarking fast-ion effects. This portion of the model is more difficult to test because of the difficulty in evaluating the fast-ion population, where orbit effects are important. Data for multimewatt injection in low-density TFTR plasmas have been the most critical test of the model. Agreement on penetration depth between experiment and model is within a few centimeters out of about 45 cm.

The success of including fast-ion effects from neutral beam injection led to a more extensive evaluation of possible fast-alpha effects in a compact ignition experiment. It was found that because of the differing role of electrons (establishment of the dense neutral gas shield) and fast alphas (weak attenuation of particles entering the neutral cloud normal to the magnetic field and thus dominating the surface evaporation), the penetration depth becomes a stronger function of velocity. While increasing velocity from 2 to 5 km/s does not greatly improve penetration in a thermal plasma, this increase can nearly overcome the

reduction in penetration caused by fast alphas. At 10 km/s and higher, the alphas do not play a significant role.

Several additions were made to the library of modular computer codes and routines being developed, with intercompatibility of units and computer input/output for transport analysis of toroidal-plasmas. The routines for initial development have been chosen on the basis of high demand by the transport modeling efforts at ORNL and elsewhere, where coupling MHD equilibrium (geometric effects of plasma shaping) and transport is important. They are being used for analysis of existing experiments and are incorporated in analysis codes to be used in proposed experiments. Among the computer codes added to the library are MOMCON, a general 2-D or 3-D MHD equilibrium solver, and TRACK, a code for the geometry of arbitrary chords in nonaxisymmetric plasmas.

3.6.1 Abstract of *The "Box Method" for Finding Intersections of a Trajectory with Axisymmetric Flux Surfaces*⁷⁹

S. E. Attenberger and W. A. Houlberg

Computer routines have been developed for handling a difficult geometry problem which frequently occurs in the analysis of toroidal plasmas. It is assumed that the plasma may be described in terms of a set of nested toroidal magnetic flux surfaces. Presented here is a set of algorithms that determine the intersections of a given trajectory with a given set of flux surfaces.

3.6.2 Abstract of "MOMCON 3-D Equilibrium Moments Code for Transport Calculations"⁸⁰

S. E. Attenberger, W. A. Houlberg, and S. P. Hirshman

A front-end routine and postprocessing routines have been written for the MOMCON 3-D MHD equilibrium code which facilitate coupling to transport codes. Internal documentation has been improved and modifications have been made to improve clarity and ease of use. Postprocessor capabilities include performing the "inverse" transformation from cylindrical coordinates to flux coordinates; evaluating local magnetic field components for field line following, rf ray tracing, etc.; evaluating metric coefficients and other flux-surface-averaged quantities for radial transport codes; and DISSPLA graphics routines. This suite of codes is a versatile tool for obtaining detailed information about the plasma equilibrium and geometry, particularly for predictive or diagnostic transport codes. Both Cray and VAX versions are available. Examples of capabilities will be shown for both 2-D and 3-D applications.

3.6.3 Abstract of "Magnetic Wells in Axisymmetric Hot Electron Plasmas"⁸¹

C. L. Hedrick and L. W. Owen

The failure to achieve reversal of the gradients of B and $\oint dl/B$ in EBT has raised questions about the reversal inferred from experimental measurements on at least three

axisymmetric mirror devices (ELMO, the Canted Mirror Facility, and SM-1). A re-examination of the data from SM-1 shows that the sensitivity of the experimental measurements to ring parameters leaves very little doubt about gradient reversal in axisymmetric mirror cells.

Improved fitting techniques, some of which have general application to the magnetic equilibria of hot electron plasmas, will be discussed. The sensitivity to all the parameters characterizing the ring pressure tensor (and current) will be shown. Of particular interest is the sensitivity to ring width. For example, in SM-1 one of the highest stored energy cases would not exhibit reversal if the ring width were doubled. However, doubling the ring width causes the calculated signals at more than half the magnetic probe positions to differ from the experimental measurements by 25 times the experimental error.

3.6.4 Abstract of "Self-Consistent Plasma Sheath Calculations"⁸²

J. Hogan, J. Whealton,^{*} R. J. Raridon,[†] and M. A. Bell[†]

The boundary conditions for scrapeoff layer models depend sensitively on the sheath condition at the divertor/limiter bounding surface. It is usually assumed that flow becomes sonic at this point, to satisfy the (generalized) Bohm criterion.

We have initiated a computational study of this question, using the Whealton Vlasov-Poisson code. We compare the sheath solutions obtained for steady parallel flow into a plate, without secondary electron emission, with the analytic results previously obtained. The physical model in the code is easily generalized to more complex geometries and processes.

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3.6.5 Abstract of "Plasma Engineering Assessments of Compact Ignition Experiments"⁸³

W. A. Houlberg

Compact, high-field tokamaks have been proposed as an economic means of examining the physics of ignition in the magnetic confinement program. Studies are in progress to examine the relationship of such an experiment to the rest of the U.S. and international program; outline the ignition physics which can be explored and its relevance to power reactors (both tokamak and other magnetic concepts); and carry out the detailed physics and engineering evaluations necessary for design and construction. The physics issues explored here include confinement, the possibility of achieving ignition by ohmic heating alone, startup sequences such as simultaneous toroidal field and plasma current ramps to compress the pulse length and minimize resistive heating in the copper TF coils, volt-second requirements for the poloidal field (PF) system, and alpha particle physics. Zero-dimensional (0-D), 1-D, and 1½-D steady-state and time-dependent confinement models, checked against results of several experiments, have been used in the analyses. The

demonstration that neoclassical resistivity enhancement exists in the JET device leads to the conclusion that ohmic heating can be used as the dominant heating source. If high-Z impurities can be controlled and alpha heating does not degrade confinement and internal MHD activity such as sawtooth oscillations allows high enough central current density, there is a possibility that ignition can be reached without additional auxiliary heating. A backup ICRF heating system adds greater flexibility and reasonable margin for ignition for a wide range of confinement models.

3.6.6 Abstract of "Transport Studies of Compact Tokamak Ignition Experiments"⁸⁴

W. A. Houlberg

The physics of ignition is one of the major areas which remain to be addressed by the experimental plasma physics program. A compact, high-field tokamak appears to be an attractive means (from both economic and physics standpoints) of exploring the ignition regime. Comprehensive transport analyses (at MIT, ORNL, and PPPL) examining the potential for ignition in several candidate devices are reviewed. These include 0-D, 1-D, and 1½-D models tested against data from existing experiments. The possibility of achieving ignition by ohmic heating alone is found to depend strongly on sawtooth activity and neoclassical enhancements to the electrical resistivity as well as on ion and electron thermal conduction. Neoclassical corrections to Spitzer resistivity seem to be justified by experimental data; this provides an additional boost to ohmic heating. The presence of frequent sawteeth limiting $q(0) \leq 1$ reduces ohmic heating in the plasma center, which is partially compensated by the conversion of poloidal magnetic energy to kinetic energy, and reduces both density and temperature on axis and therefore the fusion rate. Is the extension of existing sawtooth models to this very low resistivity regime valid? Starting sequences such as simultaneous B_ϕ and current ramping are found to ameliorate the effects of sawteeth (if otherwise present) as well as avoiding tearing mode activity in the compressed startup times (~ 8 – 10 MA in 3–5 s) for these devices. The addition of 5–10 MW of auxiliary heating ($P_{\text{aux}} < P_{\text{OH}}$) provides greater experimental flexibility and an increased margin against uncertainties in extrapolating from existing tokamak data.

3.6.7 Abstract of "Injection of Large Pellets in TFTR: Theory Issues"⁸⁵

W. A. Houlberg and H. C. Howe

Injection of large pellets ($\Delta \bar{n}/\bar{n} \geq 1$) into neutral-beam-heated TFTR plasmas is planned for late March and early April 1985. The electron temperature is significantly higher [$T_e(0) \simeq 3$ – 4 keV] than that encountered by pellets in earlier experiments on other devices, while the beam energy in these first experiments will be comparable to that in DIII pellet injection experiments ($E_0 \simeq 70$ – 80 keV). The main issues to be resolved by these experiments are (1) how does the large mass perturbation affect the plasma on both short and long timescales and (2) how far do the pellets penetrate into the plasma with its higher electron temperatures and fast ion heat fluxes; that is, is the ablation model used for simulating earlier experiments still adequate?

The short-time plasma response to a large mass perturbation depends on the penetration depth. The largest pellets are expected to penetrate to or beyond the magnetic axis, so these pellets may not resolve the question of rapid mixing inside the $q = 1$ region. The long-time plasma response is primarily governed by the density decay, which is limited by recycle, and the possibility of triggering disruptions.

Penetration of large pellets is greatly enhanced by plasma cooling during the ablation process, or "self-limiting" ablation. The fundamental assumption in current models for self-limiting ablation is that the electrons remain Maxwellian, that is, that the electron-electron collision timescale is short compared to the time for the pellet and its surrounding cloud to pass through a flux surface. In TFTR plasmas this assumption is not generally valid, so models are being examined which include an approximation to the time-dependent electron Fokker-Planck equation.

The assessment of fast-ion effects hinges on the degree to which the pellet cloud is ionized. Neutral cloud models predict greatly enhanced ablation from fast ions, whereas fractional ionization of 10–20% greatly increases attenuation of fast ions in the cloud and therefore reduces the ablation of the pellet surface.

3.6.8 Abstract of "ATFSR—A Small Torsatron Reactor"⁸⁶

W. A. Houlberg, J. T. Lacatski,^{*} and N. A. Uckan

A small (average minor radius $\bar{a} \approx 1$ m), moderate-aspect-ratio torsatron reactor based on the ATF is proposed as a starting point for improved stellarator reactor designs. The major limitation of the compact size is the lack of space under the helical coils for the blanket and shield. Neoclassical confinement models for helically trapped particles show that a large electric potential (radial electric field) is necessary to achieve ignition in a device of this size, although high- Q operation is still attainable with more modest potentials.

^{*}The University of Tennessee, Knoxville.

3.6.9 Abstract of "Self-Limiting Pellet Ablation in Hot, Collisionless Plasmas"⁸⁷

W. A. Houlberg, D. D. Schuresko,^{*} and S. K. Combs^{*}

Penetration of large pellets (local $\Delta n/n \geq 1$) is enhanced by plasma cooling during the ablation process, i.e., "self-limiting" ablation. This short timescale plasma response ($\tau \sim 10$ μ s) was first observed on early ISX-A experiments as a decrease in the pellet ablation rate as pellets neared the magnetic axis. The fundamental assumption in current models for self-limiting ablation is that the plasma electrons remain Maxwellian, that is, that the electron-electron collision timescale is short compared to the time for the pellet and its surrounding cloud to pass through a flux surface. In more recent experiments on DIII and TFTR this assumption is not generally valid, so a revised model has been developed which

treats the short-term electron response as a depletion of the more energetic plasma electrons. The models are compared and their predictions for planned experiments on TFTR and JET are reviewed.

*Plasma Technology Section.

3.6.10 Abstract of *Plasma Engineering Analysis of a Small Torsatron Reactor*⁸⁸

J. T. Lacatski,* W. A. Houlberg, and N. A. Uckan

This study examines the plasma physics and reactor engineering feasibility of a small, medium-aspect-ratio, high-beta, $\ell = 2$, deuterium-tritium (D-T) torsatron power reactor, based on the magnetic configuration of ATF. Plasma analyses are performed to assess whether confinement in a small, average-radius plasma is sufficient to yield an ignited or high- Q driven device. Much of the physics assessment focuses on an evaluation of the radial electric field created by the nonambipolar particle flux. Detailed transport simulations are done with both fixed and self-consistent evolution of the radial electric field. Basic reactor engineering considerations taken into account are neutron wall loading, maximum magnetic field at the helical coils, coil shield thickness, and tritium breeding blanket-shield thickness.

In stellarator/torsatron transport modeling, the transition from an intermediate collisionality regime in which confinement decreases with increasing temperature (decreasing collisionality) to the low-collisionality regime in which confinement improves with lower collisionality is shown to be of particular importance. Using a model that joins these two collisionality regimes to the resonant transition regime, where the $E \times B$ and $B \times \nabla B$ drifts cancel, electron losses are found to be the dominant factor in determining the radial electric field once the low-collisionality regime is entered. Since ion confinement is greatly improved once electron losses dominate, investigations are made into the impact of the magnitude of the radial electric field, the varying effects of ion cyclotron and electron cyclotron heating methods, and the influence of additional anomalous electron losses during startup.

These analyses lead to a small, steady-state torsatron reactor of 100-cm average plasma radius, aspect ratio 7, 9% beta, 5-T axial magnetic field, with 2.3-MW/m² wall loading. At an assumed net efficiency of 33%, ~ 306 MW(e) is produced. Plasma performance in this small reactor is found to be very dependent on the presence of a moderate-to-strong radial electric field. Nominal values of all basic reactor engineering parameters are found which satisfy feasibility constraints, with the possible exception of completely adequate shielding directly under the helical coils.

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3.6.11 Abstract of "Physics Analyses of Compact Tokamak Ignition Experiments"⁸⁹

N. A. Uckan, W. A. Houlberg, J. Sheffield,^{*} and E. Selcow[†]

We present a comprehensive method of examining the plasma performance of small ($R_0 \sim 1\text{--}2\text{ m}$), high-field ($B_0 \sim 8\text{--}13\text{ T}$) tokamak ignition experiments with detailed 0-D and 1½-D transport models. By using an analytic model, steady-state contour plots of ignition, auxiliary power requirements, heating and operating windows, etc., are generated in terms of a small number of parameters (such as aB_0^2/q_* , $\langle n \rangle R_0/B_0$, $\langle T \rangle$, etc.). These are then used for rapid assessment of a particular device and/or classes of devices with equivalent performance. Physics design space and sensitivity to variations in confinement models are examined in several candidate devices with a 0-D code. The 1½-D WHIST transport code is used for detailed analyses of these devices, including startup and burn dynamics. Steady-state plasma performance contours obtained from 0-D and 1½-D models are in reasonable agreement. Results will be reviewed.

^{*}Associate division director.

[†]Fusion Engineering Design Center.

3.6.12 Abstract of "FEDC Physics Systems Code and Application to Tokamak Experiments"⁹⁰

E. C. Selcow,^{*} N. A. Uckan, Y-K. M. Peng,^{*} and W. A. Houlberg

A new 0-D code has been developed to explore the physics design space for tokamak ignition experiments. The unique features of this code include a two-fluid model of the ion-electron power balance and a comprehensive treatment of the confinement modeling. Steady-state plasma operating and heating regimes are presented for both high-field copper and superconducting ignition devices showing sensitivities to various equilibrium, stability, and confinement assumptions. The imposition of simple engineering constraints then allows one to isolate specific regions in parameter space for candidate ignition devices. Results are compared with those from the 1½-D WHIST code.

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3.7 COMPUTING SUPPORT

J. W. Wooten

The computing support group provides the division with computing resources in two areas: operation of the User Service Center (USC) and development of the experimental

data acquisition network. The USC provides (1) access to the large computing systems at the National Magnetic Fusion Energy Computing Center (NMFECC) at the Lawrence Livermore National Laboratory (LLNL) and (2) immediate analysis and storage of experimental data. The data handling network supports the division's experimental efforts by providing a rapid, state-of-the-art means of acquiring and storing experimental data.

3.7.1 USC Operations

E. T. Blair, D. N. Clark, C. E. Gibson, D. E. Greenwood, J. V. Hughes, D. M. Jollay, H. E. Ketterer, A. C. Matas, R. W. McGaffey, D. R. Overbey, D. E. Wooten, and M. R. Wright

The major events of the past year included the following.

1. A VAX 8600 was obtained with 12 RA81 disk packs, resulting in a total of 10 gigabytes of online storage available to the division.
2. The VAX 8600 was added to the VX1-EBS cluster, which simplifies data sharing and availability on different machines. In addition, experiments are converting from PDP-11-based systems to microVAX systems, which can use full DECnet over Ethernet to communicate with the cluster. We have been able to eliminate the locally developed FEDnet software support and rely on vendor-supported DECnet for a cost saving.
3. A second HSC50 disk controller was added to the cluster. This provides dual porting and increases the reliability of the cluster for experimental data handling.
4. The PDP-10 and its disks and other peripherals were removed and shipped to the Energy Division. This has resulted in increased space, lower cooling requirements, and lower maintenance and operating costs.
5. Software support for Versatecs with NMFECC output was developed and is currently being used. This software has been distributed to the other NMFECC sites.
6. VAX software for support of VDRIVE on Versatecs is in operation. This supports T_EX and TEDI output.
7. T_EX has been installed and is being used with the Versatec printers and the T_Elaris laser printer.
8. ISSCO software has been transferred to the VAX 8600 and installed.
9. System 1032 was obtained to replace System 1022. Many crucial databases were converted and are now available under System 1032.
10. Fiche and movie support was developed by T. C. Jernigan and installed on the VAXes.
11. Version 4 of VMS was installed on all clustered VAXes.
12. NMFECC software for VAXes is installed and operational.
13. The DECnet link to X-10 is installed and operational and is much more reliable than the older PDP-10 link.
14. Tymnet X-25 external links were installed.
15. EZLink external mail was installed.
16. Ethernet cables were extended and DEC Server 100s installed.
17. Xerox 6085 PCs were added to Ethernet and are supported by the USC group.
18. Xerox-to-VAX software was installed to enable communications between networks.

19. A budgeting procedure was developed to realistically cost items and distribute to users.
20. System accounting and monitoring packages were installed and used.
21. Operator procedures for better control, tracking, analysis, and management were developed and documented and are operational.
22. Security procedures were developed and implemented.
23. Metafile plotting support on the VAX systems was upgraded to be compatible with VMS Version 4. The functionality of the plot software was enhanced to allow a single plot queue to dispatch to any of the Versatec printer/plotter devices available within the VAX cluster. The PLOT command and the metafile conversion software are upwardly compatible with the previous versions.
24. The Versaplot graphics software was upgraded to run in the VAX environment. This software was used in the development of data display applications for the International Fusion Superconducting Magnet Test Facility (IFSMTF).

3.7.2 Data Handling

E. T. Blair, R. D. Burris, J. M. Friend, D. E. Greenwood, K. L. Kannan, H. E. Ketterer, J. B. Mankin, D. R. Overbey, K. A. Stewart, J. S. Tolliver, and K. G. Young

This year a new computer system for the IFSMTF was completed. The system runs on a VAX-11/780 computer system linked to various subsystems at the facility via fiber-optic cables. Data acquisition is performed either by dedicated subsystems connected to the VAX via CAMAC equipment or directly by CAMAC. The data architecture developed to support the new system provides analysis and plotting capabilities for all data (raw or analyzed, old or new) via a consistent and uniform user interface. We expect to use a similar architecture for the ATF data system.

During the year, the repeating pneumatic injector (RPI) was constructed for TFTR. The control system for the RPI, also developed at ORNL, consisted of an Allen-Bradley programmable logic controller (PLC) and a VAX-11/730. All the control activities were ultimately the responsibility of the PLC, while the VAX system provided the software by which the operator could select actions to be taken by the PLC (such as valve closures). The VAX software included a color mimic panel display system, a menu system for data entry, data acquisition capability, and various communications tasks and mechanisms.

Another pellet injector, the deuterium pellet injector (DPI), is under development and slated for installation on TFTR in April 1987. The control system for this injector is architecturally similar to that of the RPI, but many enhancements have been made.

A new data acquisition and analysis system for the Negative Ion Test Facility was completed. The purpose of the system is to acquire several transients, digitized at 1 MHz, archive the data, and provide a vehicle for analysis. The system was implemented using CAMAC and a VAX-11/730 system.

The Radio-Frequency Test Facility (RFTF) was also completed this year. The control system for this device was implemented using an Allen-Bradley PLC-3. The data acquired by the PLC are sent to a VAX-11/780 system (the former EBT main computer) and displayed to the operator via a color mimic panel system derived from the RPI system.

A data acquisition system was developed for the ICH antenna being sent to GA Technologies (GA). This system includes data acquisition (via CAMAC) of transient waveforms and of data acquired by Hewlett Packard spectrum analyzers and other devices. These data are to be sent to GA computers via CAMAC dual-port memory modules for analysis on their computers.

One of the requirements for the ATF data system will be an excellent database management system (DBMS). In order to make an informed decision, a study was made of the available relational VAX products, including Rdb, INGRES, System 1032, and DATATRIEVE.⁹¹ The study was carried out in two parts, the first in conjunction with other fusion laboratories (Princeton and Texas) and the second to address the needs of other segments of Martin Marietta Energy Systems, Inc.

3.7.3 IFSMTF Data System Upgrade

The IFSMTF data system was upgraded to use one of the Fusion Energy Division VAX-11/780 systems. At the end of FY 1985, the VAX-based data system was nearly operational, after a year of concentrated software development and system integration efforts by E. T. Blair and L. R. Baylor (Y-12 Engineering), with additional assistance from J. B. Mankin and J.K. Munro. This development effort changed the entire complexion of the IFSMTF data acquisition and data analysis environment, providing a high level of data accessibility along with user and operations support tools. Features of the VAX-based IFSMTF data system include the following.

- A collection of interactive database utility programs patterned after a common model that provides the user with menus and a command-oriented interface.
- An integrated VAX data acquisition environment based on a common event-driven model that supports command interaction with an operator control utility; data are acquired from four independent front-end systems, each having different interface requirements, data formats, and signal acquisition modes.
- An integrated data retrieval/management environment that allows IFSMTF users to access the experimental data from any sensor, pseudo-sensor, and/or front-end channels distributed across any combination of the front-end data acquisition systems.
- A collection of programs satisfying most of the user needs for data display, including forms-oriented displays of real-time data, real-time strip charts, and various graphical presentations of scan and transient data.

Notable enhancements to the VAX CAMAC system included upgrading the software for compatibility with VMS Version 4 and adding support for the Kinetic Systems 2060-series CAMAC interfaces on the DEC microVAX II. With the announcement of the microVAX II system by DEC, a great deal of interest in a compatible implementation of the ORNL CAMAC software package was expressed by DOE in Washington. The package is currently distributed to more than 30 sites throughout the world.

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Chapter 4

Plasma-Materials Interactions

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4. PLASMA-MATERIALS INTERACTIONS

SUMMARY OF ACTIVITIES

The Plasma-Materials Interactions (PMI) Program addresses issues related to the interaction of hot plasmas with surrounding material surfaces. This includes characterization of the plasma edge, heat and particle removal, and studies of plasma-surface phenomena.

Over the past year, plasma edge diagnostics for the Advanced Toroidal Facility were identified, and the design of hardware was initiated. These diagnostics are being prepared to study plasma density and temperature, particle and heat fluxes, impurity production and transport, and the configuration of the edge plasma. The conceptual design of top and bottom rail limiters, configured as calorimeter arrays, was finished, and detailed engineering design is under way.

In collaboration with Japan and the Federal Republic of Germany, the PMI group is responsible for H_α measurements and the helium removal experiment on the ALT-II experiment in TEXTOR. Initial H_α measurements and calibration of the detectors were performed on TEXTOR. Several concepts have been investigated and the associated diagnostics identified to implement a helium removal experiment on ALT-II. In collaboration with France, a pump limiter system for steady-state operation is being designed for Tore Supra. In the past year, several concepts for pump limiter systems were studied, and predictive modeling of heat and particle removal was performed.

Studies of plasma-surface phenomena were concentrated during the past year on graphites, wall carbonization, erosion/redeposition, and postmortem analysis of limiters. Fundamental characteristics of graphites suitable for fusion technology were documented in a white paper. Four bulk graphites and one composite material were identified as reference materials. For a first evaluation of these materials, an outgassing experiment was set up and is operational. The second step will be a conditioning experiment that combines outgassing with plasma bombardment and in situ surface analysis. Wall conditioning studies were concentrated on the production and characterization of carbon films, which have been applied so successfully on the Joint European Torus (JET) and TEXTOR. The low-energy deceleration system in the Solid State Division was tested for sputtering measurements of several fusion-relevant materials. Erosion/redeposition experiments were performed with the ^{13}C technique in collaboration with the JET and TEXTOR groups. Other initiatives include studies of surface effects on the performance of rf components.

4.1 PLASMA EDGE CHARACTERIZATION: DIAGNOSTICS FOR ATF

D. L. Hillis and T. Uckan

A number of standard diagnostics will be used to characterize edge plasma properties in the Advanced Toroidal Facility (ATF). The main parameters of interest are density, temperature, heat and particle fluxes, and their scale lengths for various modes of operation. In a stellarator configuration, the edge plasma has a very complicated structure and must be analyzed as a function of radius as well as of poloidal and toroidal angle. The initial set of edge diagnostics has been identified as a movable Langmuir probe, H_α detectors, and IR and charge-coupled device (CCD) cameras. Initially, the Langmuir probe will be used to measure plasma density and temperature as functions of radius at one toroidal and one poloidal location. The H_α detectors will measure recycling at gas puff, wall, and limiter locations. Spatial distributions of H_α light will be measured with a CCD camera, and heat fluxes to limiters and wall locations will be measured with an IR camera. Heat fluxes at the top and bottom of one toroidal location will also be measured with top and bottom instrumented limiters. These limiters are configured as calorimeter arrays consisting of 11 separate tiles, each equipped with a thermocouple. This concept worked very successfully during the beryllium limiter experiment on the Impurity Study Experiment (ISX-B).¹ The central tile of this limiter is also equipped with a gas puff nozzle and with Langmuir probes embedded in the limiter tile. Fluxes of hydrogen and impurities in the plasma edge will also be measured with collection probes. Except for the CCD camera, all of these techniques were used on ISX-B.

4.2 HEAT AND PARTICLE REMOVAL

4.2.1 Limiters and Pump Limiters for ATF

T. Uckan, P. H. Edmonds, and D. L. Hillis

For initial plasma startup and for the protection of the vacuum vessel, a set of rail limiters has been designed for ATF. The limiter system consists of movable top and bottom rails, each made of 11 individual graphite tiles, attached to water-cooled stainless steel base plates. The system is designed to dissipate 3 MW of neutral beam injection for a duration of 0.25 s. Figure 4.1 shows a cutaway view of the top rail limiter inside the ATF vessel. The bottom rail limiter will be located one field period (30°) away from the top one, as shown in Fig. 4.2. The limiters have a stroke of 25 cm and can be retracted all the way to the vessel wall. This will enable us to study and compare limiter plasmas and separatrix plasmas. At least one of the limiters can be biased, which will allow us to influence edge potentials and study the plasma performance as a function thereof. The limiters also serve as edge diagnostics, as discussed in Sect. 4.1.

The modified ISX-B pump limiter, which was used on Heliotron-E, will be installed in a midplane port to study heat and particle flow pattern in a stellarator configuration. These results will serve as a database for the design of pump limiters or divertors for long-pulse operation of ATF.

Fig. 4.1. Cutaway view of the top rail limiter inside the ATF vessel. The limiter consists of two graphite tiles with thermocouples embedded. The inset shows the removable central tile with two Langmuir probes (LP) and a gas puff (GP) injection nozzle.

Fig. 4.2. The locations of the instrumented limiters on ATF. The top and bottom limiters are 30° apart toroidally.

4.2.2 Pump Limiter Studies on the Heliotron-E Device

D. L. Hillis, P. K. Mioduszewski, F. W. Baitj, O. Motojima, T. Mizuuchi, T. Mutoh, H. Zushi, K. Uo, and A. Iiyoshi

Particle control with pump limiters has been successfully demonstrated in a variety of tokamaks. The main purpose of the Heliotron-E studies was to obtain, for the first time, experimental data on pump limiter operation in a stellarator configuration. A movable pump limiter module was installed on a horizontal midplane port of Heliotron-E (see Fig. 4.3). The limiter assembly consisted of a TiC-coated graphite head with single-sided particle collection. Active pumping of the system was achieved by a ZrAl getter pump, which provided a pumping speed of 1500 L/s for thermal hydrogen. The pump limiter had a stroke of 58 cm, so that the limiter could be positioned either near the vacuum vessel wall or up to 8 cm inside the last closed magnetic flux surface. This flexibility in positioning the pump limiter permitted the study of stellarator plasmas that were limited either naturally by the magnetic separatrix or by a material limiter. The pump limiter was exposed to about 350 plasma discharges on Heliotron-E, in which the plasma conditions were $n_e \sim 2-8 \times 10^{13} \text{ cm}^{-3}$ and $T_e = T_i = 600 \text{ eV}$ and the absorbed neutral beam heating power was about 1.5 MW. Measurements of the scrapeoff layer decay lengths for electron temperature and electron density were performed by Langmuir probes located inside the throat of the pump limiter, while the total energy scraped off by the limiter head was measured by an embedded thermocouple. The particle flux to the limiter was deduced by using a fast ionization gauge to measure the pressure rise observed in the limiter pumping chamber.

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Fig. 4.3. Experimental arrangement of the pump limiter module on the Heliotron-E device. The limiter head consists of TiC-coated graphite.

For optimum pump limiter operation in a stellarator, the limiter should be located outside the last closed magnetic flux surface, thereby minimizing the influence of hydrogen recycling and impurity generation from the limiter. The pressure observed in the pump limiter as a function of the limiter position is shown in Fig. 4.4. The limiter position is measured from the last closed flux surface (positive values refer to positions inside the last closed flux surface). In the configuration investigated, only very low pressures were observed in the pump limiter when the limiter was located outside the last closed flux surface. This observation indicates that the scrapeoff layer density is very low ($<5 \times 10^{11} \text{ cm}^{-3}$). For limiter positions inside the last closed flux surface, pressures of 2–6 mtorr were observed with corresponding scrapeoff layer densities ($>5 \times 10^{12} \text{ cm}^{-3}$). These pressures and densities are quite similar to those observed during tokamak operation with comparable line-averaged densities. Other observations on Heliotron-E, including hot spots on the limiter head, indicate that the particle flow out of the confined plasma exhibited complicated divertor-like patterns of high particle density. These measurements suggest that for efficient particle collection in a stellarator, pump limiters must be matched to the particle flow patterns. A more detailed analysis of the experimental data obtained with the pump limiter is now under way.

Fig. 4.4. Pressure in the pump limiter module measured before and after activating the ZrAl pump in neutral-beam-injected (NBI) discharges.

4.3 PLASMA-SURFACE PHENOMENA

4.3.1 Postmortem Analysis of the Heliotron-E Pump Limiter

R. A. Zuhr, R. E. Clausing, D. L. Hillis and P. K. Mioduszewski

A postmortem analysis was performed on the pump limiter head that had been exposed to more than 350 neutral-beam-heated discharges in Heliotron-E. The limiter head

consisted of ATJ graphite coated with 25 μm of TiC. The elemental composition of the surface following operation was determined by Rutherford backscattering (RBS) and Auger electron spectroscopy (AES). Multimono-layer coverages of stainless steel (FeCrNi) were found on all exposed surfaces of the limiter. Traces ($\sim 10^{14} \text{ cm}^{-2}$) of molybdenum, indium, and tantalum or tungsten were also found. The amounts of stainless steel components retained on the limiter were greatest on the cylindrical sides of the limiter (up to $9 \times 10^{16} \text{ cm}^{-2}$) and lower on the surface that faced the plasma center ($1-3 \times 10^{16} \text{ cm}^{-2}$). Substantial deposits were also found on the inside of the limiter throat ($\sim 6 \times 10^{16} \text{ cm}^{-2}$). Such results indicate that the properties of the operating limiter surface are substantially altered from those of clean TiC.

Photomicrographs taken of the limiter surface showed characteristic melt patterns distinctly different from those of tokamak limiters. Salient surface features appear to represent strongly inhomogeneous particle and heat flow patterns outside the confined plasma in Heliotron-E. Figure 4.5 shows the pattern on the back side of the pump limiter and the front side of the limiter, where a similar heat flow pattern can be found inside the pumping slot.

4.3.2 Abstract of "Hydrogen Recycle and Isotope Exchange from Dense Carbon Films"²

R. E. Clausing and L. Heatherly

Several types of dense carbon films were prepared by deposition from hydrogen plasmas to which methane was added. The films were characterized by AES, X-ray photoelectron spectroscopy, and thermal desorption of hydrogen and deuterium, as well as by their recycling behavior and isotope changeover behavior. The structure and properties of the films, as well as their hydrogen content and recycle characteristics, are strongly dependent on deposition conditions and the composition and surface preparation of the substrates. The initial hydrogen recycle coefficient from the films ranges from two to less than one. The films typically contain very large amounts of hydrogen (up to 50 at. %), so they adjust themselves slowly to provide recycling coefficients near unity. Isotope changeover times tend to be long. The reservoir of hydrogen instantly available to the plasma to maintain or stabilize the recycle coefficient and isotopic composition of the plasma is 10^{15} cm^{-2} or more, depending on film preparation, temperature, and prior plasma exposure conditions. Simulator observations tend to support and improve the understanding of the observations in TEXTOR and the Joint European Torus (JET). However, they also point out the need for control of film deposition parameters to provide desirable and reproducible properties. The films and the hydrogen isotopes they contain can be removed easily by plasma processes. Since the hydrogen in these films is relatively immobile except in the zone reached by energetic particles, or at temperatures above 400°C, dense carbon films may be useful in managing the tritium recovery from near-term fusion experiments. Figure 4.6 shows, as an example, the difference in recycling behavior between a stainless steel wall and a carbonized wall.

Fig. 4.5. Heliotron-E pump limiter. (a) Front side of limiter, exposed to 250 NBI discharges. (b) Back side of limiter, exposed to 350 NBI discharges.

Fig. 4.6. Comparison of recycling behavior of stainless steel wall and carbonized wall.

4.3.3 Graphite Studies

R. E. Clausing, J. E. Simpkins, P. W. King, W. P. Eatherly, and R. A. Strehlow

As pulse duration and power loads to first-wall components increase, graphite is finding increasing use in fusion experiments. Hundreds of existing graphite grades are available, and it is important to optimize choice and conditioning procedures for present devices and future reactors. To improve the understanding and expand the database on graphites, various activities have been initiated. Some of the relevant graphite expertise available at ORNL has been documented in a white paper.³ The most important recommendation of this study is the need for experimentalists and engineers to recognize the variability of manufactured graphite and to use standardized, well-characterized material as a baseline or reference. Several bulk graphites and composite materials have been identified as reference materials for further evaluation.

As a first step for the evaluation of graphites, an outgassing experiment was set up to allow comparative studies. The system is operational, and first results confirm large amounts of H_2O , CO , and CO_2 in unbaked samples. At present, the maximum sample temperature is about $800^\circ C$. The facility will be upgraded to allow outgassing studies up to $2000^\circ C$. For further evaluation of graphites, conditioning studies under plasma bombardment will be performed in the PMI test facility (see Sect. 4.4).

4.3.4 Erosion/Redeposition Studies

R. A. Zuhr, J. B. Roberto, J. Dietz, J. Roth, and K. G. Tschersich

The low-energy ion deceleration lens system of the Solid State Division was tested with beams of several materials pertinent to the problems of erosion, redeposition, and sputtering in fusion devices. Beams of carbon, oxygen, silicon, iron, and isotopically pure ^{13}C and ^{57}Fe were obtained. The isotopic selectivity of the system can be particularly useful in providing markers for erosion/redeposition studies on graphite and stainless steel surfaces. Useful beams of these elements can be obtained at energies from 1 keV down to ~ 20 eV, although accurate current integration becomes difficult below 100 eV. Maximum currents achievable with the system appear to be limited to $\sim 35 \mu\text{A}$, with available currents of minority isotopes being somewhat lower. Beam uniformity ($\pm 10\%$) can be achieved over a 1.5-cm-diam target with some loss in current ($\sim 50\%$) by scanning the beam. Preliminary tests for carbon self-sputtering measurements and the fabrication of isotopically pure ^{13}C and ^{57}Fe films have been successfully completed.

The long-term erosion/deposition of carbon samples at the wall position in JET has been monitored using an isotopic marker technique. These measurements, the result of collaboration between the Max Planck Institute for Plasma Physics (IPP Garching) and ORNL, indicate strong surface erosion at the inner wall of JET. Twelve samples were implanted with deep (4400 Å) and shallow (280 Å) ^{13}C markers and exposed at various poloidal wall locations in octants 3 and 5 during a 6-month period of ohmic operation. The ^{13}C technique, originally developed at ORNL, allows the determination of carbon and impurity erosion and deposition at surfaces exposed to high heat fluxes. Preliminary analysis of the JET samples at IPP Garching indicates a strong poloidal asymmetry with carbon erosion of up to 6000 Å for inner wall samples and carbon deposition of up to 6000 Å on the outer wall samples. These results are being correlated with other measurements and operational details of JET.

Carbon samples implanted with a single ^{13}C marker (1000 Å) have been installed at different poloidal locations in the TEXTOR tokamak as a part of an ORNL-KFA Jülich collaboration. These samples were exposed to 77 ohmic discharges with elevated wall temperatures (200°C). Analysis of these samples is currently under way at ORNL.

4.4 TEST FACILITY FOR EDGE DIAGNOSTICS AND PMI STUDIES

T. Uckan and T. F. Rayburn

A test chamber has been set up in which a plasma is produced and heated with microwaves at 2.45 GHz. At power levels of 1 kW, plasma densities between 10^{11} and 10^{12} cm^{-3} , with electron temperatures of 10–20 eV, have been observed with a Langmuir probe. As shown in Fig. 4.7, the system consists of a magnetic mirror with high-field injection that operates at steady state. So far, the system has been used to test and calibrate edge diagnostics such as H_α detectors and Langmuir probes.

Fig. 4.7. Plasma chamber of the PMI test facility.

It is planned to combine the plasma test stand with a sample transfer system and a surface analysis station, as well as a residual gas analyzer in the path of the plasma exhaust. In this configuration, the system will be used for conditioning studies on graphite and for studies of other plasma-surface interaction phenomena.

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Chapter 5

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Plasma Technology

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5. PLASMA TECHNOLOGY

SUMMARY OF ACTIVITIES

The Plasma Technology Section has continued and significantly expanded its work in the areas of pellet fueling and rf heating during the past year.

Several new pellet injector concepts were initiated. A simplified injector was built as a prototype of a tritium injector with a minimal tritium inventory and no moving parts in the cryostat. A scale-up of the centrifuge pellet injector can accelerate multiple large pellets for extended pulse lengths; this device uses a punch to generate pellets of controlled variable size. Arc heating of the propellant gas in a pneumatic pellet injector has been shown to increase the pellet speed by up to ~40%. The cryogenic expertise developed in the pellet program has resulted in the development of a cryopump with continuous throughput.

The ORNL repeating pneumatic injector was used to inject single large (4-mm-diam) or multiple smaller (2.7-mm-diam) pellets into the Tokamak Fusion Test Reactor. Injection of single pellets results in a high density on the axis ($n_e = 1.8 \times 10^{14} \text{ cm}^{-3}$). At lower density the pellets result in significantly longer energy confinement times in the plasma center.

Future applications of rf heating in plasma devices will require higher power at higher frequencies with improved coupling efficiency from the source to the plasma. These needs are being met through collaboration with other laboratories and with industry in developing higher power sources, components, and antennas. For example, vacuum feedthroughs designed and built at ORNL are now in use at several U.S. and European laboratories. The compact loop antenna will be used on Doublet III-D, and the resonant double loop and folded waveguide antennas are candidates for use on the Tore Supra experiment at Cadarache, France.

Construction of the RF Test Facility was completed, and initial plasma operation began in 1985. This device uses superconducting magnet coils from the ELMO Bumpy Torus Proof-of-Principle project and has created a plasma using 10.6-GHz power for electron cyclotron heating.

Microwave sintering of ceramic materials will be investigated using a test facility constructed in 1985. The microwave source for these experiments will be a 28-GHz gyrotron oscillator, which can provide up to 200 kW.

5.1 PLASMA FUELING

5.1.1 Pellet Fueling Development

5.1.1.1 Tritium injector

S. L. Milora

A new single-shot pneumatic pellet injector was built in 1985 as a prototype tritium pellet injector, in support of future applications on the Tokamak Fusion Test Reactor (TFTR). The device, which is similar to the tritium proof-of-principle (TPOP) injector shown in Fig. 5.1, is patterned after a concept proposed by Claudet.¹ The potential advantages of this design for tritium service include:

1. the absence of moving parts in the cryostat,
2. minimal tritium inventory and waste,
3. a simple operating cycle, and
4. tritium-compatible materials.

The ORNL injector has been operated at up to 1400 m/s with 4-mm-diam, 8-mm-long deuterium pellets. We are proceeding with the design and construction of the apparatus shown in Fig. 5.1 to test tritium pellet acceleration limits.

5.1.1.2 Centrifuge injector

C. A. Foster

Activities in developing the centrifuge pellet injector centered on the construction and testing of a new injector, the centrifuge upgrade, which is designed to accelerate large (3-mm-diam) deuterium pellets at a rate of 30 pellets per second for extended pulse lengths. This injector will be a flexible fuel source for large, long-pulse tokamaks such as TFTR, Doublet III-D (DIII-D), Tore Supra, the Jcat European Torus (JET), JT-60, and the ASDEX upgrade. Variable-mass pellets will be accelerated to speeds of 800–1500 m/s for pulse lengths of 30 s. Program activities are divided into two projects: (1) construction and testing of the centrifuge accelerator and injection line and (2) development of a new type of pellet fabrication and feed system.

The pellet accelerator shown in Fig. 5.2 is a 50% scale-up of the centrifuge used for the DIII pellet injection experiments in 1984. The larger (1.5-m) spin diameter of the new accelerator will accommodate the larger pellets and reduce the stresses on them by extending the acceleration length. The rotor is spun in vacuum by a ball-bearing spindle driven by an induction motor. The rotor was spun to its full design speed of 9600 rpm, which is equivalent to pellet speeds of 1500 m/s.

The new pellet fabrication system will produce specially shaped pellets to reduce the stresses in the pellets during acceleration. A rim of frozen deuterium is formed on a rotating disk (cooled with liquid helium), from which pellets of variable size are punched. The voids left in the rim after the pellets are punched are refilled as the disk rotates through an

Fig. 5.1. Schematic of the TPOP pellet injector.

ORNL-PHOTO 7439-85

Fig. 5.2. Pellet accelerator for the centrifuge upgrade.

injection caliper. The device, which has been named the Zamboni machine, is shown in Figs. 5.3 and 5.4. To date, a rim of solid deuterium has been formed on the cold disk, and individual pellets have been successfully punched from it. An automated punch mechanism is being developed to allow production of 30 pellets per second and changes in the pellet size from one pellet to the next. This feature will allow, for example, increasing the pellet mass during the auxiliary heating phase of a tokamak shot or, conversely, decreasing the mass during current rampup. Integrated testing of the injector with the pellet feed mechanism is scheduled for late 1986.

5.1.1.3 Tore Supra injector

C. A. Foster

The upgraded centrifuge pellet injector (Sect. 5.1.1.2) has been selected as the basis for a pellet injector for Tore Supra, primarily as a result of the injector's potential for fueling at high repetition rates for up to 30 s. The Tore Supra injector is shown in Fig. 5.5.

Fig. 5.3. Schematic of the Zamboni machine.

ORNL-PHOTO 7436-85

Fig. 5.4. Photograph of the Zamboni machine.

Fig. 5.5. Schematic of the Tore Supra pellet injector.

Construction of the accelerator and injection line will be completed in 1986, with fabrication of the pellet feed system in 1987 and delivery to Tore Supra in the spring of 1988.

5.1.1.4 High-throughput continuous cryopump

C. A. Foster

A cryopump with the unique feature of a mechanism that continuously regenerates the cryocondensation surfaces during operation has been demonstrated. The principle of the regeneration method is illustrated in Fig. 5.6. A scraper head, named the Snail, travels over the cryopump surface and removes the frost buildup, which is trapped inside the head, sublimed, and pumped by a forepump. The pump has been tested to remove 30 torr·L/s of deuterium at a speed of 1600 L/s and a compression ratio of 100. The pump geometry allows stable operation at inlet pressures up to 0.1 torr; bursts of pumping 100 torr·L/s up to 5 torr every 5 s are possible.

5.1.1.5 High-velocity pellet injector

D. D. Schuresko

We are developing a prototype high-velocity pneumatic pellet injector, shown in Fig. 5.7, that uses arc discharge heating for increased propellant pressure and sound speed. A cylindrical arc chamber has been interposed between the propellant inlet valve and the

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Fig. 5.6. Schematic of the Snail mechanism. The scraper travels over the cryopump surface, removing frost.

Fig. 5.7. Prototype gun arc chamber.

gun breech and fitted with a ceramic insert to generate swirl in the incoming gas stream. The arc is initiated after the propellant valve opens and the breech pressure starts to rise; a typical discharge lasts 300 μs with peak currents up to 1.2 kA at arc voltages ranging from 100 to 400 V. The system is instrumented with piezoelectric pressure transducers at the propellant valve outlet, the gun breech, and the gun muzzle. The gun has been operated with 4-mm-diam polyurethane foam pellets (density of 0.16 g/cm³), an 80-cm-long gun barrel, and various gas propellants at pressures exceeding 70 bar. At $I_{\text{arc}} = 1.2$ kA and $V_{\text{arc}} = 300$ V, with hydrogen as the propellant gas ($P = 55$ bar), the arc produces a two-fold increase in P_{breech} and a small (15%) increase in P_{muzzle} when the pellet exits the gun (250 μs earlier than with gas propellant and no arc).

The pellet speed increases from 1150 m/s without the arc to 1600 m/s with the arc; this represents a 6-J increment in the pellet kinetic energy for 90 J of electrical dissipation in the arc. At present the arc discharge is driven by a single-stage capacitor bank power supply, which is not well suited to driving the 2- Ω load. Efforts are under way to design a pulse-forming network arc power supply for increased arc power.

The velocity increase and the arc dissipation are sensitive to the amount of time between the propellant gas puff and the arc initiation. Figures 5.8 and 5.9 show the electrical and pressure characteristics for two shots with different time intervals between gas puff and arc initiation. These curves show that the breech pressure increase coincides with the

Fig. 5.8. Effect on discharge of arc initiated at ~ 785 μs . (a) Arc voltage, (b) arc current, (c) pressure in valve region, (d) pressure in breech region. At the arrow, $V_{\text{arc}} = -147$ V, $I_{\text{arc}} = 1.70$ kA.

Fig. 5.9. Effect on discharge of arc initiated at $\sim 1065 \mu\text{s}$. (a) Arc voltage, (b) arc current, (c) pressure in valve region, (d) pressure in breech region. At the arrow, $V_{\text{arc}} = -314 \text{ V}$, $I_{\text{arc}} = 1.68 \text{ kA}$.

arc current pulse and that larger values for the arc dissipation (V_{arc}) and breech pressure increment are obtained for the shot in Fig. 5.9, in which the arc was delayed until P_{valve} had peaked. The pellet velocities for these shots are 1420 m/s in Fig. 5.8 and 1550 m/s in Fig. 5.9, compared with 975 m/s for a comparable shot with no arc. We can draw two conclusions: first, from the gas pressure dependence of V_{arc} , the electrical power is dissipated in the propellant gas; second, the impurities resulting from electrode erosion may reduce the propellant sound speed, but the possible reduction is more than offset by the arc heating of the propellant.

5.1.2 Pellet Fueling Applications

5.1.2.1 Pellet fueling research

S. L. Milora

In 1985 the initial phase of the pellet injection program on TFTR was concluded.^{2,3} This set of experiments featured the ORNL repeating pneumatic injector (RPI)⁴ shown in Fig. 5.10, a device that can deliver 4-mm hydrogen (or deuterium) pellets at speeds up to

Fig. 5.10. The repeating pneumatic injector installed on TFTR.

1900 m/s and repetition rates of 6 Hz. A series of experiments was performed with single 4-mm deuterium pellets injected into 80-cm ohmically heated discharges at $I_p = 1.6$ MA and with multiple 2.7-mm pellets injected before the start of neutral beam heating (6 MW of 80-kV D^0) at $I_p = 2.2$ MA. Typical results for the line density and Thomson scattering density profile for both types of discharges are shown in Fig. 5.11. In the low-current ohmic case, the single 4-mm pellet penetrates to the axis, where it produces a highly peaked density profile with $n_e(0) = 1.8 \times 10^{14} \text{ cm}^{-3}$, $n_e(0)\tau_E(a) = 6.7 \times 10^{13} \text{ cm}^{-3} \cdot \text{s}$, and $n_D(0)T_i(0)\tau_E(a) = 8.8 \times 10^{13} \text{ cm}^{-3} \cdot \text{s} \cdot \text{keV}$. The global energy confinement time for these discharges saturates at 0.45 s.

The energy confinement properties of intermediate-density [$n_e(0) = 1 \times 10^{14} \text{ cm}^{-3}$], full beam power discharges have been studied in detail using the TRANSP code. The line-average density and central electron temperature for the pellet-fueled and gas-fueled comparison shots are shown in Fig. 5.12. The high-density pellet-fueled plasmas are contrasted to gas-fueled cases by a strongly inverted beam power deposition profile (calculated) shown in Fig. 5.13. Both discharges exhibit L-regime global confinement, but the energy confinement times in the central plasma region are considerably higher for the pellet-fueled case, as demonstrated by Fig. 5.14a. This improvement in central τ_E has been attributed to a corresponding reduction in the electron thermal diffusivity χ_e . Values of χ_e

Fig. 5.11. Central line density and density profile of TFTR for injection of (a) a single 4-mm pellet and (b) three 2.7-mm pellets.

inferred from the measurements of T_e , n_e , input power, and P_{rad} are contrasted in Fig. 5.14b for ion heat diffusivities between one and three times the Chang-Hinton neo-classical value. Inside the half-radius, χ_e for the pellet-fueled case attains values that are comparable to or less than the ion heat diffusivity values.

Fig. 5.12. Evolution of the electron temperature profile during neutral beam injection for (a) pellet-fueled and (b) gas-fueled discharges.

Fig. 5.13. Power deposition profiles for the high-density, pellet-fueled discharge in Fig. 5.12a. P_{OH} = ohmic power density, P_{be} = beam power to electrons, P_{bi} = beam power to ions, P_{therm} = fast-ion thermalization power.

Fig. 5.14. Comparison of energy confinement time τ_E and electron thermal diffusivity χ_e profiles for discharges in Fig. 5.12. (a) Radial variation of τ_E at 2.6 and 2.8 s for gas-fueled (14771) and pellet-fueled (14773) discharges. (b) Profiles of χ_e for gas-fueled (14771) and pellet-fueled (14773) discharges. The upper and lower limits of χ_e correspond to calculations based on ion heat diffusivities one and three times the neo-classical value, respectively

5.1.2.2 Deuterium pellet injector

S. L. Milora

The RPI is scheduled to operate on TFTR through May 1986, after which it will be replaced by a new eight-shot pneumatic device, the deuterium pellet injector (DPI), that can be upgraded to tritium operations. The DPI was designed and built at ORNL. It differs from the RPI in that its eight single-shot guns are inside a single cryostat and in its extruder-type pellet feed system. The guns can be individually programmed for pellet speed and firing intervals, and three pellet diameters (4, 3.5, and 3 mm) are available. The DPI gun mechanism is illustrated in Fig. 5.15; the overall assembly is shown in Fig. 5.16.

ORNL-PHOTO 7089-86

Fig. 5.15. Deuterium pellet injector for TFTR.

ORNL-PHOTO 7088-86

Fig. 5.16. Assembly view of the deuterium pellet injector.

5.2 RF DEVELOPMENT

5.2.1 High-Power RF Supply System Development

W. R. Becraft, G. L. Barber, R. E. Wright, and D. O. Sparks

With the increased interest during the last five years in ion cyclotron resonance heating (ICRH) of plasmas, there has been an attendant emphasis on developing high-power rf systems to operate for longer pulses. Previous experiments with ICRH have been for short (≤ 1 -s) pulses and most often for magnetic fields of ≤ 2 T. The rf frequency requirements of such experiments have usually been < 35 MHz. To produce these frequencies, existing broadcast or radar systems were adapted, since they were in the appropriate frequency range and could meet the short-pulse requirements. However, near-term fusion devices require high-power ICRH at high frequencies (≥ 80 MHz) for pulses of up to 30 s. New designs or extensive redesigns of existing circuitry are required to produce transmitters with such characteristics.

As the Department of Energy's lead laboratory for rf technology, ORNL is developing a high-power (1.5-MW), long-pulse rf power system that is tunable over the frequency range from 40 to 80 MHz. This unit is to be completed in June 1986 and will serve as the design basis for modular building blocks of future ICRH power systems. The unit's power output is limited by the output power tube now available. However, all aspects of the system are designed to use a 2-MW power tetrode being developed by Varian.

Tests of this unit require a special load system to receive the variable-frequency rf power at up to 2 MW. Because these systems were not needed before fusion applications pushed single-unit power levels to new highs, ORNL has had to develop a load for use with the high-power system.

The high-power rf supply will be used principally to drive test antennas undergoing research and development in the ORNL RF Test Facility (RFTF), described in Sect. 5.2.4. In the future, units similar to this design will probably be used to power ICRH antennas on DIII-D, the Advanced Toroidal Facility (ATF), and TFTR.

5.2.2 RF Component Development

5.2.2.1 Antenna

D. J. Hoffman

Last year, we compared five compact loop antennas to optimize their power capacity. The results of these experiments, summarized in ref. 5, led us to continue optimization of two concepts, the cavity antenna and the resonant double loop (RDL). Four major problems were studied:

1. What are the capacitor requirements?
2. Where are the losses?
3. What is the optimum feed port for the RDL?
4. Can the current strap in the cavity antenna be shaped to optimize currents and voltages in the structure?

To determine the capacitor requirements, a full-scale development effort is needed. We are developing capacitors to meet ~ 45 kV and 1200 A rms at 30 MHz.

The losses were located by building a medium-power (20-kW) cavity antenna. We have found, through electrical and thermal measurements, that the antenna structure losses (which are $\leq 1\%$) are mostly in the capacitor, as expected. We have designed a full-scale, high-power cavity antenna for the RFTF to prove this.

Determination of the optimum feed point for the RDL is addressed in ref. 6.

The shape of the current strap was investigated by placing various current straps in a recessed cavity and then measuring coupling and inductance to infer the current and voltage requirements. It is possible to tailor the shape of the current strap to maximize power, depending on the loading and capacitor capabilities.

A new antenna, the folded waveguide, was conceived and bench tested this year. We have confirmed that this antenna allows operation with frequencies near 60 MHz with a 60-cm port. We have made preliminary estimates of power and determined the means of coupling to the waveguide, as discussed in Sect. 5.2.2.5. Tests of this antenna on RFTF are planned for next year.

5.2.2.2 Faraday shields

D. J. Hoffman

The main activities in developing Faraday shields were summarizing the configuration studies (see ref. 7) and initiating rf-plasma-material interaction experiments. We tested a variety of Faraday shield shapes with a single current strap and also measured the redistribution of current when a Faraday shield was added. Our next studies in this area will measure how Faraday shields modify the current distribution in other current straps.

We also began experiments on the means by which rf heating enhances plasma erosion of Faraday shield elements. The experiments involved tests of graphite elements exposed to (1) no rf heating and no dc bias, (2) rf heating and no dc bias, and (3) rf heating and dc bias. In general, we saw that erosion increased fourfold with rf heating. The reason for this increase will be addressed in experiments next year.

5.2.2.3 Capacitors

F. W. Baity

High-current, continuous-wave (cw) vacuum variable capacitors will be needed for high-power operation of compact loop antennas. The compact antenna being designed by ORNL for the Tore Supra tokamak will require twice the current available from commercially available capacitors; in addition, reducing the internal inductance of the commercial units for the higher frequency (~ 100 MHz) of Tore Supra is desirable. Modification of the internal structure of the capacitors is required.

To provide these capacitors, ORNL has initiated a development program in cooperation with industry. Existing designs were modeled to determine the limiting components and to ascertain how much the limits can be extended. Both major manufacturers of vacuum variable capacitors, Jennings in the United States and Comet in Switzerland, are

working to produce high-performance capacitors for testing during 1986. Both companies have designs for 1200-A units that will operate at 30 MHz and have bid on delivering the high-current, high-voltage capacitors for testing. A facility will be built at ORNL to test the capacitors under full load with a steady-state magnetic field.

5.2.2.4 RF vacuum feedthroughs

T. L. Owens

A barrier is required between pressurized transmission line and evacuated transmission line in rf heating of fusion plasmas. In the past, high-voltage breakdown at this barrier frequently limited the power handling of rf systems. To overcome this limitation, we have developed vacuum feedthroughs that can withstand the voltage levels, current levels, and pulse lengths of today's experiments.^{8,9}

Figure 5.17 is a schematic diagram of a water-cooled feedthrough built during the past year. Four of these feedthroughs, designed for connection to 8-in.-diam coaxial transmission line, are now in operation on the TEXTOR tokamak in Jülich, Federal Republic of Germany. They have operated at power levels up to 1.2 MW for 1.5 s at 28 MHz without breakdown or overheating. Uncooled versions of this feedthrough have also been supplied to the Tandem Mirror Experiment upgrade (TMX-U) at Lawrence Livermore National Laboratory (LLNL) and are currently operating at power levels of

Fig. 5.17. ORNL rf vacuum feedthrough.

≈ 300 kW. A feedthrough on loan to the Massachusetts Institute of Technology (MIT) has operated at 70 kV and 183 MHz in an antenna system on Alcator-C.

Bench tests of the feedthrough indicate pulse voltage capability of >100 kV with modest pressurization on the pressurized side. The cw capability of the feedthrough has been tested to voltage levels of 55 kV (which was the limit of the test apparatus rather than of the feedthrough).

A feedthrough is now being built for the ASDEX experiment at the Max Planck Institute for Plasma Physics in Garching, Federal Republic of Germany. This feedthrough will operate with a characteristic impedance of 25Ω , half the value of earlier feedthroughs. It will have a double barrier to provide a buffer vacuum region and additional mechanical strength.

5.2.2.5 A folded waveguide coupler for ion cyclotron heating

T. L. Owens

The folded waveguide cavity coupler is a new concept in high-power waveguide couplers for ICRH.¹⁰ A periodic array of interleaved metallic vanes is placed in a rectangular enclosure, analogous to a wide rectangular waveguide that has been "folded" several times to form a more compact structure, as shown by the conceptual diagram in Fig. 5.18. The frequency range of the waveguide is determined by the number of folds in the structure.

At the mouth of the waveguide, alternate spaces between vanes contain shorting diaphragms. This produces a unidirectional field polarization in the remaining openings. A

Fig. 5.18. Folded waveguide antenna with seven folds.

shorting plate is placed approximately half a waveguide wavelength from the mouth of the coupler to ensure that the electric fields in the apertures of the waveguide mouth are small and that most of the wave energy is contained in the H_1 fields, which couple strongly to the plasma. The structure can be designed to have its narrow dimension in the toroidal sense, so that it will fit easily between toroidal field coils. The principal advantages of this coupler are its large radiating surface area and the small electric field at the coupler-plasma interface.

For a three-vane coupler suitable for use on Tore Supra, with a frequency $f = 120$ MHz and a 10-cm coupler-plasma separation, the estimated plasma-loaded quality factor is ≈ 480 , with a coupling efficiency of 98%. For a power input of 10 MW, the peak electric field in the coupling aperture is 3.4 kV/cm, and the peak field in the center of the waveguide (~ 70 cm from the coupler mouth) is 31 kV/cm. These values are below vacuum sparkover field levels.

A small-scale model has been assembled and tested to verify the general principle. Full-scale tests at high power will be performed on the RFTF.

5.2.2.6 RF theory and analysis

Analysis of high-power rf vacuum feedthroughs

J. H. Whealton, M. A. Bell, K. E. Rothe, R. J. Raridon, T. L. Owens, F. W. Baity, D. J. Hoffman, and W. R. Becraft

To validate the analysis of the electrical characteristics of transmission lines and components, a notched coaxial line was assembled with several step changes in conductor radius. The voltage standing-wave ratio (VSWR) for this transmission line was measured and compared with analysis, as shown in Fig. 5.19.

Fig. 5.19. Calculated and measured frequency variation of the VSWR for a notched 0.5-in.-diam coaxial line.

In addition, the equipotential surfaces of two rf vacuum feedthroughs were calculated. Figure 5.20 shows the potential contours for an ORNL design with smoothly varying inner and outer conductor radii with an insulator separating the two; this design is in use on TMX-U. Figure 5.21 shows the equipotential contours for the rf vacuum feedthrough used on ASDEX, which has a more complex geometry.

Figure 5.22 shows the calculated VSWR as a function of frequency for 17 feedthrough designs, including several from U.S. laboratories and some German designs. This

Fig. 5.20. Calculated potential contours for the ORNL rf vacuum feedthrough.

Fig. 5.21. Calculated potential contours for the ASDEX rf vacuum feedthrough.

Fig. 5.22. Calculated VSWR for several rf vacuum feedthroughs.

analytic capability has led to improved designs for rf feedthroughs, and this has fostered collaboration with other laboratories.

Analysis of ICRH waveguide antenna

G. L. Chen, J. H. Whealton, and T. L. Owens

Waveguide antennas are widely considered for use as couplers of the ion cyclotron range of frequencies (ICRF) for plasma heating, so the design of a small, high-power waveguide antenna is a useful activity. Our three-dimensional (3-D) algorithm¹¹ can calculate all possible eigenmodes and their fields in a cavity of arbitrary shape. Figure 5.23

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Fig. S.23. H field distribution for the three lowest TE eigenmodes in a rectangular cavity.

shows the three lowest TE eigenmodes (TE_{011} , TE_{101} , and TE_{111}) and their H_z fields for a rectangular cavity.

We have analyzed the recently proposed folded waveguide¹² (see Sect. 5.2.2.5) and found that the resonant frequency is lower for a folded waveguide than for a linear waveguide of the same total width; the more folds, the lower the resonant frequency. The minimum resonant frequency occurs when the gap between the vane and the wall is equal to the distance between adjacent vanes, as illustrated by Fig. 5.24, which shows the resonant frequency as a function of the gap for a vane-to-vane distance of 5 cm. The H field of a typical folded waveguide is shown in Fig. 5.25; the E field is shown in Fig. 5.26.

Analysis of ICRF cavity antenna and Faraday shield

G. L. Chen, J. H. Whealton, F. J. Hoffman, F. W. Baity, T. L. Owens, R. W. McGaffey, R. J. Raridon, and M. A. Bell

The analysis of ref. 11 has been applied to the cavity antenna for DIII-D. The current strap, which is at the center of the vertical segment in Fig. 5.27, is driven in the fundamental mode. The resonant frequency determines the antenna inductance, and the magnetic field per unit current at the plasma edge determines the plasma loading. From these

Fig. 5.24. Resonant frequency for a folded waveguide antenna as a function of the gap distance.

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Fig. 5.25. H field in a folded waveguide antenna with three folds.

Fig. 5.26. H field in a folded waveguide antenna with three folds.

Fig. 5.27. H field in a cavity antenna with no Faraday shield. The current strap is the vertical element at the center.

parameters, we determine the current and voltage requirements and thereby the antenna characteristics. The addition of a Faraday shield reduces both the inductance of the antenna and the magnetic field passing through the shield. The contours of H_z are shown in Fig. 5.27 for the antenna without a shield and in Fig. 5.28 for the antenna with a shield. Figure 5.29 shows the calculated and measured variation of H_z with distance from the current strap with and without a Faraday shield.

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Fig. 5.28. H_z field in a cavity antenna with a Faraday shield.

Fig. 5.29. Variation of H_z with distance from the current strap with and without a Faraday shield. (a) Calculated variation. (b) Measured variation.

5.2.3 Collaborative RF Programs

5.2.3.1 ICRH project on DIII-D

D. J. Hoffman

A 9-MW ICRF experiment has been proposed to heat the DIII-D plasma. The initial experimental program on DIII-D includes ohmic plasma and neutral beam studies; high-power rf experiments will follow. Compact loop antennas, which fit in a 35- by 50-cm port, have been chosen to convey this rf power because of their ease of maintenance, high efficiency, and versatility (see Fig. 5.30). To verify that the antenna will have sufficient loading, a prototype has been designed and will be installed in January 1986; measurements will be made through September 1986. The prototype is a cavity antenna that will operate from 20 to 74 MHz with a 50- Ω match for a load resistance of $\sim 1 \Omega$. The antenna is surrounded by a fixed graphite-covered limiter (not shown in Fig. 5.30) and can be extended from 3 cm behind this limiter to 2 cm in front of it, in order to adjust the coupling to the plasma. Characteristics of this antenna system will be determined and extrapolated to ignited tokamaks. Experimental explorations of coupling have been supplemented by investigations of wave propagation and absorption in DIII-D using a cold collisional plasma model in straight tokamak geometry with rotational transform. Although loading is dependent on plasma position, the predictions are compatible with the antenna design.

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Fig. 5.30. Prototype cavity antenna to be used on DIII-D.

5.2.3.2 ICRF heating on Tore Supra

F. W. Baity

The United States and France have agreed to collaborate in several key technology areas, including ICRF heating, on the Tore Supra tokamak now under construction at Cadarache, France. The United States will design and construct hardware to be installed on Tore Supra in return for the opportunity to participate in experiment planning and operation.

Tore Supra is a large superconducting tokamak that will have a plasma pulse length of 30 s, longer than any other major fusion experiment. Its main parameters include a major radius of 2.38 m, a minor radius of 0.8 m, a magnetic field on axis of 4.5 T, a total of 12 MW of ICRF power, and 30-s pulses every 4 min. Each rf antenna must deliver 2 MW at a frequency between 35 and 80 MHz. The first antenna will be installed in 1988.

The ICRF heating launchers will be designed by ORNL, which was chosen for this task based on the development of the resonant double loop and the folded waveguide. Work began in 1985 on the conceptual design for an antenna to be installed on one of the large (60- by 70-cm) outer ports of Tore Supra. Each ICRF heating port will contain two loops behind a common Faraday shield. The capacitor development program described in Sect. 5.2.2.3 was initiated to produce the vacuum variable capacitors that will be needed for these antennas.

5.2.4 The ORNL RF Test Facility

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The ORNL RFTF^{13,14} is a national facility for testing high-power rf components and systems being developed for applications on plasma confinement devices. The facility's flanged opening and volume are large enough to accommodate prototype antenna structures for any existing U.S. fusion device. Components can be tested at rf power levels up to ~1.0 MW steady-state in an environment with a controlled vacuum and magnetic field and a warm hydrogen plasma. The facility is located in Bldg. 9201-2, in the eastern end of the Y-12 Plant. Tests in the facility are conducted in a coordinated effort by RFTF operations staff and the related rf development program.

5.2.4.1 Facility description

Figure 5.31 shows a cutaway view of the RFTF. The test cell, which is located in a lead-shielded enclosure, is a simple mirror that uses two superconducting magnets built for the ELM/ Bumpy Torus (EBT) Proof-of-Principle project. A vacuum vessel between the

Fig. 5.31. Isometric view of the RFTF.

magnets serves as the test chamber for antennas and other rf components. A steady-state plasma can be formed in the chamber using microwave power from 10.6-GHz or 28-GHz systems salvaged from the EBT-Scale (EBT-S) device. Auxiliary equipment for the magnets is located outside the shielded enclosure on the adjacent equipment platform.

Vacuum vessel and associated subsystems

Figure 5.32 shows the vacuum vessel sandwiched between the magnets with pumping extensions through the magnet throats to the vacuum system. The vessel is made of non-magnetic stainless steel; the enclosed volume between magnets roughly is 1.5 m³. The main access opening is 74 by 163 cm, which is large enough to allow testing of the launching structures proposed for DIII-D and TFTR. Additional access for components to be tested and for diagnostics is available through a large (35- by 25-cm) port on top of the vacuum vessel; through several smaller, round ports on the three exposed sides; and through the ends of the pumping extensions.

Fig. S.32. Elevation views of the RFTF, showing the vacuum pumps and superconducting magnets.

The vacuum pumping system¹⁵ consists of two 2000-L/s Balzer turbopumps backed by a Roots-type blower mechanical pump. Vacuum diagnostics and controls are based on standard ion and thermal convection gauges with logic interlocks controlling electropneumatic valves for system protection. The long vacuum extensions place the turbopumps in a greatly reduced magnetic field (110-G maximum); magnetic shields protect ion gauges and the magnet dewar vacuum pump. Microwave screens made of stainless steel honeycomb, coated with copper and with openings 0.125 in. wide by 0.25 in. thick, prevent microwave power from reaching the pumps with minimal reduction in throughput. A base pressure of $<1 \times 10^{-6}$ torr is readily obtained by this system.

A hydrogen gas injection system provides controlled pressure for maintaining and adjusting plasma characteristics. Information about the pressure in the vessel is compared with the desired pressure and used to control a piezoelectric valve that adjusts gas input.

Energy is removed from the vacuum vessel by internal water-cooled panels attached to the vessel walls and extending through the magnet throats to pumping tees. The cooling panels are expected to handle a power density of 15 W/cm² on the walls in the region of maximum loss, and the demineralized water cooling system is sized to remove 1 MW of heat (steady-state).

Magnet system

The operating maximum field is ~ 3 T on axis in the magnet throats, corresponding to a coil current of 1200 A. The separation between the magnets, centerline to centerline, is 112 cm, providing a mirror ratio of 4.8. The magnetic geometry is shown in Fig. 5.33.

A closed-loop cryogenic system using a Koch 1630 refrigerator/liquefier supplies liquid helium to the magnets at a rate matched to their combined boiloff rate of ~ 15 L/h. The boiloff gas is recovered, stored, and reliquefied; the closed-loop system piping is liquid nitrogen-traced.

A power supply rated at 2000 A and ± 10 V, with $<1\%$ ripple, supplies current to the magnets, which are connected in series. Across each magnet is a dump resistance of 0.24 Ω , and electrically between the magnets is a dump breaker. When the dump breaker is activated by the magnetic protection system, the magnet current (stored energy) is forced through the dump resistance, where the energy is dissipated. The magnet protection systems acts whenever a quench (magnet windings going normal) is detected or when a manual dump command is given. The magnets have their own support structures, which are independent of the vacuum vessel and are designed to accommodate 2.3 times the expected attractive force of 57,850 N (13,000 lbf).

Microwave systems

Continuous plasmas with temperatures from 1 to 10 eV and densities on the order of 10^{11} to 10^{12} cm⁻³ can be formed in the RFTF by electron cyclotron resonance breakdown, using microwave power from the 200-kW, 28-GHz gyrotron system, and less intense plasmas can be formed with the 20-kW, 10.6-GHz gyrotron system. Microwave power is piped to the facility through an evacuated 2.5-in.-diam waveguide and enters the vacuum vessel through a microwave-compatible vacuum valve. The evacuated waveguide, rather than a gas-filled waveguide, was chosen to eliminate the need for a waveguide window at the vacuum vessel.

Fig. 5.33. Flux lines and mod-B surfaces for RFTF.

RF sources

Existing high-power, cw rf sources available to support the first year of RFTF operation are as follows:

Frequency range (MHz)	Power (kW)
2-200	1.5
2-30	100
3-18	200
30-60	50
40-80	50
175-215	50

A modified rf source from the Fusion Materials Irradiation Test (FMIT) facility that can operate at >1 MW cw and 40–80 MHz will be installed and should be operational near the end of 1986. The locations of the existing 200-kW ICRH transmitters and the electron cyclotron heating (ECH) waveguide from the EBT-S experiment area are shown in Fig. 5.34. The rf power is fed to the facility through large-diameter coaxial cable, matching circuitry, and a 50- Ω vacuum feedthrough.

Instrumentation and diagnostics

Virtually all of the RFTF instrumentation is monitored by a programmable logic controller (PLC). System information is available to the operator through a series of display "pages" on a color monitor; each page corresponds to a major subsystem. System controls are, for the most part, separate from the PLC.

Diagnostics for the initial phase of RFTF operation are minimal. Power loading on vessel walls and rf components is measured by calorimetry consisting of ΔT transducers and flowmeters on the water-cooling circuits. Plasma characteristics and profile are measured with a movable Langmuir probe. Additional information on plasma density is

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Fig. 5.34. Plan view of the RFTF, showing the ICH transmitter and the ECH waveguide.

obtained with a 4-mm microwave interferometer. Data acquisition is handled through CAMAC by a VAX-780. Additional diagnostics will be added as they are needed for experiments and tests.

5.2.4.2 Status

Preoperational tests were successfully performed on the vacuum systems for the vessel and for the magnets in June 1985. The cryosystem was checked out in late July 1985, and the magnets were tested to full operating current (1200 A), corresponding to the design field requirement of 3 T on axis in the magnet throats, in early August 1985. Assembly and installation of the basic facility hardware were completed in August 1985. Initial operation of the 28-GHz system was delayed when a large power supply failed, so the waveguide was tested into a dummy load using 20 kW of 10.6-GHz microwave power, of which 14 kW was measured at the load. The completed facility is shown in Figs. 5.35 and 5.36.

The first plasma in the RFTF was obtained on December 18, 1985. Facility parameters included a magnetic field of 0.25 T at the test cell center, a vessel pressure of 4×10^{-6} torr (hydrogen), and 10 kW of 10.6-GHz injected microwave power. The pulse

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Fig. 5.35. Exterior of the RFTF enclosure.

Fig. 5.36. RFTF control room.

length was 4.5 h, and the Langmuir probe data indicated a plasma density of slightly less than 10^{11} cm^{-3} and an electron temperature of $\approx 8 \text{ eV}$.

At the end of this reporting period, the RFTF is considered ready for low-power antenna tests. Repair of the power supply for the 28-GHz gyrotron is nearly completed, and checkout testing with plasmas formed using 28-GHz ECH is expected to start in early 1986 and to continue at increasing power levels. Incremental X-ray shielding will be added and energy removal provisions modified as needs are identified. The facility should be ready for high-power antenna tests by mid-1986.

5.2.5 Microwave Utilization and Development

5.2.5.1 Microwave sintering of ceramic materials using high-power gyrotron oscillators

H. D. Kimrey, T. L. White, T. S. Bigelow, and P. F. Becher

Microwave sintering of ceramics^{16,17} offers several distinct advantages over sintering using conventional furnaces. First, ceramic heating rates are substantially faster than in

conventional sintering, because the ceramic is heated directly by microwave power according to its loss tangent¹⁸ and the amount of microwave power incident on the ceramic. As a result of the rapid heating, nonisothermal processes, such as segregation of impurities to grain boundaries,¹⁹ are minimized. Since the sintering time can also be lowered, the possibility of secondary recrystallization (exaggerated grain growth) may be reduced. The minimization of impurity segregation and exaggerated grain growth should lead to a strengthening of the ceramic's mechanical properties.

A second advantage is that, because most ceramic materials have a low loss tangent in the microwave frequency range, fairly uniform microwave heating can be achieved even for large, irregularly shaped samples. Conventional sintering techniques heat the surface of the ceramic from external heat sources and thus produce nonuniform heating until the entire furnace and sample reach an equilibrium temperature. Uniform internal microwave heating should result in a more homogeneous sintered product.

Finally, in conventional thermal sintering, a substantial amount of heat is needed to bring the sample and its enclosure (e.g., furnace walls or linings) to the desired processing temperature, and heat is generally lost through the enclosure to the surrounding area. In microwave sintering, these problems are largely avoided because the microwave cavity or "oven" is made of materials with a high electrical conductivity; thus, the majority of the power is absorbed by the ceramic sample. The result is more efficient use of power to heat the sample.

Previous microwave sintering experiments have been performed with small microwave applicators at 2.45 GHz, with a few kilowatts of available cw power. Small ceramic samples had to be specially shaped to fit into the small, single-mode cavities or applicators.

We now propose to use the newly developed high-power gyrotron oscillators²⁰ that produce hundreds of kilowatts at 28 GHz for microwave sintering. The loss tangents of most materials increase with frequency, so that power absorption should be enhanced even further. This factor, in combination with the considerably higher power available, leads us to expect that the time required to sinter the ceramic will be greatly reduced. This will improve the mechanical strength of the ceramic over that obtained in earlier microwave experiments. Our technique, which uses a large cavity with dimensions much greater than a free-space wavelength, will allow us to heat large, irregular samples uniformly. The power density in the cavity is made uniform by allowing the microwave energy to reflect off the cavity walls many times, thus exciting a very large number of cavity modes.

In order to demonstrate proof of principle, we plan to share the RFTF 28-GHz, 200-kW cw gyrotron with the rf development program. Most of the the equipment needed to construct a microwave furnace for heating the ceramic samples is available as surplus from the EBT-S experiment. We plan to use an available 76-cm-diam, 100-cm-long cavity as an evacuated microwave cavity; this will allow us to sinter the ceramic samples without atmospheric gases or vapors that might contaminate the samples.

The heating process consists of a low-temperature bakeout phase followed by rapid heating to a 1600°C sintering phase. Special high-temperature thermocouples will monitor the ceramic temperature in several locations, and the power delivered to the sample will be measured using system calorimetry. A residual gas analyzer will monitor the vacuum quality in the cavity during all phases of heating.

Because the loss tangents of ceramics increase with temperature, we expect the samples to "run away" thermally at constant power. To correct for this effect, we can roll back the duty cycle using modulated output power, thus stabilizing the temperature.

Tantalum is a prime candidate for the sample holder because of its high operating temperature in vacuum and because it does not react with high-purity alumina samples at high temperatures.²¹ In addition, it has low microwave losses compared to graphite or carbon crucibles. The sample holder will be carefully designed without sharp corners or points on metal surfaces to minimize arcing, and a high-quality vacuum will be maintained in the furnace.

With very small ceramic samples, appreciable microwave power can flow back toward the gyrotron sources. Protective devices called mode absorbers have been installed to absorb this power.

If the experiment is successful, we plan to submit a proposal for outside funding to the Department of Energy for research into improved ceramics for the Heat Engine Program.

5.2.5.2 Gyrotron development

G. R. Haste

The principal effort in the gyrotron development program during 1985 was the development of a 140-GHz cw source with 200-kW capability. The approach taken by Varian was to use a standard magnetron injection gun to provide an electron beam to either a single cavity or a stepped interaction cavity. This resulted in successful operation at the proper frequency and in the proper mode (TE_{03} circular electric). The initial device produced over 100 kW in pulsed operation with a 1.8% duty cycle. Management of this contract passed from ORNL to LLNL at the end of the fiscal year.

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Chapter 6

4

Superconducting Magnet Development

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6. SUPERCONDUCTING MAGNET DEVELOPMENT

SUMMARY OF ACTIVITIES

A major milestone in the international Large Coil Task (LCT) was reached in October 1985, when experimental operation of a six-coil toroidal array of superconducting test magnets was begun. During the first months of the year, work in the International Fusion Superconducting Magnet Test Facility (IFSMTF) consisted of installing the five completed test coils and the system of remotely movable pulse coils. Meanwhile, Westinghouse encountered a technical problem with one component of its coil, which delayed delivery until August. This placed extreme schedule pressure on its installation in the IFSMTF. Foreign participants in the LCT expressed grave concern to the Department of Energy, which was allayed only when extraordinary effort by the ORNL Fusion Energy Division, the Engineering Division of Martin Marietta Energy Systems, Inc., and the Y-12 Maintenance Division resulted in installation of the Westinghouse coil and remaining equipment and closing of the vacuum tank by the date that had been promised before the setback at Westinghouse occurred.

Following the closure of the 11-m-diam tank housing the test array, a satisfactory vacuum was achieved, and cooldown to the required cryogenic operating temperature was started in December. There were some delays due to problems with the helium compressor system, and the cooldown was interrupted for replacement of a leaking helium heat exchanger. On the whole, however, at the end of the year, prospects were good for achievement of all the goals of the LCT test program within a reasonable period of time.

Concurrently with the mainline effort on the LCT, development of advanced superconductor concepts continued at a modest level.

6.1 INTRODUCTION

During 1985, the ORNL Fusion Energy Division continued to have a leading role in the development of superconducting magnet systems for toroidal fusion reactors. Since 1977, most of the effort has been devoted to the Large Coil Program (LCP) and its outgrowth, the international Large Coil Task (LCT). The International Fusion Superconducting Magnet Test Facility (IFSMTF) at Oak Ridge is the focal point for the collaboration by the United States, the European Atomic Energy Community (EURATOM), Japan, and Switzerland in the LCT under the terms of an International Energy Agency agreement. In the IFSMTF, six different toroidal field (TF) coils will be operated as a toroidal magnet system, with each coil tested in turn to determine its capabilities. The United States is providing the facility and three of the test coils; the other participants, one coil each. On

behalf of the United States, ORNL acts as the operating agent for the LCT. The facility is operated by qualified members of the Fusion Energy Division staff. On-site representatives of all LCT participants work together in planning and carrying out the coil tests.

At the beginning of 1985, test facility construction had been finished, five of the six test coils had been completed and delivered, and shakedown operation with three coils in place had been accomplished. Westinghouse Electric Company had finished installation of conductor in the sixth coil and was machining the structure in preparation for final assembly. The year's activities are described in Sect. 6.2.

In addition to the LCT, work continued in the division's Magnetics and Superconductivity Section to advance superconducting magnet technology for fusion by developing better analysis techniques, generating basic design information, and evaluating small quantities of experimental conductor embodying advanced concepts. These activities are described in Sect. 6.3.

6.2 LARGE COIL TASK

6.2.1 Test Coils

The only substantial work under the coil subcontracts this year was carried out by the Westinghouse Electric Company at East Pittsburgh, Pa. The coil halves had been brought together in October 1984, and in January 1985 the job of bolting them together was finished. Machining of structural interfaces with the test facility was completed in February. Final assembly was not completed until August. Although joining of conductors between coil halves went more slowly than planned, the principal cause of delay was failure of the high-current feedthroughs.

Early development tests had indicated that the design of the feedthroughs was sound, and one of the two feedthroughs was fabricated and successfully tested at liquid helium temperature. However, the second could not be made leaktight and finally failed completely on May 9. Staff from ORNL and Westinghouse reviewed the situation and worked out a major design change. The problem received top-level management attention within Westinghouse, with the result that the job was given highest priority and intensive effort. (The coil work was already proceeding 24 hours a day, 7 days a week.) Westinghouse quickly developed the manufacturing steps and rebuilt both feedthroughs. The remainder of the assembly was soon completed, and the results of helium leak tests led ORNL to authorize shipment. The coil left the Westinghouse plant on August 8 and was installed in the IFSMTF as described in Sect. 7.2.3.

Westinghouse retained only the equipment that might be needed if rework of the coil proved necessary. Other government property was disposed of. Airco shipped selected materials to ORNL and inventoried the remainder for disposal.

6.2.2 Program Management

Plans for testing in a six-coil array were reevaluated in light of experience with the partial-array tests and in the context of the priorities and budget of the national fusion program. In consultation with the Department of Energy (DOE), a proposal by the

Operating Agent to the other LCT participants was worked out. This proposal was presented at the LCT semiannual meeting April 29-May 1. Discussions led to an agreement on a "baseline" program with scope reduced by elimination of the certain tests on selected coils.

Technical monitoring of Westinghouse work at East Pittsburgh continued until the coil was shipped in August. An especially intensive period of management activity followed the failure of the high-current feedthrough in May, which had major impacts on costs and schedules. The LCP assistant manager stayed in East Pittsburgh until coil assembly was completed. Agreements on final leak testing were worked out. Shipment to Oak Ridge was arranged by Martin Marietta Energy Systems, Inc.

Managers from ORNL and Westinghouse reviewed the cost history of the Westinghouse activities and agreed upon appropriate allocations between fabrication and other activities. Agreement was also reached on how direct and indirect costs should be charged. A final cost proposal was accepted by Energy Systems after review by ORNL.

When foreign LCT participants formally expressed concerns about schedule and implementation of the program, ORNL staff prepared responses for use by DOE. Managers from ORNL participated in an international meeting of fusion program directors in Paris and in semiannual meetings of project officers and executive committee in Europe in September. These interactions led to understandings and agreements, including a commitment by DOE to complete the "baseline" program.

Weekly meetings of the international steering committee at the test site continued throughout the year. Coordination meetings of installation and operations personnel were also held each week.

Possibilities for use of the IFSMTF after completion of the Large Coil Task were surveyed. Those that appeared feasible were identified in an interim report to DOE.

6.2.3 Facility Installation and Modifications

As expected, shakedown operation of the facility in 1984 had shown the need for numerous minor repairs, adjustments, and modifications. In general, however, it appeared that the facility, with the notable exception of the helium system, would perform satisfactorily during the six-coil tests. All participants agreed that the performance of the helium refrigerator/liquefier and distribution/storage system would have to be upgraded. Plans for upgrading the helium system, worked out with participation of consultants and foreign LCT participants, were reviewed in January by DOE and experts from other laboratories.

Vacuum-jacketed helium transfer lines that had excessive heat leakage during shakedown were removed, repaired, and replaced. The anomalously high boiloff from the vapor-cooled leads and dewars was investigated by a laboratory test of two leads and dewars at high currents.

The additional liquefaction capability that tests showed to be necessary was provided using a CTI-2800 refrigerator system obtained from the Bonneville Power Authority, which had used it in demonstrating superconducting magnet storage of energy. The equipment was brought to the IFSMTF in March and installed during the next four months. Helium gas storage capacity was greatly increased by connection of two large,

intermediate-pressure tanks and a bank of high-pressure helium cylinders, with the CTI-2800 system compressors and one additional surplus compressor.

The General Electric (GE) coil, which had been delivered from the Oak Ridge Gaseous Diffusion Plant shops on December 20, 1984, was equipped with external sensors during January and February. On March 4 it was set in the test stand.

Installation of the pulse coil system had begun in 1984 with the coil supports, half of the segmented circular track (through the bores of the three test coils already in place), and the coil/torque beam assembly. Installation of electrical buses, contact assemblies, nitrogen supply hose and takeup assemblies, and drive mechanisms continued into October 1985.

Substantial redesigning and reworking were required on the pulse coil system. Tests showed that the forces involved in the original movable contacts and the stresses in bolts holding the assembly together were excessive. Higher-strength bolts and nuts and specially adapted tensioning equipment were required. The installation of the new bolts was finished in June.

Installation of special equipment for the Swiss and European coils and connection of these coils was a major activity through March 1985.

Preparation of the Westinghouse coil began immediately upon delivery on August 10, and it was set in place on August 15. Its installation was integrated with that of the final elements of the pulse coil system. This work was performed by craft forces from the Y-12 Plant organization, with active participation of Energy Systems Engineering and ORNL Fusion Energy Division staff, all working extended hours.

Intensive work on installation, removal of access equipment, and cleanup of the vacuum tank interior continued through October 23. The assembly and checkout of the movable pulse coil system proved particularly arduous, but finally this task was satisfactorily completed. On October 24, the date that had been promised to LCT participants at the meeting on May 1, the vacuum tank lid was installed.

6.2.4 Facility Operation

The helium refrigerator/liquefier was operated extensively from April through June to establish optimal operating parameters and to verify the effectiveness of some modifications. Results were gratifying. Sustained operation with subatmospheric pressure in the auxiliary cold box (as required for the European coils) was demonstrated for the first time. Some deficiencies requiring correction were identified. However, analysis indicated that performance corresponding to acceptable duty cycles should be achievable in six-coil tests.

Continuous operation of the facility resumed on October 24 with the closure of the vacuum tank. After a few air leaks at tank penetrations were found and repaired, the tank pressure was quickly brought low enough for sensitive helium leak tests of the helium and nitrogen systems within the tank. These tests showed intolerable leakage from the Swiss coil and a much smaller amount of leakage from the GE coil. The tank was opened and the Swiss coil leak located and repaired. Air leaks into the tank that had not previously been repaired were located and stopped at this time. No leak could be located in the GE coil, so the tank was closed.

By November 16, the tank pressure had reached the criterion for starting cooldown (0.8 mtorr). Cooldown could not proceed, however, because preparation of the helium system had not yet been completed. After additional equipment to remove oil from the compressed helium had been installed and leak tested, circulation of helium to remove impurities was begun. Persistence of moisture led to discovery of water leakage in a compressor oil cooler. Location and repair of several leaks (due to gradual deterioration of old equipment) extended through December 5. While this critical-path work was going on, the vacuum tank was opened and the small leak in the GE coil was found and repaired.

Operations to eliminate moisture, nitrogen, and oxygen from the helium system achieved satisfactory purity on December 15. Cooldown of the test stand, by circulation of helium cooled with liquid nitrogen, began on that date. Losses of helium from the circulating system soon became excessive, and the operation had to be interrupted on December 19, after the test stand had been cooled only to 270 K.

The helium leak was found to be in the cooldown heat exchanger, into the nitrogen space, which was vented through a check valve to the outside atmosphere. The cooldown heat exchanger was removed and taken to the manufacturer's plant for inspection. Evidently air had leaked back through a malfunctioning check valve in the nitrogen vent line, and freezing of moisture had caused extensive damage and an indeterminate number of leaks. Repairs were deemed impracticable, manufacture of a replacement would take several months, and no satisfactory substitute was available for purchase. Fortunately, an identical heat exchanger was located in an idle refrigerator at Lawrence Berkeley Laboratory. Arrangements were made to obtain this heat exchanger, enabling resumption of IFSMTF operations.

6.2.5 Testing and Analysis

Analysis of the results of the partial-array tests continued throughout the first half of the fiscal year, resulting in comprehensive reports at the American Nuclear Society's 6th Topical Meeting on the Technology of Fusion Energy.^{1,2} Testing and analysis personnel produced plans for coil tests, estimated the operating times required for each of the tests originally proposed, and participated in reviews that led to adoption of the "baseline" test program, which was estimated to require 40 weeks with the coils cold. After the international meetings in early September, they began drafting a revision of the Plan for Implementation of the LCT Test Programme.

A substantial amount of analysis and engineering was required to establish the pattern of bolt tensioning in the pulse coil structure that would optimize allowable operating cycles. Much effort was also expended in 1985 on developing computer models of the six-coil array for use in stress analysis of various operating modes.

6.3 ADVANCED CONDUCTOR DEVELOPMENT AND MAGNET TECHNOLOGY

In reporting the magnet technology work accomplished in 1985, we include work funded by other projects.

6.3.1 Abstract of "Quench Energies of Potted Magnets, III"³

L. Dresner

Several different formulas have been proposed for estimating quench energies of potted magnets. These formulas, although based on plausible approximations, disagree with one another by factors of 2 to 6. By the use of maximum and minimum principles for the heat balance equation, it has been possible to determine whether each estimate is an upper or lower bound and to show that some estimates majorize others.

6.3.2 Abstract of "Studies of Heat Transport to Forced Flow He-II"⁴

L. Dresner, A. Kashani,^{*} and S. W. Van Sciver^{*}

Analytical and experimental studies of heat transport to forced-flow He-II are reported. The work is pertinent to the transfer of He-II in space. An analytical model has been developed which establishes a condition for two-phase flow to occur in the transfer line. This condition sets an allowable limit to the heat leak into the transfer line. Experimental measurements of pressure drop and flowmeter performances indicate that turbulent He-II can be analyzed in terms of classical pressure drop correlations.

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6.3.3 Abstract of "EBT-P Magnet Development Program Summary"⁵

T. J. McManamy,^{*} J. W. Lue, J. K. Ballou,[†] T. L. Mann,[‡] S. S. Shen, and C. T. Wilson

The ELMO Bumpy Torus Proof-of-Principle Magnet Development Program, which began in 1979, has fabricated and tested three NbTi superconducting magnets. The magnets are metastable with a maximum winding pack current density of 10,000 A/cm² at a peak field of 7.4 T. Each magnet has been installed in a close-fitting dewar with a warm bore. Previous publications have reported successful tests of the first two windings in a large dewar, tests of the first complete magnet in its own dewar, and construction details of the dewars. This paper gives procedures and results for the test of the third winding in an open dewar, the test of the second single magnet in its own dewar, and two tests in which two magnets were run in series to generate a 73,000-lb side load at 5 T. The last tests verified the mechanical integrity of the design and qualified two of the magnets for use in the National Radio Frequency Test Facility.

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6.3.4 Abstract of "Thermal and Electrical Joint Test for the Helical Field Coils in the Advanced Toroidal Facility"⁶

R. L. Brown and R. L. Johnson*

Initial feasibility studies of a number of configurations for the Advanced Toroidal Facility (ATF) resulted in the selection of a resistive copper continuous-coil torsatron as the optimum device considering the physics program, cost, and schedule. Further conceptual design work was directed toward optimization of this configuration and, if possible, a shorter schedule. It soon became obvious that in order to shorten the schedule, a number of design and fabrication activities should proceed in parallel. This was most critical for the vacuum vessel and the helical field (HF) coils. If the HF coils were wound in place on a complete vacuum vessel, the overall schedule would be significantly (12 months) longer. The approach of parallel schedule paths requires that the HF coils be segmented into parts of 180° of poloidal angle and that joints be made on a turn-by-turn basis when the segments are installed. It was obvious from the outset that the compact and complex geometry of the joint design presented a special challenge in the areas of reliability, assembly, maintenance, disassembly, and cost. Also, electrical, thermal, and force excursions are significant for these joints. A number of soldered, welded, brazed, electroplated, and bolted joints were evaluated. The evaluations examined fabrication feasibility and complexity, thermal-electrical performance at approximately two-thirds of the steady-state design conditions, and installation and assembly processes. Results of the thermal-electrical tests were analyzed and extrapolated to predict performance at peak design parameters. The final selection was a lap-type joint clamped with insulated bolts that pass through the winding pack.

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Chapter 7

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Fusion Engineering Design Center

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7. FUSION ENGINEERING DESIGN CENTER

SUMMARY OF ACTIVITIES

The Fusion Engineering Design Center is a major engineering resource of the national fusion program. Representatives from industry work with staff members from the national laboratories involved in fusion research to provide conceptual designs and cost estimates for future fusion reactor projects. For the past several years, activities have focused on the design of next-generation tokamak projects.

In the spring of 1985, the Department of Energy (DOE) directed the Design Center to take a lead responsibility in assessing the engineering feasibility of a very compact tokamak experiment with copper coils. A national effort addressed key issues for the proposed design options. The assessment was concluded in June with a workshop that endorsed the feasibility of this class of machine.

Following this assessment, the Design Center studied the Ignitor concept at the request of DOE and arrived at a design configuration. Many features of this configuration have been incorporated into the national baseline conceptual design for a Compact Ignition Tokamak (CIT).

The Design Center continued to participate in the mirror program by contributing to the Minimars design effort, a two-year program to develop and describe an attractive tandem mirror reactor concept. The Design Center's principal role is in configuration definition of the candidate concepts.

The Design Center continues to lead the engineering activities for the International Tokamak Reactor program. This work may form the basis for an expanded international project through the U.S.-U.S.S.R. summit process. The Design Center also provides the primary point of U.S. interaction with the Next European Torus (NET) and Fusion Engineering Reactor (FER) teams in Europe and Japan. Collaborative programs with the FER team are under way.

Advanced commercial tokamaks were studied by the Design Center as part of the Tokamak Power Systems Studies project coordinated by the DOE Office of Fusion Energy. The Design Center also provided design integration of the U.S. effort.

A cost accounting system¹ is applicable to all magnetic fusion reactor design studies was developed and applied to different confinement concepts and types of projects. The system provides the structure for development of a fusion cost database and validated cost estimating procedures.

7.1 ACTIVITIES IN SUPPORT OF THE COMPACT IGNITION TOKAMAK

Ignition studies carried out during the past several years have arrived at an impasse between the goals of the proposed devices and the ability to finance any device beyond a paper study. The earliest studies concentrated on devices that not only would achieve ignition but also would use the burning plasma for the engineering development required for a demonstration reactor. The very long operating times and engineering needed to support a reactor prototype led to the design of large (typical major radius ≈ 5 m), expensive devices with superconducting magnet systems.

A study undertaken in 1984 for a device called the Tokamak Fusion Core Experiment (TFCX) had less ambitious goals. The TFCX was proposed as an equilibrium burn ignition experiment, in which the burn would be held for 300 s to allow study of fully developed ignition for times that were long compared with the predicted skin times of the plasma. Configurations with both copper and superconducting coils were studied, the latter to represent the engineering needed for future reactor prototypes. The major radii of these devices ranged from 2.6 m for the smallest copper-coil device to 4.2 m for the largest superconducting device. The cost ranged from \$1000 million to \$1500 million, even if existing facilities were used, and exceeded the perceived resources of the U.S. program.

Studies undertaken by the Fusion Engineering Design Center (FEDC) in 1985 have pursued ignition as a scientific goal independent of engineering for a reactor prototype. The Compact Ignition Tokamak (CIT) has as its goal an ignition burn time of 10–12 times the energy confinement time, in the range of 3–5 s, rather than the 300-s burn time of the TFCX device. This type of device is expected to cost less than \$300 million, which is affordable in the present program.

The Design Center carried out a feasibility assessment of compact, high-field ignition devices, as described in Sect. 7.1.1, and developed an alternative version of the Ignitor concept discussed in Sect. 7.1.2, as a candidate option for the U.S. CIT program.

7.1.1 Abstract of *Interim Report on the Assessment of Engineering Issues for Compact High-Field Ignition Devices*²

C. A. Flanagan, J. R. Haines, D. B. Montgomery,^{*} F. A. Puhn,[†] C. A. Trachsel,[‡]
W. R. Hamilton, T. G. Brown, V. D. Lee, and H. Furth^{**}

Several concepts have been under investigation as candidates for a near-term tokamak experiment. All concepts are compact (1- to 2-m major radius) and feature high-field, high-plasma-current operation. During approximately the past year, scoping studies have been under way by three different groups; these include the Ignitor concept¹ guided by Prof. B. Coppi of the Massachusetts Institute of Technology (MIT), the LITE concept guided by Dr. Dan Cohn at MIT, and concepts studied at the Princeton Plasma Physics Laboratory (PPPL) in the Ignition Studies Project guided by Dr. John Schmidt. In June 1985, an Ignition Design Point Workshop was hosted by the United States, at which the status of the studies was summarized for an international review panel. The advocate groups presented the status of the design efforts. The current understanding of the physics

database was reviewed. Also, a session was devoted to a preliminary assessment of the engineering issues associated with such compact high-field devices.

The engineering issues addressed at the workshop included the overall configuration, layout, and assembly; limiter and first-wall energy removal; magnet system structure design; fabricability; repairability; and costs. In performing the assessment, the primary features and characteristics of each concept under study were reviewed as representative of this class of ignition device. The emphasis was to understand the key engineering areas of concern for this class of device and deliberately not attempt to define an optimum design or to choose a best approach.

The assessment concluded that compact ignition tokamaks, as represented by the three concepts under study, are feasible. A number of critical engineering issues were identified, and all appear to have tractable solutions. The engineering issues appear quite challenging, and to obtain increased confidence in the apparent design solutions requires completion of the next level of design detail, complemented by appropriate development programs and testing.

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7.1.2 FEDC Version of Ignitor

The Ignitor concept¹ has been advocated for several years as a means to achieve ignition in a compact device using state-of-the-art technology and little or no auxiliary heating. The Design Center developed an alternative version of the Ignitor design as a candidate for the U.S. CIT program. The projected incremental cost of the device, if it is located at PPPL, is attractive. The principal engineering features of the concept are as follows.

1. Precompression of the toroidal field (TF) coil inner leg is achieved by a hydraulic press system. The system applies a preload of about 178 MN (40 million lb) vertically and operates at room temperature.
2. Out-of-plane loads are carried by an external torque structure.
3. All poloidal field (PF) coils are external to the TF coils.
4. The TF coils are high-strength copper plates.
5. The ohmic heating (OH) solenoid operates at uniform current density and is in contact with the TF coils.
6. The magnet system operates at liquid nitrogen (LN₂) temperature.

The device design, the rf launcher design, the expected operating and performance characteristics, and representative system studies carried out in support of the design are described in this section.

7.1.2.1 Configuration design

The FEDC version of Ignitor is shown in Figs. 7.1 and 7.2. Key parameters are given in Table 7.1.

Fig. 7.1. FEDC Ignitor concept.

Precompression structure

The compactness of the design is achieved in part by precompression of the TF coil inner legs using a hydraulically driven press. The vertical preload applied to the coil legs is about 178 MN. Three options were considered for applying the preload: a disk concept, sand cast from stainless steel (SS), as shown in Fig. 7.1; a titanium thrust cone; and a built-up picture-frame SS plate. All three options are feasible, and none adds a large cost to the overall device. The low-cost, commercially available hydraulic press system consists of 24 cylinders around the upper periphery of the preload assembly. Each cylinder contributes about 7.4 MN (1.7 million lb) [at about 69 MPa (10,000 lb/in.²)]. The system operates at room temperature. Thermal insulation isolates the cold reactor module from the warm press.

TF coils

The TF coil system consists of 24 coils, each made of 10 plates of 1.7-cm-thick, high-strength copper. The coil bore is about 1.0 by 1.6 m. The radial span of the inner leg is

Fig. 7.2. Exploded view of FEDC Ignitor reactor module.

Table 7.1. Key parameters of the FEDC Ignitor

Major radius, m	1.16
Minor radius, m	0.42
Aspect ratio	2.7
Elongation	1.8
Triangularity	0.2
Plasma current, MA	10.0
Field on axis, T	11.7
Safety factor	2.6
Plasma beta, %	6.1
Fusion power, MW	400.0
Neutron wall load, MW/m ²	10.5
Volt-seconds required, V·s	25.0
Coil materials	
TF coils	OFHC copper
OH solenoid	BeCu
PF coils	OFHC copper

≈ 20 cm; that of the outer leg, ≈ 25 cm. The coils wedge together in the nose region and buck against the OH solenoid. The pulse length is on the order of 3–5 s, during which the coils are inertially cooled. The device is cooled to LN_2 temperature (about 77 K) after each pulse.

OH solenoid

The OH solenoid is wound onto the central compression support structure and is radially divided into two parts, as shown in Fig. 7.3. The current density is uniform throughout the solenoid, and single entry and exit electrical leads are located at the top. The inner section turns are explosively bonded to stainless steel rings, as shown in Fig. 7.4. This part of

Fig. 7.3. Split solenoid concept.

TURN-TURN TRANSITION OF THE BITTER PLATES

Fig. 7.4. Support of split solenoid Bitter plate turn from stainless steel rings.

the solenoid experiences higher loads because it operates in the high-field region and must be made of a high-strength material. The outer solenoid winding, which operates in a lower field, can be made of a softer material with lower resistivity to minimize the electrical losses.

A preliminary design of a solenoid for an Ignitor-type device has an ampere-turns requirement of 334 MAT; inside and outside radii of 25.4 and 45.4 cm, respectively; a height of 200 cm; and a current density in the conductor of 10 kA/cm^2 (overall current density of 8500 A/cm^2). The magnetic field in the solenoid bore is 21.4 T, giving an average stress in the solenoid winding of 380 MPa. If only one conductor material could be used for this winding, it would have to be CDA-175 (beryllium-copper), which has a yield strength of 820 MPa and an ultimate strength of 860 MPa. This material has 50% higher electrical resistivity than CDA-155 (silver-copper). With the two-part design approach,

however, the inner part of the solenoid can be made of CDA-175, and the outer part, which undergoes much lower stress (500 MPa), can be made of CDA-102 copper, which has 10% lower resistivity than CDA-155. The electrical loss in the two-part CDA-175/CDA-102 solenoid is only 10% higher than it would be in a CDA-155 winding.

PF coils

The PF coil system consists of ring coils, all external to the TF coils. They are separate from the external torque frame, which carries the out-of-plane loads.

Access ports

Access ports are located between each pair of TF coils at the outboard horizontal midplane. These ports are designed for installation of auxiliary heating systems [ion cyclotron resonance heating (ICRH)], fueling, and diagnostics. Minimum penetration size is about 20 by 80 cm.

Limiter requirements

The FEDC Ignitor device will generate between 100 and 400 MW of fusion power; the corresponding neutron wall loads range from 5 to 15 MW/m². The detailed design of limiters to withstand such an environment is quite challenging. However, preliminary assessments, as described in ref. 2, indicate that design solutions exist.

Maintenance approach

Two basic approaches to maintenance and disassembly have evolved: an arrangement with no shielding at the device and location of most peripheral equipment outside the test cell, as shown in Fig. 7.5, and an arrangement with shielding close to the reactor and around peripheral equipment, which is located in the test cell, as shown in Fig. 7.6. The first approach provides easier access to the plasma chamber and minimal activation of peripheral equipment. The second approach provides close-in bulk shielding and minimizes activation of the equipment in the test cell. The final arrangement will depend on the location of peripheral equipment and the penetrations needed and may contain features from both approaches.

Structural analysis

In a contained magnet, the design-allowable stresses³ are 70% of ultimate stress or 85% of yield stress. A NASTRAN finite element model that represented all key components [center post, OH solenoid, TF coils, upper inner equilibrium field (EF) coil, and outboard load reacting ring] was used to perform an in-plane analysis,⁴ which included the preload, TF and OH forces, and temperature effects (including nuclear-induced contributions). In-plane and out-of-plane stresses are below allowable values. Figure 7.7 is a stress contour plot of the maximum shear stress within the TF coil that shows average Tresca stresses at three cross sections for a vertical preload of 178 MN.

Fig. 7.5. Test cell arrangement with unshielded reactor.

Fig. 7.6. Test cell arrangement with shielding around reactor.

Fig. 7.7. Preload and TF forces calculated with NASTRAN.

7.1.2.2 RF system design

One of several candidate rf launchers is shown in Fig. 7.8. This launcher design⁵ covers the range of frequencies and loading impedances expected for CIT machines and yields reasonable component values and power handling characteristics. A version of this launcher is being designed and integrated into the port configurations for proposed CIT machines, including the FEDC version of Ignitor.

7.1.2.3 Operation and performance

About 3,000 full D-T pulses and about 50,000 half-force pulses are planned. Each pulse has a flat-top period of 3–5 s, during which the temperature in the TF coils increases to perhaps 350 K. Cooling times of 1–2 h between pulses may be required.

Current waveforms for the TF coils, the plasma, the OH solenoid, and the EF coils are shown in Fig. 7.9. The temperature of the conductor in the inner and outer legs of the TF coils, the energy and power consumed by each TF and PF coil, the megavolt-amperes required by each TF and PF coil, and the voltages at the terminals of each coil are calculated as functions of these waveforms.

Typical performance curves for CDA-102 TF and PF coils are shown in Figs. 7.10 and 7.11. Figure 7.10 shows coil current waveforms for both coil sets and instantaneous temperature in the inner and outer TF coil legs, and Fig. 7.11 shows power (I^2R) loss and energy consumption. Figure 7.12 shows the impact of the flat-top burn period on peak temperature in the TF coil inner leg. An appropriate burn time can be selected based on physics and structural considerations.

Fig. 7.8. ORNL resonant double loop antenna.

Fig. 7.9. Current waveforms for TF coils, plasma, OH solenoid, and EF coils.

Fig. 7.10. (a) Cell current waveforms for TF and PF coils and (b) temperatures in the inner and outer legs of TF coils and in PF coils made with CDA 102 conductor.

Fig. 7.11. (a) Coil power (I^2R) and energy consumption for TF and PF coils made with CDA 102 conductor.

Fig. 7.12. Temperature of TF coil inner leg as a function of burn time for CDA 102 and CDA 155 conductor. EOB: end of burn. EOC: end of cycle.

7.1.2.4 System studies

The design of the CIT involves trade-offs of engineering and physics variables and constraints. Change in one requirement often affects the ability to satisfy several constraints, which greatly increases the difficulty of identifying feasible designs.

To facilitate studies of the design space, the FEDC Tokamak Systems Code (TSC)⁶ has been coupled to a numerical optimization routine, VMCON.⁷ The coupled code simultaneously varies prescribed design space variables and searches for a combination that satisfies the equations for the physics and engineering constraints and also maximizes (or minimizes) a prescribed figure of merit. Up to 32 physics and engineering variables can be simultaneously iterated subject to 21 constraints; some of these variables and constraints are listed in Table 7.2. This setup provides flexibility for performing trade studies and quickly assessing the impact of design changes.

During the definition of the FEDC Ignitor, system studies were conducted using the TSC-VMCON coupled code and an improved version of the TSC. The TSC-VMCON code was used to select a given region of parameter space. In Fig. 7.13, the Kaye-Goldston ignition contours for an Ignitor candidate are shown in R - B space, where R is major radius and B is magnetic field, with lines of constant aspect ratio A and constant figure of merit $B^2 a(\kappa)^{1/2}/q$ superimposed. For ignition (Kaye-Goldston ignition margin = 1), the minimum major radius is at $R = 0.99$ m with $A = 3.1$ and $B = 14.5$ T. (This analysis is

Table 7.2. Variables and constraints used in the new version of the Tokamak Systems Code

Variables	Constraints
Physics	
Field on axis	Poloidal beta
Poloidal beta	Plasma pressure
Ignition margin	Energy balance
Aspect ratio	Density limit ^a
Major radius	Beta limit ^a
Electron (ion) temperature	Nonthermal beta
Engineering	
Solenoid bore	Radial build
Solenoid thickness	Volt-seconds ^a
TF coil thickness	TF coil stress ^a
TF coil current density	Solenoid stress ^a
Solenoid current density	Wall load limit ^a
Burn time	Field at TF coil
Ramp time	Energy requirement ^a
Power requirement ^a	
Peak MVA requirement ^a	

^aInequality equations with limits.

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Fig. 7.13. Contours of constant Kaye-Goldston ignition margin for a candidate version of Ignitor in R - B space with aspect ratio and figure of merit curves superimposed.

not for the final FEDC Ignitor configuration, but the procedure to find the minimum R is the same.)

An upgraded version of the TSC code was used to find final values of current, beta, safety factor, and other parameters after the parameter space was narrowed using the TSC-VMCON coupled code. A flow diagram for the TSC is shown in Fig. 7.14. The FEDC equilibrium code has been incorporated as a subroutine to compute PF coil currents. An option has been added to return the value of the plasma current, determined with the MHD equilibrium code, to the plasma physics module so that self-consistency can be attained. (Previously, the plasma physics module used a plasma current based on scaling relationships.) A new plasma physics subroutine⁸ performs a steady-state, two-fluid power balance for the ions and electrons subject to constraints imposed by the machine configuration and the plasma equilibrium, stability, and confinement. One of the most comprehensive sections of this subroutine contains the energy transport models; the Chang-Hinton neoclassical formulation and several empirical scalings (including neo-Alcator, Mirnov, Kaye-Goldston, Perkins, ASDEX, and Pfeiffer-Waltz) make it possible to determine the impact of specific confinement models on performance and design.

The plasma physics subroutine, which is used in both the improved TSC code and the coupled code, can also be used in a standalone mode to compare the plasma performance of specific design concepts by generating plasma operating contour (POPCON) plots. Steady-state auxiliary power is plotted in density-temperature space, showing the steady-state heating and operating windows and minimum auxiliary power requirements for a

Fig. 7.14. Flow diagram for revised TSC.

given device, as indicated by Fig. 7.15. Figure 7.16 is a POPCON plot for a candidate Ignitor design.

7.2 ACTIVITIES IN SUPPORT OF MINIMARS

The Design Center is involved in a collaborative two-year study directed by Lawrence Livermore National Laboratory to describe an attractive tandem mirror reactor called Minimars, which would produce a net electrical power of 600 MW. The mission of Minimars is to provide an inherently safe fusion reactor with a competitive cost of electricity (COE), <50 mill/kWh, and a mass utilization of more than 100 kW/tonne. Minimizing the risk of major investment loss if the reactor should fail is a third major goal.

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Fig. 7.15. Plasma operating regimes.

Fig. 7.16. Contours of auxiliary heating power for Kaye-Goldston scaling.

The first year of the study has emphasized innovative concepts and trade studies that lead to good cost vs performance ratings.⁹ A set of baseline parameters, a preliminary engineering description, and a first cost estimate have been generated.

7.2.1 Minimars System

One conclusion of the Mirror Advanced Reactor Study (MARS) program was that an improved end-cell stabilization technique was needed to make the tandem mirror more competitive with other fusion reactor types. To this end, Minimars uses an octopole magnet system rather than the MARS quadrupole. Other stabilization techniques have been investigated, but none has replaced the octopole as the most desirable. The magnetic profile of Minimars in Fig. 7.17 shows the vacuum magnetic field at the important regions of the reactor: in the central cell, at the choke and transition coils, in the anchor mirror cell, and at the halo scraper/direct converter.

Other features of Minimars are low-ripple central-cell magnetics, a high-field choke coil, a negative-ion neutral beam injector for sloshing ions, an rf-heated electron mantle, a halo scraper, and a direct converter. Table 7.3 lists some important parameters.

Fig. 7.17. Magnetic profile for the Minimars reactor.

Table 7.3. Minimars characteristics

Fusion power, MW	1237
Net electrical power, MW	600
Neutron wall loading (center), MW/m ²	3.3
Total launched rf and NBI power, MW	38
Q (fusion power/absorbed heating power)	34
Recirculating power, MW	135
Engineering Q (electrical power out/recirculated power)	5.4
Machine length, m	145
Central cell	
Solenoid length, m	90
First-wall radius, m	0.57
Vacuum vessel principal diameter, m	9.3
Availability, %	75
Blanket	
Cooling	Helium
Structural material	Ferritic steel
Breeder material	LiPb
Tritium breeding ratio	1.067
Energy multiplication factor	1.44
Maximum heating to coil case, mW/cm ³	0.1
Maximum dose to electrical insulation, rad	10 ¹²
Number of central cell modules	24
Choke coil inner radius, m	0.17
Field on axis, T	24
Power (dc) from direct converter, MW	84
Large octopole	
Structure	Four window-frame
Cooling	Forced-flow Nb ₃ Sn
Thermal power from direct converter, MW	152

7.2.2 Reactor Configuration

As shown in Fig. 7.18, the Minimars reactor is 145 m long and 9.3 m in diameter along its principal length. Its centerline is approximately 9 m above the floor. The reactor is assembled from central-cell modules, a fueling and instrumentation sector, and two end cells, each containing a magnet set, a beam injector, rf systems, a halo scraper, and a direct converter.

7.2.2.1 Central-cell arrangement

The 90-m-long central cell is assembled from 24 interchangeable central-cell modules, each 3.725 m long, and a fueling/instrumentation module, which is 0.4 m long.

The general arrangement of a central-cell module is shown in Fig. 7.19. A solenoid coil surrounds nuclear radiation shielding and a blanket module. The ≈ 0.38 -m-thick nuclear shield, which is made from stainless steel, B₄C, and lead, protects the solenoid from neutron damage and limits the nuclear heating of the windings to 0.1 mW/cm³.

ORNL-PHOTO 7516-86

Fig. 7.18. Typical Minimax configuration.

ORNL-PHOTO 7518-86

Fig. 7.19. Arrangement of central-cell module.

The sheet-type solenoid has six winding layers and a 3000-A/cm^2 current density. Pool-boiling NbTi was chosen as the baseline superconductor to minimize busbar size and to simplify the liquid helium (LHe) piping. The central cells are placed in individual vacuum dewars to enhance reliability and maintainability. The primary plasma vacuum chamber extends from the magnet dewar and encloses the shield, blanket, and poloidal coolant manifolds.

7.2.2.2 End-cell arrangement

Each end cell is configured around an octopole-based magnet set, which consists of a choke coil, a transition octopole, the large octopole, and an outer mirror coil set, as shown in Fig. 7.20. The choke coil consists of a background superconducting coil cooled with helium II and a water-cooled copper winding insert, which provides about 8 T of the total 24-T field on axis. The large octopole field is produced by an assembly of four window-frame windings. The end mirror coil consists of four large-diameter, nearly concentric short solenoids.

In addition to the coil set, each end cell contains negative-ion neutral beams, electron cyclotron resonance heating (ECRH) and drift pumping systems, a halo scraper and direct

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Fig. 7.20. Octopole-based magnet set.

converter, shielding, a mantle limiter, and a microwave reflecting surface. The arrangement is shown in Fig. 7.21.

The neutral beam system is positioned to produce the sloshing-ion distribution in each end plug. All three ECRH systems are located in the end plug. The drift pumping system, located under the small octopole in the initial configuration, removes particles trapped in the core plasma between the inner mirror point and the potential peak.

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Fig. 7.21. End-cell systems assembly.

7.2.2.3 Halo scraper and direct converter

Minimars is bounded on each end by systems designed to convert the terminal core and halo plasmas into electricity through both direct potential and thermal energy recovery conversion. Helium-cooled plates made of TZM (a molybdenum alloy containing titanium and zirconium) intercept the plasma approximately 12 m beyond the end-cell coil set. Several materials, including B_4C , BN, and BeO, are being considered for a sputter layer that must be bonded to the base material. Power is directly recovered at the 170-kV potential between the core plasma and the halo. Thermal power is transported to steam generators through a helium cooling system.

7.2.3 Fueling

A laser-driven pellet injector system is proposed for fueling Minimars. Laser beams are combined 20 to 30 m from the reactor and transmitted perpendicular to the central-cell plasma axis. Two pneumatic gun injectors are provided to inject 3.1-mm-diam pellets at a maximum rate of 10 pellets per second in the same plane as the laser beam and crossing the beam axis at an angle of about 5°. The laser beam is focused near the crossover point. When the pellet crosses the beam axis, the lasers are triggered to provide a 100-ns pulse of energy, which rapidly ablates the rear part of the pellet. The reaction thrust increases the velocity of the front part of the pellet, which reaches the desired velocity of 46 km/s.

7.2.4 Facilities

A conceptual facilities layout, shown in Fig. 7.22, was developed for Minimars based on the assumption that the security requirements and the support facilities (e.g., the non-radioactive maintenance shop and administration building) for the fusion plant will be similar to those for a light-water reactor (LWR) plant. The total area shown for this plant is approximately the same as that for an LWR plant.

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Fig. 7.22. Minimars facility layout.

Crane access requirements, minimum process piping and electrical cable length, and proximity of a building to areas served by building systems dictate the arrangement of the cryogenics, power conditioning, turbine, tritium, and steam generator buildings around the central reactor vault. Safety and efficiency dictate the centralization of the decontamination, hot cell, activated waste, and storage pool areas.

The primary services required to operate the reactor and convert power from the plasma to a useful form include the steam generators, the rf heating systems, and the neutral beam systems. Figure 7.23 shows the principal piping required to circulate coolant between the blanket modules and the steam generators; the LHe, LN₂, and water piping; and the vacuum and electrical lines to the magnets.

7.2.5 Cost

The FEDC costing methodology (see Sect. 7.5) was applied to Minimars. The first scoping estimate of the total projected capital cost was \$1487 million in mid-1985 dollars, based on (1) a Minimars interim Baseline III, (2) an undeveloped, non-government-owned site, and (3) a tenth-of-a-kind reactor (which eliminated design engineering costs). With leveled costing methods, this yields a COE of <50 mill/kWh. This estimate is by nature

ORNL-PHOTO 7519-86

Fig. 7.23. Reactor operation and power conversion services on Minimars.

speculative because it is based on concepts, not designs. Even when final parameters for Minimars become available in mid-1986, estimating costs for a reactor to be implemented 50 years from now will be speculative and will remain so until 2035.

7.2.6 Conclusions

The first year of the Minimars study has produced an established set of system concepts and parameters integrated into an overall design. The output of the tandem mirror system optimization code was used to establish a minimum-cost configuration. Some design studies and analyses that identify hardware requirements and establish spatial relationships and interface requirements were performed. The design concept will be used as the starting point for next year's more detailed engineering studies, which will incorporate the design guidelines and requirements of the Minimars program and achieve a self-consistent overall design.

7.3 ACTIVITIES IN SUPPORT OF INTOR

The Design Center continued to play a major role in the International Tokamak Reactor (INTOR) program during the Phase Two A, Part 2 Workshop completed in 1985. In the U.S. INTOR program, the Design Center led the efforts on the critical issue of maintainability, on the compilation of the physics and engineering databases for INTOR, and on the assessment of the technical benefit of an international project such as INTOR; in addition, the Design Center performed the design integration and systems engineering for the development of a modified INTOR design.¹⁰ Highlights are presented here.

7.3.1 Maintainability

The Design Center's evaluation of the critical issue of maintainability compared an INTOR designed for totally remote maintenance with one designed for limited personnel access to allow "hands-on" maintenance in the test cell (external to the device itself). The study developed the basis for both designs, identified the R&D requirements, and compared the designs in terms of induced activation levels, shielding requirements, maintenance equipment, capital and operating costs, and flexibility of operation. The conclusion was that a design based on limited personnel access is the most effective, given the maintenance considerations. The impact of the choice on reactor configuration, size, and cost was not dramatic. The only clear consideration was flexibility for normal and off-normal operations, which is best satisfied for an experimental device by the design with limited personnel access.

7.3.2 Database Assessments

The physics database assessment compiled the current understanding relevant to INTOR in the areas of stability limits, confinement, neutral beam heating and current drive, operating scenario, burning plasma considerations, and plasma diagnostics.

The engineering database assessment compiled the current developments relevant to INTOR in the areas of systems engineering, magnets, vacuum enclosures and technology,

neutral beam heating systems, pellet injection, and radiation-hardened diagnostics and instrumentation.

7.3.3 Technical Benefit of an International Project

The Design Center evaluated the technical feasibility of partitioning the design and component fabrication of an international project such as INTOR so that all participants would benefit from the development of all advanced technologies. Cost and schedule implications that were developed showed the degree of benefit from the partitioning. The conclusion was that a significant cost benefit could be realized; for four participants, the cost per participant was 30–40% of the cost incurred by a single participant building the entire project. A construction schedule was developed to forecast the delays that would arise in an international project rather than a national project.

7.3.4 Modified INTOR Design

The current understanding of fusion physics and technology led to the development of a slightly modified INTOR design. The aim was to reduce the major radius to about 5.0 m from 5.3 m and increase the plasma current to about 20 MA from 6.3 MA.

The modified design incorporates the following features:

1. A thinner inboard shield (70 cm, down from 83 cm).
2. A smaller TF coil radial thickness (0.547 m, down from 0.813 m) and bore.
3. Single torus sectors with split blanket modules.
4. A split divertor module (12 to 24 sectors).
5. A smaller major radius (5.0 m, down from 5.3 m).
6. A larger OH bore (1.24 m, up from 1.15 m).
7. A modified PF coil system, to satisfy the MHD requirements.

The modified design is shown in Figs. 7.24 and 7.25. Table 7.4 lists key parameters for the modified design.

7.4 OTHER ACTIVITIES

7.4.1 Abstract of *Advanced Commercial Tokamak Study*¹¹

S. L. Thomson, A. E. Dabiri, D. C. Keeton, T. G. Brown, and G. T. Bussell*

Advanced commercial tokamak studies were performed by the Design Center as a participant in the Tokamak Power Systems Studies project coordinated by the Office of Fusion Energy. The FEDC studies addressed the issues of tokamak reactor cost, size, and complexity. A scoping study model was developed to determine the effect of beta on tokamak economics, and it was found that a competitive COE could be achieved at a beta of 10 to 15%. The implications of operating at a beta of up to 25% were also addressed. It was found that the economics of fusion, like those of fission, improve as unit size increases. However, small units were found to be competitive as elements of a multiplex plant, provided that unit cost and maintenance time reductions are realized for the small units. The

Fig. 7.24. Elevation view of modified INTOR with 5.0-m major radius.

modular tokamak configuration combined several new approaches to develop a less complex and lower cost reactor. The modular design combines the TF coil with the reactor structure, locates the primary vacuum boundary at the reactor cell wall, and uses a vertical assembly and maintenance approach.

*Stone & Webster Engineering Corporation, Boston.

Fig. 7.25. Plan view of INTOR torus sector and split divertor module.

Table 7.4. Modified INTOP. parameters

Inboard shield thickness, cm	70
TF coil current density, A/cm ²	2900
Minor radius a , m	1.2
Major radius R , m	5.0
Aspect ratio A	4.167
Plasma beta β (DT)	0.041
Safety factor q	2.4
Fusion power P_{fus} , MW	~570
W_L , MW/m ²	~1.3
High-field current I_{HB} , MA	8.04
Low-field current I_{LB} , MA	7.56
Toroidal field B_t , T	5.5
Maximum field on coil B_{max} , T	10.96
OH coil bore, m	1.244
TF coil thickness, m	0.547

7.4.2 Abstract of *Cost Accounting System for Fusion Studies*¹²

W. R. Hamilton, D. C. Keeton, and S. L. Thomson

A cost accounting system that is applicable to all magnetic fusion reactor design studies has been developed. This system provides (1) definitions of the elements of cost and methods for the combination of these elements to form a cost estimate, (2) a code of accounts that uses a functional arrangement for identification of the plant components, and (3) definitions and methods to analyze actual cost data so that the data can be directly reported into this cost accounting system.

The purpose of the cost accounting system is to provide the structure for the development of a fusion cost database and for the development of validated cost estimating procedures. This system has been developed through use at the FEDC and has been applied to different confinement concepts (tokamaks and tandem mirrors) and to different types of projects (experimental devices and commercial power plants). The use of this cost accounting system by all magnetic fusion projects will promote the development of a common cost database, allow the direct comparison of cost estimates, and ultimately establish the cost credibility of the program.

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Chapter 8

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Materials Research and Development

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8. MATERIALS RESEARCH AND DEVELOPMENT

SUMMARY OF ACTIVITIES

This chapter covers research on and development of alloys for fusion reactor structural applications, special-purpose materials such as ceramics, and materials for superconducting magnets. Research on plasma-materials interactions is presented in Chap. 4.

Our Alloy Development for Irradiation Performance (ADIP) and Damage Analysis and Fundamental Studies (DAFS) programs are concerned primarily with fusion reactor structural materials and have as their objectives:

1. developing an understanding of the mechanisms involved in radiation damage and corrosion phenomena;
2. providing information on materials properties necessary for the design of near-term fusion devices and for fusion power reactor conceptual design and trade studies; and
3. developing a sound technological basis for the development of materials that will allow fusion to reach its full potential.

Our efforts to develop austenitic stainless steels are carried out in collaboration with Japan under the agreement for U.S.-Japan Collaborative Testing of First Wall and Blanket Structural Materials. In advanced austenitic stainless steels such as PCA and JPCA (U.S. and Japanese prime candidate austenitic stainless steels, respectively), compositions have been developed that are more stable to the formation of massive second phases than are the standard austenitic steels (such as type 316 stainless steel) and in which a fine dispersion of matrix and grain boundary MC precipitates can be formed. These precipitates trap helium on a very fine scale, thus extending the incubation period for void swelling and increasing the temperature at which helium embrittlement limits performance or use. Spectral tailoring irradiation experiments in the Oak Ridge Research Reactor and the High-Flux Isotope Reactor are being used to explore irradiation response at He/dpa levels of the fusion spectrum and at temperatures (60–400°C) and damage levels (up to 50 dpa) appropriate to near-term reactors such as the Fusion Engineering Reactor. Other experiments explore higher damage levels to determine the maximum useful life of such alloys and to provide understanding on which further development can be based. Exploratory research—including basic physical metallurgical studies, Fast-Flux Test Facility (FFTF) and dual-ion irradiations, and liquid-metal coolant compatibility studies—has continued on manganese-stabilized austenitic stainless steels. These alloys are possible low-activation structural alloys.

Radiation effects in vanadium alloys are being explored employing the elevated-temperature tritium trick to introduce fusion-relevant helium levels and subsequent irradiation in the FFTF to create displacement damage. Three alloys (V-15 Cr-5 Ti, V-3 Ti-1 Si, and VANSTAR 7) were included in the initial experiments. The initial results show that the V-15 Cr-5 Ti alloy is most susceptible to irradiation hardening and helium

embrittlement, followed by VANSTAR 7 and V-3 Ti-1 Si. VANSTAR 7 exhibited relatively high swelling, nearly 6% at 520°C and 40 dpa, when preinjected with 70 appm He. The other two alloys exhibited much lower swelling (<0.3%). The vanadium alloys appear to be subject to the same forms of radiation damage as other alloy systems. The present results provide a starting point for alloy development.

In ferritic steels we have continued to scope the response of our reference alloys HT-9 and 9 Cr-1 Mo to neutron damage, with particular emphasis on impact and tensile properties. Our approach to developing low-activation versions of the ferritic steels is to explore the metallurgical consequences of substituting tungsten or vanadium for molybdenum and niobium in an experimental set of alloys with chromium ranging from 2% to 12%. Although our initial studies are promising, a great deal remains to be done to investigate the ductile-to-brittle transition, high-temperature strength, corrosion properties, and radiation response of this new class of materials.

Corrosion studies have the objectives of (1) establishing compatible structural material, coolant, and breeder combinations and (2) understanding the mechanisms of corrosion and mass transfer to provide a basis for predicting kinetics in reactor systems and developing techniques for control. Emphasis has shifted to low-activation alloys, with continuing research on standard alloys to aid development of corrosion-resistant materials and possible inhibitive techniques for liquid-metal environments.

The program on insulators and stabilizers for superconducting magnets (the Special-Purpose Materials Program) has focused on providing facilities at the National Low-Temperature Neutron Irradiation Facility for research in the area of radiation effects on magnet materials. This facility will provide temperatures and irradiation conditions similar to those at the magnets of a fusion reactor.

8.1 ALLOY DEVELOPMENT FOR IRRADIATION PERFORMANCE

8.1.1 Austenitic Stainless Steels—Abstracts

8.1.1.1 *Effects of Helium Content on Microstructural Development in Type 316 Stainless Steel Under Neutron Irradiation*¹

P. J. Maziasz

This work investigated the sensitivity of microstructural evolution, particularly precipitate development, to increased helium content during thermal aging and during neutron irradiation. Helium (110 appm) was cold preinjected into solution-annealed (SA) DO-heat type 316 stainless steel (316) via cyclotron irradiation. These specimens were then exposed side by side with uninjected samples. Continuous helium generation was increased considerably relative to Experimental Breeder Reactor (EBR-II) irradiation by irradiation in the

High-Flux Isotope Reactor (HFIR). Data were obtained from quantitative analytical electron microscopy (AEM) in thin foils and on extraction replicas.

Normal aging produced small amounts of $M_{23}C_6$ (τ) precipitation at 560 to 600°C and τ plus large amounts of Laves phase at 650 to 700°C. Preinjected helium eliminated τ and replaced it with Laves at 600 and 700°C. Interstitial "black-dot" Frank loops were introduced via preinjection. These grew and coarsened upon long-term aging at 400 to 600°C, and then unfaulted at 700°C, apparently to provide vacancies for growing helium-vacancy clusters and bubbles.

Normal EBR-II irradiation produced dislocation loops and network, precipitate-associated and matrix voids, and precipitation of mainly M_6C (η) plus a minor assortment of irradiation-produced phases [G , γ' (Ni_3Si) plus A and B phosphides] at 500 to 630°C and 8.4 to 36 dpa (except at 630°C and 36 dpa, where Laves plus η formed). Helium preinjection eliminated void formation and eliminated radiation-induced solute segregation (RIS)-induced or RIS-modified precipitation at 500 to 625°C and 8.4 dpa. Early fine bubble nucleation apparently caused an overwhelmingly bubble-dominated sink structure that also diluted RIS to eliminate precipitation at 500°C and yet allow enhanced thermal precipitation at 625°C.

The results of HFIR irradiation at 500 to 650°C were intermediate to those described above due to intermediate levels of bubble nucleation and evolution. Void formation was initially suppressed but then, at higher fluences, enhanced; maximum RIS was observed at 425 to 450°C, but RIS intensity was reduced at 515 to 555°C and virtually eliminated at higher temperature. Enhanced thermal precipitation dominated above 550°C, and void swelling appeared cut off above 650°C.

This thesis includes an extensive literature survey of data on SA 316 and of relevant theoretical work to provide the perspective necessary to discuss and understand the data.

8.1.1.2 "A Perspective on Present and Future Alloy Development Efforts on Austenitic Stainless Steels for Fusion Application"²

P. J. Maziasz

High-fluence neutron data on advanced titanium-modified austenitic stainless steels have only recently become available, and experiments on helium effects in these alloys are currently in progress. Helium-enhanced void swelling and grain boundary embrittlement potentially impose the most severe temperature and lifetime limits for fusion first walls. However, controlled precipitation of MC/bubble dispersions can improve the resistance to both swelling and embrittlement of the advanced titanium-modified alloys. Fine stable bubble microstructures also cause dilution of RIS, which affects swelling and possibly other properties as well. Lower fluence data from the HFIR and the Oak Ridge Research Reactor (ORR) suggest that helium does not degrade the properties of fatigue and creep in these alloys under irradiation, but further study is required. Finally, a new class of alloys is being developed for lower induced radioactivity in which manganese replaces nickel while the concentrations of molybdenum, copper, and nitrogen are severely restricted.

8.1.2 Ferritic Steels—Abstracts

8.1.2.1 "Tensile Behavior of Irradiated 12 Cr-1 MoVW Steel"³

R. L. Klueh and J. M. Vitek

The effect of irradiation on the tensile properties of 12 Cr-1 MoVW steel given two different normalized-and-tempered heat treatments was determined for specimens irradiated in EBR-II at 390, 450, 500, and 550°C to ~13 dpa. Tensile tests were conducted at the irradiation temperature and room temperature on the irradiated specimens, as-heat-treated specimens, and as-heat-treated specimens thermally aged 5000 h at the irradiation temperature.

Irradiation at 390°C increased the 0.2% yield stress and ultimate tensile strength above the strength of the unaged and aged controls for both heat-treated conditions. The effect was greater for the steel heat treated with the shorter austenitization and tempering times. The increased strength was accompanied by a slight decrease in ductility. After irradiation at 450, 500, and 550°C, the yield stress, ultimate tensile strength, and ductility were similar to the as-heat-treated and the thermally aged controls. Strength changes were discussed in terms of microstructural changes observed in other studies.

8.1.2.2 "Tensile Properties of 2½ Cr-1 Mo Steel Irradiated to 23 dpa at 390 to 550°C"⁴

R. L. Klueh and J. M. Vitek

The effect of neutron irradiation on the tensile properties of normalized-and-tempered 2½ Cr-1 Mo steel was determined for specimens irradiated to 23 dpa in EBR-II at 390, 450, 500, and 550°C. Two types of unirradiated control specimens were tested: as-heat-treated specimens and as-heat-treated specimens thermally aged for 10,000 h at the irradiation temperatures. Aging caused a loss of strength at all four temperatures. Irradiation at 390°C increased the strength and decreased the ductility compared to the control specimens. At 500 and 550°C, softening occurred relative to the thermally aged specimens, with the amount of softening greatest at 550°C. At 450°C, the strength of the irradiated specimens was similar to a specimen aged 10,000 h at 450°C. The changes in tensile properties resulted from displacement damage caused by the irradiation and by irradiation-induced changes in carbide precipitates that occurred during elevated-temperature exposure.

8.1.2.3 "Elevated-Temperature Tensile Properties of Irradiated 9 Cr-1 MoVNb Steel"⁵

R. L. Klueh and J. M. Vitek

Normalized-and-tempered 9 Cr-1 MoVNb steel tensile specimens were irradiated in EBR-II at 390, 450, 500, and 550°C to ~ 2.1 and 2.5×10^{26} neutrons/m² (>0.1 MeV), which produced displacement damage levels of ~10 and 12 dpa, respectively. Tensile tests were conducted at the irradiation temperature and at room temperature. In addition to the irradiated specimens, as-heat-treated specimens and as-heat-treated specimens thermally aged at the irradiation temperature for 5000 h were also tested.

Thermal aging had no effect on the unirradiated tensile properties. Irradiation at 390°C increased the 0.2% yield stress and the ultimate tensile strength above those of the unirradiated control specimens. The ductility decreased slightly. After irradiation at 450, 500, and 550°C, the tensile properties were essentially the same as the unirradiated values. The hardening at 390°C was attributed to the dislocation and precipitate structure formed during the irradiation. The lack of hardening at 450°C and higher correlates with absence of an irradiation-induced damage structure.

8.1.3 Refractory Metals and Alloys—Abstracts

8.1.3.1 "Tensile Properties of V-15 Cr-5 Ti Following Neutron Irradiation and Helium Implantation"⁶

M. L. Grossbeck and J. A. Horak*

Miniature specimens of V-15 Cr-5 Ti were prepared in the annealed condition and with 10, 20, and 30% cold work. The annealed specimens were cyclotron injected with 80 appm He and irradiated in sodium in EBR-II. The cold-worked (CW) specimens were irradiated in EBR-II but not helium injected. The specimens were irradiated at 400, 525, 625, and 700°C and received a fluence of 4.1 to 5.5×10^{26} neutrons/m² ($E > 0.1$ MeV). Helium injection alone appeared to have a slight embrittling effect at 400°C which was absent at 700°C. However, neutron irradiation resulted in a reduction in ductility by about a factor of 5 to 10 in annealed material. The effect of cold work was to strengthen unirradiated material. Irradiation strengthened both annealed and CW material, but the strength level essentially saturated at 10% cold work, the lowest value examined. For 30% CW material, the strengthening in the irradiated material recovered rapidly with increasing temperature, returning to the annealed strength at 700°C. Ductility was reduced by cold working but remained above 2% for all conditions examined.

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8.1.3.2 "The Use of Refractory Metals in Fusion Reactors"⁷

F. W. Wiffen

Conceptual studies of fusion reactors have identified many of the design details and materials issues of these future electric power systems. These studies show a number of components where the use of refractory metal alloys offers potential performance advantages. If superconducting magnets are selected, they require alloys of niobium or vanadium. For most components, the refractory alloys must compete with other candidate materials. The properties attractive to fusion include superconductivity, high strength at elevated temperatures, compatibility with liquid metals, and high melting point. Limitations on the use of these alloys include an inadequate database, difficult fabrication and joining technologies, and sensitivity to oxygen and other interstitial impurities. The potential applications of refractory alloys in fusion reactors will be identified and the properties of the metals compared to the application requirements.

8.1.3.3 "The Effects of Irradiation Temperature on the Properties of Niobium and Nb-1 Zr"⁸

F. W. Wiffen

Niobium alloys have been proposed as candidate materials for a number of reactor components. Currently they are under consideration for high-heat-flux components in fusion reactors and for cladding and structural applications for space power fission reactors. Few data are available on the effect of reactor neutron irradiation on the properties that determine successful service performance. The experiments discussed in this paper were designed to yield swelling and tensile properties for niobium and Nb-1 Zr irradiated at temperatures between 70 and 950°C. Some material was prepared with ¹⁰B added, for helium production during irradiation.

Cylindrical rod tensile specimens with gage sections 2.0 mm in diameter and 18 mm long were contained in experimental capsules for exposure in the peripheral target positions (PTP) of the HFIR reactor flux-trap zone. The specimens were in a helium atmosphere. Specimen-to-holder annular gaps were designed to produce gradients giving the desired irradiation temperatures between 450 and 950°C. Data from other specimens irradiated in HFIR at 70°C and in EBR-II at higher temperatures are also used to establish trends in the temperature dependence of the effects of irradiation on niobium.

The maximum fluence achieved in the experiment was 2.2×10^{27} neutrons/m² (total). This produced approximately 27 dpa in niobium. Transmutation of the ¹⁰B, essentially complete in the first few days of irradiation, produced 133 appm He in the main sample set.

Measured swelling for irradiation at 450°C was almost 4%; near-zero swelling resulted for irradiation temperatures 550 to 950°C. All irradiations between 70 and 950°C resulted in strength increases shown during tensile tests. Tensile elongation was decreased by all irradiations, in inverse relationship to the strengthening. However, significant ductility was displayed for all irradiation conditions. Plastic instability, resulting in near-zero uniform tensile elongation, was only observed in samples irradiated at 70°C. All tensile tests for irradiations between 450 and 950°C showed some component of work hardening after yield. Uniform elongation values for these irradiation conditions ranged from 1.0 to 13%. The lack of plastic instability implies that dislocation channeling is not a dominant deformation mode for any of the material irradiated at 450°C or higher. Macroinspection and scanning electron microscope (SEM) fractography showed only ductile fracture modes for all tested material.

These results show that swelling at temperatures in a range including 450°C could produce some limitations on the use of niobium alloys in components for high-fluence reactor operation. At higher temperatures, including at least the range 550 to 950°C, these results do not indicate any additional, radiation-related constraints on the use in reactor service.

8.1.4 Copper and Copper Alloys—Abstract of “Microstructure of Copper Following High Dose 14-MeV Cu Ion Irradiation”⁹

S. J. Zinkle, G. L. Kulcinski,^{*} and R. W. Knoll[†]

High-purity copper has been irradiated with 14-MeV copper ions to peak doses of 40 dpa over the temperature range of 100–500°C. Examination of the foils in a transmission electron microscope revealed that no significant amount of void formation had occurred, in conflict with previous irradiation studies. Instead, a high density of stacking fault tetrahedra (SFT) was observed. The defect cluster density is constant for irradiation temperatures $\leq 200^\circ\text{C}$, and the density decreases rapidly with irradiation temperature $\geq 300^\circ\text{C}$. It is postulated that the absence of voids is due to the low oxygen content of the copper foils (<5 ppm by weight). The depth dependence of the ion damage was studied by using the cross-section technique. Evidence of displacement damage in the form of “black-spot” defect clusters (SFT and small dislocation loops) was observed at depths well beyond that expected from ion damage range calculations. The discrepancy between the observed and calculated damage profiles is most likely due to the use of too large of an electronic stopping power value in the calculation, but diffusional spreading effects may also be playing a role.

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8.1.5 Innovative Materials—Abstracts

8.1.5.1 “Elevated-Temperature Heavy-Ion Damage in $(\text{Co}_{0.78}\text{Fe}_{0.22})_3\text{V}$ Ordered Alloy”¹⁰

D. N. Braski and K. Farrell^{*}

Irradiation-induced changes in microstructure were studied in a $(\text{Co}_{0.78}\text{Fe}_{0.22})_3\text{V}$ long-range-ordered (LRO) alloy after bombardment with 4-MeV nickel ions at temperatures from 843 to 1023 K and damage levels from 10 to 70 dpa. Small cavities and Frank loop were developed, and particles of preexisting VC phase were redistributed by the irradiation. Long-range order was retained under all irradiation conditions, but domain size was effectively reduced by the Frank loops that served as antiphase boundaries. Simultaneous injection of helium at a rate of 3 appm/dpa and deuterium at a rate of 12 appm/dpa during irradiation produced gas bubbles in the grain boundaries, increased the dislocation and cavity densities, reduced the mean cavity size, but had little effect on the ordered domain size or the redistribution of VC particles. Even with injected gases, swelling remained below 0.25%. It is deduced that swelling did not reach the rapid bias-driven state because the critical cavity size was large.

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8.1.5.2 "Structure and Properties of Welded Long-Range-Ordered Alloys"¹¹

S. A. David,* D. N. Braski, and C. T. Liu*

Sound autogenous welds can be produced readily in thin sheets of the $(\text{Fe,Ni})_3(\text{V,Ti})$ and $(\text{Fe,Co})_3(\text{V,Ti})$ LRO alloys. Welds of $(\text{Fe,Ni})_3(\text{V,Ti})$ were disordered, but the tensile properties were not adversely affected. A reordering heat treatment restored the weldment properties to those of the base metal. Welds of $(\text{Fe,Co})_3(\text{V,Ti})$ remained ordered after the weld thermal cycle. Tensile properties similar to those of the base metal and a more uniform hardness profile can be produced by using a postweld heat treatment. In many potential applications at elevated temperatures, the use of a postweld reordering heat treatment would be unnecessary for both alloys because the weldment would reorder itself during service. Factors that seem to influence the weldment properties include ordered domain size and precipitation of carbides.

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8.1.6 Testing and Analysis—Abstracts

8.1.6.1 *Scanning Electron Microscope Facility for Examination of Radioactive Materials*¹²

J. R. Gibson* and D. N. Braski

An AMRAY model 1200B scanning electron microscope was modified to permit remote examination of radioactive specimens. Features of the modification include pneumatic vibration isolation of the column, motorized stage controls, improvements for monitoring vacuum, and a system for changing filaments without entering the hot cell.

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8.1.6.2 "Preparation of Ion-Irradiated Foils for Cross-Section Analysis"¹³

S. J. Zinkle and R. L. Sindelar*

A simple, routine method has been developed that allows ion-irradiated foils to be examined in cross section. The method has been successfully demonstrated on three alloy systems (stainless steel, a low-chromium steel, and some high-strength copper alloys) and is generally applicable to most metals. The high success rate that is obtainable using this procedure allows several specimens to be examined from a single irradiated sample. This implies that statistically significant void and precipitate distributions may be obtained from irradiated specimens as a function of depth (damage level). This technique also allows multiple testing techniques (such as microscopy, scratch hardness, and microhardness) to

be simultaneously performed on a single irradiated sample. The preparation technique is described in detail and some typical applications are illustrated.

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8.1.6.3 "Miniature Tensile Test Specimens for Fusion Reactor Irradiation Studies"¹⁴

R. L. Klueh

Three miniature sheet-type tensile specimens and a miniature rod tensile specimen are being used in irradiation-damage studies for fusion reactor materials. Miniature specimens are also being used to determine design properties of irradiated materials. Consequently, a comparison of the tensile properties determined with these different specimens was made by testing CW and SA type 316 stainless steel sheet and rod at room temperature, 300°C, and 600°C. Good agreement was observed when the properties obtained with the different specimens were compared, and all of the specimens can be used for irradiation-damage studies. However, there were small differences in the properties obtained using the different specimens, and it was recommended that before any miniature specimens are used to determine design-type properties, further studies be conducted.

8.1.6.4 "Neutron Irradiation Facilities for Fission and Fusion Reactor Materials Studies"¹⁵

A. F. Rowcliffe

The successful development of energy conversion machines based upon nuclear fission or fusion reactors is critically dependent upon the behavior of the engineering materials used to construct the full containment and primary heat extraction systems. The development of radiation-damage-resistant materials requires irradiation testing facilities which reproduce, as closely as possible, the thermal and neutronic environment expected in a power-producing reactor. The ORNL reference core design for the Center for Neutron Research (CNR) reactor provides for instrumented facilities in regions of both hard and mixed neutron spectra, with substantially higher fluxes than are currently available. The benefits of these new facilities to the development of radiation-damage-resistant materials are discussed in terms of the major U.S. fission and fusion reactor programs.

8.1.7 Low-Activation Materials—Abstract of "Fusion Materials Activation Characteristics as Related to Waste Disposal Requirements"¹⁶

R. W. Conn* and F. W. Wiffen

Lower activation materials for fusion reactors are technically possible, may be important to the public acceptance of fusion energy, and are a main goal of fusion energy development. Actual levels of radioactivation in a fusion reactor will depend on the

material itself and on its alloying constituents and impurity content.¹⁷⁻²⁰ The equilibrium activation level and post-shutdown decay rates affect three primary areas: long-term waste material disposal; reactor maintainability, specifically, the prospects for limited hands-on or contact maintenance; and reactor safety. Goals relating to each of these three areas are proposed and used to develop four particular categories which classify both reactors and materials.

It is recommended that as a minimum, one goal of the fusion program should be a *reference fusion reactor* which: (1) meets Class C requirements of the Code of Federal Regulations, 10CFR61, for near-surface disposal of radioactive waste; (2) permits limited hands-on maintenance outside the reactor shield within two days of shutdown; and (3) meets all requirements for engineered reactor safety. A higher level goal could lead to a *low activation reactor* which would meet Class B requirements for waste disposal in 10CFR61 and the same criteria for maintenance and safety as the reference reactor. A *very low activation reactor* would meet Class A requirements in 10CFR61, would permit limited hands-on maintenance behind the reactor blanket, and would meet criteria for both engineered and intrinsic reactor safety. A fourth classification, the *standard reactor*, is one which will meet requirements for engineered safety and may meet requirements for limited contact maintenance outside the reactor shield but does not meet the minimum requirements for near-surface burial of waste (Class C) as presently provided for in federal regulations. A reactor constructed of type 316 stainless steel will be a standard reactor. More generally, it is necessary to avoid the use of highly activating materials, such as nickel, molybdenum, or niobium, even as low-percentage alloying constituents.

Very low activation reactors would have to be constructed almost entirely of ceramic materials, examples of which include graphite and silicon carbide.²⁰ Selective use of metal alloys such as aluminum is possible. All materials, whether ceramic or metallic, in a very low activation reactor must have an extremely low impurity content, lower in fact than is considered reasonably probable for practical engineering materials. Even then it is not possible to achieve limited hands-on maintenance within the plasma chamber.

With more reasonable impurity levels, use of these materials would lead to a reactor in the low activation class.²⁰ Quality commercial vanadium and titanium alloys are examples of metal alloys that can also potentially lead to a low activation reactor. A definitive judgment on the engineering practicality of such reactors, especially when using ceramics, could not be reached, but we note that essentially all components must be constructed of these materials.

Reference reactors can be achieved with metal alloys provided care is taken to control the elemental composition of the alloy and its impurity content. Elemental tailoring of stainless or ferritic steel to avoid the use of natural nickel, molybdenum, or niobium is an example. A more speculative alternative is isotopically tailored molybdenum or nickel for use as a low-percentage alloying constituent. Technical background material and a detailed discussion of the engineering issues associated with the use of low-activation materials are given elsewhere.^{17,21}

From more extensive work than is reported here,^{17,21} we find that quantitative definitions and goals are required in the areas of waste material disposal, reactor maintenance, and safety to guide fusion and materials development toward a product with advantageous characteristics. As a minimum objective, the national fusion program should aim to

develop a reference reactor which: (1) meets requirements for Class C waste material disposal; (2) permits limited hands-on maintenance outside the magnet's shield within two days of shutdown; and (3) meets all requirements for engineered safety. The goals regarding waste management and maintenance for the reference reactor appear technically reasonable to achieve and, in the long run, may be important to the public acceptance of fusion power. The activation characteristics of a material should be a primary consideration in the development of a fusion reactor material.

Further study is needed to evaluate the relationship between the choice of a low-activation structural material and other design selections or constraints (e.g., the coolant selection, the breeder selection, or the selection of operating parameters). Together with this, better understanding and quantification of the safety and the related cost-risk-benefit aspects of low-activation reactor designs are needed. There is a strong relationship between material selections, design choices, and safety. A material's activation characteristics, afterheat, temperature capability, thermal capacity, and chemical release mechanisms relative to its behavior under accident conditions may, in the end, determine a material's true attractiveness for use in a fusion reactor.

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8.1.8 Materials Compatibility—Abstract of "Liquid-Metal Corrosion"²²

O. K. Chopra,* D. L. Smith,* P. F. Tortorelli, J. H. DeVan,[†] and Dai Kai Sze*

A review of corrosion and environmental effects on the mechanical properties of candidate structural alloys for use with liquid metals in fusion reactors is presented. The corrosion/mass transfer behavior of austenitic and ferritic steels and vanadium-base alloys is evaluated to determine the preliminary operating temperature limits for circulating and static liquid lithium and Pb-¹⁷Li systems. The influence of liquid-metal environment on the mechanical properties of structural materials is discussed. Corrosion effects of nitrate and fluoride salts are presented. Requirements for additional data are identified.

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8.1.9 U.S.-Japan Collaboration—Abstract of "Program of United States-Japan Collaborative Testing in HFIR and ORR"²³

J. L. Scott, M. L. Grossbeck, A. Hishinuma,* T. Kondo,* A. F. Rowcliffe, and M. Tanaka*

The objective of this collaborative program is to design, conduct, and evaluate joint irradiation experiments in HFIR and ORR for the purpose of investigating the irradiation response of Japanese and U.S. structural alloys to high levels of atomic displacement and helium content. Results will be used to establish the behavior of such alloys and to evaluate their use for first-wall and blanket structural applications in future fusion reactors.

The experimental program consists of eight HFIR capsules and two ORR spectral tailoring capsules, with space equally shared between the U.S. and Japanese programs. Four HFIR capsules were irradiated to 30 dpa, while the remaining four capsules were irradiated to 50 dpa. The test matrix consisted of submini-tensile and hourglass fatigue specimens of U.S. and Japanese prime candidate alloys (PCAs) and type 316 stainless steels, together with sheet-tensile specimens of a variety of austenitic and ferritic alloys. Each capsule contained more than 100 transmission-electron-microscopy (TEM) disks of both structural and base-research alloys. Specimen temperatures ranged from 65 to 600°C. The two spectral tailoring capsules each contain two temperature zones, so that specimens are being irradiated at 65, 200, and 400°C. The test matrix for these capsules includes both structural alloys and base-research alloys, so that data will be obtained on irradiation creep, embrittlement mechanisms, microstructural evolution in model alloys, and ceramics, together with the engineering database and annealed and CW austenitic structural alloys and welds.

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8.2 FUSION MATERIALS HANDBOOK AND MATERIALS SELECTION—ABSTRACTS

8.2.1 "Structural Materials Data Base Assessment for Blanket Comparison and Selection Study"²⁴

J. W. Davis,* T. A. Lechtenberg,[†] D. L. Smith,[‡] and F. W. Wiffen

The Blanket Comparison and Selection Study (BCSS) had as its primary goal the selection of a limited number of blanket concepts for fusion power reactors, to serve as the focus for the U.S. Department of Energy (DOE) blanket R&D program. To help provide a common basis for evaluation of all candidate blanket concepts considered by the BCSS, a structural materials database assessment was performed which included a compilation of available materials properties data, specification of limiting criteria for materials performance, and determination of design-allowable parameters.

Three classes of alloys are currently considered as leading candidates for the first-wall/blanket structure of a fusion power reactor. For the BCSS, one reference or baseline alloy was selected from each class and one low-activation counterpart to each reference alloy was identified for evaluation. The alloy classes, reference alloys, and low-activation analogs selected were: austenitic stainless steels (PCA; manganese-stabilized steel); ferritic or martensitic steels (HT-9; Fe-11 Cr-2.5 W-0.3 V-0.15 C); and vanadium base alloys (V-15 Cr-5 Ti; reference alloy is low activation).

The critical nuclear, thermophysical, and mechanical properties of the three reference alloys were reviewed. Where insufficient data exist for a reliable assessment, best estimates were provided for use in the blanket concepts development. For the low-activation analogs, the same properties as their respective reference alloys were assumed, including radiation damage resistance. The design stress limits, maximum allowable operating temperature,

and lifetime were set primarily by radiation damage considerations. Critical design issues associated with each of the reference alloys and low-activation analogs were identified, together with limiting criteria for materials performance.

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8.2.2 "Materials Engineering for TFCX"²⁵

F. W. Wiffen

Four point designs for the Tokamak Fusion Core Experiment (TFCX) resulted from a U.S. national effort that was completed in July 1984. The goal of these devices is to achieve D-T ignition and long-pulse operation. They differ primarily in the assumptions for the toroidal field coils. Two use copper coils, the other two Nb₃Sn superconducting coils. Differences in each pair derive from the assumed coil performance parameters, including stress levels, current density, and coil cooling. The bigger single materials question is the radiation limits on the coil insulation, but a large number of materials questions were identified by the study. The TFCX is reactor relevant in terms of neutron, particle, and heat flux values; its limited design lifetime makes fluence levels far short of levels projected for power reactors. These and other materials issues raised by this project are discussed.

8.3 SPECIAL-PURPOSE MATERIALS—ABSTRACTS

8.3.1 "A Report of the Workshop on Low-Temperature Neutron Irradiation"²⁶

R. R. Coltman, Jr., H. R. Kerchner,* C. E. Klabunde, and F. W. Young, Jr.*

A workshop was held at ORNL May 29-31, 1985, to aid in preparing for the use of the National Low-Temperature Neutron Irradiation Facility (NLTNIF) being built at ORNL's Bulk Shielding Reactor and scheduled for experiments early in 1986. The workshop encompassed areas of both basic and technology research. Presentations and committee discussions dealt with low-temperature neutron irradiation effects on the mechanical properties of metals and alloys, on polymeric material, on superconductivity, on microstructure in nonmetals, on phase stability, and on the rates of diffusion and precipitation. Ion beam mixing and dielectric breakdown in irradiated liquid helium in magnetic fields were also discussed. Summaries of the committee reports and the status of the NLTNIF construction will be given. Researchers interested in the workshop report or in utilizing NLTNIF for irradiation experiments should contact one of the authors.

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8.3.2 "Special Purpose Materials for Fusion Application"²⁷

J. L. Scott, F. W. Clinard, Jr.,* and F. W. Wiffen

Originally in 1978 the Special-Purpose Materials Task Group was concerned with tritium breeding materials, coolants, tritium barriers, graphite and silicon carbide, ceramics, heat-sink materials, and magnet components. Since then several other task groups have been created, so that now the category includes only materials for superconducting magnets and ceramics. For the former application copper-stabilized $\text{Nb}_3\text{Sn}(\text{Ti})$ insulated with polyimides will meet the general requirements, so that testing of prototype components is the priority task. Ceramics are required for several critical components of fusion reactors either as dielectrics or as a structural material. Components near the first wall will receive exposures of $5\text{--}20 \text{ MW}\cdot\text{year}/\text{m}^2$. Other ceramic applications are well behind the first wall, with lower damage levels. Most insulators operate near room temperature, but ceramic blanket structures may operate up to 1000°C . Because of a meager database, one cannot identify optimum ceramics for structural application, but MgAl_2O_4 is an attractive dielectric material.

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Chapter 9

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Neutron Transport

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R. G. Alsmiller, Jr.¹

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9. NEUTRON TRANSPORT

SUMMARY OF ACTIVITIES

The neutron transport program includes both experimental and analytic phases. The experimental program is designed to provide the data necessary for verifying the analytic method and cross-section data used for fusion reactor neutronics design calculations. Experiments are carried out to determine neutron and photon transport in typical fusion reactor shield materials and configurations and to determine the effects of penetrations in these shields. The analytic program supports the design of the integral experiments and compares calculated data with data obtained experimentally. During the past year, neutronics experiments were conducted at ORNL and, in collaboration with the staff at Princeton Plasma Physics Laboratory, on the Tokamak Fusion Test Reactor. Work was also performed in the area of nuclear data evaluation and cross-section processing, and the Radiation Shielding Information Center continued to supply a broad range of services to the fusion energy research community.

9.1 ABSTRACTS FROM EXPERIMENTAL AND ANALYTIC PROGRAMS

9.1.1 "Monte Carlo and Discrete Ordinates Calculations of 14-MeV Neutrons Streaming Through a Stainless Steel Duct: Comparison With Experiment I"¹

R. T. Santoro, J. M. Barnes, R. G. Alsmiller, Jr., and J. D. Drischler

Measured and calculated neutron and gamma-ray energy spectra from ~14-MeV neutrons streaming through a stainless steel duct having a length-to-diameter ratio of 4.6 are compared. The 1.45-m-long duct is imbedded in a concrete block. The spectra were measured with an NE-213 liquid scintillator as a function of detector location relative to the mouth of the duct. The calculated data were obtained using the Monte Carlo code MCNP and the discrete ordinates code DOT 4.3. The calculations were performed using a two-dimensional cylindrical model of the experiment with symmetry about the duct axis. The measured and calculated neutron and gamma-ray spectra are compared at two distances from the mouth of the duct and at detector locations on and off the duct axis. The neutron spectra calculated with MCNP agree with the measured data within ~5 to 50% at all detector locations. The data calculated using the discrete ordinates method are in good agreement with the experiment for the cases where the detector is on axis but are in poor agreement at the off-axis detector locations. The gamma-ray spectra calculated with

both radiation transport methods are in good agreement (~ 5 to 25% depending on photon energy) with the measured spectra.

9.1.2 "Monte Carlo and Discrete Ordinates Calculations for 14-MeV Neutrons Streaming Through a Stainless Steel Duct: Comparisons With Experiment II"²

R. T. Santoro, J. M. Barnes, R. G. Alsmiller, Jr., and J. D. Drischler

Measured and calculated neutron energy spectra from 14-MeV neutrons streaming through stainless steel ducts having length-to-diameter ratios of 14.8 and 38.2 are compared in this paper. The measured data were obtained using an NE-213 liquid scintillator and the calculated data were obtained using the Monte Carlo code MCNP and the discrete ordinates code DOT 4.3. For both duct geometries, the spectra calculated using the MCNP code are in reasonable agreement with the measured data ranging from a few percent to a factor of two, depending on the detector location and duct. The spectra calculated using the DOT code are in fair agreement (30%) for cases where the detector is on the duct axis but are in poor agreement, up to an order of magnitude, when the detector is off axis.

9.1.3 "Measurements of the Neutron and Gamma-Ray Fluences in the TFTR Test Cell Due to a Point Source Simulating D-T Fusion Plasma Neutron Production"³

J. K. Dickens, J. W. McConnell, K. M. Chase, H. W. Hendel, E. B. Nieschmidt, and F. Y. Tsang

Spectral distributions of high-energy neutrons ($0.9 \leq E_n \leq 14.5$ MeV) and of high-energy gamma rays ($0.4 \leq E_\gamma \leq 9.4$ MeV) due to a deuterium + tritium (D + T) neutron point source simulating the extended fusion plasma neutron source in the Tokamak Fusion Test Reactor (TFTR) at the Princeton Plasma Physics Laboratory are reported. A D-T neutron generator was positioned inside the vacuum vessel at ten different locations around the torus. Neutrons and gamma rays were detected by a liquid-scintillator-based detector (4.65-cm diameter by 4.22-cm height) with electronic pulse shape discrimination to differentiate between events in the detector due to incident neutrons and those due to incident gamma rays. The detector was placed on the median plane of the reactor 8.85 m from the geometric center of the TFTR. Two spectral distributions, one for neutrons and the other for gamma rays, were obtained for each measurement location. The neutron data exhibit a high-energy peak dominated by uncollided primary-energy neutrons and a low-energy contribution from the scattered neutrons. The gamma-ray data exhibit a high-energy contribution due to neutron capture gamma rays and a low-energy contribution due to gamma rays following neutron inelastic scattering and reactions.

9.1.4 "Comparisons of Calculated and Measured Spectral Distributions of Neutrons from a 14-MeV Neutron Source Inside the Tokamak Fusion Test Reactor"⁴

R. T. Santoro, J. M. Barnes, R. G. Alsmiller, Jr., M. B. Emmett, and J. D. Drischler

A recent paper presented neutron spectral distributions (energy ≥ 0.91 MeV) measured at various locations around the TFTR at the Princeton Plasma Physics Laboratory. The neutron source for the series of measurements was a small D-T generator placed at various positions in the TFTR vacuum chamber. In the present paper the results of neutron transport calculations are presented and compared with these experimental data. The calculations were carried out using Monte Carlo methods and a very detailed model of the TFTR and the TFTR test cell. The calculated and experimental fluences per unit energy are compared in absolute units and are found to be in substantial agreement for five different combinations of source and detector positions.

9.2 DATA EVALUATION AND PROCESSING FOR FUSION NEUTRONIC DATA NEEDS

D. C. Larson, C. Y. Fu, D. M. Hetrick, and R. W. Roussin

Evaluated and processed neutron cross-section data are produced to meet the needs of fusion reactor designers. Nuclear data needs for the Magnetic Fusion Energy Program are given in ref. 5 and include evaluated cross sections for copper, nickel, and chromium. The $^{63,65}\text{Cu}$ evaluations completed last year in the new ENDF/B-VI formats were the first evaluations available in the new formats and as such were used as input to check the modified processing codes. During this process, a long-standing difficulty concerning energy balance was present for some reactions near threshold. The problem occurred because the energy bin widths used in the calculations were too wide. Our evaluation codes were reconfigured to incorporate variable energy bin widths. This allows use of narrow energy bin widths near threshold and other regions where the cross sections change rapidly with energy, while retaining wider bin widths where possible to minimize the number of bins and thus lessen the processing burden. Calculations pertinent to evaluations for $^{58,60}\text{Ni}$ and ^{52}Cr are being redone with this new feature. A paper describing this work⁶ was presented at the International Conference on Nuclear Data for Basic and Applied Science, Santa Fe, N.M., in May 1985.

Evaluated cross sections must be processed into forms that can be used in radiation transport computer codes. The focus of this effort has been the development of VITAMIN-E,⁷ a 174-neutron, 38-gamma-ray group cross-section library. It can be used in conjunction with the AMPX-II system⁸ to derive cross-section data suited to a particular application. A paper describing this work⁹ was presented at the International Conference on Nuclear Data for Basic and Applied Science, Santa Fe, N.M., in May 1985.

A continuing effort is under way to maintain the VITAMIN-E library and the relevant AMPX-II computing technology on the computers of the National Magnetic Fusion Energy Computing Center at Lawrence Livermore National Laboratory. The VITAMIN-E data and AMPX-II computing technology are also available from the Radiation Shielding Information Center at Oak Ridge National Laboratory.

9.3 RADIATION SHIELDING INFORMATION CENTER

R. W. Reussin, D. K. Trubey, J. E. White, and J. Bartley

The Radiation Shielding Information Center (RSIC) serves an international community by responding to inquiries about radiation transport problems. Staff members provide guidance by drawing on a technical database that includes a computerized literature file, a collection of complex computer programs, and a substantial body of nuclear data libraries pertinent to the solution of such problems.

Acquiring the needed computer-based technology base requires the collaboration of the neutronics community with RSIC staff members to collect, organize, process, evaluate, and package relevant technology developed in the community. This technology is disseminated throughout the community with a mechanism for feedback of experience through use, which results in an improved product. The resulting technology base provides an overall advancement of the state of the art.

A sample of some recent products of this information cycle illustrates the international character of the RSIC community. The Los Alamos National Laboratory contributed latest versions of the one- and two-dimensional discrete ordinates codes CCC-428/ONEDANT¹⁰ and CCC-456/TWODANT.¹¹ A new version of the MORSE Monte Carlo code CCC-474/MORSE-CGA¹² was contributed by ORNL and featured in a workshop sponsored by RSIC. The United Kingdom Atomic Energy Authority contributed another version, CCC-471/MORSE-H,¹³ that is used primarily for fusion neutronics at its Harwell facility. The DLC-113/VITAMIN-E multigroup library, developed for use in computer codes like those listed, was updated by RSIC. The CCC-478/ACFA¹⁴ system for analyzing neutron-induced activation, heating, transmutation, and gas production was contributed by the Institut für Reaktorentwicklung, Jülich, Federal Republic of Germany.

This selection represents a small portion of the total activity of the center. Information processing (including evaluation and packaging) is a daily function. In addition to a comprehensive literature database, RSIC-packaged products include 118 data packages (DLC), 489 neutronics and shielding code packages (CCC), and 226 data processing and other miscellaneous code packages (PSR)

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Chapter 10

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Management Services Quality Assurance Safety

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Management Services

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10. MANAGEMENT SERVICES, QUALITY ASSURANCE, AND SAFETY

SUMMARY OF ACTIVITIES

Administrative support for the programmatic research and development activities of the Fusion Energy Division is provided by the Management Services Section and by the division's quality assurance (QA) and safety programs.

The Management Services Section provides support in the following areas:

- general personnel administration, material and service procurement, subcontracting, and coordination of national and international agreements;
- nonprogrammatic engineering services for support systems and equipment, planning and coordination of general plant project and facility improvements, coordination of maintenance and machine shop work, labor relations, and telecommunications;
- financial management;
- library services and resources; and
- publications services.

Support is provided through interpersonal relations with division programmatic staff and through coordination of resources from disciplines outside the division.

The QA activity in the division places particular emphasis on the development and documentation of a QA program that conforms to national standards, the review and approval of engineering documents, supplier surveillance, identification and documentation of nonconforming items, audits, and QA assessments/plans. The number of QA staff members was increased in 1985 to carry out activities in these areas.

The division's safety activities include a formal safety program, emergency planning activities, and environmental protection services. During 1985, significant effort was devoted to the removal of hazardous waste from division facilities.

10.1 MANAGEMENT SERVICES

10.1.1 General Administration Services

10.1.1.1 Personnel

On December 31, 1985, the Fusion Energy Division had 177 employees, compared with 172 on December 31, 1984. Figure 10.1 shows the composition of the division's scientific staff. Attrition from the division during 1985 consisted of 6 retirements, 9 resignations, 1 leave of absence, and 2 transfers to other divisions. The standby condition imposed on the Oak Ridge Gaseous Diffusion Plant (ORGDGP) resulted in the addition of 16 employees to the division staff: 6 science technologists, 3 secretaries, 1 computing specialist, 1 materials procurement officer, 2 engineers, and 3 administrative services personnel. Approximately 55% of the new employees in the division were women or members of minority groups, which demonstrates the division's commitment to affirmative action.

The University and College Co-Op Program sponsored one electrical engineering student from Virginia Polytechnic Institute and State University and three computer communications students from The University of Tennessee. Summer program activity was reduced to three technical students, one of whom was sponsored by Oak Ridge Associated Universities.

Long-term foreign and domestic assignments were made for 25 staff members during 1985. Apartments near the Princeton Plasma Physics Laboratory (PPPL) were rented to accommodate personnel involved in the ORNL-PPPL collaboration described in Chap. 1.

Fig. 10.1. Fusion Energy Division scientific staff by discipline.

The division was host to 94 scientific guests and subcontractor personnel; 46 of these were foreign participants [36 with the Large Coil Program (LCP) and 7 with the U.S.-Spain collaboration described in Chap. 1].

Essential personnel development activities received continued emphasis. These included job-related seminars, scientific workshops, educational studies, and employee training courses.

10.1.1.2 Procurement and subcontracts

During 1985, the division procurement group processed 1735 requisitions with a total value of approximately \$5.2 million. In addition, 316 shipping orders were processed. These efforts included major equipment loans to PPPL, the Massachusetts Institute of Technology, and TRW, Inc. The group also assisted in special procurement activities for the Japanese and Swiss site representatives to the ICP.

Industrial participation through subcontracts to the division in 1985 cost approximately \$12.6 million. This amount included the coil contracts for the LCP and industrial contracts for the Fusion Engineering Design Center, as well as contracts for other DOE programs. Approximately \$9.5 million of the subcontract cost was applicable to the ORNL Fusion Program.

10.1.1.3 International agreements and collaboration

Two implementing agreements were signed in 1985: one between the U.S. DOE and the European Atomic Energy Community (EURATOM), for cooperation in developing the stellarator concept, and one between the U.S. DOE and the U.S.-Spain Joint Committee for Scientific and Technological Cooperation for the design of a new fusion facility in Spain, the TJ-II flexible heliac.

During this reporting period, there have been key exchanges and collaborative activities on previous agreements. For example, the division has begun to build a pellet injector for the Tore Supra and has continued work on the pump limiter and ion cyclotron resonance heating component development for this device, which is located in Cadarache, France. As part of the U.S.-U.S.S.R. Fusion Research Exchange, a concept originated at ORNL for aligning torsatron coils was tested for the first time at the Kharkov Physico-Technical Institute in the Soviet Union. Pump limiter studies were performed on the Heliotron-E device at Kyoto, Japan, as part of the U.S.-Japan collaboration. The division also contributed to project and engineering activities and participated in experiments on the Advanced Limiter Test II (ALT-II) in the TEXTOR device in a collaborative effort between the United States, the Federal Republic of Germany, and Japan.

The division was instrumental in promoting an International Operations Workshop, hosted by the DOE Office of Fusion Energy (OFE) in Washington, D.C., during December 1985. The workshop provided an opportunity for U.S. laboratory staff members, university personnel, government subcontractors, OFE officials, and representatives from DOE field offices to discuss mutual problems, suggest improvements to legal and administrative processes, and designate task teams to investigate major issues.

10.1.2 Engineering Services

The Engineering Services group coordinates all engineering work performed in the division. Highlights of activities in 1985 are as follows.

1. Phase I of the construction and installation of a new cooling tower and heat exchanger was completed. This activity is funded as a Multi-General Purpose Facilities (MGPF) improvement and will improve the reliability of the demineralized cooling water system in Bldg. 9201-2. Phase II has been designed, and a fixed-price contract has been awarded to expand the cooling capacity to 40 MW.

2. The upgrade of the Bldg. 9201-2 electrical system was completed by Rust Engineering. This MGPF improvement consisted of adding an additional 50-MW transformer, connecting line 2 (a 161-kV line) from the Elza switchyard to the new transformer, installing new switchgear, and connecting heavy pulsed loads to the new switchgear. This will increase the power available to Bldg. 9201-2 and improve the overall reliability of the electrical system. This project is in the checkout phase, which will be carried out by Martin Marietta Energy Systems, Inc.

3. Hookup of the broadband video-data link and its internal wiring in Bldg. 9201-2 was completed. This improves both computer and audio-video communication between the X-10 and Y-12 sites and within division facilities.

4. The general appearance of division facilities was improved as part of a continuing effort that involves installing new tile in hallways, painting selected areas, and removing obsolete equipment. Removal of equipment left in Bldg. 9201-2 after the 86-in. cyclotron was shut down allowed relocation of the building electrical shop and provided additional space for constructing components for the Advanced Toroidal Facility (ATF). Space has also been added for expansion of the pellet fueling laboratory.

5. Shop fabrication, consisting of 97 jobs that required 25,000 hours of effort and cost approximately \$1 million, was overseen. Work was placed in shops at X-10, Y-12, and ORGDP. Several small jobs were placed in local shops outside the organization. (This shop work did not include the large fabrication efforts on the LCP and ATF, which were awarded through purchase orders to outside vendors.)

10.1.3 Financial Services

The Finance Office is a functional part of the Management Services Section and provides financial management support for administrative, engineering, and research personnel in areas that include budget preparation, cost scheduling, and variance analysis. The office also provides meaningful and appropriate accounting and cost control. Interaction with division management is an essential part of administering the budget, accounting policies, and procedures.

Table 10.1 and Fig. 10.2 show the funding trends for the ORNL Fusion Program. These funding levels create a need for great efficiency and continued improvement in specific areas of financial control. The Finance Office must continue to develop innovative techniques to meet the financial needs of the division.

Table 10.1. Fusion Program expense funding (in thousands of dollars)—budget outlay (B/O)

Activity	FY 1983 actual cost	FY 1984 actual cost	FY 1985 actual cost	Funding as of Feb. 1986	FY 1987 budget submission	FY 1988 budget submission
Applied plasma physics						
Fusion plasma theory	\$3,237	\$3,424	\$3,116	\$2,496	\$2,660	\$2,830
Experimental plasma research	1,344	1,405	1,410	1,155	1,190	1,255
National MFE computer network	352	336	443	200	460	510
Technical program support	38	185	189	0	0	0
	\$4,971	\$5,350	\$5,158	\$3,851	\$4,310	\$4,595
Confinement systems						
Toroidal systems						
Research operations	\$9,639	\$9,233	\$12,589	\$12,244	\$20,152	\$21,025
Major device fabrication	54	1,826	6,768	10,389	1,332	0
Magnet mirror systems						
Research operations	10,866	12,515	1,134	0	0	0
Technical program support					100	100
	\$20,559	\$23,574	\$20,491	\$22,633	\$21,584	\$21,125
Development and technology						
Magnetic systems	\$12,839	\$12,794	\$13,589	\$9,940	\$7,786	\$3,240
Plasma engineering	9,348	4,522	3,896	3,870	5,900	8,088
Reactor systems	765	569	478	352	340	368
Environment and safety	270	201	10	92	195	125
Reactor materials	5,615	6,292	6,928	5,243	5,646	6,212
Fusion Engineering Design Center	4,110	5,083	4,781	4,510	4,678	4,814
	\$32,947	\$29,461	\$29,682	\$24,007	\$24,545	\$22,847
TOTAL ORNL FUNDING	\$58,477	\$58,385	\$55,331	\$50,491	\$50,439	\$48,567

Fig. 10.2. Fusion Program expense funding (B/O).

10.1.4 Library Services

During 1985, the Fusion Energy Library processed 2762 circulation transactions, provided assistance with 2616 references, performed 235 online literature searches, and conducted 81 training sessions for use of the Library Information Online Network (LION). Users of LION have access to complete information about the location and availability of items in every collection in the Information Resources Organization (IRO) library system, which has increased sharing of resources among libraries. The Fusion Energy Library requested 309 items from other branches of the library system and sent 1025 items to users outside the division in 1985.

New materials received during the year included 131 books, 1200 journal issues, 733 paper reports, and 2508 microfiche reports. This enriched the collection but presented a serious storage problem. With the assistance of Library Committee members and the results of a survey of users, the library staff withdrew 2 journal indexes, 200 books, and 1500 reports from the collection. To complete the conversion to the LION system, staff

members computerized the circulation system for books and applied bar code labels to each one.

The results of online literature searches were made available on diskettes. This supplement to the traditional paper copy is a new option that became possible when the librarian began performing searches on the library's IBM-PC.

10.1.5 Publications Services

The Fusion Energy Division Publications Office provides the division with editing, composition, graphic arts, and reproduction services. It also serves as liaison with the IRO Laboratory Records Department for clearance and processing of all documents prepared by division staff.

Publications office staff handled a total of 363 jobs during 1985; these included journal articles, conference papers, reports, posters, viewgraphs, and full-color renderings. In addition, 146 abstracts were cleared. Much of this material is listed in Appendix 1.

Editors in the publications office use both the Fusion Energy Division computer systems and the IRO Text Management System for editing and writing. Compositors use principally the IRO Text Management System but are scheduled to begin using division computers during 1986 to produce draft and final copy. Graphic artists produced more than 1000 publication-quality drawings and visuals using the MAPPER program on the division computer systems. These capabilities reflect the commitment of the publications office to aid division staff in the production of high-quality documents in an expeditious and cost-effective manner.

10.2 QUALITY ASSURANCE

10.2.1 QA Program Development

A primary objective for the division's quality assurance (QA) effort in 1985 was the development of a QA program that conforms not only to standards set by the DOE Oak Ridge Operations Office but also to those of the American National Standards Institute and the American Society of Mechanical Engineers as set out in NQA-1.¹ Documentation of the division's QA program will include a QA Program Description, a QA Program Index, and QA procedures specific to division activities. During 1985, 12 QA procedures were issued:

- QA-FE-2-1 QA Assessments/Plans
- QA-FE-3-1 Deviation Requests
- QA-FE-4-1 Purchase Requisitions
- QA-FE-5-1 Work Plans
- QA-FE-5-2 Control of Shop Fabrication
- QA-FE-7-1 Supplier Nonconformances
- QA-FE-7-2 Supplier Deviation Requests
- QA-FE-7-3 QA Surveillances
- QA-FE-11-1 Test Control

QA-FE-12-i Control of Measuring and Test Equipment
QA-FE-15-1 Nonconformances
QA-FE-18-1 Audits

In addition, drafts of the program description and the program index were developed.

10.2.2 QA Program Implementation

The division QA staff has expended considerable effort in reviewing engineering documents, principally drawings and specifications, which define technical requirements for items to be fabricated, to ensure that they clearly define the QA requirements to be met.

Division QA staff made a number of trips during the year to various suppliers to ensure that major components for division projects were being manufactured correctly. The services of ORNL QA inspectors were also enlisted in this area.

Nonconformances must be identified and documented so that problems can be resolved. During the year, 93 nonconformance reports, 6 quality investigation reports, and 36 deviation reports were written and acted on.

Five external audits, also called supplier or vendor audits, and two internal audits were performed during the year by division QA staff in accordance with the new QA procedure (see Sect. 10.2.1). In addition, ORNL QA inspectors performed an audit of the division. Corrective actions were successfully implemented for all findings from a DOE audit conducted in 1984.

The QA Assessment/Plan (QAA/P) is used to identify and evaluate the risk of failure in quality or other areas, to determine if special QA activities are required to monitor the risk, and to reduce the risk of failure to an acceptable level. During the year, five QAA/Ps were completed and closed out; there were seven active QAA/Ps at year's end. The division QA staff is working with other division staff members and with Engineering personnel to develop more effective QAA/Ps for large projects.

10.3 SAFETY, EMERGENCY PLANNING, AND ENVIRONMENTAL PROTECTION

10.3.1 Safety and Emergency Planning

The division places a high priority on developing and implementing a strong safety program. Safety meetings were held every two months, and attendance was mandatory for division staff. The division safety officer or a designate attended the monthly Y-12 Plant safety meetings, and reports on these meetings were made to the division director, the section heads, and the program managers. Monthly safety inspections were conducted by each section, and quarterly inspections were made by the division director.

Several new members were added to the division Emergency Squad. All squad members were certified or recertified in first aid and cardiopulmonary resuscitation.

10.3.2 Environmental Protection

Efforts devoted to environmental protection and management of hazardous materials expanded in 1985. During the year, 2653.25 gal of liquid hazardous waste and 6180.2 lb of solid hazardous waste were disposed of through ORNL's Waste Management Organization. Much of this material was located in the cyclotron area and was removed from transformers that contained high levels of PCB-contaminated oil. A large volume of oil and some capacitors and transformers remain in this area, and plans were developed for the disposal of these items in 1986.

REFERENCE

1. *Quality Assurance Program Requirements for Nuclear Facilities*, ANSI/ASME NQA-1, 1983 ed., American Society of Mechanical Engineers, New York, 1983.

Appendix 1

PUBLICATIONS AND PRESENTATIONS

BOOKS AND JOURNAL ARTICLES	A-3
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PRESENTATIONS

January

Second Symposium on Space Nuclear Power Systems, Albuquerque, New Mexico, January 14-16, 1985

M. L. Grossbeck and F. W. Wiffen, "Swelling and Tensile Properties of EBR-II-Irradiated Tantalum Alloys for Space Reactor Applications"

Ignition Physics Workshop, Washington, D.C., January 15-16, 1985

N. A. Uckan, J. Sheffield, and W. A. Houlberg, "Physics Assumptions/Models for Ignition Experiments: Options for Small, High-Field Devices"

Sixth International ECR Ion Source Workshop, Berkeley, California, January 17-18, 1985

F. W. Meyer, "Status of the ORNL ECR Source"

Ignition Studies Mission 2 Meeting, University of California, Los Angeles, January 28-31, 1985

A. Dabiri, P. T. Spampinato, and S. L. Thomson, "TPSS Configuration Scoping Studies"

C. A. Flanagan, "FEDC Trade Studies—Upgraded Capability Exists to Examine Cost/Performance Sensitivities for All Tokamak Options"

N. A. Uckan, "Physics Scaling Studies for Tokamak Ignition Experiments—Basis for Trade-off Studies"

February

Seminar, Kernforschungszentrum, Karlsruhe, Federal Republic of Germany, February 25, 1985

J. M. Vitek, "Research at ORNL on Ferritic Steels for Fusion Reactor Applications"

Fifth APS Topical Conference on Atomic Processes in High Temperature Plasmas, Pacific Grove, California, February 25-28, 1985. Abstracts published in *Bull. Am. Phys. Soc.* 30 (1986).

C. F. Barnett, R. A. Phaneuf, H. T. Hunter, M. I. Kirkpatrick, W. L. Wiese, E. W. Thomas, M. S. Pindzola, and R. K. Janev, "Atomic Data for High Temperature Plasmas," p. 1131

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- J. M. Vitek, "The Effect of Cooling Rate on the Structure of Austenitic Stainless Steels"
- Seminar, Mannesmann Forschungsinstitut, Duisburg, Federal Republic of Germany, March 7, 1985**
- J. M. Vitek, "Research at ORNL on Ferritic Steels for Fusion Reactor Applications"

Seminar, Kernforschungsanlage, Jülich, Federal Republic of Germany, March 8, 1985

J. M. Vitek, "Research at ORNL on Ferritic Steels for Fusion Reactor Applications"

U.S.-Japan Seminar on Heliotron-E Results, Kyoto University, Gokasho, Uji, Japan, March 11-15, 1985

J. F. Lyon, "ORNL Stellarator Program"

Ignition Design Point Workshop, Princeton Plasma Physics Laboratory, Princeton, New Jersey, March 12, 1985

J. Sheffield, "WHIST Code Calculations of Ignition Margin in an Ignition Tokamak"

Workshop on High Current Electrical Joints for Tokamaks, Princeton Plasma Physics Laboratory, Princeton, New Jersey, March 14, 1985

G. E. Gorker and S. S. Kalsi, "Electrical Sliding Joint Literature Review"

Ignition Physics Workshop, Washington, D.C., March 20-21, 1985

N. A. Uckan, W. A. Houlberg, and J. Sheffield, "Ignition Requirements and Plasma Performance of a High-Field, Small Ignition Device"

12th Energy Technology Conference, Washington, D.C., March 25-27, 1985. Proceedings published.

J. H. Harris, "The Stellarator Approach to Fusion Plasma Confinement," p. 1082

Data Base Workshop, Rockville, Maryland, March 28-29, 1985

D. Keeton, "Availability/Reliability Modeling"

April

Ignition Studies Physics Modeling Workshop, Princeton Plasma Physics Laboratory, Princeton, New Jersey, April 3, 1985

N. A. Uckan and W. A. Houlberg, "0-D and 1½-D Modeling of Compact Ignition Devices"

1985 Sherwood Theory Conference: Annual Controlled Fusion Theory Meeting, Madison, Wisconsin, April 15-17, 1985. Abstracts published by the University of Wisconsin, Madison.

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- W. A. Houlberg and H. C. Howe, "Injection of Large Pellets in TFTR: Theory Issues," abstract 1R-15
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5th Tennessee Valley Symposium, American Vacuum Society, April 30-May 2, 1985

R. L. Livesey, "Clean Vacuum Systems for RFTF"

May

Sixth Topical Conference on Radiofrequency Plasma Heating, Pine Mountain, Georgia, May 13-15, 1985. Manuscripts submitted at the conference were published in *Radiofrequency Plasma Heating: AIP Conf. Proc. 129*, ed. D. G. Swanson, American Institute of Physics, New York, 1985.

F. W. Baity, D. J. Hoffman, and T. L. Owens, "ICRF Heating on ATF," p. 32

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R. C. Goldfinger and D. B. Batchelor, "Propagation and Absorption of Electron Cyclotron Waves in ATF"

D. J. Hoffman, F. W. Baity, and T. L. Owens, "ICRF Technology" (invited paper)

E. F. Jaeger, D. B. Batchelor, and H. Weitzner, "Fast Wave Heating in Tokamak and Stellarator Magnetic Fields"

T. L. Owens, F. W. Baity, and D. J. Hoffman, "ICRF Antenna and Feedthrough Development at the Oak Ridge National Laboratory," p. 95

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J. J. Yugo and D. T. Blackfield, "Analysis and Design of the RF Systems for the Minimax Tandem Mirror Reactor"

Particle Accelerator Conference, Vancouver, British Columbia, Canada, May 13-16, 1985

J. H. Whealton, R. J. Raridon, M. A. Bell, S. Y. Ohr, W. L. Stirling, D. W. Wooten, and R. W. McGaffey, "3-D Accelerator and Ion Source Design for the ORNL Low Temperature Negative Ion Source"

International Conference on Nuclear Data for Basic and Applied Science, Santa Fe, New Mexico, May 13-17, 1985. Proceedings to be published by Gordon and Breach, New York.

D. M. Hetrick, C. Y. Fu, and D. C. Larson, "Calculated Neutron-Induced Cross Sections for $^{63,65}\text{Cu}$, $^{58,60}\text{Ni}$, and ^{52}Cr from 1 to 20 MeV and Comparisons with Experiments"

R. W. Roussin, J. E. White, and J. L. Bartley, "VITAMIN-E: 174n, 38g, General Purpose Cross Section Library"

Workshop on Analysis and Comparison of Surface Probe and Related Measurements Made on TEXTOR, KFA Jülich, Federal Republic of Germany, May 14-15, 1985

R. E. Clausing, K. G. Tschersich, J. P. Fleischauer, L. Heatherly, F. Waelbroeck, J. Winter, P. Wienhold, Y. Sakamoto, S. Ishii, and K. Yano, "Surface Composition Changes of Inconel 625 During RG and ECR Discharge Cleaning of TEXTOR at 300°C"

R. E. Clausing, F. Waelbroeck, J. Winter, P. Wienhold, L. Könen, and N. Noda, "Effects of Wall Conditioning on Plasma Parameters, Impurities, and Hydrogen Recycling in TEXTOR"

Tokamak Power Systems Studies Review Meeting, Atlanta, May 15-16, 1985

S. K. Borowski, A. Dabiri, and D. J. Strickler, "TPSS Plasma Engineering Studies: Profile Effects and Plasma Power Balance"

A. E. Dabiri, "Material Options for a Commercial Fusion Reactor First Wall"

A. E. Dabiri and D. C. Keeton, "Modular Tokamak Power Plant"

S. L. Thomson, G. T. Bussell, and T. G. Brown, "Modular Tokamak Reactor Configuration"

S. L. Thomson and D. C. Keeton, "Economy of Scale"

32nd International Technical Communication Conference, Houston, May 19-22, 1985. Proceedings published by the Society for Technical Communication, Washington, D.C., 1985.

J. Heiken, M. N. Johnson, L. C. McFarland, and F. S. Cost, "Communicating our Stress," pp. WE-36-WE-38

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Ignition Studies Physics Modeling Workshop, Princeton Plasma Physics Laboratory, Princeton, New Jersey, May 21-22, 1985

N. A. Uckan, "Analytic Formulation of Tokamak Ignition Conditions and Plasma Performance Contours—Application to Compact Ignition Experiments"

16th Annual Meeting of the Division of Electron and Atomic Physics, American Physical Society, Norman, Oklahoma, May 29-31, 1985. Abstracts published in *Bull. Am. Phys. Soc.* 30 (1985).

F. W. Meyer, "Low Energy Electron Capture by Fully Stripped Light Ions from Atomic Hydrogen" (invited paper), p. 846

M. R. Strayer (in collaboration with C. Bottcher), "Finite Element Studies of the Dirac Equation for Heavy Ion-Atom Collisions" (invited paper), p. 854

June

1985 IEEE International Conference on Plasma Science, Pittsburgh, June 3-5, 1985. Proceedings to be published by the Institute of Electrical and Electronics Engineers.

K. Felch, R. Bier, L. Fox, H. Huey, L. Ives, H. Jory, N. Lopez, and S. Spang, "Initial Test Results on the Varian 140-GHz Gyrotron"

D. J. Hoffman, F. W. Baity, T. L. Owens, C. C. Tsai, and J. H. Whealton, "Antennas, Faraday Shields, and Feedthroughs for Ion Cyclotron Resonance Heating in Fusion Devices"

C. C. Tsai, W. R. Becraft, S. K. Combs, C. A. Foster, H. H. Haselton, D. J. Hoffman, M. M. Menon, S. L. Milora, D. E. Schechter, and F. Sluss, "Active Cooling Technology for Fusion Engineering Applications"

Annual Meeting of the American Nuclear Society, Boston, June 9-14, 1985

Y.-K. M. Peng, S. K. Borowski, G. T. Bussell, G. R. Dalton, J. D. Galambos, G. E. Gorker, J. R. Haines, W. R. Hamilton, S. S. Kalsi, M. H. Kunselman, D. C. Lousteau, V. D. Lee, J. B. Miller, R. L. Reid, B. W. Reimer, E. C. Selcow, D. J. Stricker, and N. A. Uckan, "Compact D-T Fusion Spherical Tori at Modest Field"

Seminar, Los Alamos National Laboratory, Los Alamos, New Mexico, June 10, 1985

M. S. Pindzola, "Electric-Field Effects on Dielectronic Recombination"

IEEE 5th Pulsed Power Conference, Arlington, Virginia, June 10-12, 1985. Proceedings to be published.

M. M. Menon and C. C. Tsai, "Energy Dissipation on Electrodes During High-Voltage Breakdown"

DOE/MIT Compact Ignition Experiments Design Point Workshop, Cambridge, Massachusetts, June 11-13, 1985

J. R. Haines and G. R. Dalton, "Limiter and First Wall Energy Removal for Compact Ignition Tokamaks"

W. R. Hamilton, "Cost Assessment of Compact Ignition Devices"

N. A. Uckan and R. A. Parker, "Ignition Condition and Plasma Operating Regimes—0-D Modeling" (invited paper)

Gordon Research Conference on Plasma Physics, New London, New Hampshire, June 12-14, 1985

H. R. Hicks, "Numerical Techniques for Nonlinear Plasma Computations"

J. A. Holmes, "Physical Processes Involving MHD Turbulence in Current Carrying Plasmas"

U.S./U.S.S.R. Topical Workshop on 3-D Equilibrium and Transport in Tokamaks, Stellarators, and Tandem Mirrors, Kurchatov Institute, Moscow, U.S.S.R., June 17-21, 1985

H. R. Hicks, "MHD Equilibrium and Stability for Stellarator/Torsatron Nonlinear MHD Plasma Evolution"

J. T. Hogan, "3-D Equilibria for Transport Calculations and Flux Averaged Transport in Stellarators and Tokamaks with Ripples"

U.S.-Japan Workshop on Pump Limiter Physics and Experiments, Los Angeles, June 18-21, 1985

D. L. Hillis, "Pump Limiter Experiments on Heliotron-E and ATF"

P. K. Mioduszewski, "Experimental Pump Limiter Results on ISX-B"

NBS Workshop on Theory of Relativistic Effects in Heavy Atoms, Gaithersburg, Maryland, June 24-25, 1985

C. Bottcher, "Pair Production at GeV/nucleon"

U.S.-Japan Workshop on Plasma-Wall Interaction Data Needs Critical to a Burning Core Experiment (BCX), Livermore, California, June 24-27, 1985. Proceedings published as CONF-8506177 by Sandia National Laboratories, Livermore, California. 1985.

R. E. Clausing, "Carbonization Studies"

P. Mioduszewski, "The Beryllium Limiter Experiment on ISX-B"

P. Mioduszewski, "Summary of the Working Group on Beryllium"

U.S./U.S.S.R. Workshop on Stellarator Research, Kharkov Physico-Technical Institute, Moscow, U.S.S.R., June 24-28, 1985

R. J. Colchin, "Status of ATF; Latest Results of TFTR"

H. R. Hicks (presented by J. T. Hogan), "Equilibrium/Stability in Stellarators and Torsatrons"

H. R. Hicks (presented by J. T. Hogan), "3-D Nonlinear Tearing Mode Stability Calculations"

J. T. Hogan, "3-D Moment Equilibrium: Comparison with BETA Code"

J. T. Hogan, "3-D Evolution Using Grad's Algorithm"

J. T. Hogan, "3-D Calculations of Current Generation in Stellarators"

11th International Conference on Numerical Simulation of Plasmas, Montreal, June 25-27, 1985

L. A. Charlton, B. A. Carreras, J. A. Holmes, H. R. Hicks, and V. E. Lynch, "Resistive MHD Numerical Calculations in Toroidal Geometry"

R. C. Goldfinger and D. B. Batchelor, "Propagation and Absorption of Electron Cyclotron Waves in ATF"

H. R. Hicks and V. E. Lynch, "An Introduction to Multitasking in CTSS"

V. E. Lynch, H. R. Hicks, and B. A. Carreras, "Interactive Analysis of the Output from a System of Codes"

Workshop on the Relation Between Mechanical Properties and Microstructure Under Fusion Irradiation Conditions, Ebeltoft, Denmark, June 27-July 2, 1985

M. L. Grossbeck and J. A. Horak (presented by N. H. P. Jkan), "Tensile Properties of V-15 Cr-5 Ti Following Neutron Irradiation and Helium Implantation"

July

Seminar, Max Planck Institut für Plasmaphysik, Garching, Federal Republic of Germany, July 1-2, 1985

J. Hogan, "Application of Grad's Algorithm to Calculation of Stellarator Currents"

IAEA Technical Committee Meeting on Mirror Fusion Research, Tsukuba, Japan, July 8-12, 1985

R. A. Dory for the EBS Team, "ELMO Bumpy Square"

Workshop on Tokamak Startup: Problems and Scenarios Related to the Transient Phases of Long-Burn Operations, Erice, Sicily, Italy, July 14-20, 1985. Proceedings published by Plenum Press, December 1985.

D. Mueller, M. Bell, F. Boody, C. Bush, J. L. Cecchi, S. Davis, H. F. Dylla, P. C. Efthimion, R. J. Hawrylak, K. W. Hill, S. Kilpatrick, P. H. LaMarche, D. Manos, D. McCune, S. S. Medley, S. L. Milora, M. Murakami, D. K. Owens, J. Schivell, G. Schmidt, S. Sesnic, B. Stratton, G. Tait, M. Ulrickson, K-L. Wong, R. D. Woolley, and M. C. Zarnstorff, "Discharge Control and Evolution in TFTR"

J. Sheffield, "Tokamak Startup—Problems and Scenarios Related to the Transient Phases of Ignited Tokamak Operations"

N. A. Uckan and J. Sheffield, "A Simple Procedure for Establishing Ignition Conditions in Tokamaks"

MINIMARS Review Meetings, General Dynamics, San Diego, July 17-19, 1985

G. Gorker, "MINIMARS Power Conversion"

S. S. Kalsi, "MINIMARS Center Cell Magnet Design Options"

J. Yugo, "RF Systems for MINIMARS"

Satellite Meeting on the Atomic Physics of Highly Charged Ions, Palo Alto, California, July 22-23, 1985

C. Bottcher, "Ion Atom Collision at GeV per Nucleon" (invited paper)

F. W. Meyer, "The Electron Cyclotron Resonance Ion Source for Producing High Charge State Ions" (invited paper)

XIV International Conference on the Physics of Electronic and Atomic Collisions, Palo Alto, California, July 24-30, 1985

C. Bottcher, "Towards Numerical Solution of the Threshold Ionization Problem"

D. C. Gregory, "The Role of Indirect Processes in Electron-Impact Ionization of Multicharged Ions"

D. C. Griffin, M. S. Pindzola, and C. Bottcher, "Theoretical Calculations of Electric Field Effects on Dielectronic Recombination by Matrix Diagonalization"

C. C. Havener, H. F. Krause, and R. A. Phaneuf, "Multicharged Ion-Atom Merged-Beams Apparatus"

F. W. Meyer, A. M. Howald, C. C. Havener, and R. A. Phaneuf, "Low Energy Electron Capture by Fully Stripped Light Ions from H and H₂"

A. Miller, K. Tinschert, C. Achenbach, E. Salzbom, R. Becker, and M. S. Pindzola, "Single and Multiple Ionization of Multiply Charged Ions by Electron Impact"

Seminar on Atomic Processes in Fusion Plasmas, Santa Cruz, California, July 31-August 2, 1985

C. F. Barnett, "The Next Decade of Atomic Physics in Fusion" (invited paper)

R. A. Phaneuf, "Single Ionization in Electron-Ion Collisions" (invited paper)

M. S. Pindzola, "Theoretical Studies of Electric Field Effects on Dielectronic Recombination" (invited paper)

August

Industrial Participation Workshop for STX, Oak Ridge, Tennessee, August 2, 1985

Y-K. M. Peng, D. J. Strickler, T. C. Hender, B. A. Carreras, J. A. Holmes, L. A. Charlton, W. A. Houlberg, S. E. Attenberger, S. Borowski, S. P. Hirshman, J. A. Rome, R. H. Fowler, C. C. Tsai, G. H. Neilson, E. A. Lazarus, J. Sheffield, T. C. McManamy, E. Hill, D. Lousteau, W. Wright, R. L. Brown, G. R. Dalton, W. R. Hamilton, S. S. Kalsi, D. K. Lee, S. Litherland, J. Luton, J. Mayhall, K. Shipley, and P. B. Thompson, "The Spherical Torus Experiment (STX): Briefing for the Office of Fusion Energy"

ISIAC VIII Conference, Flagstaff, Arizona, August 2-3, 1985

C. Bottcher, "Numerical Solution of the Time Dependent Dirac Equation" (invited paper)

Ignition Physics Workshop, San Diego, August 8-9, 1985

N. A. Uckan and W. A. Houlberg, "Transport Modeling of Compact Ignition Experiments—0-D and 1½-D Confinement Analysis"

Tokamak Power Systems Studies Review Meeting, Boston, August 12-13, 1985

T. G. Brown, "Fusion Engineering Design Center Advanced Tokamak Reactor Configuration"

A. E. Dabiri, "Material Options for a Commercial Fusion Reactor First Wall"

A. E. Dabiri, "System Studies of Various Concepts"

S. L. Thomson, "The Effect of Beta Limits on Tokamak Economics"

Cryogenic Engineering Conference, Cambridge, Massachusetts, August 12-16, 1985. Papers to be published in *Adv. Cryog. Eng.* 31 (1986).

F. J. Cogswell, Y. Iwasa, J. W. Luc, and J. N. Luton, Jr., "Monitoring Large Superconducting Magnets Using Acoustic Emission Technology"

L. Dresner, "Quench Energies of Potted Magnets, III"

J. W. Luc, W. A. Fietz, R. E. Stamps, G. R. Zahn, and J. R. Miller, "Test Results of the Vapor-Cooled Leads for the IFSMTF"

S. W. Schwenterly, W. A. Fietz, R. E. Stamps, and J. F. Ellis, "Performance of the IFSMTF Helium Refrigerator in Partial-Array Tests"

S. S. Shen, C. T. Wilson, and J. N. Luton, Jr., "Acoustic Emission Measurements for Locating High-Voltage Breakdowns in Large Superconducting Magnet Systems"

R. E. Stamps, J. F. Ellis, W. A. Fietz, and S. W. Schwenterly, "Description and Experience with the Operation of a 300 g/s Helium Purifier in a Fusion Superconducting Magnet Facility (FSMTF)"

Conference on Helium Transfer in Space, Boulder, Colorado, August 20-21, 1985

L. Dresner, A. Kashani, and S. W. Van Sciver, "Studies of Heat Transport to Forced Flow He-II"

1985 International Symposium on Antennas and Propagation, Kyoto, Japan, August 20-22, 1985

C. H. Ma, D. P. Hutchinson, P. A. Staats, K. L. Vander Sluis, D. K. Mansfield, H. Park, and L. C. Johnson, "Submillimeter Wave Propagation in Tokamak Plasmas"

Seminar, Department of Materials Science and Engineering, Cornell University, Ithaca, New York, August 22, 1985

P. J. Maziasz, "Precipitation Behavior and Control in Austenitic Stainless Steels Under Neutron Irradiation"

Annual Review of the Fusion Engineering Design Center by DOE/OFE Officials, Oak Ridge, Tennessee, August 26, 1985

J. D. Galambos, "Optimizer Tokamak Systems Code Trade Studies Results"

D. J. Strickler, "Plasma Magnetics and Equilibrium Modeling"

1985 International Symposium on Antennas and EM Theory, Beijing, China, August 26-28, 1985. Proceedings to be published in *Space Solar Power Review*.

C. H. Ma, D. P. Hutchinson, P. A. Staats, K. L. Vander Sluis, D. K. Mansfield, H. Park, and L. C. Johnson, "Submillimeter Wave Propagation in Tokamak Plasmas"

Workshop on Basic Physical Processes of Toroidal Fusion Plasmas, International School of Plasma Physics, Varenna, Italy, August 26–September 3, 1985

- J. A. Holmes, B. A. Carreras, L. A. Charlton, L. Garcia, H. R. Hicks, V. E. Lynch, and T. C. Hender, "MHD Stability of Toratrons Using the Average Method"
- V. K. Paré, K. McGuire, R. J. Colchin, P. C. Efthimion, E. Fredrickson, K. W. Hill, G. Taylor, and N. R. Sauthoff, "Varieties of Sawtooth Behavior in TFTR Plasmas"
- D. A. Spong, W. A. Cooper, D. J. Sigmar, D. E. Hastings, and K. T. Tsang, "Effect of Trapped Alphas on Ballooning Modes in Tokamak" (invited paper)
- D. J. Sigmar, "The Viscosity Tensor for a Tokamak Plasma with Strong Toroidal Rotation" (invited paper)

September

12th European Conference on Controlled Fusion and Plasma Physics, Budapest, September 2–6, 1985. Contributed papers published in the proceedings, *Europhysics Conference Abstracts Series* 9F, edited by L. Pócs and A. Montvai, European Physical Society, Budapest, 1985. Invited papers published in *Plasma Phys. Controlled Fusion* 28 (1986).

- J. D. Callen, W. X. Qu, and K. C. Shaing, "Neoclassical MHD Instabilities and Transport in Tokamaks," Pt. II, pp. 397–400
- J. Guasp, J. L. Alvarez Rivas, J. A. Fábregas, J. A. Lopez Fraguas, A. P. Navarro, A. Perea, E. R. Solano, J. L. Cantrell, B. A. Carreras, J. H. Harris, T. C. Hender, H. R. Hicks, T. C. Jernigan, V. E. Lynch, J. F. Lyon, and J. A. Rome, "Physics Studies for the TJ-II Device," Pt. I, pp. 441–44
- K. McGuire, J. D. Callen, R. J. Colchin, P. C. Efthimion, E. Fredrickson, K. Hill, V. K. Paré, W. Park, N. Sauthoff, G. Taylor, R. White, and H. Yamada, "Compound Sawteeth and Heat Pulse Propagation in TFTR," Pt. I, pp. 134–37
- S. S. Medley, M. Bitter, A. C. England, E. Fredrickson, S. von Goeler, R. T. Goldston, G. W. Hammett, R. J. Hawryluk, H. W. Hendel, K. W. Hill, R. Kaita, S. M. Kaye, K. McGuire, D. M. Meade, M. Murakami, E. B. Nieschmidt, A. L. Roquemore, S. D. Scott, J. D. Strachan, G. D. Tait, C. E. Thomas, K.-L. Wong, K. M. Young, and M. C. Zarnstorff, "Ion Behavior in TFTR during Ohmic and Neutral Beam Heating," Pt. I, pp. 343–46
- G. H. Miley, W. R. Sutton, and D. J. Sigmar, "Alpha-Driven Fast Magnetosonic Wave Heating in Tokamak Plasmas," Pt. II, pp. 288–91
- M. Murakami, V. Arunasalam, J. D. Bell, M. G. Bell, M. Bitter, W. R. Blanchard, F. Boody, N. Bretz, R. Budny, C. E. Bush, J. D. Callen, J. L. Cecchi, S. Cohen, R. J. Colchin, S. K. Combs, J. Coonrod, S. L. Davis, D. Dimock, H. F. Dylla, P. C. Efthimion, L. C. Emerson, A. England, H. P. Eubank, R. Fonck, E. Fredrickson, H. P. Furth, L. R. Grisham, S. von Goeler, R. J.

- Goldston, B. Grek, R. Groebner, R. J. Hawryluk, H. Hendel, K. W. Hill, D. L. Hillis, W. Heidebrink, R. Hulse, D. Johnson, L. C. Johnson, R. Kaita, R. Kamperschroer, S. M. Kaye, S. Kilpatrick, H. Kugel, P. LaMarche, R. Little, C. H. Ma, D. Manos, D. Mansfield, M. McCarthy, R. T. McCann, D. C. McCune, K. McGuire, D. M. Meade, S. S. Medley, S. L. Milora, D. R. Mikkelsen, D. Mueller, E. Nieschmidt, D. K. Owens, V. K. Paré, H. Park, B. Prichard, A. Ramsey, D. A. Rasmussen, M. H. Redi, A. L. Roquemore, P. H. Rutherford, N. R. Sauthoff, J. Schivell, G. L. Schmidt, S. D. Scott, S. Sesnic, M. Shimada, J. E. Simpkins, J. Sinnis, F. Stauffer, J. Strachan, B. Stratton, G. D. Tait, G. Taylor, C. E. Thomas, H. H. Towner, M. Ulrickson, R. Wieland, J. B. Wilgen, M. Williams, K-L. Wong, S. Yoshikawa, K. M. Young, M. C. Zarnstorff, and S. J. Zweben, "Confinement Studies of Neutral Beam Heated Discharges in TFTR" (invited paper), *Plasma Phys. Controlled Fusion* **28**, 17-27 (1986)
- A. Perea, R. Martin, J. Botija, A. Pueblas, J. R. Cepero, B. A. Carreras, K. K. Chipley, T. C. Hender, T. C. Jernigan, J. F. Lyon, and B. E. Nelson, "Physics Issues in the Design of TJ-II," Pt. I, pp. 433-35
- G. L. Schmidt, S. L. Milora, M. G. Bell, M. Bitter, C. E. Bush, R. J. Colchin, S. K. Combs, A. England, E. Fredrickson, R. J. Goldston, B. Grek, L. Grisham, R. J. Hawryluk, W. Heidebrink, H. W. Hendel, D. Johnson, P. LaMarche, R. Little, D. Mansfield, D. C. McCune, K. McGuire, D. M. Meade, D. Mueller, M. Murakami, E. B. Nieschmidt, H. Park, A. T. Ramsey, D. Rasmussen, J. F. Schivell, S. Sesnic, M. Shimada, F. Stauffer, B. L. Stratton, and G. Taylor, "Production of High Density Plasma by Pellet Injection in TFTR," Pt. II, pp. 674-77
- J. D. Strachan, F. P. Boody, C. Bush, B. Grek, R. J. Hawryluk, D. Heifetz, H. W. Hendel, K. W. Hill, D. W. Johnson, A. T. Ramsey, S. Sesnic, J. F. Schivell, M. Shimada, G. Taylor, and M. C. Zarnstorff, "Formation of Detached Plasmas in TFTR," Pt. I, pp. 339-342
- Course and Workshop on Applications of RF Waves to Tokamak Plasmas, Villa Monastero, Varenna, Italy, September 5-14, 1985**
- K. Felch, R. Bier, L. Fox, H. Huey, L. Ives, H. Jory, N. Lopez, J. Shively, and S. Spang, "Recent Operating Experience with Varian 70-GHz and 140-GHz Gyrotrons"
- E. F. Jaeger, D. B. Batchelor, and H. Weitzner, "Full Wave Finite Difference Calculation for ICRF Wave Fields in Tokamak and Stellarator Geometry"
- D. W. Swain, F. W. Baity, D. J. Hoffman, T. L. Owens, and C. C. Tsai, "The Technology of ICH Systems for Advanced Fusion Experiments"
- 9th International Conference on Magnet Technology (MT-9), Zurich, September 9-13, 1985**
- J. N. Luton, J. A. Clinard, J. W. Lue, W. H. Gray, L. T. Summers, and R. Kershaw, "Internal Pressure Effects in the Airco-LCT Conductor Sheath"
- S. S. Shen, P. L. Walstrom, and C. T. Wilson, "Performance of the Quench Detection System During the Partial-Array Test in IFSMTF"
- S. S. Shen, P. L. Walstrom, C. T. Wilson, and J. S. Goddard, "The Design and Preliminary Test Results of the Quench Detection System for IFSMTF"

Third European Workshop on Problems in the Numerical Modeling of Plasmas, Varenna, Italy, September 10-13, 1985. Proceedings published in *Comput. Phys. Commun.* **40** (1986).

E. F. Jaeger, D. B. Batchelor, H. Weitzner, and J. H. Whealton, "ICRF Wave Propagation and Absorption in Tokamak and Mirror Magnetic Fields—A Full-Wave Calculation," pp. 33-64

MINIMARS Design Review, Livermore, California, September 17, 1985

W. R. Hamilton, "MINIMARS Scoping Cost Assessment"

Workshop on Physics of Highly Charged Ions, CRYISIS Facility, Stockholm, September 23-24, 1985

R. A. Phaneuf, "Collisions Between Highly Charged Ions and Electrons" (invited paper)

NATO Advanced Study Institute on Physics of Electron-Ion and Ion-Ion Collisions, Han sur Leze, Belgium, September 30-October 12, 1985

R. A. Phaneuf, "Experiments on Electron-Impact Excitation and Ionization of Ions"

October

U.S.-U.S.S.R. Stellarator Workshop, Oak Ridge, Tennessee, October 1-2, 1985

R. J. Colchin, "Magnetic Field Alignment Studies on the URAGAN-3 Torsatron"

T. C. Jernigan, "Status of ATF Project"

Tokamak Innovations Workshop, Atlanta, October 3-4, 1985

S. L. Thomson, "The Modular Tokamak Configuration"

Ignition Technical Oversight Committee and MFAC Panel XIV Meetings, Princeton Plasma Physics Laboratory, Princeton, New Jersey, October 9-10, 1985

N. A. Uckan and W. A. Houlberg, "Confinement Analysis of Compact Ignition Experiments"

U.S.-Japan Workshop on Next Step Machine Design, Mito, Japan, October 21-22, 1985

T. G. Brown, "System Studies Overview"

N. A. Uckan, "Physics Design Options for Compact Ignition Experiments"

N. A. Uckan and W. A. Houlberg, "Plasma Operation in Compact Ignition Tokamak Design: Startup and Heating to Ignition"

Second U.S.-Japan Exchange Workshop on Superconducting Magnet Requirements for Long-Pulse Tokamaks, Tokai, Japan, October 29-November 1, 1985

J. W. Lue, "Superconducting Magnet Research and Development Program"

International Pellet Fueling Workshop, La Jolla, California, October 30–November 3, 1985

- S. K. Combs, S. L. Milora, C. R. Foust, G. L. Schmidt, and T. McBride, "Operation of the Repeating Pneumatic Injector on TFTR and Design of an Eight-Shot Deuterium Pellet Injector"
- C. A. Foster, "Centrifuge Injector Development at Oak Ridge National Laboratory"
- C. A. Foster and M. M. Menon, "Electron Beam Rocket Pellet Accelerator"
- S. L. Milora, G. L. Schmidt, S. K. Combs, C. E. Bush, R. J. Goldston, B. Grek, R. J. Hawryluk, D. Johnson, D. Mansfield, H. Park, M. Shimada, and F. Stauffer, "Production of High Density Plasma by Pellet Injection in TFTR"
- D. D. Schuresko, "Results of Pellet Injection into ISX-B"
- D. D. Schuresko, "Studies of Pellet Acceleration with Arc Discharge Heated Propellants"

November

APS/DPP Satellite Meeting, U.S./Israel Plasma Physics Workshop, San Diego, November 3, 1985

- E. C. Selcow, Y-K. M. Peng, N. A. Uckan, and W. A. Houlberg, "FEDC Physics Systems Code and Application to Tokamak Ignition Experiments"

Ignition Technical Oversight Committee Meeting, San Diego, November 3, 1985

- N. A. Uckan, J. Sheffield, and H. P. Furth, "Figure-of-Merit Parameters Under Different Scaling Assumptions"

Ignition Physics Study Group Meeting, San Diego, November 4, 1985

- J. Sheffield et al., "Ignition Physics Study Group Status Report"

27th Annual Meeting of the Division of Plasma Physics of the American Physical Society, San Diego, November 4–8, 1985. Abstracts published in *Bull. Am. Phys. Soc.* 30, No. 9 (1985).

- V. Arunasalam, P. C. Efthimion, R. J. Goldston, B. Grek, D. W. Johnson, D. C. McCune, G. Taylor, M. Murakami, D. Rasmussen, J. Wilgen, and F. J. Stauffer, "Electron Power Balance Studies of H^0 into D^+ and D^0 into D^+ Neutral Beam Injection Experiments in TFTR," p. 1521
- S. E. Attenberger, W. A. Houlberg, and S. P. Hirshman, "MOMCON 3-D Equilibrium Moments Code for Transport Calculations," p. 1398
- D. B. Batchelor, A. Fruchtman, and H. Weitzner, "Electron Cyclotron Resonance Heating in Parallel Stratified Plasmas," p. 1591
- W. R. Becraft and J. H. Whealton, "Neutral Beam Concept: 150 mA/cm² at 1,000,000 Volts," p. 1596
- M. A. Bell, J. H. Whealton, R. J. Raridon, R. W. McGaffey, and D. E. Wooten, "Optimization of Accelerator for Dudnikov Negative Ion Source," p. 1609

- T. S. Bigelow, "Improved High Power Microwave Launchers for Lower Hybrid Heating and Current Drive," p. 1624
- D. T. Blackfield, L. J. Perkins, and R. B. Campbell, "Tandem Mirror Reactor Engineering and Physics Trade Studies," p. 1475
- F. P. Boody, C. E. Bush, S. S. Medley, H. Park, and J. Schivell, "Macroscopic Edge Phenomena in TFTR: MARFes, Moving MARFes, and Detached Plasmas," p. 1518
- N. L. Bretz, M. Bitter, P. C. Efthimion, B. Grek, K. W. Hill, S. M. Kaye, G. Taylor, C. E. Bush, H. W. Hendel, and F. J. Stauffer, "Size Scaling in TFTR Ohmic Discharges," p. 1517
- C. E. Bush, J. Schivell, F. P. Boody, H. F. Dylla, P. C. Efthimion, K. W. Hill, R. A. Hulse, P. H. LaMarche, S. S. Medley, S. Sesnic, B. C. Stratton, M. Ulrickson, and M. C. Zarnstorff, "Radiated Power from Ohmically Heated TFTR Plasmas," p. 1517
- J. D. Callen and K. C. Shaing, "Neoclassical MHD Equations for Tokamaks," p. 1424
- A. Carnevali, J. Misium, K. A. Connor, J. R. Goyer, R. L. Hickok, W. C. Jennings, J. C. Glowienka, and J. K. Munro, "Heavy Ion Beam Probe Diagnostic System for ATF-1," p. 1419
- B. A. Carreras, "Finite Beta Magnetic Surfaces in a Fixed Toroidal Plane for Stellarator Configurations," p. 1574
- M. D. Carter, J. D. Callen, and D. B. Batchelor, "Fokker-Planck Heating Coefficients Utilizing Nonlinear Wave-Particle Interactions at Cyclotron Harmonics," p. 1592
- J. B. O. Caughman II, F. W. Baity, D. J. Hoffman, T. L. Owens, D. N. Ruzic, D. E. Schechter, C. C. Tsai, and T. H. Zerguini, "The Effects of RF on the Sputtering Rate of ICRH Antenna Components," p. 1590
- L. A. Charlton (in collaboration with B. A. Carreras, T. C. Hender, H. R. Hicks, J. A. Holmes, and V. E. Lynch), "Tokamak M.H.D. Activity: Calculations Using a Full Set of Nonlinear Resistive M.H.D. Equations," p. 1479 (invited paper, see also p. 1423)
- G. L. Chen, J. H. Whealton, R. W. McGaffey, R. J. Raridon, M. A. Bell, E. F. Jaeger, and D. J. Hoffman, "Electromagnetic Fields in 3-D for Various Cavity Antennas and Faraday Shields," p. 1589
- R. J. Colchin, S. I. Fedotov, J. H. Harris, D. K. Lee, F. Ozherel'ev, O. S. Pavlichenko, D. L. Pogozhev, J. A. Rome, J. Treffert, and V. M. Zalkind, "Measurements of Error Fields on the URAGAN-3 Torsatron," p. 1576
- S. K. Combs, S. L. Milora, C. R. Foust, G. L. Schmidt, and T. McBride, "Performance of Repeating Pneumatic Pellet Injector on TFTR," p. 1520
- W. A. Cooper, "ICRF Wave Equations in Stellarator Configuration," p. 1587
- E. C. Crume, Jr., K. C. Shaing, S. P. Hirshman, C. O. Beasley, Jr., and W. I. van Rij, "Finite-Aspect-Ratio Tokamak Viscosities in the Deep Banana Regime," p. 1425
- W. K. Dagenhart, W. L. Stirling, C. C. Tsai, and D. E. Schechter, "A Two Cathode S₁TEX Plasma Generator," p. 1609

- M. C. Depassier, W. A. Cooper, and S. P. Hirshman, "Helically Symmetric Equilibria," p. 1580
- H. F. Dylla, P. H. LaMarche, J. E. Simpkins, M. Ulrickson, W. R. Blanchard, F. P. Boody, C. E. Bush, R. J. Groebner, D. Manos, P. J. McCarthy, J. Schivell, and B. C. Stratton, "Chromium Getter Studies in TFTR," p. 1517
- P. C. Efthimion, M. Bitter, N. L. Bretz, H. F. Dylla, B. Grek, R. J. Hawryluk, K. W. Hill, D. W. Johnson, H. K. Park, A. T. Ramsey, S. Sesnic, B. C. Stratton, G. Taylor, M. C. Zarnstorff, C. E. Bush, M. Murakami, D. Rasmussen, M. Shimada, and R. J. Groebner, "Confinement Studies of Ohmically Heated Deuterium and Helium Plasmas in TFTR," p. 1385
- A. C. England, D. L. Hillis, and P. H. Edmonds, "The Migration of Radioactive Beryllium in the ISX-B Tokamak," p. 1440
- K. Felch, R. Bier, L. Fox, H. Huey, L. Ives, H. Jory, N. Lopez, and S. Spang, "First CW Tests on the Varian 140 GHz Gyrotron," p. 1548
- C. A. Foster, "A High Throughput Continuous Cryopump for Fusion Applications"
- C. A. Foster, D. E. Schechter, S. K. Combs, S. L. Milora, C. C. Tsai, and M. M. Menon, "Actively Cooled and Pumped Pump Limiter Design," p. 1627
- R. H. Fowler and J. A. Rome, "Studies of the Plasma Edge Region in Stellarators Using a Field Line/Orbit Code," p. 1367
- E. Fredrickson, J. Callen, R. J. Colchin, K. McGuire, V. Paré, N. Sauthoff, and M. C. Zarnstorff, "MHD Activity and Heat Pulse Propagation on TFTR," p. 1518
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- M. D. Galloway, G. H. Neilson, L. Garcia, J. H. Harris, and G. R. Dyer, "Magnetic Configuration Diagnostics for ATF," p. 1575
- L. Garcia, B. A. Carreras, and P. H. Diamond, "Numerical Studies of Resistive Pressure Driven Turbulence," p. 1394
- W. L. Gardner, D. J. Hoffman, H. C. McCurdy, T. J. McManamy, J. A. Moeller, and P. M. Ryan, "The Oak Ridge RF Test Facility," p. 1590
- R. C. Goldfinger and D. B. Batchelor, "Electron Cyclotron Heating in ATF," p. 1590
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Appendix 2

ABBREVIATIONS AND ACRONYMS

ADIP	Alloy Development for Irradiation Performance
AEM	analytical electron microscopy
AES	Auger electron spectroscopy
ALT-II	Advanced Limiter Test II
appm	atomic parts per million
ASM	average statistical model
ATF	Advanced Toroidal Facility
BCSS	Blanket Comparison and Selection Study
CCD	charge-coupled device
CFADC	Controlled Fusion Atomic Data Center
CFR	Code of Federal Regulations
CIT	Compact Ignition Tokamak
CNR	Center for Neutron Research
COE	cost of electricity
CPU	central processing unit
CW	cold-worked
cw	continuous wave
CXE	charge-exchange excitation
DAFS	Damage Analysis and Fundamental Studies
DOE	U.S. Department of Energy
dpa	displacements per atom
DPI	deuterium pellet injector
DR	dielectronic recombination
D-T	deuterium-tritium
DIII	Doublet III
DIII-D	upgrade of DIII
EA	excitation-autoionization
EBR-II	Experimental Breeder Reactor upgrade
EBS	ELMO Bumpy Square
EBT	ELMO Bumpy Torus
EBT-I	Initial EBT device and configuration (microwave power of ≤ 60 kW at 18 GHz, magnetic field of 5 kG)
EBT-S	EBT-Scale device and configuration (microwave power of ≤ 200 kW at 28 GHz, magnetic field of 7.2 kG)
ECH	electron cyclotron heating
ECR	electron cyclotron resonance
ECRH	electron cyclotron resonance heating
EF	equilibrium field
EURATOM	European Atomic Energy Community
FEDC	Fusion Engineering Design Center
FER	Fusion Engineering Reactor
FFTF	Fast-Flux Test Facility
FIR	far-infrared
FLR	finite Larmor radius
FMIT	Fusion Materials Irradiation Test facility
FWHM	full width at half-maximum
FY	fiscal year
GE	General Electric Company

HF	helical field
HFIR	High-Flux Isotope Reactor
HFPS	helical field power supply
HIBP	heavy-ion beam probe
HWHM	half-width at half-maximum
ICH	ion cyclotron heating
ICRF	ion cyclotron range of frequencies
ICRH	ion cyclotron resonance heating
IFSMTF	International Fusion Superconducting Magnet Test Facility
INTOR	International Tokamak Reactor
IR	infrared
IRO	Information Resources Organization
ISX	Impurity Study Experiment
ISX-B	upgrade of ISX
JAERI	Japan Atomic Energy Research Institute
JEN	Junta de Energia Nuclear
JET	Joint European Torus
JILA	Joint Institute for Laboratory Astrophysics
LCP	Large Coil Program
LCT	Large Coil Task
LCTF	Large Coil Test Facility (now IFSMTF)
LEED	low-energy electron diffraction
LH	lower hybrid
LION	Library Information Online Network
LLNL	Lawrence Livermore National Laboratory
MARS	Mirror Advanced Reactor Study
MGPF	Multi-General Purpose Facility
MHD	magnetohydrodynamic
MIT	Massachusetts Institute of Technology
NET	Next European Torus
NLTNIF	National Low-Temperature Ion Irradiation Facility
NMFECC	National Magnetic Fusion Energy Computing Center
ODE	ordinary differential equation
OFE	Office of Fusion Energy
OH	ohmic heating
O-mode	ordinary mode
ORGDP	Oak Ridge Gaseous Diffusion Plant
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Research Reactor
PCA	prime candidate alloy
PCB	polychlorinated biphenyl
PDX	Poloidal Divertor Experiment
PF	poloidal field
PIG	Penning ion gauge
PLC	programmable logic controller
PLT	Princeton Large Torus
PMI	Plasma-Materials Interactions
PMT	photomultiplier tube

POPCON	plasma operation contour
PPPL	Princeton Plasma Physics Laboratory
PTP	peripheral target position
QA	quality assurance
QAA/P	QA Assessment/Plan
RBS	Rutherford ion backscattering
RDL	resonant double loop
rf	radio frequency
RFP	reversed-field pinch
RFTF	Radio-Frequency Test Facility
RIS	radiation-induced solute segregation
RPI	repeating pellet injector; Rensselaer Polytechnic Institute
RSIC	Radiation Shielding Information Center
R&D	research and development
SA	solution-annealed
SEM	scanning electron microscope
SFT	stacking fault tetrahedra
SPM	Special-Purpose Materials
STX	Spherical Torus Experiment
TEM	transmission electron microscopy
TEXT	Texas Tokamak
TF	toroidal field
TFCX	Tokamak Fusion Core Experiment
TFTR	Tokamak Fusion Test Reactor
TMX	Tandem Mirror Experiment
TMX-U	TMX Upgrade
TPOP	tritium proof-of-principle
TSC	Tokamak Systems Code
USC	User Service Center
VF	vertical field
VSWR	voltage standing-wave ratio
X-mode	extraordinary mode
YAG	yttrium-aluminum-garnet
0-D	zero-dimensional
1-D	one-dimensional
1½-D	one-and-one-half-dimensional
2-D	two-dimensional
3-D	three-dimensional

Appendix 3

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