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OAK
RIDGE
NATIONAL
LABORATORY

MASTER

**FUSION ENERGY DIVISION
ANNUAL PROGRESS
REPORT**

Period Ending December 31, 1979

OPERATED BY
UNION CARBIDE CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

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FUSION ENERGY DIVISION ANNUAL PROGRESS REPORT

Period Ending December 31, 1979

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Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

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INTRODUCTION

During the year covered by this report the Fusion Energy Division of Oak Ridge National Laboratory (ORNL) made significant accomplishments in a number of existing programs and initiated several new ones that will have an important and far-reaching impact on the future of fusion research at the laboratory. Our program reflects the status of fusion research, which can now anticipate scientific feasibility and is concentrating on engineering and technology. We attempt to make our activities relevant to the most pressing needs of the fusion program by relating them to the next major step required in the development of fusion, the Engineering Test Facility (ETF). We are fortunate to have the ETF Design Center located at ORNL, and we strive for a very close working relationship between this national staff and our research and development (R&D) activities.

Fusion research at ORNL is characterized by unusual breadth in the several scientific disciplines that contribute to this international effort toward a long-term solution to the world energy problem. The ORNL program includes major efforts in the categories of experimental fusion physics, fusion theory, and technology development. However, it also has well-developed activities in fusion systems analysis, materials science applications in fusion, the neutronics of fusion reactors, related atomic and molecular physics, and the environmental science aspects of future fusion reactors. This breadth of scientific competence, coupled with the extensive resources of the UCC-ND Engineering organization, has allowed the ORNL program to undertake a number of complex multidisciplinary tasks for the Department of Energy (DOE).

Accomplishments in experimental fusion physics center in the two major confinement experiments of the program, the ELMO Bumpy Torus-Scale (EBT-S) experiment and the Impurity Study Experiment (ISX-B) tokamak, both located in the experimental fusion facilities in Building 9201-2 of the Y-12 Plant in Oak Ridge. For much of the year normal operation of the EBT-S experiment was discontinued to allow refurbishment of the two large motor-generator sets that provide power for the magnetic field coils. However, this operational gap provided the opportunity for a number of improvements in the supporting facilities for EBT-S. Included were increases in the power level, stability, and reliability of the 50-kV, 18-GHz microwave source; improvements in the controls, mounting support, and cooling system for the 28-GHz microwave source; and significant improvements in the diagnostic instrumentation and data-handling equipment for the experiment. Work continued at Varian, Inc., subcontractor for the development of the 28-GHz gyrokystrons, on suitable output structures for the tubes and on increases in the power level.

The ISX-B tokamak experiment provided a year of excellent operation with significant results in spite of troublesome failures in new high-voltage power supplies. Experiments seeking higher beta values achieved a root-mean-square value of 3.3%, which is above the theoretical threshold for pressure-driven MHD instabilities. Plasma cross-section elongations of 1.5 were achieved in the first use of the shaping coil system for noncircular plasmas. The newly developed pellet fueling system injected 1-mm-diam frozen hydrogen pellets into the plasma at a velocity of 1 km/s, producing highly useful five-fold increases in plasma density. The application of 85 kW of 35-GHz microwave energy from a gyrotron developed by the Naval Research Laboratory to the plasma for electron cyclotron resonance heating produced a 50% increase in the electron temperature of the central plasma and allowed a 40% decrease in initial loop voltage through preionization. Neutral beam injection power was increased to 1 MW, with confirmation of expected plasma heating. Further investigations were made of impurity effects and the effects of magnetic field ripple. The interpretation of the ISX-B experiments was enhanced greatly by the addition of a new formic acid far-infrared laser system and a new low-energy particle spectrometer.

Work in plasma theory at ORNL not only encompasses the fundamental understanding of confinement and transport in tokamak and EBT configurations, but also includes analysis of fundamental problems in technology development, contributions to the analysis of reference designs for future fusion reactor systems, and involvement in the design and interpretation of major fusion experiments. Among the achievements of this year were the formulation of a model for predicting energy loss in the hot electron rings of EBT, the enhancement of resistive MHD tokamak stability codes to model more conductive plasmas and finite beta and aspect ratio effects, and the development of codes for analyzing and minimizing losses caused by magnetic field ripple and bundle divertors. In addition, major theoretical studies were made in support of proposals to build EBT-P (proof of principle) and to upgrade ISX-B to ISX-C.

The computing and data-handling facilities of the program were improved by upgrading the DEC KL-10 central computer to a Model B processor, as well as by completing the installation of an Ampex 256K core memory. The PDP 11/45 software was made fully operational as a front-end processor for the KL-10. Furthermore, various improvements and additions were made in the disk storage, terminal, and graphics display capabilities of the computing system.

Continued progress was made in the area of neutral beam injection systems. The two injection systems for the ISX-B tokamak were completed and operated at the 1-MW level. Also, the ion source system under development for the Poloidal Divertor Experiment device at Princeton Plasma Physics Laboratory was successfully operated at 100 A through a 30-cm-diam grid. Investigations into energy recovery methods for neutral injection systems resulted in the recovery of 60% of the energy of the nonneutralized beam. Studies and facility proposals were prepared for the development of deuterium beam injection systems, and studies were initiated on advanced positive-ion injection systems suitable for ETF.

The program also continued its management of the international Large Coil Program, the design and manufacturing development of the three U.S.-produced coils continuing with some modifications in the pace of the program. The Large Coil Test Facility, located in Building 9204-1 at the Y-12 Plant, entered the construction phase, and much of the basic structure of the vacuum vessel is now in place. R&D proceeded on advanced 12-T conductors and coils, and a major design effort was completed on prototype coils for EBT-P.

Furthermore, the ORNL fusion program was assigned host and management responsibilities for the ETF Design Center, where a reference design for this major fusion experiment scheduled for the early 1990s will be developed. The staff of the Center has been assembled, with participants from other fusion research laboratories and from industry. The organization of the work has been completed, and the design work is now under way.

Also, the materials science and neutronics activities of the program continued work in the categories of alloy development, permeation studies, damage analysis, neutronics analysis, and radiation effects on organic materials.

During the period of this report the ORNL fusion program also accepted the responsibility for management of the EBT-P program. EBT-P will be a major fusion experiment to establish proof of principle of the ORNL-developed EBT as a suitable alternative to tokamak and mirror configurations in future fusion reactors. The task of the EBT-P program is to develop, with large-scale participation from private industry, a preliminary design for the experiment and to evaluate proposals for its construction and operation. ORNL will recommend one proposal based on the construction of the facility at the Y-12 site and another based on siting elsewhere. From these two DOE will select the proposal to be implemented.

Further, the program has prepared and is pressing acceptance of a proposal to upgrade the ISX-B tokamak to a significantly enhanced new experiment - ISX-C. This tokamak would address, among others, problems of long-pulse duration and advanced divertor technology. ISX-C is an important step in the near-term national fusion research program, and its approval is vital to the ORNL fusion program as a vehicle for major future participation in fusion research.

The future impact of ISX-C, EBT-P, and perhaps ETF on the ORNL fusion program can be enormous. The leadership and staff of the program are proceeding with careful optimism in the planning and management activities that will bring these major tasks to the program. The successful efforts of this talented, multifaceted staff in recent years provides a suitable momentum and managerial base for these new opportunities. We will not forfeit, through lack of initiative or effort, these opportunities to make significant and permanent contributions to our chosen field of endeavor and through these to the energy needs of coming generations.

Lee A. Berry, Director
ORNL Fusion Program

O. B. Morgan, Jr., Director
Fusion Energy Division

4. EXPERIMENTAL CONFINEMENT

J. Sheffield, Section Head
J. L. Dunlap, Associate Section Head Tokamak Group
G. R. Haste, Associate Head for EBT
H. O. Eason, Microwave Development

G. B. Adams ¹	M. Davis ¹⁵	K. E. Hackett ²²	C. M. Loring	J. W. Pearce	D. J. Strickler ¹
B. A. Appieton ²	W. A. Davis	C. B. Haizlip ¹⁵	R. Lucy ²⁰	Y.-K. M. Peng ¹³	D. W. Swain
D. Anuricus ²	S. M. DeCamp	S. L. Halstead	R. V. Lunsford, Jr. ¹	G. F. Pierce	C. E. Thomas ²²
F. W. Bailey	H. De La Fuente ¹⁵	C. E. Hammons ¹³	V. E. Lynch ³	J. L. Ping ¹	D. M. Thomas ²³
C. F. Barnett ⁴	R. D. Donaldson	H. H. Haselton ¹⁴	J. E. Lyon	N. S. Pontel ¹⁴	D. H. Thompson ¹
D. D. Bates ¹	R. A. Dory ²²	L. Heatherly ¹⁰	C. H. Ma ²⁷	B. L. Pope	P. E. Thompson ¹
S. C. Bates	J. L. DeCap	R. E. Hill ¹⁵	L. Maddox ³	C. C. Quen ¹	M. Tillack ²²
M. C. Becker ⁵	G. R. Dyer	D. L. Hillis	J. M. Madison ⁵	E. J. Rappaport ²²	J. E. Tracey ²²
J. D. Bell ³	R. B. Easter ¹	J. T. Hagan ¹⁵	L. A. Massengill	T. F. Ravhurn	C. C. Isa ¹⁴
F. M. Bientosek ⁶	P. H. Edmonds	H. C. Howe ¹³	D. Mattox ¹⁵	M. E. Read ²	T. C. Tucker ³
E. Blair ²³	O. C. Eldridge ¹⁷	B. Hui ⁹	H. L. McCurdy	W. J. Redmond	T. Uckan
A. Broverman ¹	J. F. Ellis ¹⁸	W. R. Husinsky	M. C. McCurfin	R. K. Richards	K. L. Vander Sluis ⁴
P. F. Brown	L. C. Emerson ¹⁰	D. P. Hutchinson ¹⁴	M. M. Meoni ¹⁴	J. B. Roberto ²	R. D. Wall ¹
W. E. Bryson	A. C. England	R. C. Isler	J. T. Mihalczol	J. A. Rome ¹³	J. E. Warwick ¹
P. B. Burn ⁷	C. A. Foster ¹⁴	D. L. Jassby ²¹	S. L. Milora ¹⁴	P. M. Ryan ¹⁴	J. A. White ¹
K. E. Burrell ⁸	R. H. Fowler ³	T. C. Jernigan ²⁴	D. B. Montgomery ²²	M. J. Saltmarsh	T. L. White
R. D. Burris ³	J. E. Francis ¹³	O. V. Jett	E. C. Moore	H. C. Sanderson	P. V. Whitfield ³⁴
C. E. Bush	P. Gelpi	R. L. Johnson ¹	C. H. Muller ³	C. R. Schaich	J. B. Whitley ¹⁵
G. L. Campen ¹	K. A. Geoffroy	K. Kasai ²⁵	J. L. Munro ³	D. E. Schechter ¹⁴	K. M. Wieland ³
L. A. Charlton ³	R. M. Gilgenbach ²³	G. G. Kelley ¹	M. Murakami ²⁸	B. Schweer ³²	J. B. Hilgen
K. R. Chu ⁹	J. C. Glowienka	H. E. Ketterer	C. W. Murphy ¹	S. D. Scott ²²	W. R. Wing
R. E. Clausinger ¹⁰	R. Goldston ²¹	H. J. Kim ¹⁴	L. E. Murray ²³	J. E. Simpkins	R. C. Wintenberg ⁵
J. A. Cobble	G. G. Gonzalez	P. W. King	A. P. Navarro ³⁰	E. Slaughter, Jr. ¹	S. P. Withrow ²
R. J. Colchin ¹⁴	V. L. Granatstein ²	A. Komori	R. V. Neidigh ³¹	S. J. Sosnowski, Jr. ¹	S. K. Wong ⁶
W. A. Cooper ¹²	D. E. Gray ¹	A. G. Kulcha ¹⁷	G. H. Nelson	P. A. Staats ¹⁴	K. E. Worsham ³⁵
R. L. Deland	D. H. Gray ¹	M. H. Kunselmann ¹	B. E. Nelson ¹	J. S. Stanton ³	R. E. Wright ¹⁴
E. C. Crume, Jr. ¹²	D. Greenwood ³	R. A. Langley ²⁶	J. A. O'Toole ¹³	K. A. Stewart ¹¹	W. L. Wright ¹
J. S. Culver ¹¹	A. D. Guttery	E. A. Lazarus	D. R. Overbey ¹⁵	W. L. Stirling ¹⁴	R. B. Wysor ¹
W. K. Darenhart ¹⁴	M. P. Hacker ⁸	R. L. Livesey	V. K. Paré ¹	J. D. Stout ¹	T. F. Yang ²²
		J. W. Yarbrough ¹	R. A. Zuhrt ¹	C. Zurro ³¹	

^{*}Part-Time

[†]Retired

1. UCC-ND Engineering
2. Solid State Division
3. Computer Science Division
4. Physics Division
5. Instrumentation and Controls Division
6. Rensselaer Polytechnic Institute
7. Quality Assurance and Inspection Division
8. General Atomic Company, San Diego, California
9. Naval Research Laboratory, Washington, D.C.
10. Metals and Ceramics Division
11. EBT-S Coordinator
12. University of Michigan, Ann Arbor, Michigan
13. Plasma Theory Section
14. Plasma Technology Section
15. Sandia Laboratory, Albuquerque, New Mexico
16. Visiting Scientist, Mexico
17. University of Tennessee, Knoxville, Tennessee
18. Magnetics and Superconductivity Section
19. Grumman Aerospace Corporation, Bethpage, New York

20. JAYCOR, Alexandria, Virginia
21. Princeton Plasma Physics Laboratory, Princeton University, Princeton, New Jersey
22. Massachusetts Institute of Technology, Cambridge, Massachusetts
23. Technical University of Vienna, Vienna, Austria
24. Coordinator, ISX-C Program
25. JAERI, Tokai, Japan
26. Coordinator, Surface Physics Program
27. University of Mississippi, Oxford, Mississippi
28. Coordinator, ISX-B Experiment
29. Grinnell College, Grinnell, Iowa
30. Visiting Scientist, Spanish Nuclear Energy Commission, Madrid, Spain
31. Consultant, FED
32. KFA, Jülich, Federal Republic of Germany
33. University of Texas, Austin, Texas
34. ARAMCO, Saudi Arabia
35. Project Manager, Diagnostics

1. EXPERIMENTAL CONFINEMENT

ABSTRACT. ELMO Bumpy Torus (EBT) operations were halted for much of the year because of a major overhaul of the motor-generator sets that provide power for the steady-state toroidal magnets. At peak steady-state output, the generators supply 10 MW. The system had operated for over 35 years without problem, but in recent years, it became apparent that a major overhaul was necessary. The two motor-generator sets were refurbished, one set at a time, ultimately requiring 4½ months per set to complete.

The long down time for generator repair provided ample opportunity for work on a series of machine upgrades that had been contemplated for some time, one of which particularly concerned the 18-GHz power. It had been noted that the level of reliable steady-state 18-GHz power had been dropping because one of the klystrons was weak. The down time allowed the klystron to be replaced and, in addition, the overall drive power to be increased. The net result was reliable operation above 50 kW steady state and a fairly uniform distribution of 18-GHz power around the torus. In addition, considerable progress was made in improving the instrumentation and control for the 28-GHz gyrotron. Furthermore, the support structure, water cooling, etc., for an axial output gyrotron are nearly complete and in place.

Several improvements have been made to the plasma diagnostic array, and the details can be found in the description of each diagnostic. Much time and effort were spent discerning and improving the limits of accuracy for the diagnostics. One of the additions, although only partially implemented, is computer-aided data acquisition.

The ISX-B (Impurity Study Experiment) program emphasized the study of several issues relevant to the development of fusion reactors. These studies included high-beta tokamak operation, plasma shaping, pellet fueling, electron cyclotron heating (ECH), magnetic field ripple effects, and impurity analysis and control. An rms beta of 3.3% has been achieved; this is above theoretical thresholds for pressure-driven MHD instabilities. Plasmas that are D-shaped with an elongation of up to 1.5 have been produced, and density rises of up to a factor of five were observed when 1-mm-diam pellets were injected into the plasma at a velocity of ≈ 1 km/s. A 35-GHz gyrotron injected 85 kW of microwaves into a plasma, resulting in a 50% increase in the central electron temperature. Initial loop voltage was reduced $\approx 40\%$ as a result of ECH preionization. Magnetic field ripples of up to 1.5% have produced large fast ion losses but much smaller thermal ion losses and flux surface perturbations. Characteristics of the edge plasma were studied by surface analysis (ion scattering, channeling, and nuclear reaction techniques) of samples placed in the shadow of the limiter. Laser-induced fluorescence diagnostic systems developed by General Atomic (GA) and Jülich have successfully measured impurity density near the plasma edge for the first time in tokamaks.

The successes in the ISX-B program were made possible by significant improvements in tokamak operations, neutral beam systems, and data handling systems. Neutral beam power of ≈ 1 MW was reliably provided, and computer interfacing of new and existing diagnostics continued to expand and improve.

A formic-acid FIR laser was used successfully as a multichannel interferometer and for Faraday rotation measurements. Details of this new diagnostic are presented along with results from other diagnostics.

Design studies for two future programs (ISX-C and bundle divertor) are presented.

1.1 EBT RESEARCH

1.1.1 Heavy Ion Beam Probe

E. M. Bielesek

Measurements with the heavy ion beam probe have continued on EBT-I during 1979, and the first potential measurements have been made on EBT-S. The EBT-I is generally characterized by a positive potential in the surface plasma on the order of 100 V and a negative potential well in the core plasma of up to 300 V with respect to the surface potential. The potential structure of EBT-S closely resembles that of EBT-I. The well depth and peak potential are similar to or greater than those of EBT-I, given the same power input.

A Cs^+ beam was used. Figure 1.1 shows a calculation of cross sections for the production of Cs^{2+} and Cs^{3+} for a fixed beam energy. This calculation shows a sensitivity to all three electron distributions - surface, bulk, and hot electron ring. Turning off the microwave power to the beam probe cavity caused a 20-50% drop in the Cs^{3+} signal in the region of the expected hot electron ring location. This drop is not observed on the Cs^{2+} signal, implying the existence of hot electrons in the presence of local microwave power. The observed drop is localized and consistent with the electron ring measured with other diagnostics.

Fig. 1.1. Effective cross sections for electron impact ionization.

If it is determined where in the plasma the Cs^{2+} signal originates, an estimate of the profile of $nf(T_e)$ can be constructed. In Fig. 1.1, $f(T_e)$ is the effective cross section. Although this estimate cannot be transformed directly into a density profile, variations in the profile of $nf(T_e)$ have been observed. For example, the observed profiles in the C- and T-modes are different.

Scaling studies of the potential structure on EBT-I show (a) variations in peak potential and well depth as the filling pressure is changed (Fig. 1.2) but little change in $nf(T_e)$ at the center and (b) variations in the size of the potential structure as the toroidal magnetic field is changed (Fig. 1.3). The rim of the potential well moves approximately with the expected position of the hot electron ring (shaded areas). The peak potential and well depth change dramatically as the toroidal field current increases from 4000 to 5500 A, but the sum of these two quantities remains approximately constant.

Fig. 1.2. Peak electric potential, potential well depth, and $nf(T_e)$ vs fill gas pressure.

Fig. 1.3. Horizontal variation of the space potential at four toroidal magnetic fields.

1.1.2 Hard and Soft X-Ray Measurements

D. L. Hillis, G. R. Haste

The primary resonant heating of the plasma in EBT-I is provided by klystrons that deliver up to 60 kW of 18-GHz power and 30 kW of 10.6-GHz power. At the end of 1979 additional ECH was added to EBT using a 28-GHz cw gyrotron that has operated at power levels up to 54 kW. The effects of this additional ECH on the plasma of EBT-S have been studied via both hard and soft x-ray measurements.

Hard x-ray measurements have been performed by observing the energetic x rays emitted from the hot electron annulus of EBT in a NaI(Tl) detector 12.7×12.7 cm surrounded by a lead collimator. The detector views the plasma in the midplane of one of the 24 cavities. From these measurements the temperature and density of the hot electron annulus in EBT were deduced. Before the introduction of additional 28-GHz ECH power, typical values in EBT-I for the hot electron annulus were $T_e \approx 100$ keV and $n_e \approx 2.5 \times 10^{11} \text{ cm}^{-3}$. With the additional 28-GHz power (EBT-S) the values of T_e increased

significantly, yielding $T_e \sim 500$ keV and $n_e \sim 2.5 \times 10^{11} \text{ cm}^{-3}$. A comparison of data obtained from the hard x-ray measurements in the EBT-I and EBT-S configurations is shown in Fig. 1.4.

To measure the soft x-ray component of the bulk plasma, a windowless Si(Li) detector views the midplane region of the EBT-S through an evacuated tube. The low energy soft x-ray component from the bulk plasma is superimposed on a high energy x-ray component that arises from the hot electron annulus. These two components of the soft x-ray spectrum are unfolded and a value of T_e for the bulk plasma is obtained. An analysis of the data taken during the initial operation of EBT-S with the 28-GHz gyrotron shows higher peak electron temperatures for the bulk plasma than were observed in EBT-I. Typical values of T_e for EBT-I as deduced from the soft x-ray data range from 200-300 eV, while T_e values for EBT-S with 40-50 kW of 28-GHz power vary between 250-600 eV. Figure 1.5 shows a comparison between the values of T_e determined from the soft x-ray data and those observed in Thomson scattering. (See also Sect. 1.1.3.)

Fig. 1.4. Hard x-ray spectra comparing EBT-S with EBT-I.

Fig. 1.5. A comparison of the bulk plasma electron temperature determined by the soft x-ray and Thomson scattering diagnostics. The temperature is plotted vs microwave power; however, the neutral pressure was not held constant.

Overall, there is agreement between the values of T_e determined by soft x-ray analysis and those found in Thomson scattering.

1.1.3 Thomson Scattering

J. A. Cobble, L. Bigiel

The first iteration on the Thomson scattering experiment on EBT is complete. In this phase of operation, data were taken for one wavelength at a time, with ten to twenty laser shots in a single data set. Thus, for this to produce a useful data set the plasma must be time independent. For EBT-I operation with 40 to 45 kW of microwave power, the electron temperature was 100-200 eV, ΔT_e equaled T_e due to statistics, the density was in the mid- 10^{11} - cm^{-3} range, and the collisionality parameter ν/ω_0 was less than or equal to one. The discrepancy with soft x-ray measurements is not understood.

A polychromator has been incorporated into the system; it allows simultaneous data acquisition for up to five wavelength channels. Single-shot spectra were obtained that validated the earlier laser temperature measurements. Shot-to-shot density fluctuations have been observed but not explained.

Finally, a scanning mirror assembly was installed on EBT for radial information. This system was first used to probe the plasma core with 75° scattering. On EBT-S, T_e is seen to increase with 28-GHz microwave power; T_e is 200-400 eV for 40-50 kW of input power (in agreement with soft x-ray data); and the electron density is $1-2 \times 10^{12} \text{ cm}^{-3}$.

1.1.4 Ion Cyclotron Resonance Heating

F. W. Baitz, G. C. Eldridge, A. C. England, R. E. Wintenberg

During 1979 a program was instituted for conducting ion cyclotron resonance heating (ICRH) experiments on EBT-I/S. Collaboration with industry was sought in the form of a two-phase subcontract. In November, subcontracts were signed with both McDonnell-Douglas Astronautics Company and TRW for four-month studies of ICRH. The studies were in two parts: theoretical modeling of ICRH under EBT-I/S conditions and development of a program plan for carrying out experiments over a one-year period. The goal of the ICRH program is to study EBT confinement when the power delivered to the bulk plasma ions is comparable to the power delivered to the toroidal core electrons by ECH.

The most critical problems to be addressed are the design of the antenna and the determination of the mode structure. Antennas for EBT must operate in steady state and be compatible with the ECH power. There are basically two different methods for ICRH; these are referred to as fast-wave and slow-wave heating. Because of the low magnetic field and the low density in EBT-I/S, fast-wave ICRH experiments may be impossible or inconclusive. For EBT-P, however, heating with the fast wave should be the more attractive. Both methods will be investigated on EBT-I/S.

Initial antenna design and testing have already begun at Oak Ridge National Laboratory (ORNL). A simple, 3/8-turn loop copper antenna was constructed and installed on EBT. The

copper antenna proved incompatible with the soft x-ray diagnostics; thus, design of an all aluminum antenna was begun. Matching circuits have been designed, and tests have begun on design of an ECH-compatible Faraday shield.

1.1.5 Charge Exchange Measurements

F. W. Baity, B. Davis

Three parallel plate, electrostatic charge exchange analyzers were used on EBT during 1979. Two of these employed conventional nitrogen stripping cells to produce positive ions, and the other used a cesium heat pipe to form negative ions.

The cesium analyzer viewed a single chord through the center of the plasma in the midplane of one of the mirror cavities. A four-channel nitrogen analyzer viewed four chords, each separated by 3.2 cm, in the midplane of a different mirror cavity. A discrepancy in the measured values of T_i between the cesium and nitrogen analyzers has been removed by reducing the diameter of entrance aperture of the cesium heat pipe to the same as that of the nitrogen analyzer (2 mm). Although this sacrificed considerable signal strength because of reduced detector area, it supported the idea that the discrepancy resulted from the fact that the cesium detector was calibrated with a neutral ion much smaller in cross section than the entrance aperture of the detector (1.2 cm).

The third analyzer views a single chord through the midpoint of a mirror throat region at an angle of 45° to the minor axis of the torus. This analyzer views passing particles in the torus. An $E \times B$ mass filter is fitted to this analyzer, and mass scans with this filter show hydrogen and aluminum neutrals emerging from the torus. This observation can be contrasted with similar measurements in the midplane of a cavity section that have shown no evidence of any other mass species. Because relevant charge exchange cross sections are not known for aluminum, it is impossible to determine a temperature. The aluminum signal

displays a different distribution and falls off more rapidly, versus analyzer voltage, than the hydrogen signal.

To allow a charge exchange spectrum of hydrogen neutrals to be measured with this analyzer, a stepping circuit was constructed to vary the voltage on the $E \times B$ filter and the energy analyzer simultaneously. This will be used in 1980.

Figure 1.6 shows a typical charge exchange neutral spectrum obtained by one channel of the four-channel analyzer after a constant background has been subtracted. The dashed lines represent three Maxwellian ion temperature distributions that are fitted to this data. The solid line is the sum of the three fitted spectra.

1.1.6 Fluctuation Measurements

A. Komori

Measurements of plasma fluctuations were initiated at the end of the year. The intent

Fig. 1.6. The neutral spectrum measured by the charge exchange diagnostic showing the three ion temperature distributions.

c. this work is to gain a detailed knowledge of plasma fluctuations; this is required for understanding the role of the hot electron rings or transport. Fluctuations in the EBT plasma are theoretically expected to be stabilized by the minimum B field produced by the hot electron rings. Thus, the stability of the hot electron ring itself is also very important in obtaining a stable plasma.

The fluctuations in the surface plasma were measured by Langmuir probes. It was difficult to obtain the fluctuations in the toroidal plasma without disturbing its confinement or enhancing the fluctuations. However, these fluctuation measurements are required to prove the stabilizing effect of hot electron rings. Thus, an optical measurement system is being prepared for obtaining the data without disturbing the plasma. This will give the spatial distributions of the fluctuations from spatial scans of the H_α line; a modified algebraic reconstruction technique will be used.

Some preliminary results obtained by the Langmuir probes are as follows: (1) the low frequency fluctuation (<100 kHz) is generated in the surface plasma in all modes, although the percentage fluctuation of density has a minimum in the T-mode; and (2) another fluctuation of 150-250 kHz is found near the T-M mode transition, whose frequency and amplitude increase as the M-mode is approached. This fluctuation seems to play a very important role in destroying hot electron rings.

1.1.7 EBT Computing Support

R. Burris

Considerable improvements in EBT computing support were made in 1979 in all areas, including data acquisition, experiment control, computer-to-computer communications, and the usability and availability of the computers.

Improvements in data acquisition and experiment control derive from the decision to support all future diagnostics with CAMAC hardware and FORTRAN software and to convert existing diagnostics as time and personnel

permit. As a result of this decision, CAMAC software was developed for the PDP 8/A, CAMAC hardware and software were added to the PDP 8/E, and CAMAC hardware and software for a PDP 11/24 computer, to be discussed shortly, were acquired. Software has been written to support several diagnostics, including Thomson scattering, the RPI heavy ion beam probe, a H_α monitor, and field probes.

A Nuclear Data 6600 computer-based analyzer that will be used to monitor x-ray signals has been purchased and installed. Hardware and software to permit the transfer of data from this computer to others in the Division is now under development.

A plan to upgrade computer-to-computer communications for EBT has nearly been completed. A new computer, a PDP 11/34, has been purchased and installed; its chief purpose is to permit very fast, low overhead, easily accessible data-transfer capabilities from EBT to the Fusion Energy Division (FED) network. The software to permit such transfer has already been written and is in production use on similar computers in the Division. Total implementation awaits the delivery of four hardware boards.

The availability and usability of the PDP 8/A have been improved through the purchase of a DECTape for the PDP 8/A, permitting the convenient physical interchange of files between the two PDP 8's, and through the addition of CAMAC support to both computers. The use of CAMAC hardware through FORTRAN programs has been emphasized.

1.1.8 Neutral Hydrogen

R. K. Richards

Because the plasma in EBT is created through the collisional ionization of neutrals, the penetration length of neutrals into the plasma is an important parameter. This length is proportional to the thermal velocity of the neutrals.

Neutral temperature measurements were obtained by spectroscopically observing the

Doppler broadening of H_α with a 1-m monochromator capable of 0.05-Å resolution. The measured spectral line shape of H_α is given in Fig. 1.7. This measurement indicates the presence of a bi-Maxwellian distribution for the neutrals.

The data profile fits the sum of two components with Doppler broadening from 0.5-eV and 4-eV temperatures. However, other broadening mechanisms will also alter the line shape; these result from fine structure, Zeeman splitting, and the monochromator. When these mechanisms are included, the temperatures of the two components are 0.2 ± 0.2 eV and 4.0 ± 0.5 eV. The density ratio of the two components can also be determined from Fig. 1.7, which indicates that the two densities are equal.

From these neutral temperature measurements the path length for ionization can be determined, and thus the spatial profile of neutral density can be estimated. For the 0.2-eV component the ionization scale length is comparable to the plasma diameter, producing a hollow profile. For the 4-eV component the ionization scale length becomes several times the plasma diameter, producing a flat spatial profile.

Fig. 1.7. The spectral profile of H_α showing the two neutral temperature distributions.

The sum of these two components will produce a spatial profile with a 20% reduction in central neutral density. Such a profile is in good agreement with previous results,¹ which indicate a constant value of $n_e n_0$ across the plasma core.

1.1.9 Multichannel Microwave Interferometer T. Uckan

The design of a multichannel microwave interferometer (see Fig. 1.8) at 70 GHz for the density profile measurements in EBT-1/S necessitates the use of standard-gain horns. In this frequency range the directivity of the horn is not adequate, and as a result multiple reflections in the aluminum cavity are a severe problem. Microwave metallic lenses, designed to focus the energy into a plane wave, can improve the directivity considerably. Therefore, we have implemented the horn with a type of delayed hyperbolic lens,² as shown in Fig. 1.9; it consists of a solid metallic disk with a number of equal-sized small holes and has indicated a gain of 15 dB over the no-lens

Fig. 1.8. The 70-GHz multichannel microwave interferometer for EBT-1/S.

Fig. 1.9. The geometry of the 70-GHz horn-lens combination and the hole pattern. The dimensions are: $L = 9.164$ cm, $A = 4.064$ cm, $B = 3.363$ cm, $C = 1.39$ cm, $D = 5.08$ cm, $d = 3.174$ mm, $s = 5$ mm, and $w = 0.127$ cm. There are 123 holes on the lens surface.

case on the EBT cavity, which has a diameter of 56 cm.

1.1.10 Field Error Effects on EBT

T. Uehara

As a result of closed field lines and the lack of rotational transform in EBT, a toroidal-current-free plasma equilibrium is possible. But inevitable magnetic field asymmetries in the system ($\Delta B/B \approx 10^{-4}$) cause the field lines to spiral out and also give rise to a toroidal current along the field lines, which may be given by $J_{\parallel} \approx \sigma_{\parallel} E_{\perp} (\Delta B/B)$. Here, $E_{\perp}$ is the ambipolar electric field and $\sigma_{\parallel}$ is the Spitzer conductivity. Externally introduced transverse error fields with two pairs of current loop, as shown in Fig. 1.10, may nullify the toroidal current. The cancellation of this current provides a poloidally symmetrical ambipolar potential and reduces the density fluctuations considerably (see Figs. 1.11 and 1.12).

It is observed that the toroidal-current-free EBT equilibrium can be maintained for a range of error field $\Delta B/B < \rho_e/R$, where ρ_e is the Larmor radius and R is the major radius. But for $\Delta B/B > \rho_e/R$, the drift surfaces can be changed from closed to open topology, thus allowing a toroidal current to flow. As is shown in Fig. 1.13, the radial component of this current, $J_r \approx J_{\parallel} (\Delta B/B)$, can strongly affect the formation of a negative ambipolar

Fig. 1.10. Location of the global field correction loops. Each loop consists of a 40-turn coil capable of carrying a current up to 40 A.

Fig. 1.11. The effect of error fields on potential profiles.

Fig. 1.12. The variation of toroidal current, central potential, and density fluctuations with horizontal correction current.

Fig. 1.13. Electron and ion particle flux vs ambipolar electric field.

potential well, which is necessary for stable operation.

The external fields B_H and B_V , which are functions of the loop currents I_H and I_V , respectively, are also utilized to study plasma parameters such as toroidal current, ambipolar potential, density fluctuations, etc. A typical result is shown in Fig. 1.12 for the EBT-I plasma.

1.1.11 EBT-S Microwave Systems Development

R. G. Eason, T. L. White, G. F. Pierce,
H. W. McGuffin

Development of the 28-GHz, 200-kW cw ECH system^{3,4} for EBT-S was continued. Numerous system modifications were made for the purpose of improving reliability and ease of operation. Additional modifications were incorporated to accommodate gyrotrons having an axisymmetric rf output configuration. High-power cw operation was not possible until near the end of the period because of the lack of a suitable cw gyrotron. A gyrotron was received in early December; this allowed EBT-S operation at 28 GHz of cw power levels in excess of 50 kW.

Development of 28-GHz oversized waveguide components⁵ was continued with emphasis on bench tests at very low rf power levels. This activity was supplemented by high-power tests including calorimetric measurements of rf loss in system components during operation, first into a dummy rf load and then into the EBT-S plasma.

The development of 18-GHz and 10.6-GHz ECH systems for EBT-I⁶ was continued in the areas of reliability and rf distribution. The purpose of this effort was to ensure maximum system availability for machine operation and to provide operating characteristics that conform to evolutionary experimental needs.

1.1.12 EBT Parameters

R. L. Colclough, J. C. Fournier

An interesting facet of the data runs is a documentation of the three different operating modes (C, T, and M) as a function of the two normal experimental parameters (power and background pressure). These parameters, average line density $n_e l$ and perpendicular energy $W_\perp$ (in arbitrary units), are plotted in Fig. 1.14 as the pressure is varied at fixed microwave power. At very high pressure (i.e., in the C-mode), $n_e l$ is high but there is no $W_\perp$ and no well. When $W_\perp$ and the potential well depth ΔV_p become nonzero, $n_e l$ stops falling. The break in the curves as this happens delineates the C-T transition. The T-mode continues through lower pressures until $n_e l$ can no longer be maintained in steady state and ΔV_p decreases. This, along with large fluctuation levels, delineates the T-M transition. A combined plot of $n_e l$ versus pressure at various power levels is shown in Fig. 1.15. Most striking is the expanding envelope of the T-mode as microwave power increases.

Concentrated run time became available shortly after repair work ended in late summer. The major thrust was to duplicate the results of the past with the improved diagnostic array and thereby create a complete data set. Table 1.1 displays two such data sets in T-mode.

1.2 EBT-P DEVELOPMENT

Current DOE policy on the development of fusion energy places increased emphasis on the testing of promising alternate concepts to

Fig. 1.14. The EBT-I parameters $n_e l$, $W_\perp$, and ΔV_p vs fill gas pressure.

Fig. 1.15. The variation of $n_e l$ with fill gas pressure and microwave power.

Table I.1. Two typical data sets for EBT-I in the T-mode

	Case 1	Case 2
B_{res} (T)	0.65	0.65
Power at 18 GHz (W)	4×10^4	5×10^4
Power at 10.6 GHz (W)	1×10^4	2×10^4
Average core radius (m)	0.070	0.077
Major radius (m)	1.5	1.5
Central density (m^{-3})	7×10^{17}	10^{18}
Central i_e (eV)	200	250
Maximum T_i (eV)	70-80	80-90
Minimum T_i (eV)	~ 30	~ 30
Neutral density (m^{-3})	$\sim 5 \times 10^{15} \times 2 \div 2$	5×10^{15}
Background pressure (T)	$6-7 \times 10^{-6}$	$6-7 \times 10^{-6}$
Potential well depth (V)	165	190
Potential shape	$\sim$ parabolic	$\sim$ parabolic
Perpendicular energy (relative)	150	300

permit comparisons with tokamaks and mirrors in the mid-1980s. The EBT concept was selected by DOE in December 1978 as one of the alternate concepts to be advanced through a proof-of-principle experiment.⁹ The ELMO Bumpy Torus Proof-of-Principle (EBT-P) experiment will be a hydrogen plasma confinement device whose purpose is to extend most of the important physics parameters into or near the reactor regime and to resolve the outstanding physics and technology issues in a way that will establish the technical viability of continuing the development of the EBT concept.

The responsibility for managing the EBT-P project, including the project research and development, lies with ORNL. The project will be conducted in three phases:

- Phase I -- Project Definition;
- Phase II -- Design, Fabrication, Installation;
- and
- Phase III -- Operations.

Industrial participation to broaden the EBT program base is an important aspect of this project. It is intended that industry have a major role not only in the fabrication of EBT-P but also in the project definition,

design, and subsequent experimental operation. A principal goal is the selection of a primary industrial participant (IP) whose role will lead to broad corporate responsibilities for, and identification with, the ultimate success of the EBT program.

The IP will be selected from among several organizations that are first competitively selected to conduct project definition studies. Based on the results of the industrial project definition studies and continuing studies by ORNL, a reference design for EBT-P will be proposed by ORNL; it will likely be a combination of design features from all of the studies. The reference design, after approval by DOE, will be used as a basis for soliciting competitive proposals and selecting the IP that will design and build the EBT-P (Phase II) and have a major role in its operation.

The site for EBT-P will be determined by DOE. Each candidate IP will be requested to submit a proposal for the government-owned ORNL site (Building 9201-2). In addition, proposers may submit proposals for a site or sites other than ORNL, but submission of alternate site proposals will not be required. All proposals will be evaluated by ORNL, which

will then select the best proposal for the ORNL site and the best proposal for an alternate site. The DOE will then determine the site that it considers to be the most advantageous for the EBT program and for the fusion program generally.

Progress in 1979

A Memorandum of Agreement (MOA) completed in March 1979 documents agreements between DOE and ORNL on plans and responsibilities for initiating the EBT-P project. The plan and schedule for initiating the project are shown in Fig. 1.16.

Phase I (Project Definition) was essentially completed in 1979. Four industrial teams (Ebasco Services, Inc.; Grumman Aerospace Corporation; McDonnell-Douglas Astronautics Company; and Westinghouse Electric Corporation) were awarded ORNL fixed-price subcontracts of \$200,000; and each performed

four-month project definition studies, including a proposed reference design. Because a principal objective was to obtain industry views and input into the definition of the EBT-P project, precise criteria for the proof-of-principle experiment were not specified in order to allow maximum latitude for industry recommendations. Concurrently, ORNL refined the preconceptual design that was presented in 1978 to the Concept Evaluation Committee as EBT-II.^{9,10} Following completion of the project definition studies, extensive discussions were held among the subcontractors, ORNL, and DOE to identify agreed-on physics and technological constraints and to establish goals for the project that are consistent with perceived schedule and budget requirements.

Using the information derived from the preconceptual design studies and from subsequent discussions with the participants, ORNL prepared a proposed reference design for

Fig. 1.16. EBT-P project initiation schedule.

EBT-P.¹¹ The DOE convened an EBT-P Review Panel and will review the proposed reference design and reassess the EBT program early in 1980. Assuming that a decision to continue the project is forthcoming, the approved reference design will be used by ORNL as a basis for soliciting proposals from the candidate IPs to design and build EBT-P.

Proposed reference design

The proposed reference design¹¹ was selected to give EBT-P in its initial operation the ability to achieve a significant advance over EBT-S and to satisfy the scaling requirements of proof-of-principle. In addition, the device will be constructed with a degree of built-in flexibility to allow future improvements that will extend the performance parameters toward the condition necessary for containing a fusion reactor plasma. For example, it is proposed that EBT-P start operation at a 60-GHz ECH frequency as determined by realistically expected advances in microwave source technology; the magnets will, however, be capable of accommodating 90-GHz microwave sources, which are being developed.

This increase in frequency, along with additional ring power and heating power, should lead to substantial improvement in plasma parameters. Approximately one year after initial operation, structures will be added for coupling 1 MW of ICRH to the plasma, and high-power experiments will be started that are aimed at demonstrating stable operation at high beta in the core plasma. Other potential improvements that will be added as justified by experimental results include: additional microwave power, microwave feed from the high-field side, additional ICRH, lower hybrid heating, neutral beams, molecular ion beams, aspect ratio enhancement (ARE) coils, particle control (divertor), and alternate fueling methods.

Table 1.2 gives the principal machine characteristics and estimated plasma parameters for the base EBT-P device as initially constructed before and after ICRH is added; the projected parameters achievable with improvements are also shown.

The proposed schedule for EBT-P design and construction is shown in fig. 1.17; it is expected that the device can be completed by

Table 1.2. Machine characteristics and projected plasma parameters for proposed reference design, EBT-P

Machine characteristics		Physics parameters			
		Parameter	Base		Upgrade ^a
Major radius, R (m)	4.5	n_e (cm ⁻³)	1.7×10^{13}	3×10^{13}	5×10^{13}
Clear bore radius (m)	0.17	T_e (keV)	2.0	3.5	5
Mechanical aspect ratio, A	15.5	T_i (keV)	0.4	2	3
Number of sectors	36.0	$B(\text{annulus})^b$ (%)	1.5	5	10
Superconducting magnet system		τ_E (ms)	30	75	150
Initial field, B_{max} (T)	5.6	$n\tau$ (cm ⁻³ s)	5×10^{11}	2×10^{12}	7×10^{12}
Design field, B_{max} (T)	7.4	P_{ECH} (MW) [GHz]	0.8 [60]	1.2 [60]	(a)
Microwave frequency (GHz)			0.4 [28]	0.4 [28]	
Initial	60/28	P_{ICRH} (MW)	0	1.0	(a)
Later	90				
Mirror ratio	2.2				

^aImprovements include additional ICRH power, ECH at 90 GHz, higher field operation, and ARE coils.

^b $B(\text{annulus})$ is defined as the plasma pressure at the midplane of the ring and the center of the torus divided by the magnetic pressure at the outside of the ring.

Fig. 1.17. Proposed schedule for EBT-P design and construction.

the end of FY 1984. A preliminary estimate is that the cost (not including improvements) will be in the range of \$75 to 85 million.

Figure 1.18 is an artist's conception of the EBT-P device.

Technology development

Successful realization of EBT-P requires significant advances in some areas of technology. The ongoing EBT-S experimental program will be essential in extending the technology by providing important operating experience and serving as a test bed for subsystems and components; other needs will require development efforts specifically for EBT-P.

Two major subsystems, the superconducting mirror magnets and the microwave sources, were identified early as requiring intensive development efforts, and development tasks for these components have been initiated. The microwave distribution and supporting sub-

systems also require technological advances, and a development program is being formulated to focus on specific needs for EBT-P. Consistent and vigorous efforts are clearly required in these areas to support a credible design for EBT-P in a time frame consistent with scheduled goals. In addition, it is recognized that as the design proceeds, other components and subsystems will require fabrication development or confirmatory testing.

The principal element of the mirror magnet development task is a two-magnet test based on the EBT-P reference design and is described in Sect. 6.4. Development of the microwave sources is being carried out as part of the DOE National Gyrotron Program and is described in Sect. 1.4.

1.3 TOKAMAK RESEARCH

Progress in the ISX-B experimental program during 1979 was characterized by expansion in

Fig. 1.18. Artist's conception of EBT-P.

breadth and depth. Nearly all program elements have produced significant results, thanks to the efforts of the numerous contributors listed in Table 1.3. Highlights of the results are given below.

In the high-beta studies with beam power up to 1 MW in circular plasmas:

- A rms beta β^* of 3.3% [with central beta $\beta(0)$ of 8% and volume-averaged beta $\langle\beta\rangle$ of 2.3%] has been achieved; beta continues to increase with beam power.
- Although experimental beta values are above theoretical thresholds for pressure-driven magnetohydrodynamic (MHD) instabilities, no catastrophic loss of confinement has been observed.
- In almost completely beam dominated discharges ($P_{inj}/P_{OH} \geq 10$), values and scalings of electron and ion energy confinement times

are still similar to those obtained with ohmic heating alone.

- New and interesting phenomena correlated with high-power injection are being studied: marked change in the character of MHD oscillations; apparent deterioration of particle confinement time; and apparent inhibition of impurity inward transport.

In the plasma shaping studies, D-shaped plasmas with an elongation of up to 1.5 have been produced and maintained throughout the discharge.

In ECH studies with the use of a 35-GHz NRL gyrotron with power levels of up to 85 kW:

- Bulk electron heating with an increase in $T_e(0)$ of up to 50% has been demonstrated with a constant heating rate (~ 5.5 eV/kW).
- ECH preionization has resulted in a large ($\sim 40\%$) reduction of initial loop voltage,

Table 1.3. Contributors to the IST-B Program, Experimental Confinement Section, Fusion Energy Division
 J. Sheffield, Section Head
 J. L. Duniap, Associate Section Head

<u>Experimental Coordinator</u>	<u>High-Beta Equilibria, Shaping</u>	<u>Data Handling</u>
M. Murakami	D. W. Swin	W. R. Ming
<u>Operations</u>	S. C. Bates	D. R. Overbey
M. J. Saltmarsh	G. H. Neilson	J. D. Bell
S. M. DeCamp	<u>Electron Cyclotron Heating</u>	E. Blair
D. E. Ketterer	A. C. England	J. E. Francis
T. P. Rayburn	C. M. Loring	L. Maddox
W. J. Redmond	G. L. Campen	K. A. Stewart
H. C. Sanderson	W. L. Wright	<u>Theory Computation</u>
G. P. Dyer - Electronics	D. H. Gray	L. Charlton
J. M. Madison	O. C. Eldridge (U. of Tennessee)	A. Cooper
J. W. Pearce	A. G. Kulchar (U. of Tennessee)	E. C. Crune
<u>Diagnostics</u>	<u>Pellet Injection</u>	R. H. Fowler
R. C. Isler - Spectroscopy	S. L. Milora	C. E. Hammons
S. Kasai (JAERI)	C. A. Foster	J. T. Hogan
L. Murray	C. E. Thomas - Holography	H. C. Howe
R. Neidigh (Consultant)	<u>Ripple</u>	J. K. Munro
R. K. Richards	J. F. Lyon	Y-K. M. Peng
C. E. Bush - Radiometers	J. T. Mihalcz - Neutrons	J. A. Rome
J. L. Duniap - MHD Fluctuations	G. H. Neilson - Charge Exchange	D. J. Strickler
T. D. Burnis	S. D. Scott	T. C. Tucker
A. P. Navarro (JEN, Spain)	S. L. Halstead	R. M. Wieland
V. A. Paré	R. E. Worsham	<u>Collaborative Programs</u>
E. A. Lazarus - Thomson Scattering	<u>Plasma-Wall/Surface Physics</u>	<u>General Atomic - Flow Reversal</u>
P. W. King	R. A. Langley	K. H. Burrell
B. Zurro (JEN, Spain)	B. P. Appleton	C. H. Muller - Edge Diagnostics
M. De La Fuente (Consultant)	R. E. Clausung	M. P. Hacker
J. B. Wilgen - Microwaves	L. C. Emerson	G. K. Wong - Theory
R. E. Worsham	L. Heatherly	<u>NRL - Electron Cyclotron Heating</u>
<u>Neutral Injection/Beam Upgrade</u>	H. R. Husinsky (Tech. U. of Vienna)	M. E. Read
P. M. Diamonds	J. B. Roberto	R. M. Gilgenbach (JAYCOR)
S. C. Bates	J. E. Simpkins	R. F. Lucey (JAYCOR)
C. M. Loring	H. B. Schmeer (KFA, GUTsch)	K. Heckett (MIT)
L. A. Massengill	S. R. Withrow	<u>Sandia - Coated Limiter</u>
C. W. Murphy	R. A. Zehr	M. J. Davis
J. Pearce	<u>Diagnostic Development</u>	D. M. Mattox
W. L. Stirling - Development	C. F. Bennett	J. B. Whitley
J. Kim	D. P. Hutchinson - FIR Lasers	C. B. Hazlip
N. C. Ponte	P. A. Staats	<u>PPPL - Ripple Injection</u>
R. E. Wright	K. L. Vander Sluis	R. Goldston
C. C. Tsai	C. H. Ma (U. of Mississippi)	D. L. Jassby
M. M. Menon	D. M. Thomas (U. of Texas)	
R. E. Worsham - Beam Line Upgrade	<u>Bundle Divertor</u>	
H. C. McCurdy	T. C. Jernigan	
R. L. Johnson	W. R. Ding	
D. H. Gray	R. B. Wyser	
R. E. Hill	W. L. Wright	
A. Y. Broverman	J. A. O'Toole (Grunner)	
D. E. Schecter	D. E. Hill	
E. Slaughter	B. E. Nelson	
S. J. Sosnowski		
W. L. Wright		

Table 1.4. Range of plasma characteristics in ISX-B

Major radius, R_0 (cm)	93
Minor radius, a (cm)	27
Elongation, b/a	1-1.6
Toroidal field, B_T (T)	0.8-1.5
Plasma current, I_p (kA)	75-185
Central electron temperature, $T_e(0)$ (keV)	0.5-2.2
Central ion temperature, $T_i(0)$ (keV)	0.3-1.5
Line-averaged electron density, $\bar{n}_e$ (10^{13} cm^{-3})	0.5-7
Injection power, P_{inj} (MW)	0-1
Ohmic heating current, P_{OH} (kW)	50-250
Effective charge, Z_{eff}	1-3
Gross energy confinement time, τ_E (ms)	6-30
Discharge duration, τ_{disch} (ms)	150-250

leading to substantial savings of weber expenditure in the OH transformer during the current initiation phase ($t < 2$ ms).

In the pellet fueling studies with ≈ 1 -mm-diam pellets injected by a pneumatic injector at a velocity of ≈ 1 km/s:

- A density rise of up to a factor of five has been observed in ohmic and beam heated discharges without adverse effects on stability and confinement.
- Holographic interferometry and shadowgraphy of pellets in flight have aided in developing the physics of pellet ablation.

In the ripple studies:

- Ripples of up to 1.5% have produced large fast ion losses, but much smaller thermal ion losses and flux-surface perturbations.

In diagnostic development:

- A formic-acid FIR laser has been used successfully as a multichannel interferometer and for Faraday rotation measurements.
- Two laser-induced fluorescence diagnostic systems developed by GA and Jülich have successfully measured the density of impurities such as iron near the plasma edge, for the first time in tokamaks.

These successes in the ISX-B program have been made possible by significant improvements in both tokamak and neutral beam operation and

by improved capabilities of the diagnostics/data handling systems. Table 1.4 lists the ranges of plasma characteristics attained thus far in ISX-B.

Tokamak operation has been scheduled to optimize the time available for various experimental programs while allowing installation of new equipment and diagnostics in addition to repair and maintenance. The reliability of the operation has significantly improved (with an average of 70 shots per operational day), and a total of over ten thousand shots were fired in the course of the year. The distribution of shots among the major program elements is listed in Table 1.5.

Table 1.5. Distribution of tokamak shots by program objectives for 1979

Program	Number of shots	Percent
General plasma experiments, baseline and exploration	3643	35
High-beta program	2506	24
Pellet injection	928	9
ECH studies	914	9
Impurity flow reversal	872	8
Ripple studies	474	5
Plasma shaping studies	466	4
Power supply tests, etc.	434	4
Surface physics ²	136	1

²Does not include shots primarily devoted to other programs.

The capability of neutral beam injection has been increased to 1 MW reliably with the use of two 20-cm, 60-A ion sources, as shown in Fig. 1.19. The first 1-MW injection was achieved even though the experiment was handicapped by the availability of only one 100-A accel supply (borrowed from the ORNL Plasma Technology Section) to feed the two 60-A sources. The two 100-A accel supplies originally planned for use were finally delivered a month before the device was shut down to upgrade the beam line to accept the

large, 30-cm, 100-A sources with improved beam transmission and instrumentation.

A large number of diagnostics in support of the various programs were installed and made operational during this period. Table 1.6 lists the diagnostics in use to date. Equally valuable has been added capability of the data handling system, which interfaces nearly all the diagnostics listed with the central archival storage on the PDP 10.

More detailed descriptions of experimental results from ISX-B are given in the following

Fig. 1.19. History of tokamak and neutral beam injection operation. Points show the maximum power injected into the plasma for the months.

Table 1.6. Diagnostics in operation on ISX-8

Magnetics	
I_p , V, equilibrium measurement (Δ_H , Δ_V , B_p , $L_i/2$);	
64 magnetic probes (configuration and fluctuation);	
fast-response magnetic probes (fluctuations);	
ϕ -loops; diamagnetic measurement	
Electron density	
Horizontal and vertical microwave interferometer; four-channel submillimeter interferometer; Thomson scattering (four pulses, radial and limited vertical scans)	
Electron temperature	
Thomson scattering; fast x-ray spectrometer; second-harmonic cyclotron measurements (fixed 72 GHz and 58 GHz); four pin-diode, soft x-ray monitors (fluctuation)	
Ion temperature	
Mass energy, charge exchange analyzer (parallel, scannable); velocity filter, charge exchange analyzer (perpendicular, scannable); neutron detectors (D-D)	
Spectroscopy	
Grazing incidence spectrometer (100 Å-1200 Å); normal incidence spectrometer (1200 Å-8000 Å); visible spectrometer with optical multichannel analyzer; six H_α monitors (toroidal asymmetry)	
Radiative power	
Three wide-angle pyroelectric monitors (average wall power, pellet); six-channel vertical, 15-channel horizontal pyroelectric arrays	
Surface studies	
In situ surface analysis station (M&C); UHV portable sample transport system (SS); fast rotating sample wheel (GA)	
Residual gas analyzer	
Plasma TV camera	
Infrared camera (heat deposition on limiters)	
Drum camera (100,000 frames/s)	
Beam profile monitor	
Pellet holography and shadowgraphy	
Langmuir probes	
Four-channel Faraday rotation system	
Laser-induced fluorescence measurements (GA; Jülich)	
Resonance cone probe (GA)	
Gridded energy analyzer (GA)	

sections, along with reports of progress in various related activities.

1.3.1 High-Beta Studies

High-beta, confinement, and neutral beam heating studies

S. C. Bate, C. E. Bush, L. J. Charlton,
R. J. Colchin, W. A. Cooper, H. de la Puente,
S. M. DeCamp, J. L. Dunlap, G. R. Dyer, P. H.
Edmonds, A. C. England, J. T. Hogan, H. C.

Howe, R. C. Isler, T. C. Jernigan, P. E.
Ketterer, J. Kim, P. W. King, L. A. Lazarus,
C. M. Loring, J. P. Lyon, H. C. McCurdy,
M. M. Menon, J. T. Mihalcz, M. Murakami,
A. P. Navarro, R. V. Neidigh, G. H. Neilson,
D. R. Overbey, V. K. Paré, Y.-K. M. Peng,
N. S. Ponte, M. J. Saltmarsh, D. E. Schechter,
J. Sheffield, J. E. Simpkins, W. L. Stirling,
D. W. Swain, C. C. Tsai, T. C. Tucker, R. M.
Wisland, J. B. Wilgen, W. R. Wing, R. E.
Worham, B. Zurro

The so-called "high-beta studies," the major thrust of the ISX-B program, involve the integration of high-beta, confinement, and neutral beam heating studies; they range from theory to technology. Last year's annual report¹² discussed results of the initial experiments, which successfully reproduced all major results obtained in the Oak Ridge Tokamak (OPMAK) and ISX-A. In this report period, the studies have entered new stages, largely due to the increased injection power. With injected power of up to 1 MW, rms beta (β^*) values above 3% have been achieved. These are above the theoretical MHD beta limit predicted for ballooning modes. The injected power has become an order of magnitude larger than the ohmic heating power. Thus we have been provided a unique opportunity to verify the confinement scalings, which have been based primarily on ohmically heated discharges. High injected power has produced markedly different plasma phenomena that may be related to very basic issues of heating physics (such as the limit of heating resulting from plasma rotation). Pursuing these issues requires further experimentation with increased power and improved diagnostics as well as further theoretical work.

In the sections below, we describe the discharges in which highest beta values were obtained and discuss beta results and their correlation with theory. Confinement scalings in beam dominated discharges are then presented. Finally, we describe the phenomena in which particle confinement is affected by injection.

Typical discharge characteristics. Figure 1.20 shows the behavior of some plasma parameters for a typical shot with 0.95 MW of neutral beam injection. Plasma pulse duration is about 200 ms, with the beam being injected from 70 ms to 170 ms. Simultaneously with injection, the plasma current is ramped up from 110 kA to 150 kA with feedback control of plasma current and position. The loop voltage drops from about 2 V before injection to approximately 0.5 V during the beam pulse

Fig. 1.20. Time histories of plasma parameters in a typical ISX-B beam-heated discharge: (a) plasma current I_p and loop voltage V ; (b) neutral beam power P_b and line-averaged electron density $\bar{n}_e$; (c) wall energy flux from a pyroelectric radiometer P_{rad} and central ion temperature $T_i(0)$ from charge exchange analysis.

due to electron heating. Therefore, during injection the beam power is more than ten times higher than the ohmic heating power, in essence eliminating the role of the plasma current in plasma heating. The central ion temperature, obtained from charge exchange analysis, rises from 0.35 keV to approximately 1.3 keV. Figure 1.21 shows the electron temperature and density profiles for this discharge along with those for similar experimental conditions but without injection. Table 1.7 compares plasma parameters deduced for these two cases. Both the central and density-averaged electron temperatures [$T_e(0)$ and $\langle T_e \rangle$, respectively] increase by a factor of 2.5 with injection. The central ion temperature rises by a factor of three, with a heating rate of 3.8×10^{13} eV cm^{-3}/kW , which is about equal to that obtained in the

Fig. 1.21. Radial distributions of electron temperature and density from Thomson scattering for discharges with characteristics listed in Table 1.7.

Table 1.7. Comparison of plasma parameters for the beam heated discharge (shown in Fig. 1.20) and the ohmically heated discharge with similar experimental conditions ($H^0 \rightarrow D^+$, $B_T = 1.1 \text{ T}$, $I = 150 \text{ kA}$)

Parameter	Injection	No injection
$\bar{n}_e$ (10^{13} cm^{-3})	4.6	3.8
P_b (kW)	950	0
P_{OH} (kW)	84	260
$T_e(0)$ (keV)	1.5	0.58
$\langle T_e \rangle$ (keV)	0.81	0.31
$T_i(0)$ (keV)	1.1	0.29
Z_{eff}	2.2	2.6
τ_E (ms)	17	18
$\langle \beta_p \rangle$	1.4 ± 0.5 (beam)	0.5
$\langle \beta_T \rangle$ (%)	1.1 ± 0.6	0.5
$\beta_T(0)$ (%)	5.5 ± 2.8	1.4
β^* (%)	3.3	0.7

Princeton Large Torus (PLT) neutral beam heating experiments.^{13,14}

Beta results. High injection power with favorable confinement at a low toroidal field has resulted in relatively high beta values. Figure 1.22 shows a plot of the rms-averaged

beta $\beta^* [= 2\mu_0 \langle p^2 \rangle^{1/2} / B_T^2]$ as a function of beam power. For an injection power of 1 MW, we have achieved a β^* value of $\geq 3\%$. As indicated in the figure, β^* increases linearly with beam power for the highest density cases. The figure also shows that higher values of β^* are usually achieved at higher

Fig. 1.22. Root-mean-square beta β^* as a function of injection power for discharges operated at 11.5 kG.

values of density for a constant beam power. Extrapolating to a beam power of 1.5 MW, we should achieve a β^* value of about 5%, if no limiting instabilities occur.

Figure 1.23 shows two plots of beta versus line-averaged density. Figure 1.23a is the volume-averaged beta $\langle\beta\rangle$, and Fig. 1.23b is the peak beta β_0 . Average beta values greater than 2% have been obtained, with peak beta

values of up to 8%. The solid points on the graph are the values of beta due to the plasma electron and plasma ion contributions alone. The open points include the contribution of the nonthermal fast ion tail (due to beam injection) as well as the plasma contribution. As can be seen from the figure, the calculated fast ion component provides a significant fraction of the total pressure in the plasma at low density and high beam power. However, at higher densities the contribution is $\leq 25\%$. The pressure contribution of the fast ions is computed by a Fokker-Planck code developed at ORNL¹⁵ to calculate beam deposition and slowing down. The results of the calculations agree, within a few percent, with the results obtained from a modification of a Monte Carlo code developed at the Princeton Plasma Physics Laboratory (PPPL);¹⁶ it allows for diffusion of the fast ions in both velocity and real space.

The equilibrium measurements are in reasonable agreement with the volume-averaged β_p values based on the laser and charge exchange measurements and the calculated fast ion contribution. At low s_p , the Shafranov formula relates B_z/I_p (the ratio of vertical

Fig. 1.23. Dependences of (a) volume-averaged beta and (b) peak beta on line-averaged density. Solid points are the plasma contribution to beta, while open points are the plasma plus fast ion component from injection.

field to plasma current) with $\beta_p + \epsilon_i/2$, where β_p is given by $2\mu_0(\langle P \rangle_{\text{plasma}} + \langle P \rangle_{\text{beam}})/B_p^2(a)$ and ϵ_i is the normalized internal inductance. However, a modification to the linear relationship is required as β_p increases. Figure 1.24 compares the laser-based β_p values with the β_p estimate based on fixed- and free-boundary equilibrium calculations using profiles similar to those deduced from the experiment. The rough agreement seen in the figure gives some confidence in our estimate of the total pressure.

Fig. 1.24. Comparison of β_p values (including fast ion contribution) deduced from two different measurements. Solid points are based on laser profiles, charge exchange ion temperature, and calculated fast ion contribution. Values of β_p from the equilibrium measurement are shown by the solid curve based on numerical calculations as well as by the broken line based on the Shafranov formula.

Comparison of the β^* values achieved in ISX-B with theoretical limits deduced from calculations for low- n (toroidal mode number), pressure-driven instabilities using the PEST code¹⁷ indicates that we are above the beta limit predicted by this code. Detailed calculations done with the ERATO code,¹⁸ using plasma parameters that closely simulate

the measured parameters in our experiment, also indicate that the ISX-B values are above threshold for the ballooning instability. High- n -mode stability analysis¹⁹ using an anisotropic pressure tensor and based upon experimental data also shows a broad range of instability in the plasma interior. Results from an ideal MHD ballooning code²⁰ with an isotropic pressure tensor also agree qualitatively with these calculations.

In spite of these theoretical predictions, no catastrophic loss of confinement has been observed in the experiment. Certainly there have been no phenomena analogous to disruptive instabilities. The energy confinement in high-power injection is similar to that with ohmic heating alone or with low-power injection, indicating no evidence of saturation (of β^*) at the highest injection power levels.

However, the pattern of MHD activities (detected by soft x-ray emission and Mirnov loop signals) is observed to undergo rather marked changes as beam power is increased, as detailed later in this section. Identifiers of sawtooth oscillations and $m = 1$ modes are lost, and frequencies of both the interior (soft x-ray) and exterior (Mirnov signals) oscillations are distinctively high (~ 20 kHz, compared with a few kHz at low injection power). In addition, the rotation direction of the Mirnov signals is observed to reverse from the usual electron-diamagnetic direction to the ion-diamagnetic direction, implying that the modes are shifted due to beam-driven plasma rotation. Are these oscillations manifestations of the theoretically predicted pressure-driven modes or resistive modes modified by plasma rotation? The answer awaits more detailed documentation with improved diagnostic capabilities.

Energy confinement. The measured central ion temperature and its scaling are found to be in good agreement with the predictions of neoclassical theory for ion energy transport, as shown in Fig. 1.25. The central ion temperature is obtained from analysis of the charge exchange neutral atoms by assuming the

Fig. 1.25. Comparison of charge exchange measurements of $T_i(0)$ with neoclassical theory predictions for different values of $\bar{n}_e$, beam power, and plasma current.

T_i profile to be parabolic. The theoretical ion temperature was calculated by computing a 1-D ion power balance that uses measured electron parameters [$n_e(r)$, $T_e(r)$, Z_{eff} , etc.] along with classical beam-ion¹⁵ and electron-ion energy transfer rates. The neutral density is estimated from a neutral transport model for which the neutral density at the plasma edge corresponds to 2.5 times the gas feed rate. The ion thermal conductivities used in these calculations are 1.5 times the neoclassical values.²¹ As can be seen from the figure, good agreement is obtained between theory and experiment for both magnitude and scaling (with density and plasma current). The two lowest density points were taken with virtually no gas flow into the tokamak. However, if a gas recycling rate from the wall sufficient to raise $n_0(0)$ by a factor of three is included, the theoretical calculations

of $T_i(0)$ will then agree with experimental values at the lowest density cases. Under present ISX-B experimental conditions, the calculations indicate that the effects of halo neutrals²² (i.e., the neutrals generated by the injected beam) and the ion loss region in velocity space¹⁵ are negligible in spite of low B_T and I_p operation. Based on the theoretically calculated ion temperature profile, the ion energy confinement time (τ_{Ei}) scales roughly with $I_p/\bar{n}_e$. However, the values of τ_{Ei} are nearly equal to those of ohmically heated discharges with similar conditions, contrary to expectations that τ_{Ei} would improve due to increased ion temperature with injection and the resulting reduction of the thermal conductivity. Apparently the improvements are offset by increased losses due to charge exchange and convection at higher neutral density.

Electron energy confinement times derived from the experimental measurements of beam dominated discharges are also similar to those of ohmically heated discharges. Figure 1.26 shows the dependence of the electron energy confinement time τ_{Ee} (within the half radius) on the line-averaged density $\bar{n}_e$ for three different levels of injection power: τ_{Ee} continues to increase linearly with density. The slope indicated in the figure corresponds to the usual Alcator scaling: $\tau_{Ee} = 5 \times 10^{-19} \bar{n}_e a^2$ (τ_{Ee} is in s, $\bar{n}_e$ is in cm^{-3} , and a^2 is in cm). Although there appears to be an apparent trend of gradual decrease (by up to ~30%) in the slope with increasing beam power, more thorough studies are needed because of uncertainties in analysis, particularly in (1) energy content with flux distortion at high β_p and (2) power input due to charge exchange loss and reabsorption of fast ions. Although the amount of data is somewhat limited at present, electron energy confinement time does not significantly decrease with plasma current or toroidal field. The latter is important in attaining high β_T values.

The total energy confinement time at high densities follows the τ_{Ee} scaling more closely

Fig. 1.26. Electron energy confinement time τ_{Ee} within radius $a/2$ as a function of $\bar{n}_e$ for different values of beam power. The line shown corresponds to the Alcator scaling.

than the τ_{Ei} scaling. This may occur because most of the injected beam power goes to the electrons and also because the electron-ion heat transfer decreases as the difference between ion and electron temperature decreases. Therefore, τ_E with injection continues to increase with density (Fig. 1.27), in contrast to the behavior observed with ohmic heating alone, where τ_E goes through a maximum,¹² as in ISX-A.²³ Consequently, it is important to operate the tokamak at high density in order to attain high β_T values.

Particle confinement. We have observed a phenomenon that occurs with high power injection that we call density clamping; it is illustrated in Fig. 1.28. Neutral beam injection into discharges with linearly rising $\bar{n}_e$ (fed by a constant gas puff) results in a "clamp" of $\bar{n}_e$ shortly after the beams are turned on. Although the n_e profiles reflect some portion of the change of $\bar{n}_e$, the major part appears to result from an

Fig. 1.27. Gross energy confinement time τ_E as a function of $\bar{n}_e$ for different values of beam power. The line shown corresponds to the Alcator scaling.

Fig. 1.28. Time histories of $\bar{n}_e$ with and without injection.

apparent deterioration of particle confinement. The clamp is correlated neither with the onset of the above-mentioned MHD behavior nor with the toroidal field (and thus high beta). However, increasing the plasma current substantially reduces the extent of the density clamp. Furthermore, a mild disruption or the application of a short, intense gas puff just before start of beam injection can prevent density clamping. Line-averaged densities of $7 \times 10^{13} \text{ cm}^{-3}$ have been obtained

in this fashion. However, more recent high-density operation (such as the highest β^* case) has been obtained with substantially high, constant gas puffing.

The deterioration of plasma particle confinement with beam injection may be related to the marked changes in impurity transport with beam injection as observed in the VUV spectroscopic measurement discussed later in this section. The emission from the 15^+ state of argon injected during the clamp does not increase strongly as it does in ohmically heated discharges, possibly indicating that inward transport is prohibited in the interior of the plasma. Observations of high ionization states of naturally occurring iron impurities substantiate the above interpretation. These observations may be due to the theoretically predicted effects of beam-induced momentum.^{24,25}

A status report on plasma shaping studies on ISX-B

D. W. Swain, M. J. Saltmarsh, S. Bates

The ISX-B device was designed to make elliptical or D-shaped plasmas with elongations ($\epsilon = b/a$) of up to 1.8 allowed by the geometry of the vacuum vessel. However, most of the experiments on ISX-B to date have been performed using nearly circular plasmas with $\epsilon < 1.2$. In particular, the high-beta experiments have been carried out with circular plasmas. If a beta limit is encountered due to plasma instabilities (which has not happened yet), the plan is to elongate and shape the plasma to see if a higher beta can be attained as predicted by theory.

In order to be prepared for this possibility, about 6% (<500) of the shots on the machine have been devoted to shaping experiments. The results are encouraging: we have been able to create a D-shaped plasma with $\epsilon \geq 1.5$ and maintain it in a stable equilibrium.

Figure 1.29 shows the poloidal coil systems used in the shaping experiments to date. Coils within each grouping are connected in series. A separate radial field coil set is

Fig. 1.29. The poloidal coil systems used for shaping experiments. All coils in a given circuit are connected in series.

not shown. The system in Fig. 1.29(a) creates elliptical plasmas with $\epsilon \approx 1.4$ with no current in the shaping coils. The arrangement shown in Fig. 1.29(b) creates nearly circular plasmas ($\epsilon \leq 1.1$) and is the system that has been used in most of our experiments. Elongations ≥ 1.5 have been attained using both systems, with the lower system producing more strongly D-shaped (as opposed to elliptical) plasmas. The system configuration can be changed to create other plasma shapes by connecting the coils in different ways. A feedback control system varies the currents in the inner and outer coil sets to maintain

the plasma current and position at desired values. After the plasma is initiated, a preprogrammed current shape is run through the shaping coils, changing the plasma shape. A typical shot is shown in Fig. 1.30. The shape of the plasma edge (defined by the last flux surface that does not intersect the limiter) is determined from magnetic probe measurements.

Fig. 1.30. Typical shaping results from May 1979. Shaping coil current is turned on at 50 ms, elongating the plasma as shown in (b).

The primary obstacle encountered in the shaping experiments has been keeping the plasma centered vertically. To create an elongated plasma, it is necessary to apply an external vertical field with a negative index of curvature:

$$n = - \frac{R}{B_z} \frac{\partial B_z}{\partial R}.$$

A plasma in this curvature field is unstable to vertical displacements; the greater the elongations, the more unstable the equilibrium. An active vertical feedback control system is used to hold the plasma in position. Vertical motion of the plasma is detected by magnetic sensor coils at the top and bottom of the vacuum vessel, and the control system changes the current in the radial field coils to drive the plasma back toward the midplane. The feedback control problem is complicated by the presence of the thick stainless steel vacuum vessel (1.9 cm thick at the top and bottom) between the radial field coils and the plasma, which shields the plasma from fast changes in the radial field outside the vacuum vessel. Considering only the lowest decay mode of eddy currents in the vessel, the relation between B_R^{in} and B_R^{out} is given by

$$\tau_v \frac{dB_R^{in}}{dt} + B_R^{in} = B_R^{out},$$

where the vessel time constant is $\tau_v = 10$ ms.

Figure 1.31 shows the vertical position Δz for two shots. In the lower picture, the plasma escaped from the control or the feedback system as the shaping current was increased. The upper picture shows the behavior during a well-controlled shot. The maximum elongation that can be successfully maintained is determined by the characteristics of the feedback system and the radial field supply. In particular, the voltage of the radial field supply is of great importance, because the rate at which current can be changed is determined by $I_{RF} = V_{RF}/L_{RF}$, where L_{RF} = inductance of the radial field coil. The data in Fig. 1.30 are typical of results obtained in May 1979 with a 36-V radial field supply. Maximum stable elongations were ≤ 1.4 . The radial field supply voltage was increased to 60 V in August, which permitted elongations ≥ 1.5 to be maintained stably. A typical shot is shown in Fig. 1.32.

Future shaping experiments will be performed using other coil configurations and in conjunction with neutral beam heating. In the future, radiometer and soft x-ray arrays

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PLASMA VERTICAL
POSITION (1 cm/div)SHAPING COIL CURRENT
(2.8 kA/div)PLASMA CURRENT
(150 kA/div) VERTICAL CONTROL
MAINTAINED
THROUGHOUT SHOT VERTICAL CONTROL
LOST AT $t = 40$ ms

Fig. 1.31. Plasma response to shaping. (a) Plasma vertical position is controlled throughout the shot. (b) Vertical feedback system loses control as shaping coil current is increased. Plasma current terminates early as plasma crashes into the top limiter. The spike in the shaping current is caused by the increase in the loop voltage as the plasma goes out.

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Fig. 1.32. Shaping results from August 1979 with $e \approx 1.6$.

will be operational on the machine; these will yield 2-D information about the plasma shape and also details of the internal structure of the plasma. The addition of these and other diagnostics will provide more information on the effects of plasma shaping on, for example, MHD activity and energy confinement time so that a study of the effects of shaping on these and other plasma properties can be made.

Studies of MHD instabilities with neutral beam injection in ISX-B

R. D. Burris,^{*} J. L. Dunlap, A. P. Navarro,[†]
V. K. Pavē[‡]

Plasma instabilities have been studied with sets of Mirnov coils around both poloidal and

^{*} Computer Sciences Division, ORNL

[†] Visiting scientist, Spanish Nuclear Energy Commission

[‡] Instrumentation and Controls Division, ORNL

toroidal perimeters of the vacuum vessel and a single soft x-ray detector collimated to view along a vertical line near the center of the vessel. The arrangement of the diagnostics in a poloidal section is shown in Fig. 1.33. The availability of only one x-ray channel seriously limited the work, because many channels are necessary to characterize the instabilities. At this time, no definitive statement can be made concerning the presence of pressure-driven modes.

Fig. 1.33. Soft x-ray and Mirnov coil diagnostics in a poloidal section.

The responses observed can be summarized in terms of three broad regions of injected neutral beam power (or, equivalently, plasma beta). Oscilloscope traces representative of the majority of the discharges studied (those that had sawtooth oscillations at the time of beam injection) are shown in Fig. 1.34. The responses are detailed in Table 1.8.

As noted in Table 1.8, increasing the injected power leads to gradual distortion of

Fig. 1.34. Representative soft x-ray and Mirnov coil signals. The x-ray detector used a beryllium absorber 2.5×10^{-3} cm.

the x-ray signal away from the classic signatures of the $m = 1$ induced internal disruption and the inside sawtooth. Part of this distortion must result from the shift of the magnetic axis to a larger major radius, a shift that accompanies increasing beta. At high powers, oscillations in the x-ray signal lose the characteristic identifiers of the $m = 1$ mode, no inside or outside sawtooth is present, and therefore no mode assignment can be substantiated by that signal.

Studies of the Mirnov signals from operation with medium and high injected powers are in progress. The tentative conclusion is that the instability structure is dominated by an $m = 2, n = 1$ mode. However, phase variation along the inside wall differs from that expected for a pure $m = 2$ mode, and the field perturbation at the outer wall is about twice that at the inner wall. These inside-outside differences are not uncommon with ohmically heated plasmas and do not necessarily indicate

Table 1.8. MHD diagnostic responses for typical discharges
(sawtoothing when NBI occurs)

Injected power	$\tilde{X}$ response	$\tilde{B}_\theta$ response
Zero	$\tilde{X}_0$ and $\tilde{X}_1$ those of inside sawtooth	Defines a forward going mode (FG, the electron diamagnetic direction) with $\omega > \omega[\tilde{X}_1]$; i.e., doing its own thing (DIOT); $\tilde{B}_\theta/B_\theta \approx 10^{-4}$
Low	$\tilde{X}_0$ and $\tilde{X}_1$ continue as above with enhanced amplitudes and period of $\tilde{X}_0$ At higher powers, $\tilde{X}_0$ and $\tilde{X}_1$ waveforms are altered by outward shift of the plasma	As above when there is only modest enhancement of the sawtooth With larger enhancement, $\tilde{B}_\theta$ starts to frequency couple to $\tilde{X}_1$ when the latter's amplitude is large just before the internal disruption; this mode at $\omega[\tilde{X}_1]$ is reverse-going (RG, the ion diamagnetic direction), $M = 1$, $1 < M \leq q_a$, and somewhat higher $\tilde{B}_\theta/B_\theta$; when $\tilde{X}_1$ is quiescent, $\tilde{B}_\theta$ is DIOT
Medium	$\tilde{X}_0$ loses the classic sawtooth shape, but outside sawteeth do not appear $\tilde{X}_1$ is sometimes still definable from frequency doubling of the positive half cycle, but this feature is also eventually lost at higher power $\tilde{X}_1(\tilde{X}_2)$ exhibits slower growth rate, and becomes continuous through the internal disruption, tending toward a steadier signal; the ω becomes high, 13-20 kHz as compared with ~7-10 kHz for the uncoupled $\tilde{B}_\theta$	DIOT when $\tilde{X}_1$ is quiescent Frequency coupled to $\tilde{X}_1$ and $\tilde{X}_2$, RG, $M = 2$, $1 < M \leq q_a$, and higher $\tilde{B}_\theta/B_\theta$
High	$\tilde{X}_2$ continuous, mostly at large amplitude but irregularly and often abruptly modulated to small amplitude; remains at high ω or $\tilde{X}_2$ reverts to mostly intermittent bursts of large amplitude, much like some behavior at medium power There is some evidence the latter is favored by higher plasma density	Like medium power; $\tilde{B}_\theta/B_\theta$ to ~0.5%

the presence of pressure-driven modes. Using an analysis based on a cos m eigenmode expansion, the differences result from phase reinforcement of the different modes along the outward-directed horizontal.

Reversal of the apparent direction of the mode rotation evident in the loop signals is

presumably Doppler shifting that results from the transfer of toroidal momentum to the plasma by the co-injected beams. In some discharges, the plasma has a well-developed $m = 2$ instability at the time of beam injection, rather than the sawtoothing. In these, the $m = 2$ usually persists as a well-defined mode

throughout the duration of the discharge and the times of reversal are easily identified. The poloidal propagation is initially in the electron diamagnetic direction, but when injection is applied, the observed frequency of rotation gradually slows to zero, the propagation switches to the ion-diamagnetic direction (equivalent to the toroidal velocity in the direction of the injected particles dominating the mode phase velocity), and the frequency increases. A second passage through zero frequency and reversal occurs after injection ceases.

The highest observed frequencies for reverse-going modes have been approximately 20 kHz. For these, a toroidal mode velocity of $\sim 1 \times 10^7$ cm/s is calculated, with the assumption that the frequency observed is due only to toroidal motion. No charge exchange or spectroscopic measurements of toroidal rotation are available for comparison.

Work is proceeding to add additional channels of x-ray diagnostics in the form of three arrays in one poloidal section. These will be useful for more detailed study of the

instabilities and for obtaining plasma shape parameters.

Impurity studies during neutral beam injection

R. C. Isler, E. C. Crume, D. Armarius,
L. Murthy

One concern about the use of neutral beam injection as a method of auxiliary heating in tokamaks is that additional impurities may be introduced into the discharge by sputtering caused by high-energy atoms or ions. However, Stacey and Sigmar²⁶ have recently emphasized another aspect of this problem, i.e., that external momentum sources can also alter impurity transport. Under the appropriate conditions, coinjection (in the direction of the plasma current) may result in a net outward transport of impurities, whereas counterinjection can produce the opposite effect. Some observations that may substantiate this conjecture have been made on impurities in the ISX-B tokamak.

Figure 1.35 shows a comparison of the radiation from Ar VII and Ar XVI during a

Fig. 1.35. Spectral emissions from Ar VII and Ar XVI. The argon is introduced into the plasma in a 2-ms burst at 100 ms after breakdown.

discharge in which no neutral beam injection was employed and during one in which approximately 500 kW of coinjection was used. The argon was introduced as a gas puff at 100 ms. The intensity of the Ar VII line quickly reaches a steady-state value in both cases, indicating that there is a great deal of recycling in the periphery of the plasma, i.e., that there is a strong anomalous outward transport that tends to counteract the usual inward neoclassical transport mechanisms.

Emissions from Ar XVI, which is produced within $a/2$, differ markedly in the two cases. Without injection the argon transports slowly toward the interior of the plasma and appears to accumulate there continuously, whereas with

injection an almost steady-state value without appreciable accumulation is obtained in 20 ms. These results seem to show that a neoclassical contribution to the outward transport may exist in the presence of neutral beam injection.

Figures 1.36 and 1.37, which illustrate a comparison of the behavior of iron radiation with and without injection, tend to corroborate the conclusions observed from the argon puffing experiments. The peripheral ions such as Fe IX always demonstrate an increase of radiation during injection. If the iron concentration were to remain constant during injection, the Fe IX intensity should actually decrease due to the overall heating of the plasma. Therefore, it is clear that the iron

Fig. 1.36. Iron line intensities: (a)-(c) without neutral beam injection, (d)-(f) with neutral beam injection.

Fig. 1.37. Iron line intensities: (a)-(c) without neutral beam injection, (d)-(f) with neutral beam injection.

concentration in the periphery increases. However, an increase of T_e should generate a strong rise in the Fe XXII radiation, and a decrease is actually observed. It appears that the iron is actually being driven from the plasma interior to the edge during injection. To explore this point further we have calculated a total iron distribution as a function of radius by fitting the intensities observed in the noninjected case to intensities calculated with the RECYCLE CODE²⁷ using measured values of $T_e(r)$ and $n_e(r)$. Then, under the assumption that the iron profile remains constant as the neutral beam heats the electrons [$T_e(0) = 900 \text{ eV} \rightarrow T_e(0) = 1250 \text{ eV}$], we have calculated the expected changes of intensity. The calculated ratios are plotted in Fig. 1.38 together with the

actually observed ratios. It is obvious that the two trends are opposite, and indications are that during injection, the impurity transport to the center of the plasma is indeed reduced.

1.3.2 Pellet Injection Studies

ISX-B pellet injection results and implications for fusion fueling

G. L. Milora, C. A. Foster, C. E. Thomas,
J. P. Wilgen, C. E. Lush, E. A. Lazarus,
H. C. Howe, G. H. Neilson, D. P. Butshinaon,
P. H. Edmonds, J. L. Dunlap

High-speed pellet fueling experiments have been performed on the ISX-3 device in a new regime characterized by a massive global density rise ($\Delta \bar{n}/\bar{n} > 300\%$) in both ohmic and

Fig. 1.38. Ratios of line intensities for injection vs no injection.

neutral beam heated discharges. Hydrogen pellets 1 mm in diameter are injected into the plasma midplane with a pneumatic injector at velocities exceeding 1 km/s. Pellets frequently penetrate beyond the magnetic axis in low temperature ohmic discharges, and in such cases a sharp decrease in ablation is observed as the pellet passes the plasma center. This behavior can only be accounted for by an ablation model that includes dynamic cooling of the target plasma while the ablation proceeds. It is postulated that the limited heat content of the plasma, particularly on small radii flux surfaces, regulates the ablation rate. Complete penetration is prevented by operating in low density regimes, where runaway electrons are thought to be responsible for anomalous ablation, or by utilizing moderate to large amounts of neutral beam injection. There is a strong enhancement in the ablation in the outer 10-cm plasma

region even for short heating intervals, and this effect can be explained qualitatively by a pellet ablation model that accounts for the presence in the discharge of multi-keV ions (which can easily penetrate the thin ablation cloud and impact the pellet surface).

Density increases of ~300% have been observed (in ~250- μ s time scale) without degrading plasma stability or confinement. Penetration to $r \sim 7$ cm is optimal. For such cases, large amplitude sawtooth oscillations are observed and the density remains elevated (detailed experimental results are presented below). Confinement time is increased from 8 ms to 25 ms, which is in agreement with the empirical scaling $\tau_E \sim n_e$. The low level of power emission from the pellet-plasma interaction region is consistent with negligible charge exchange losses, and within the experimental accuracy, nearly all of the pellet mass can be accounted for in the initial plasma density rise (as compared with the 30% of gas puffing). Gross plasma stability is dependent roughly on the amount of pellet penetration and can be correlated with the temporal evolution of the current density profile [i.e., whether $j_e(r)$ steepens or broadens]. Penetration to or beyond the magnetic axis usually results in external disruptions in ~10 ms.

For mean ablation rates of $\sim 2 \times 10^{23}$ ions/s, ruby-laser interferograms of the pellet-plasma interaction region indicate that a highly ionized ($n_e \sim 1 \times 10^{13} \text{ cm}^{-3}$) cloud (with scale length of ~1 mm) surrounds the pellet. There is no direct indication of the existence of a large, high density neutral cloud directly exposed to the tokamak plasma ions, which would precipitate large charge exchange losses.

Several conclusions can be drawn from these results concerning the practicality of pellet fueling in fusion devices: (1) pellets larger than heretofore thought possible can be used for refueling, which reduces the velocities required for deep penetration; (2) penetration to the magnetic axis has been

demonstrated to be unnecessary, if not undesirable; (3) plasma stability and confinement do not deteriorate even in the regime of large pellet mass to plasma mass ratio; (4) the efficiency of refueling by pellet injection is far greater than by gas puffing; and (5) the concern that the pellet will serve as a large source of charge exchange neutrals is unfounded. Although higher ablation and reduced penetration are observed with neutral beam injection (for up to 900 kW of injected power), the highest mean ablation rate measured exceeds the hot, high density ohmic case by only 25%.

Pellet-plasma interaction studies on ISX-B

E. A. Lazarus, S. L. Milora, C. A. Foster,
C. E. Thomas, J. B. Wilgen, C. E. Bush,
E. C. Howe, G. H. Neilson, D. P. Hutchinson,
P. H. Edmonds, J. L. Dunlap, T. C. Jernigan,
M. Marukami

We have injected very large pellets ($\Delta n/n \sim 3$) into ohmically heated discharges to determine the limits on density perturbation via pellet injection in tokamak discharges. The initial conditions were: $I_p = 120$ kA, $B_T = 15$ kG, $\bar{n}_e = 1.1 \times 10^{13} \text{ cm}^{-3}$, and $T_e(0) \approx 1500$ eV. Typical behavior of the line-averaged density is shown in Fig. 1.39 for shot #15646. Also shown is the behavior for penetration that is too shallow (15657) and for penetration to the magnetic axis (15678). Characteristic MHD behavior for these three cases is shown in Fig. 1.40 along with the H_α luminosity that is characteristic of the ablation rate. For the optimal case (15646) there is no increase in the δB_θ amplitude, and the PIN detector shows rapid temperature recovery accompanied by sawtoothing. Shallow penetration (15678) shows a large, sustained increase in $m = 2$ oscillations, and the most rapid loss of added fuel is observed. On the other hand, if the pellet penetrates very close to the magnetic axis (shots 15674 and 15657), a minor disruption will occur in ~ 20 ms. It is believed that the large shift of the centroid of the current density, which

Fig. 1.39. Line-averaged density for a variety of pellet penetration depths, obtained from both infrared laser and 2-mm microwave interferometry (which showed essentially perfect agreement). Shot 15646 represents optimal penetration to ~ 7 cm outside the magnetic axis, shot 15678 represents penetration to the magnetic axis, and shot 15657 represents too shallow penetration where large $m = 2$ oscillations are created.

must occur with this severe cooling of the center of the plasma, causes the disruption. Furthermore, once the pellet reaches the magnetic axis, it has encountered and cooled the bulk of the electrons in the discharge. Thus, little further ablation occurs, and unless ablation is virtually completed, the pellet impacts on the inner wall. This then acts as an extremely large gas puff, and the attendant edge cooling shrinks the current channel sufficiently to cause the minor disruption. In summary, the fueling is optimal when the pellet penetrates well within the $q = 2$ surface but is stopped before reaching the magnetic axis. For these experiments penetration to $r \approx 7$ cm was found to yield the best performance.

The slow rise in the line-averaged density after the initial jump reflects the filling in of an initially hollow density profile, as evidenced in the sequence of Thomson scattering profiles in Fig. 1.41. The density profiles obtained via Thomson scattering are normalized to the line-averaged density as measured by both 2-mm and FIR laser interferometry. By

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Fig. 1.40. Ablation rate, $\dot{B}_0$ response, and soft x-ray emission for three shots. Shot 15646 represents optimal penetration to ~7 cm outside the magnetic axis, shot 15674 represents penetration to the magnetic axis, and shot 15687 represents too shallow penetration.

integrating the density we determine the total number of particles in the discharge.

Just prior to injection the discharge contains 0.87×10^{19} electrons; 1.7 ms later it contains 3.65×10^{19} electrons. This shows an increase of 2.8×10^{19} particles (compared with 3.1×10^{19} particles contained in the pellet). Thus, 90% of the fuel is accounted for in the plasma subsequent to injection.

Charge exchange ion temperature measurements show that after the cooling, the ion temperature rises above its prepellet value, as expected from the increased electron-ion coupling. The entire plasma recovers from the injection, as evidenced by the steady increase in the density-averaged electron temperature (Fig. 1.42) obtained from Thomson scattering profiles. This temperature increase is still occurring

Fig. 1.41. Time evolution of temperature and density profiles, determined from Thomson scattering measurements (subsequent to pellet injection). Also shown is the average ablation rate over this sequence of shots. An inverted density profile is observed shortly after pellet injection, but it gradually fills in, consistent with the line-averaged density shown in Fig. 1.39. Plasma pressure is conserved locally.

Fig. 1.42. Density-averaged electron temperature obtained from Thomson scattering profiles shown in Fig. 1.41 and central ion temperatures measured by charge exchange analyzer.

at the end of the interval over which these measurements are made (23 ms). Furthermore, the lack of change in plasma energy content in the first 2 ms demonstrates that the process is adiabatic (Fig. 1.43).

From the Thomson scattering profiles before and after pellet injection we can estimate the improvement in confinement time using

$$\frac{dW}{dt} = \frac{W}{\tau_E} + P_{in}$$

Because dW/dt is not zero and there is large uncertainty in its value, as well as in the true ion temperature profile, we can only set limits here. From laser profiles at times from 15 to 23 ms, we can estimate that

$$0 < \frac{dW}{dt} < 63 \text{ kW}$$

Fig. 1.43. Electron energy density E_e before and after pellet injection, obtained from the integration of Thomson scattering profiles. This clearly shows the adiabatic nature of the pellet fueling process. Furthermore, integration of the density profiles accounts for 90% of the pellet mass. Also shown are the ion energy E_i , assuming that the ion temperature profile is the same as the electron temperature profile (normalized to the central ion temperature), and the radiated power $\int P_{rad} dt$, which is typical of ISX discharges at these densities.

and thus

$$25 \text{ ms} < \tau_E < 45 \text{ ms}.$$

Prior to the pellet, τ_E was 10 ms. Both the 10- and 25-ms values are typical ISX-B values at the stated densities, and the increase is consistent with Alcator scaling, indicating that the plasma has not suffered any deleterious effects from the pellet injection. The change in Z_{eff} (Fig. 1.44) is consistent with simple dilution of impurities by the added fuel.

From the time required for the central density to fill in, the diffusion constant near the center is estimated to be $\sim 1 \text{ cm}^2/\text{ms}$. The n_e profile evolution indicates (e.g., compare n_e profiles at 1.7 and 4.7 ms) that heat conduction is more rapid than particle diffusion in the central region of the plasma. The fact that the density peak narrows and appears to move inward indicates that either there is a strong radial dependence on the

Fig. 1.44. Changes in Z_{eff} upon pellet injection. At the time of pellet injection, there is a large inductive change in the plasma that creates a spike in the loop voltage; thus, plasma resistivity cannot be determined during the first 14 ms after the pellet is injected. The change in Z_{eff} is consistent with simple dilution of the impurities by the added fuel. Also shown is typical loop voltage behavior.

diffusion constant or other transport processes are important. Further experiments are planned on a much finer time scale with increased spatial resolution to further study the transport processes.

1.3.3 Electron Cyclotron Heating Studies

ECH experiments in the ISX-B tokamak

R. M. Gilgenbach,^{*} M. E. Read,[†] K. E. Hackett,[‡]
R. Lucey,^{*} B. Hui,[†] V. L. Granatstein,[†]
K. R. Chu,[†] A. C. England, C. M. Loring,
O. C. Eldridge,[§] H. C. Howe, A. G. Kulchar,[§]
E. Lazarus, M. Murakami, J. B. Wilgen

Experiments with ECH have been performed on a tokamak in which the electron temperature ($\sim 1 \text{ keV}$) and major radius (93 cm) are such that significant single pass microwave absorption is expected. Theories predict approximately 100% single pass absorption of the

^{*}JAYCOR, Alexandria, Virginia

[†]Naval Research Laboratory, Washington, D.C.

[‡]Massachusetts Institute of Technology, Cambridge, Massachusetts

[§]University of Tennessee, Knoxville, Tennessee

extraordinary wave; the ordinary wave absorption is 50% in one pass. The ohmic heating power and energy confinement time (10 ms) of the tokamak are comparable to the power (100-150 kW) and pulse length of the gyrotron microwave source. The frequency of the gyrotron (35.08 GHz) was such that the electron cyclotron resonant surface could be located at any desired minor radius. This enabled us to heat the plasma center or to perform off-axis heating. Peak electron densities in these experiments were kept at about $1.5 \times 10^{13} \text{ cm}^{-3}$, which corresponds to the ordinary wave cutoff density for the 35-GHz radiation. Microwave power was transmitted to the tokamak with the TE_{01} circular mode in oversized (6-cm-ID) copper waveguide. A quasi-optical miter bend with coupling holes was used to monitor the forward microwave power; reflected power was about 20 dB lower. Unpolarized microwaves were injected from the midplane of the high-field side of the tokamak to avoid the cutoff of the extraordinary wave. The angle of incidence was 45° from the toroidal direction at the plasma edge. A launching antenna was located in the shadow of a limiter inside the tokamak chamber and consisted of an open circular waveguide with a reflecting plate. Electron temperatures were measured by laser Thomson scattering and by two superheterodyne receivers that measured the second harmonic cyclotron emission at 70 GHz and 58 GHz. A silicon-lithium detector gave time-integrated soft x-ray energy spectra to determine whether nonthermal electrons were generated by ECH. The experimental results demonstrated an increase of about 50% in central electron temperature when heating was on axis with 80-kW power levels. Good agreement was found between the electron temperature measurements and an empirical transport code. The efficiency of ECH (60%) on axis is in good agreement with theory. Off-axis heating was performed by lowering the toroidal magnetic field to 11.1 kG. Detection of the second harmonic cyclotron emission at 70 GHz showed a smaller electron temperature increase than

when heating was at the center of the plasma; this indicates that heating occurred over a larger plasma drift surface. The second harmonic cyclotron emission at 58 GHz (10 kG) was observed to increase over the emission when heating was at the center because this frequency corresponds to the low-field side of the ECH resonant drift surface.

ECH/UHR preionization experiments in the ISX-B tokamak

R. M. Gilgenbach,^{*} M. E. Read,[†] K. E. Hackett,[‡]
R. Lucey,^{*} V. L. Granatstein,[†] A. C. England,
C. M. Loring, J. B. Wilgen, P. C. Isler,
K. E. Burrell,[§] O. C. Eldridge,^{||} M. Hacker,[§]
P. King, A. Kulchar,^{||} M. Murakami, Y.-K. M.
Feng, R. Richards

We report the results of preionization experiments in which 80-kW microwave pulses from a 35-GHz gyrotron are injected into a tokamak. The experimental configuration is discussed above. Two types of experiments were performed. In the first type, microwaves were injected with only the toroidal magnetic field applied to the tokamak. This enabled us to study the preionized plasma electron density, temperature, and confinement time. The central line-averaged electron density was measured by horizontal and vertical 2-mm microwave interferometers. By varying the toroidal magnetic field from shot to shot, we obtained a time-resolved density profile. Electron temperature was measured by several techniques, including spectroscopy of oxygen and iron line radiation, Langmuir probes, and a gridded analyzer. The temporal behavior of the preionized plasma was studied using high-speed movies (3000 frames/s).

^{*}JAYCOR, Alexandria, Virginia

[†]Naval Research Laboratory, Washington, D.C.

[‡]Massachusetts Institute of Technology,
Cambridge, Massachusetts

[§]General Atomic Company, San Diego, California

^{||}University of Tennessee, Knoxville,
Tennessee

These experiments demonstrated that significant electron densities ($\sim 5 \times 10^{12} \text{ cm}^{-3}$) can be obtained with microwave preionization. The microwave absorption appears to be localized at the electron cyclotron and upper hybrid resonance frequencies. Bulk electron temperatures are in the range of 8-13 eV; however, there are indications of higher electron temperatures in the upper hybrid layer. The preionized plasma appears to remain stable with only a toroidal magnetic field. Particle confinement time is about 10 ms, the time it takes the electron density to decay to half its value after the microwaves are turned off. In a second type of experiment, microwave preionization was applied from 1 ms to 11 ms prior to a standard tokamak shot. Preionization eliminated a large voltage spike normally required for breakdown and reduced the loop voltage early in the shot by as much as 40%. The initial rate of current rise was higher than in a standard discharge, and Thomson scattering measurements show a higher electron temperature throughout the current-rise phase of the discharge. The peaks of oxygen and iron line radiation are lower than when ohmic heating alone is employed. A pyroelectric detector shows that the total radiation losses are lower immediately after the startup of the ohmic current.

ECH experiments at 28 GHz on the ISX-B tokamak

C. M. Loring, G. L. Campen, O. C. Eldridge,
A. C. England, A. G. Kulchar,* W. L. Wright*

The ECH experiments with the Varian gyrotron were expected to start in mid-1979. This gyrotron should be capable of producing $>150 \text{ kW}$ of power at 28 GHz for pulse widths up to 100 ms.

The equipment has not yet operated due to delays in installing and testing of the high-voltage power supply. All other components

are now or have been in place, and except for details the rest of the system is operable. Initial operation of the gyrotron is expected early in 1980, and high-power experiments should start in the spring of 1980.

1.3.4 Ripple Studies

*J. F. Lyon, J. A. Rome, S. M. DeCamp, R. A. Dory,
J. L. Doolan, R. E. Fowler, S. L. Halsead,
V. E. Lynch, J. T. Mihalzo, G. E. Neilson,
B. E. Nelson, J. A. O'Toole, S. D. Scott,
J. Sheffield, E. Slaughter, D. J. Strickler,
R. E. Worsham, R. B. Wisor*

The ISX-B ripple studies program has two major objectives: (1) determining the maximum allowable toroidal field (TF) ripple for future tokamaks — both the periodic (global) field ripple from TF coils and the localized field ripple, as from a bundle divertor or ripple-producing coils and (2) testing the principle of ripple-assisted neutral beam injection in tokamaks.

Determination of the maximum allowable ripple is to be accomplished by detailed comparison between theory and experiment on ISX-B using three types of ripple-producing systems having the same on-axis ripple ($\leq 1.5\%$) but very different spatial variations of this field ripple. The physics issues studied are: (1) fast ion orbit losses — injected fast (beam) ions and the high-energy tail of the background ion population; (2) thermal ion energy losses — bulk ion heat conduction due to ripple; (3) ripple perturbation of poloidal flux surfaces — stationary islands and ergodic regions; and (4) ripple damping of beam-induced toroidal rotation.

The ripple-assisted injection experiment has been described previously.²⁸ The vertically injecting, diagnostic neutral beam line to be used for these studies is being installed and is discussed in detail in Sect. 1.3.7.

Ripple configurations studied

Three different ripple configurations with different spatial variations are being studied:

*University of Tennessee, Knoxville, Tennessee

(1) variable, periodic TF ripple obtained by operating with either 9 or 18 TF coils; (2) two eyeglass-shaped coils ("ripple coils") located under ISX-B that produce a toroidally localized ripple that decays rapidly in the vertical direction; and (3) a bundle divertor that will produce an up-down symmetric, very strong, toroidally localized ripple (especially near the plasma edge). The presently installed (i.e., the first two) configurations are indicated in Fig. 1.45 along with toroidally scanning, mass-selective charge exchange analyzers used to study the different ion populations (TF ripple trapped, ripple coil trapped, banana trapped, and

circulating ions). The properties of the different configurations are summarized in Table 1.9, where the peak-to-average field ripple δ is defined by $\delta = (B_{\max} - B_{\min}) / (B_{\max} + B_{\min})$, where $B_{\max}$ and $B_{\min}$ are the maximum and minimum values of the toroidal field at a given major radius.

The 9-TF-coil bus modification just completed allows energizing either set of alternate TF coils or all 18 TF coils. Ripple values between the two extremes shown in Table 1.9 could be obtained by adding a parallel resistive shunt on an individual TF coil or on a set of 9 TF coils. Although reducing the current in a single TF coil can

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Fig. 1.45. Partial plan view of ISX-B showing the two alternating sets of nine TF coils, the eyeglass-shaped ripple coils beneath the vacuum chamber, and the diagnostic neutral beam (solid circle) between the ripple coils and their spatial relation to the two toroidally scanning, mass selective charge exchange energy analyzers.

Table 1.9 Ripple configuration properties

Coils	9-18 TF coils	Ripple coils	Bundle divertor
Toroidal localization	Periodic	4 of 18 TF coils	2 of 18 TF coils
δ on axis (%)	0.01-1	<1.5	≈ 1
δ at plasma edge (%)	0.7-7	3 (down) 0.3 (up)	~ 100 (out) ~ 0.01 (in)
Symmetry	Up-down; toroidally periodic		Up-down
Status	Installation finished	In use	Install end of 1980

give approximately the same on-axis ripple as a 9-TF-coil set, the two cases result in very different toroidal variations of the ripple. A 9-TF-coil set produces a well with a toroidal period of 40° , and a single reduced current coil produces a well that extends 120° around the torus on either side of that coil. The ripple coil well extends $\pm 30^\circ$ toroidally, and the bundle divertor well extends $\pm 7^\circ$ toroidally.

Fast ion losses

It is well known that the small class of ions trapped in a local field well drifts vertically due to a toroidally uncompensated $B \times \nabla B$ drift and is quickly lost. Recently Kome et al.²⁹ have found that a much larger class of banana-trapped ions (locally $\sim 1/3$ of the ions) can also suffer a similar loss due to their interaction with the field maxima on either side of the field well produced by a bundle divertor (or the topologically similar ripple coils). Essentially the fast ion banana tips occur further out on each interaction with these maxima than they would in their absence, and the banana tips (where $V_{||} = 0$) follow $|B|$ contours to the wall. A ripple at the top or bottom of the tokamak (as from the ripple coils) has a larger deleterious effect than a ripple on the equatorial plane (as produced by a bundle divertor). Circulating ions, on the other hand, are relatively unaffected. Although the canonical angular momentum p_ϕ is no longer conserved because of the broken toroidal symmetry, p_ϕ does appear to be periodic in time, and a new orbit invariant presumably occurs.

Some experimental evidence for this fast ion loss is obtained using the ripple coils and fast ions from the main heating beams. Figure 1.46a shows the time behavior of the measured charge exchange flux at 3 keV and 18 keV from slowing-down injected (beam) protons. Turning on the ripple coil field produces an apparent loss of near perpendicular (7° pitch angle) fast protons that increases with increasing ripple strength and ion energy. Figure 1.46b shows the effect of this ripple on the background deuteron temperature inferred from neutron measurements, which should be more representative of the total ion behavior. However, this measurement also overestimates the effect of the ripple on the total ion losses because the neutron-producing deuterons are in the high-energy tail of the ion distribution and hence more strongly affected by the ripple. Use of the new pitch angle scanning capability for the charge exchange analyzers should permit a better evaluation of the fast ion orbit losses.

Effect of TF ripple on bulk ion heat conduction

In addition to fast ion orbit losses, and hence reduced beam energy transfer to the background ions, ripple trapping can lead to background ion energy loss through increased ion heat conduction. Although the small ripple from the 18 ISX-B TF coils has an insignificant effect on ion heat conduction, the recent 9-coil modification with its 10-100 times larger ripple can have a dramatic effect. Figure 1.47 shows the ion heat conduction coefficients calculated³⁰ for ripple trapping,

Fig. 1.46. Effect of the ISX-B ripple coil fields on (a) near perpendicular fast ion losses and (b) the neutron-inferred temperature of the background ion population.

neoclassical, trapped-ion mode, and banana drift losses for expected ISX-B parameters with 9 TF coils. Although poloidal averaging of the TF ripple will reduce the dominance of the ripple-trapping contribution to the ion heat conduction, it should still be possible to assess the effect of TF ripple on ion heat conduction in ISX-B.

Ripple perturbation of poloidal flux surfaces

Enhanced cross field transport can also be produced by ripple perturbation of poloidal flux surfaces. Figure 1.48 illustrates this

Fig. 1.47. Calculated ion heat conduction coefficients for losses from ripple trapping, neoclassical, trapped ion mode, and banana drift for nominal ISX-B parameters with nine TF coils. Parabolic density and temperature profiles were assumed with $T_i(0) = 1 \text{ keV}$, $n_e(0) = 3 \times 10^{13} \text{ cm}^{-3}$, and $Z_{\text{eff}} = 2$.

effect for the ISX-B ripple coils producing 1% peak-to-average ripple on axis. The perturbation calculated³¹ depends on the strength of the perturbing (ripple) field at the mode rational surfaces. In Fig. 1.48a, with $q(a) = 3.2$ and $q(0) = 1$, the ripple field strength at the $q = 2$ surface is large enough to produce an ergodic field line region at the outside of the plasma; at the center it produces an $m = 1$ stationary island. In Fig. 1.48b, with $q(a) = 5.2$ and $q(0) = 1.6$, no $q = 1$ surface exists in the plasma, and the $q = 2$ surface is farther from the ripple coils, resulting in a stationary $m = 2$ island. Some experimental evidence for the latter case is shown in Fig. 1.49. In the absence of the ripple coil fields, a constant amplitude $m = 2$ oscillation is seen for the entire shot duration. Although the $m = 2$ oscillation amplitude apparently ceases when the ripple field is turned on, it often resumes at essentially the same amplitude when the ripple field is turned off and the plasma density remains constant

Fig. 1.48. Perturbation of poloidal flux surfaces for 1% ripple on axis due to ripple coil fields for (a) $q(0) = 1$, $q(a) = 3.2$ and (b) $q(0) = 1.6$, $q(a) = 5.2$.

Fig. 1.49. Distorted $m = 2$ instability mode waveforms indicating mode locking with a ripple-produced island. The traces indicate a PIN diode soft x-ray signal, a $\dot{B}_\theta$ MHD loop signal, and four $\dot{B}_\theta$ loops on an expanded time scale equally spaced poloidally around the plasma.

throughout, instead of rapidly rising as it does when an $m = 2$ instability is actually quenched. Further evidence that the $m = 2$ mode has not been damped but has only stopped rotating is seen in the distorted waveforms of Fig. 1.49, which show the mode slowing down and stopping.

Ripple damping of toroidal rotation

Unequal coinjection and counterinjection in a tokamak produces a net momentum input to the plasma and may lead to large induced ion rotation velocities,³² possibly exceeding ion sound speeds. Ripple damping is a possible means of avoiding this, and it should be possible to test ripple damping on ISX-B with

the 9-TF-coil modification. For the same plasma parameters as those used for Fig. 1.47, the calculated³³ damping times vary from 0.1 to 1 ms, a factor of 10-100 times faster than expected for charge exchange damping, the fastest classical process. By contrast, ripple damping times for the 18-TF-coil configuration are greater than 10^3 s.

1.3.5 Impurity Studies

Impurity study program

R. E. Clausing, L. C. Emerson, L. Heatherly, W. R. Eustachy,* R. A. Langley,† J. E. Roberto, B. Schweer,‡ J. E. Simpkins, S. F. Withrow, E. A. Zuhre

The purpose of the Impurity Study Program is to identify the controlling impurity generation mechanisms and to minimize their effects. The approach of the ORNL program involves a vigorous, interactive coalition of surface scientists and plasma physicists that focuses on in situ tokamak experiments. It has coordinated efforts in four main task areas: (1) in situ measurements, (2) development of new diagnostics, (3) plasma materials simulation experiments, and (4) controlled laboratory investigations.

The program involves the study of hydrogen recycle and impurity introduction mechanisms; the in situ testing of special materials such as limiters, beam dumps, armor plate, and divertor plates; the measurements of the impurity charge state in the plasma scrapeoff region; the study of cleanup techniques; and the measurement of beam energy profile composition in neutral beam injectors.

To pursue these tasks, in situ experiments were performed on ISX-B and collaborative experiments were conducted with IPP (at Garching) and KFA (at Jülich). These experiments were supported by controlled laboratory

investigations. Detailed information may be found in references given later in this section.

ISX-B.³⁴⁻⁴⁰ Radial deposition profiles were determined for heavy impurities ($A > 12$) in the plasma edge of ISX-B. The species, quantity, and saturation behavior of the principal impurities (Fe-Ni, Cr, O, C) were determined by ion backscattering and channeling analysis of silicon deposition probes. The iron deposition was linear up to 12 discharges and varied from a $3 \times 10^{14}/\text{cm}^2$ discharge at 2 cm behind the limiter to a $1.5 \times 10^{14}/\text{cm}^2$ discharge at 5 cm behind the limiter. The oxygen profiles showed saturation behavior after one discharge with an estimated deposition rate of $10^{16}/\text{cm}^2$ at 2 cm behind the limiter. The radial concentrations of iron and oxygen were only slightly increased for a series of discharges with 350 kW of neutral injection.

Deuterium depth and damage profiles were determined for amorphous silicon samples exposed in ISX-B using nuclear reaction and ion scattering analysis. Near saturation of retained deuterium was observed after one tokamak discharge. The saturation of damage and deuterium depth profiling and the saturation behavior suggest deuterium edge temperatures of 50-150 eV and fluxes of the $0.5-2 \times 10^{17}/\text{cm}^2$ discharge at 2 cm behind the limiter. Samples exposed to discharges with 350 kW of neutral injection showed nearly identical behavior.

Erosion yields were determined for thin gold films exposed to 20 and 46 deuterium discharges at various locations in the limiter shadow of ISX-B. The erosion of the gold; the accumulation of deuterium, oxygen, and iron; and the damage on the same samples were measured using ion scattering, channeling, and nuclear reaction techniques. Erosion rates of 0.5-1 Å/discharge were observed 2 cm behind the limiter. Total erosion increased with the number of discharges and decreased to zero as the wall was approached. Comparable erosion rates were observed for H_2 and D_2 plasmas. The results are consistent with a dominant role for oxygen sputtering (in the absence of

* Technical University of Vienna, Austria

† Program Manager

‡ KFA (Jülich), Federal Republic of Germany

arcing) in erosion near the limiter radius with negligible erosion at the wall.

The density of neutral iron atoms in the plasma edge of ISX-B was measured by laser fluorescence spectroscopy in collaboration with KFA (Jülich). Preliminary results show a detection sensitivity of $\sim 10^6$ atoms/cm³. The optical background during the discharge was also measured and found to be somewhat lower than expected in the vicinity of the iron transition (382 nm). The results demonstrate the applicability of laser fluorescence spectroscopy for in situ impurity studies.

Simulation experiments.⁴¹ The philosophy behind all plasma-wall simulation experiments has been to model in the laboratory the impurity introduction mechanisms identified on existing tokamaks. The mechanisms are then studied in a suitable simulator where controlled experiments can be conducted.

Introduction of impurities into plasma discharges by unipolar arcing has been observed in both in situ ISX experiments and other plasma experiments. The mechanisms that lead to arc initiation are not well understood and are very difficult to study in large, complex experiments like ISX. In situ ISX studies have been supplemented with a small-scale experiment using a rf-generated plasma. Arc characteristics have been measured and appear to be similar to those observed in ISX. Arcing frequency was found to be a strong function of surface cleanliness and to decay exponentially with plasma exposure time for all conditions examined. This last observation implies that arc initiation sites are destroyed by arcing and that wall conditioning to minimize arcing should not only be possible but desirable.

Many physical processes are active in recycling from the walls of plasma devices, e.g., thermal diffusion, recombination, trapping, ion bombardment, induced diffusion and desorption, and reflection. These processes may be influenced strongly by various physical parameters, e.g., material, surface condition, temperature, and ion bombardment flux and flux

rate. Preliminary work has been completed that demonstrates the applicability of the simulator experiments to plasma devices (in particular, ISX) and has shown that the dominant recycle process is dependent on the particular physical parameters of the system. In cases studied and reported in the literature, thermal processes such as diffusion and recombination dominate at higher temperatures, and ion bombardment effects dominate at lower temperatures.

Sputtering measurements.^{42,43} Low-energy H⁺ sputtering measurements were completed for 200-2000 eV H⁺ on gold, nickel, and 304 stainless steel for H⁺ doses as low as 2×10^{13} /cm². The present results approximate more closely the dose range expected in a fusion reactor and agree well with higher dose measurements performed using the weight-loss method.

A cooperative sputtering experiment with IPP was completed. Identically prepared samples of nickel and gold were used in independent H sputtering measurements by weight-loss at Garching and by ion scattering analysis at ORNL. Agreement in all cases between 200 and 2000 eV was better than 20%.

Neutral beam studies. Impurity production and hydrogen species distributions for neutral beam injectors (NBI) developed by the Plasma Technology Section (FED, ORNL) will be measured by exposing silicon crystal samples to beam pulses and analyzing them by nuclear microanalysis and secondary ion mass spectroscopy (SIMS) techniques. These two important aspects will be studied as a function of beam power, pulse length, and spatial position. Time-resolved measurements will be implemented. The NBIs have been developed for use on PLT, PDX, and ISX. Extraction voltages of up to 50 kV are used, and maximum power injected for a 0.5-s pulse is ~ 1.2 MW with a design goal of 1.5 MW. The Medium Energy Test Facility (METF) will be used for exposure of single crystal (100) silicon samples. Preliminary results have shown that oxygen is the major atomic impurity and is present at levels of 1.5×10^{-3} O/H. Carbon is the next most

abundant impurity, and copper is at or below the sensitivity of the analysis technique. Neutral hydrogen from the injector is expected to have three major energy components, e , $E/2$, and $E/3$, arising from formation of H^+ , H_2^+ , and H_3^+ in the ion source and subsequent neutralization and dissociation in the gas cell neutralizer. The full energy component (40 keV) is easily separated in the depth analysis, but the $E/2$ and $E/3$ components overlap due mainly to range straggling. Analysis by SIMS is used to determine the depth profiles of the implanted hydrogen, and the three energy components are determined from this depth profile.

Impurity flow reversal studies

K. E. Burrell, C. E. Bush, E. de la Fuente,†
M. P. Backer,* R. C. Isler, P. W. King,
E. A. Lazarus, C. E. Muller,* M. Murakami,
L. E. Murray,‡ E. K. Richards, M. J. Saltmarsh*

Our experimental work this year was concentrated in two areas: the extension of the successful proof-of-principle demonstration of impurity flow reversal and the development of better diagnostics for the edge plasma region. Work in these areas is designed to allow us to attack the impurity problem in tokamaks by understanding and, if possible, altering both impurity production and impurity penetration of the plasma.

Because obtaining the same plasma with both signs of the toroidal field can require time-consuming tuning of the plasma, we added (in ISX-B) eight gas injectors (four on the top and four on the bottom) to the vacuum vessel. Short pulses of impurity gases were injected using piezoelectric gas valves.

The most striking difference between the ISX-A and ISX-B neon injection results is the effect of recycling on the time behavior of the various neon spectral lines. The neon

charge states observed in ISX-B (Ne III through Ne VIII) all rise to a maximum and stay there for the rest of the tokamak shot. One possible explanation for the difference in neon behavior in ISX-A and ISX-B would be a major difference in impurity transport in the two devices. To check this hypothesis, we have used our laser impurity injection system⁴⁴ to inject aluminum into ISX-B. We were able to verify that the basic impurity transport in ISX-B is (to within the experimental error) identical to that in ISX-A. Rates in both machines are within a factor of two of neo-classical values.

Because the transport for aluminum, which does not recycle, is the same in the two machines, we believe that the difference in the neon results has to do with the recycling of the neon in the edge. We have tried to match the neon data with the theory by installing edge recycling in our previous code.⁴⁵ However, we were not able to match the data until we assumed that there was also an additional source, due to neon, that was initially ionized in the limiter shadow and only gradually entered the plasma via recycling off the limiter. If we did not include the portion due to the measured pulses, then the code would not produce the rapid rise in the signal seen in the experiment. On the other hand, without the recycling background, the long time behavior of the signal is not correct.

If we add edge recycling to our code, we find that it masks the impurity flow reversal effect almost entirely. For the ISX-B experiments, we would expect less than a 5% effect, which is consistent with our observations of no effect. The reason for this masking is quite simple. Flow reversal builds a barrier to inward impurity transport through the plasma edge that can never be 100% effective. If an impurity recycles at the edge, it has infinite chances to penetrate that barrier and, ultimately, it will do so. Because the particle confinement times in the edge of the plasma are short, this penetration happens quite quickly. This masking effect occurs for

*General Atomic Corporation, San Diego, California

†Visiting Scientist, Mexico

‡Grinnell College, Iowa

any hydrogen gas injection rate, but it is more severe at low hydrogen injection rates where the flow reversal effect is small. Because of hydrogen recycling in ISX-B, we were limited to injection rates of about 25 T-liters/s. In ISX-A, we used 27 to 190 T-liters/s. Accordingly, while titanium gettering in ISX-A may have enhanced the poloidal asymmetry in the hydrogen source, its ability to suppress hydrogen and noble gas recycling apparently played a more important role. Titanium gettering is essential for flow reversal work with noble gas impurities (helium, neon, and argon).

To test our hypothesis that impurity recycling can mask the flow reversal effect, we plan to work during FY 1980 with an impurity gas with a constituent that will not recycle (e.g., SiH_4). Aluminum from the laser injection system is not suitable for these tests, because it is apparently injected as small pellets. Further impurity-related experiments are planned on ISX-B in 1981 as part of the work with the bundle divertor.

Time-resolved measurements of neutral iron density in ISX-B

C. H. Muller, K. H. Burrell

Heavy metal impurity number densities are being measured directly in the ISX-B tokamak using the laser-induced fluorescence technique (LIFT). This represents the first application of LIFT to a tokamak environment for the determination of neutral ground state densities.⁴⁶

To pump a chosen atomic or molecular transition in the species to be detected, LIFT utilizes a tunable source of laser radiation (usually a dye laser). The resulting photoluminescence intensity is, to a first approximation, directly proportional to the pumped-state number density.

The GA-designed LIFT experiment was calibrated in-house for neutral iron, nickel, and chromium using an effusive oven as a metal vapor source. After calibration, the experiment was moved to ORNL and installed on the ISX-B tokamak. A schematic diagram of the ISX-B LIFT experiment

is shown in Fig. 1 50. Output signals from the coaxial fluorescence detection system, the photodiode, and the hollow cathode lamp are digitized with a gated CAMAC ADC and stored in a VAX-11 computer. The gates for triggering the Nd:YAG laser and opening the ADCs are obtained from CAMAC delay generators that are synchronized with respect to an ISX-B trigger.

A plot of measured $[\text{Fe}]$ versus time is shown in Fig. 1.51 for a series of 0.95-T and 1.15-T toroidal field discharges. In both cases iron densities increase by roughly an order of magnitude before disruption at 180 ms. Measured electron densities, however, increase by only a factor of two between 30 and 180 ms. It appears, therefore, that thermal evaporation from stainless steel surfaces is responsible for the observed signal. Sputtering does not seem to be significant, because the $[\text{Fe}]$ signal and the electron (and presumably $[\text{D}^+]$) density do not behave in a similar way. At approximately 180 ms, control over the 0.95-T discharge is lost and a large increase in $[\text{Fe}]$ is noted. The relatively slow signal decrease beginning at 200 ms is also suggestive of a thermal process in which case the hot spots may be cooling down gradually. Finally, there does not appear to be any tendency for the west beam line to "shine" through the discharge to the vacuum walls and create excess iron.

Measurements using LIFT are continuing on ISX-B. Future experiments will include $[\text{Fe}]/[\text{Ni}]$ impurity ratios, velocity profiles for impurity flow measurements, and possibly impurity carbon or oxygen density measurements.

1.3.6 Technology and Operation

Tokamak operations

M. J. Salomoneh, D. D. Bates, S. M. DeCamp, G. E. Dyer, P. H. Edmonds, H. E. Ketterer, J. W. Pearce, J. F. Rayburn, W. J. Redmond

At the beginning of 1979 an attempt was made to establish an operating schedule for ISX-B that would optimize the time available for the various experimental programs while allowing

Fig. 1.50. Schematic diagram of the ISX-B LIFT experiment. The detected fluorescent volume element is about 3 cm long and 3 mm in diameter. Different vertical sections of the discharge can be viewed by translating the optics.

adequate time for repairs, maintenance, and installation of new diagnostics. A goal of four operating days per week was set, with one day reserved for maintenance and installation activities and every morning available for diagnostic set-up or modifications. In addition, two or three shutdowns of the order of three weeks each were anticipated for major installation work. The balance sheet for the year is indicated in Table 1.10.

The major factor contributing to the lost days was underestimation of the time required to complete tasks scheduled during the shutdowns. Those days lost due to minor equipment failures were largely recouped by rescheduling the maintenance days to coincide with the repairs.

In the course of the year, 10379 shots were fired, an average of about 70 per operational day. The distribution of shots among the major program elements is given in Table 1.11.

The operating history for 1979 is indicated in Fig. 1.52, where the number of shots per week is plotted as a function of time. Three incidents resulted in significant losses of operating time. The first occurred on December 8, 1978, when one of the 40-t/A transformers for the TF power supply was struck by lightning. Repairs were not completed until January 10, 1979, due to bad weather, but the adverse effects were minimized because the lightning coincided precisely with the beginning of a planned three-week shutdown. The second incident occurred when a 13.8-kV distribution box in the TF power supply was damaged by arcing on November 1. Repairs were completed by mid-December, but again little operating time was lost because a five-week shutdown was scheduled to begin on November 9. The third incident, at the end of April, involved the failure of the two bellows connecting each of

Fig. 1.51. Absolute iron density at 100 mm above midplane vs time in ISX-B for 0.95-T (x) and 1.15-T (△) toroidal field discharges. Approximately 400 kW of power was delivered from the west beam line for 0.95-T discharges and 1000 kW from the east and west beam lines for 1.15-T discharges.

Table 1.10. ISX-B operational days for 1979

Projected days available (4 days/week)	201
Days lost to scheduled shutdowns (June, November)	31
Days available	170
Days of actual operation	151
Net days lost	19

the beam lines to the tokamak. These components failed within two days of one another, resulting in the loss of a week for repairs and modifications.

Table 1.11. Distribution of tokamak shots by program objectives for 1979

Program	Number of shots	Percent
General plasma experiments, baseline and exploration	3649	35
High-beta program	2506	24
Pellet injection	928	9
CH studies	914	9
Impurity flow reversal	872	8
Ripple studies	474	5
Plasma shaping studies	466	4
Power supply tests, etc.	434	4
Surface physics ^a	136	1

^aDoes not include shots primarily devoted to other programs.

A number of changes were made during the year in an effort to improve the operational reliability; the most significant was a major overhaul of the aging battery banks that comprise the inner, radial, and shaping coil power supplies.

ISX-B neutral ion beam lines and system upgrade

P. H. Edmonds, S. C. Bates, A. Y. Broverman, W. K. Dagenhart

ISX-B beam lines. The major source of plasma heating for the ISX-B high-beta program is neutral beams. Powers of up to 1 MW have been injected into the plasma. This power was supplied by two 60-A ion sources in two separate beam lines, both oriented to produce similar directions of injection. To date all injection has been coinjection (with the conventional ohmic current). The ion sources and associated beam lines were developed by the Plasma Technology Section at ORNL.⁴⁷ Two high-voltage power supplies were included in the original project; they had design capability of up to 100 A at up to 60 kV with a 3% duty cycle. A description of the project is given in previous FED annual reports.^{48,49}

Fig. 1.52. Operating history of ISX-B for 1979.

The beam lines were installed on the ISX-B experiment during 1978, and the west beam line injected 0.5 MW of neutral hydrogen into the tokamak plasma in October 1978. The east beam line was brought into operation in March 1979. At that time the power supplies dedicated to the beam lines were still unfinished, and a single 100-A accel power supply that was time shared with the Plasma Technology Section was used. The interconnection between the two sources consisted of a simple link between the two modulator anodes; surprisingly few problems occurred that were associated with this simple link.⁵⁰ In June, with this system, injected powers of up to 1 MW of hydrogen neutrals were obtained.

In July an experiment to calibrate the efficiency of injected power versus calorimetric power was made. This was done by inserting an instrumented paddle into the tokamak. The results are still under analysis,⁵¹ but the general conclusion was that about 85% of the power deposited on the calorimeter during a conditioning shot was injected into the tokamak during an injection shot. The new 100-A, 60-kV power supply installation was

completed in August; however, numerous problems occurred and the two beam lines did not become available until October. At this time, one power supply was complete; the other still had a weak tap changer (step voltage regulator), which was run with a series resistance for protection. The beam lines were run during October and November and shut down for the upgrade in mid-November.⁵² The defective components in the accel power supply were returned to the manufacturer for repair.

During the summer a prototype data acquisition system for the sources was designed and implemented. The instrumentation uses a microprocessor-controlled CAMAC system to analyze and display the data. A similar system to monitor the beam line parameters and calorimetric measurements is also nearing completion. This will replace the hardwired power monitor previously developed for this system.^{53,54}

Beam line upgrade. In mid-November the two beam lines were removed from the ISX-B enclosure and modified to accept the larger 30-cm, 100-A sources. These sources are identical to

those developed for PDX by the Plasma Technology Section.⁵⁵ This upgrade is described in detail in the 1978 Annual Report.⁵⁶ The basic features of the conversion are the modification of the beam lines to accept the increased size and power of the 30-cm sources, the installation of a new front end with a cryopump (to improve the pumping of the drift duct to reduce re-ionization losses), and a general reduction of about 10% in the overall length.

Liquid helium system. The upgrade beam line will use up to four cryopumps, having a liquid helium consumption (including transfer losses) of over 200 liters per pump per day. At present, this would cost approximately \$1000 per day. Some years ago a project was initiated to install a helium liquifier system to supply liquid helium to the Plasma Technology Section and ISX. The installation of this system is now complete. The ISX portion of this project consists of a 1.7-MPa (250 psi) gaseous storage vessel with a volume of 28 m³, a 30-liter/h helium liquifier, and a screw compressor. The entire system has been checked out, the piping is expected to be completed in January 1980, and it is hoped that this system will considerably reduce the operating costs of the beam lines in 1980.

The ISX-B data system

W. R. Wing, R. D. Burris, D. R. Overbey, K. A. Stewart

Data system development highlights from 1979 are:

- completion of Phase I of MHD/PIN diode data handling project;
- interfacing of the infrared camera;
- installation of 32 channels of data handling to allow phasing out of the PDP 12;
- installation of first fiber-optic CAMAC serial link for in-service testing (installed as part of infrared camera interface system);
- installation of radiometer, ECH, and general spectroscopic data channels;

- interfacing of high-speed data acquisition for monitoring pellet injection;
- arrival (after ~2-year vendor delay) of first 32-bit processor with fully integrated CAMAC interface and software package; and
- wrap-up of bid package and contract award for next three 16-bit computers.

The year 1979 has been one of evolutionary progress, rather than one of revolutionary change.

As is implied by the highlights, progress for the year occurred primarily in two areas: interfacing new diagnostics (and extending the service provided to old ones) and purchasing new hardware. Several quite different activities are implicit in the interfacing of new diagnostics. Not only must data be acquired (in a timely and efficient way), but provision must be made for passing this data through the computer network to central archival storage on the PDP 10. Software must be written to handle the data acquisition, data transfer, and data storage, and finally to make the data recoverable. These functions must be transparent to the diagnostician-user. Problems presented by each new diagnostic in these areas have been dealt with.

A few cases are especially interesting. Phase I of the MHD/PIN diode project involved setting up our first large bank of high-speed CAMAC transient recorders. Four recorders were used, each capable of recording data on eight channels with 8000 words of storage per channel, at data rates of up to 500,000 samples per second per channel. The system was temporarily implemented using the PDP 11/34, which will become the "surface physics" computer but which was available due to the delay in completing the surface physics sample transfer system. The data acquired by this system (256,000 words per shot) were passed directly to the PDP 10 via a special link because it would have badly clogged the standard ISX data link. The system was used to acquire data from high-power injection shots in ISX-B and, in addition, to recover for analysis data from ISX-A that had been stored on

analog data tapes using an FM instrument tape recorder.

In the coming year the system will be expanded (by more than a factor of four) to handle data from the PIN arrays, and the truly massive amounts of data then acquired will have to be preprocessed by the 32-bit computer mentioned below.

The infrared camera data system is interesting in two respects. It is one of the first diagnostics to need a time base that is much longer than the experiment, and it is serving as an in-service test bed for our first experience with a completely optical-fiber-coupled CAMAC serial link. The infrared camera acquires up to 128 single-scan sweeps of its target at times that can both precede an ISX pulse and follow it by up to 30 s. Because of its data acquisition "schedule" the infrared camera becomes essentially asynchronous with ISX, and its data are stored on a disk at the front end computer for separate transmission to the PDP 10. The fiber-optic link (installed now and running continuously for over six months) is of the type on which we will be totally dependent when we start assembling diagnostics of the neutral injector source tables and the new Thomson scattering system.

The first steps were taken during 1979 to automate the control room. The first CAMAC pulse generators/sequencers were tested (although not permanently installed) and 32 channels of slow (1 ms per sample) analog data handling were installed to parallel the PDP 12 and prepare for its eventual phase-out.

Two channels of special very high-speed (2 million samples per second) data acquisition were installed for the pellet injection experiments, and the data were incorporated into the ISX data base.

Nearly standard channels of data were provided for the ECH experiments as additions to the growing numbers of spectroscopy and radiometer channels.

The usefulness of Thomson scattering as an on-line diagnostic was considerably enhanced this year by software provided at the control

room to permit on-line profile analysis during the accumulation of shot sequences.

Finally, orders were placed for the next three 16-bit computers that will arrive in early 1980. After almost a two-year delay, our first 32-bit computer arrived. The delay (caused by slow delivery of a special interface) has forced us into a number of makeshift solutions to data handling. However, we now have the largest, most powerful computer system ever assembled with a completely integrated CAMAC interface and software package.

Developments in electronic instrumentation and techniques

J. W. Pearce, J. M. Madison, G. R. Dyer

The electronics group provided new instrumentation and a significant amount of maintenance for the tokamak operations group, the neutral beam injection group, and various individual diagnostic teams during the past year.

We assisted in strain-gauge tests of the toroidal field coil supports on the tokamak. Several modules were designed and fabricated for the tokamak's plasma position control system. These included a ramp generator with calibrated amplitude and duration adjustments, a triggered linear gate with five gated channels, a summing amplifier with calibrated offset and attenuation, and a signal-conditioning amplifier with selectable time constants, gain, and output polarity. Instrumentation completed in the past year for the tokamak operating system includes an ionization-type vacuum gauge with a 1-ms time response. This fast gauge will be used to follow pressure changes in the beam drift tubes during neutral beam injection.

Considerable work was done to improve neutral beam line calorimetry and timing circuitry. New beam line diagnostic equipment is being developed to upgrade the present calorimeter.⁵⁴ It will be controlled by a microcomputer-based (MIK-11) CAMAC system and will provide an easily calibrated power monitor for all relevant targets and beam dumps in the ISX neutral beam injectors.

Several new instruments for various diagnostics were developed during the year. An interface and display unit for the infrared camera provides a three-axis plot of temperature versus time for a segment of the tokamak limiter from data acquired during each tokamak discharge. Several thermocouple amplifiers were constructed for the coated limiter experiment, and various power supplies for the charge exchange analyzers were designed and built. A digital timer for the ECH experiment was also finished. Work is now in progress on a laser calorimeter for the new Thomson scattering system; the design is essentially complete and circuit layout will begin soon.

A considerable part of the electronics program was devoted to unscheduled maintenance and repairs that were necessary for the operation of the tokamak. The capability to quickly diagnose and repair electronic faults on equipment as diverse as the Thomson scattering diagnostic, the charge exchange analyzer, or the tokamak inner field supply was a major service provided by the group.

1.3.7 Diagnostics

Power losses from the ISX-B tokamak

*C. E. Bush, R. C. Isler, J. M. Madison,
D. R. Overbey*

Several diagnostics are being used to measure the power losses from ISX-B in order to account for the total power input, including the usual ohmic plus the additional auxiliary heating, whether it be neutral beam heating or ECH. Uncollimated pyroelectric detectors are used to measure the total power losses to the walls due to radiation and charge exchange neutrals, and a 2-D array (12 x 6) of collimated pyroelectric detectors provides spatially resolved intensity data from which the emission profiles within the plasma may be deduced. An AGA 680 infrared camera monitors the temperatures of and thus the power losses to the limiters. The radiometer arrays are shown schematically in Fig. 1.53, and the technique and geometry for the infrared camera measurement are illustrated in Fig. 1.54.

Fig. 1.53. The 12-detector top array and the 6-detector side array that provide time and spatially resolved measurements of energy flux on ISX-B. Inverted emission intensity profiles give radial distribution of volume emission.

Both radiometry and spectroscopy show that ISX-B exhibits the same low impurity content as did ISX-A, indicating that on the average ~30% of the input power is lost as impurity radiation to the walls in ohmically heated discharges. The infrared camera measurements show that the remaining ~70% of the input power is deposited at the limiter. With the addition of auxiliary heating or pellet fueling, no excessive radiative or charge exchange losses are observed. However, in the case of neutral beam injection, though the major fraction of the total input power is accounted for, ~30% is not. In a typical high-beta experiment, where ~1 MW of neutral beam power was injected, about 30% of the total power input was accounted for as losses to the outer limiter, where temperatures of $\geq 1100^\circ\text{C}$ were observed. Another 10% was deposited at the outer limiter, and ~30% was lost as impurity radiation to the walls.

The evolutions of limiter temperatures and power loss to the wall, along with other parameters for the high-beta experiment are shown in

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INFRARED CAMERA: USED TO MONITOR ENERGY FLUX TO LIMITERS ON ISX-B

Fig. 1.54. Viewing of limiters through a 10-cm-diam (4-in.) calcium fluoride window using a pair of gold-plated front surface mirrors. The split mirror allows viewing of inner and outer limiters simultaneously when using an 8° lens. The wide angle lens (45°) allows direct viewing of all four limiters.

Fig. 1.55. Intensity and emission profiles for a similar high-beta discharge, as provided by data from the top radiometer array, are given in Fig. 1.56. Approximately 30% of the input power is unaccounted for, and it is speculated that the discrepancy may be due to a combination of factors including, among others, toroidally and poloidally asymmetric losses to the walls and possible errors in beam power calibration.

Time-resolved T_e and Z_{eff} measurements by x-ray energy spectrometry

G. R. Dyer, H. E. Kottler

An x-ray energy analyzer with 10-ms time resolution has been in use as a plasma diagnostic on the Oak Ridge tokamaks for some time.^{57,58} While the basic technique remains essentially the same as described previously,

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Fig. 1.55. Variation of outer and inner limiter temperatures during injection, with $\Delta t \sim 720^\circ\text{C}$ for outer limiter and 300°C for the inner limiter. Lower half of figure shows P_{beam} , P_{rad} , soft x-ray, and MHD amplitude B_{θ} .

the analytical procedure has been expanded and modified in several ways. For example, a three-axis display has been developed to plot x-ray energy spectra as a function of time into a tokamak discharge. This display allows us to make a rapid qualitative inspection of an individual shot for a nonthermal x-ray distribution or for the presence of impurity line radiation. The sample given in Fig. 1.57 shows evidence of iron line radiation at ~ 6.3 keV.

A thorough analysis of the theory of x-ray emission from a plasma in the tokamak geometry has allowed us to develop a model for extracting the peak (or central) value of T_e from a chord integral energy spectrum. This model takes into

Fig. 1.56. Energy flux intensity and volume emission data from the 12-detector array. The volume emission profiles are approximate Abel inversions of the intensity data.

Fig. 1.57. A three-axis plot of $\ln I(E)$ vs E vs time for ISX-B Shot 9090, showing characteristic iron line radiation as well as thermal x-ray spectra.

account the effects of various possible temperature and density profiles and different concentrations of plasma impurities. The computer model incorporates corrections for pulse pileup, checks for nonthermal energy distribution, and calculates errors in T_e due to statistical fluctuations. A plot of chord integral and peak T_e versus time for a discharge with neutral beam injection is shown in Fig. 1.58.

Finally, an investigation of the effects of plasma impurities on x-ray intensity* and the subsequent development of a method for determining the intensity of an experimental spectrum relative to that from pure hydrogen at the same T_e and n_e allowed us to use the x-ray data in conjunction with plasma density information to estimate Z_{eff} throughout a single discharge with neutral beam injection on ISX-B as shown in Fig. 1.59.

* This investigation relied on the transport code RECNO developed by E. C. Crume et al. We thank Crume and D. Arnurius for the special assistance they provided in adapting this code for our use.

Fig. 1.58. A plot of T_e vs time from x-ray data for ISX-B Shot 18558, demonstrating electron heating during neutral beam injection.

Fig. 1.59. A plot of Z_{eff} vs time, derived from x-ray data and plasma density information from the microwave interferometer for shot 18558.

In general, agreement between x-ray data and results from other plasma diagnostics is good. The x-ray technique provides a unique combination of T_e and Z_{eff} information with 10-ms time resolution throughout a single tokamak discharge and should prove to be a very useful on-line diagnostic in future experiments.

Diamagnetic measurements on ISX-B

A. C. England, J. T. Mihalezo, J. ...
G. H. Neilson, J. F. Ellis, W. R.

A system of diamagnetic coils and bucking coils (to subtract the toroidal field variations) was mounted on ISX-B during its construction. Other coils were added to sense the error fields from the outer field coils, the inner field coils, the shaping field coils, and the radial field coils.

These measurements have been extremely difficult to make because of the toroidal field supply. The silicon-controlled rectifier (SCR) 12-phase supply has strong ripple components at 120 Hz, 120 Hz, 60 Hz, and 34 Hz. These frequencies are all in the part of the frequency spectrum where the diagnostic coil is most

sensitive. Although special analog filters were made for these frequencies, the best results to date have used digital notch filters on the diagnostic signal and a digital low pass filter on the reconstructed paramagnetic signal.

The diamagnetic signal has been observed but is very noisy. This is believed to be due to the toroidal field supply. Motion sensors show that there is a very small amount of shaking of the pickup coils, but this does not account for the noise observed.

At low beta, the measurements, averaged over many shots, are consistent with beta values based on the laser profile. However, at high beta, the measurements give beta values almost always lower than the laser profile determined values. Work is continuing to improve the signal quality.

Submillimeter-wave interferometry

D. P. Hutchinson, K. L. Vander Sluis, R. A. Stamps

A new diagnostic system utilizing a submillimeter-wave, phase-modulated polarimeter was used to simultaneously measure the time evolution of the line-averaged electron density and the poloidal-field-induced Faraday rotation in the ISX-B tokamak. A schematic diagram of the system is given in Fig. 1.60. Measured Faraday rotation and electron density profiles obtained along four chords of the plasma column are shown in Fig. 1.61. The increase in the outer channel ($r/a = 0.85$) Faraday signals from 60 to 80 ms was correlated with a current ramp beginning at 50 ms. This increase in poloidal field near the edge is consistent with a skin current caused by the ramp. Also, visible at $t = 100$ ms is a distortion in the current profile due to the injection of 800 kW of neutral beam power beginning at $t = 80$ ms. The line-averaged electron density (Fig. 1.61b) recorded for this shot sequence varied smoothly during the entire discharge. There is a hint of a hollow density profile during the first 50 ms of the discharge.

Fig. 1.61. Measured Faraday rotation (a) and electron density (b) along four chords through the plasma at r/a of 0, 0.28, 0.56, and 0.85 and at nine different times during the plasma discharge.

A. E. Warshaw, J. F. Lyon, D. E. Gray,
D. H. Gray, D. L. Halsted, J. Kim, J. L. Perry,
N. J. Ponte, D. E. Schechter, W. C. Stirling,
A. E. Warshaw, A. A. White, R. B. Weber

A vertical neutral beam line has been installed under ISX-B and is nearly ready for operation. It will be used for: (1) a test of ripple-enhanced injection⁵⁸ in a collaborative program with PPPL using eyeglass-shaped coils on each side of the beam to produce a localized trapping well, as discussed earlier in this section; and (2) a localized (FWHM ≈ 8 cm), high-density neutral target used with scanning charge exchange analyzers and a VUV spectrometer to give local measurements of T_i , n_i , n_0 , and various impurity line intensities.⁵⁹

A general description of the diagnostic neutral beam has been given in a previous annual report.⁶⁰ Figure 1.62 shows the beam line as constructed and its relationship to ISX-B. The main elements of the beam line and related tokamak features, as enumerated in Fig. 1.62, are: (1) a duoPIGatron source with a 15-cm-diam grid; (2) magnetic shielding; (3) a source-to-tank flange; (4) a cylindrical reentrant support for the source, to be replaced

Fig. 1.62. The ISX-B diagnostic beam line.

later by a bellows that allows a 25-cm horizontal translation of the diagnostic beam and hence 2-D profile measurements; (5) a tapered gas cell; (6) a beam line tank that provides a large ballast volume to minimize the pressure rise during the beam pulse; (7) a retractable beam target calorimeter and thermocouple array for

beam power and profile measurements; (8) a translatable defining aperture plate and thermocouple array for beam position and aiming measurements; (9) a 1500-liters/s turbomolecular pump; (10) a VAT valve for isolation from ISX-B; (11) a water-cooled transition duct between the beam line and the ISX-B vacuum

chamber; (12) a support frame for toroidal field coils and ripple coils; (13) toroidal field coils; (14) ripple coils; and (15) the ISX-B vacuum vessel.

On the duopigatron ion source with a 15-cm-diam grid the aperture has been sized down to a 7-cm-diam extraction area with a tantalum insert to provide a smaller diameter beam for the diagnostics studies. The ion source and associated power supplies and modulator are the same as that used for the high-power (15-cm) beam heating studies on ORMAK in 1976. The ion source with the sized-down aperture has been tested on a beam test stand modified to mock up the actual beam line and transition duct geometry. At optimum perveance the source delivers from 4.5 A extracted at 35 keV down to 0.7 A extracted at 10 keV, with essentially the same profile (≈ 8 cm FWHM) at the nominal location of the plasma center at all energies. Higher or lower extracted currents lead to broader profiles. Operation in deuterium gives similar results with extracted currents ranging from 3.2 A at 35 keV down to 0.5 A at 10 keV.

Deuterium displacement by hydrogen neutral beam injection from neutron measurements on ISX-B

J. T. Mihalazo, S. H. Neilson

Thermonuclear production during H^0 injection into D^+ plasma has been determined using 3He proportional counters calibrated absolutely with a ^{252}Cf neutron source. The neutron production rates were not consistent with the ion temperature measured by charge exchange measurements unless it is postulated that the thermal D^+ ion density in the center of the plasma is reduced by H^0 beam injection. The central deuterium density, which was obtained by requiring the neutron data and the charge exchange data to yield the same ion temperature, is plotted in Fig. 1.63 as a function of time for 1.0-MW neutron beam injection. The central deuterium density drops when injection comes on and increases when injection terminates.

Fig. 1.63. Central ion temperature, central deuterium density, and line-averaged electron density as a function of time for ISX-B with 1.0 MW of injection.

Charge exchange measurements on ISX-B

*G. R. Neilson, S. L. Husted, J. F. Lyon,
J. T. Mihalazo, S. D. Scott, R. D. Wall,
R. E. Woreham*

Diagnostic status. Two mass energy gas-stripping neutral particle analyzers are now installed on ISX-B. The momentum energy analyzer (MEA) first used on ISX-A¹ was used on ISX-B² as a fixed-chord, quasi-perpendicular analyzer until mid-1979. At that time, a newly developed and newly calibrated velocity energy module³ (VFA for "velocity filter analyzer") was installed as a scanning perpendicular analyzer, and the MEA was reconfigured as a scanning tangential analyzer.

The MEA has been used to make ion temperature measurements and other measurements inferred from charge exchange in both its old and new orientations. A few of these results will be described in this section. Results from the VFA are not yet available because of the existence of stray magnetic fields and a large background signal due to uv photons from the ISX plasma. Recent analyzer modifications have eliminated the latter problem, so that ion temperature profile measurements will be made as soon as adequate magnetic shielding is provided. A further improvement in the diagnostic capability of both analyzers will result from the installa-

tion of the diagnostic neutral beam⁶⁴ in early 1980.

Contributions of charge exchange measurements to several of the experimental programs of ISX-B are described here. More comprehensive discussions of these studies appear in Sects. 1.3.1 through 1.3.4.

High-beta studies. Central ion temperatures of up to 1.5 keV have been measured with up to 1.0-MW H^0 beams injected into D^+ plasmas. Comparisons between the measurements and results of the usual ion power balance calculations (using measured laser profiles and classical deposition, slowing down, and power transfer processes) confirm that ion confinement is neoclassical²¹ at line-averaged densities greater than $\sim 3.0 \times 10^{13} \text{ cm}^{-3}$, as has been established on the Oak Ridge Tokamak (ORMAK),⁶⁵ PLT,^{1,3} and other injection experiments. Details of one such calculation are diagrammed in Fig. 1.64. This behavior is presumed to extend to lower densities as well, but the uncertainty there is increased due to the greater importance of charge exchange losses compared to heat conduction. The former are difficult to estimate because reliable neutral density n_0 measurements are not available; based on experience on PLT⁶⁶ and on 2-D neutrals calculations,

the n_0 values inferred from absolute measured charge exchange flux are at best local quantities that lead to underestimates of the global charge exchange losses.

Pellet fueling studies. In ISX-B, H^0 pellets are injected into D^+ plasmas. Time histories of the D^0 and H^0 flux, shown in Fig. 1.65 for an ohmic heating example, give conclusive evidence of the appearance and subsequent confinement of pellet ions in the discharge. The postpellet transient is characterized by an initial cooling of the background ions (D^+) followed by a reheating of the mixed H^+-D^+ ion population on a time scale of 20-30 ms to a temperature (300 eV in the example in Fig. 1.65) greater than the pre-pellet value (200 eV). This can be explained in terms of more efficient heating by electrons due to the increased density. Although the H^+ pellet produces about a threefold instantaneous electron density increase, the ratio of the neutral fluxes $D^0:H^0$ is roughly 2:1 and increases later on. Given the history of almost exclusive deuterium operation in ISX-B, this observation is probably a manifestation of wall and limiter recycling in the particle balance.

Ripple studies. One issue to be addressed in field ripple studies in ISX-B is the effect of ripple on fast ion confinement. To investigate this, the ISX-B ripple coils were energized during beam heated discharges and the fast ion charge exchange flux was monitored. Time histories of the flux at different energies indicate a progressive deterioration of fast ion confinement with increasing ripple. The effects on the density and the ion and electron temperatures were somewhat less severe.

ECH studies. Although electron heating was clearly demonstrated in experiments with up to 100 kW of ECH power from the NRL gyrotron, no evidence of ion heating has been seen. At line-averaged densities of $\sim 0.8\text{--}1.0 \times 10^{13} \text{ cm}^{-3}$, deuteron temperatures of ~ 220 eV were measured; no significant change resulted from the 10-ms ECH pulse. The lack of observable heating can be attributed to the fact that, compared to the pulse duration, the electron-ion equipartition time is very long (~ 100 ms) at these low

Fig. 1.64. Details of ion power balance calculation for ISX-B case with density 3.1×10^{13} and injected power 630 kW. Neoclassical heat conduction was invoked; charge exchange losses were based on a neutral density that was chosen to reproduce the measured ion temperature.

Fig. 1.65. Time histories of pellet (H) and background (D) neutral fluxes for pellet injection in ohmic discharge in ISX-B.

densities. On this basis, it is expected that higher densities and longer pulses would result in measureable ion temperature increases.

Microwave interferometers

J. B. Wilgen

Both of the 2-mm microwave interferometers in use on ISX-B have been modified during the past year. In the case of the old ORMAK interferometer, microwave deflectors were placed inside the vacuum vessel to prevent multiply reflecting signals from reaching the receiving antenna and distorting the phase information.

The second interferometer, which is still under development, is of a new design that features a large bandwidth and that is capable of following the rapid phase changes characteristic of pellet injection, 50 fringes in 100 μs . This interferometer does in fact follow the density rise resulting from pellet injection provided that the pellet density is deposited near the plasma edge, in which case regions of high density approaching cutoff density do not develop. This interferometer also benefited

from the use of microwave deflectors inside the vacuum vessel.

The two interferometers view the plasma orthogonally, with the ORMAK interferometer viewing horizontally on the midplane and the new ISX interferometer viewing vertically at the major radius. Hence, the ratio of the interferometer signals is a measure of a density-weighted plasma elongation. Such measurements have usually been in agreement with the shape of the outermost flux surface as inferred from magnetic loop measurements, except during elongation experiments.

1.3.8 Design Studies for Future Programs

The ISX-C program

*T. C. Jernigan, C. Adams, R. L. Johnson,
R. E. Hill, R. A. Dory, H. C. Howe, J. P. Lyon,
B. E. Nelson, J. Sheffield*

The ISX-C tokamak is being proposed to help speed the successful realization of an Engineering Test Facility (ETF) and to improve the viability of the tokamak reactor concept. The

ISX-C program is specifically designed to address several critical ETF physics and technology issues by (1) serving as a test bed for advanced design divertors, (2) demonstrating the ability to maintain clean, high-beta plasmas for long pulses at high power density, and (3) providing a flexible, high-parameter tokamak facility in the mid-1980s for studies related to ETF and tokamak improvement.

Scoping design studies for both the ETF and the U.S. version of INTOR have established a tokamak reference design in which a clean ($Z_{\text{eff}} \leq 1.5$), high-beta (6-7%) plasma would be maintained at high edge power densities ($\geq 60 \text{ W/cm}^2$) for long pulses ($\geq 100 \text{ s}$) by using pellet refueling and an "improved" divertor for impurity control. The two divertor options being considered for ETF are a bundle divertor and an external poloidal divertor.

Studies made of the R&D needs for the U.S.-ETF and INTOR^{67,68} have identified plasma disruption control; plasma preheating, ignition, and burn control scenarios; helium ash removal; long pulse operation of clean, high-beta, high power density plasmas; and plasma reproducibility. Areas in which only marginal data will be provided include: field ripple effects; plasma fueling scenarios; fusion alpha heating scenarios; impurity generation, transport, and burial; divertor unloading and shielding physics; and pellet ablation physics.

The ISX-C tokamak program proposes to address several of the critical ETF physics and technology issues mentioned above. Specifically, ISX-C would:

- test advanced divertor systems at high particle and power fluxes, initially for moderate pulse lengths ($\leq 3 \text{ s}$);
- demonstrate the ability to maintain a clean ($Z_{\text{eff}} < 1.5$), high-beta (3-5%) plasma at high edge power densities ($\sim 15\text{-}30 \text{ W/cm}^2$) for long pulses (10-30 s); and
- provide a flexible, low stress, good access, readily modifiable tokamak facility in the mid-1980s with significant plasma parameters for studying other ETF-related issues

(fueling, ripple, materials, current drive, divertorless operation, long pulse heating, etc.), similar to the role played by ISX-A and ISX-B in the late 1970s and early 1980s. In fulfilling this role the ISX-C program would also contribute to improving the tokamak reactor concept.

An ISX-C design study was carried out to determine the set of minimum tokamak parameters needed to accomplish the objectives stated above. These parameters, along with the corresponding values for ISX-B and ETF, are given in Table 1.12. The primary improvements in ISX-C, which remove the ISX-B limitations that reduce its usefulness, are improved access to accommodate advanced divertor designs, much longer pulse length, and higher plasma current for better confinement of near perpendicular or counterinjected fast ions. The ISX-B program has demonstrated the value of a modest field, modest stress, good access machine and high reliability. On ISX-B it has been possible to make many tests and add several features that were not foreseen when the ISX-A facility was constructed (e.g., ripple injection, nine-toroidal field coil modification, bundle divertor, coated limiter station, elongated vacuum vessel, high-power neutral beam injectors, and ECH). Thus, an important design requirement for ISX-C is that it have very good access to permit relatively easy modification to various components such as the first wall, poloidal system, TF coils, and edge control systems (advanced design divertors).

An elevation view of ISX-C is shown in Fig. 1.66. This design satisfies the minimum parameters given in Table 1.12 and incorporates the design philosophy discussed above. For example, this design features very large access ports, demountable outer legs on the toroidal field coils, and demountable external poloidal coils that permit relatively easy modification or removal of these coils. ISX-C would utilize the ISX-B site and most of its facilities (diagnostics, control areas, upgraded beam lines, and power supplies), necessitating the

Table 1.12 Comparison of tokamaks ISX-B, ISX-C, and ETF

Parameter	ISX-B	ISX-C	ETF
Major radius, R_0 (cm)	93	130	540
Minor radius, a (cm)			
To separatrix	20	40	130
Scrapeoff thickness	7	10	10
Maximum elongation, b/a	1.8	1.6	1.6
Maximum toroidal field (kG)	18	12-24	55
Maximum plasma current (kA)	180	400-800	5600
Total flux (Mb)	0.9 (iron core)	4.5 (air core)	90 (air core)
Input power (MW)	1-3	1.5-7	50
Maximum pulse length (s)	0.25	10-30	>100
Number of TF coils	18	12	12
Toroidal field ripple (%)			
On axis	0.01	0.2	<0.1
At plasma edge, a	0.7	3	1
Impurity control divertor	bundle	bundle or external poloidal	bundle or external poloidal
Maximum access, port size (cm x cm)	22 (diam)	125 x 80	

shutdown of ISX-B in 1982, nine months prior to the start of ISX-C operation in 1983. Upgrades that would be added to the basic design later are long pulse (10-30 s) auxiliary heating at 3-6 MW and a power supply that would double both the toroidal field (to 24 kG) and the plasma current (to 800 kA).

ISX-B bundle divertor construction project

T. C. Jernigan, R. B. Easter, D. H. Gray,
J. P. Lyon, D. B. Montgomery,* B. E. Nelson,
J. A. O'Toole,† E. J. Rappenport,* J. Sheffield,
J. E. Tracy,* W. R. Wing, W. L. Wright,
R. B. Weyer, T. F. Yang,* J. W. Yarbrough

The project to add a bundle divertor to ISX-B began in earnest in 1979. The goal of the project is to add a system capable of testing various bundle divertor configurations under

conditions of high plasma edge power loading. The objectives of the test are to measure the ability of the divertor to extract particles and power from the plasma edge and to shield the plasma from the wall- and limiter-generated impurities. The divertor system (Fig. 1.67) consists of an interchangeable divertor coil module, a power supply for the coils, a target for the diverted plasma, a pumping system to handle the extracted gas load, and structural elements to handle the magnetic forces between the divertor coils and the tokamak coils.

The Plasma Fusion Center at MIT has been selected to design and fabricate the initial divertor coil module. Using the groundwork laid by Culham Laboratory in England and by the Westinghouse Electric Corporation, MIT has designed a low magnetic field ripple divertor coil module with a compact mechanical design. The peak ripple at the plasma center is 1.1%. The coil temperature rises 27°C during a 200-ms pulse at 18 kG.

The remainder of the system is being designed by Union Carbide Corporation-Nuclear Division

*Massachusetts Institute of Technology,
Cambridge, Massachusetts

†Grumman Aerospace Corporation, Bethpage,
New York

Fig. 1.66. Elevation view of ISX-C.

(UCC-ND) Engineering. The vacuum pumping system is a 150,000-liters/s cryopump like the small cryopumps on the neutral beam lines. The water cooling system is a feedback-controlled system designed to keep the coil-inlet-to-coil-outlet water temperature rise below 6°C to minimize thermal stress. A 600-V, 15-kA SCR power supply identical to the one used on the ripple coils is being constructed. Installation of the

divertor system is scheduled to start in November 1980, with initial experiments to begin in April 1981.

1.4 NATIONAL GYROTRON DEVELOPMENT PROGRAM

This program was set up in 1979 as part of a consolidation of the U.S. Fusion Program's gyrotron research and development effort.

Fig. 1.67. Proposed ISX-C divertor system.

Significant progress was made in late 1979 toward the goal of producing 200 kW at 28 GHz for the EBT-S program.

On November 1, 1979, ORNL FED began managing the development portion of the OFE D&T National Gyrotron Program. The ORNL effort consists of managing the work of two subcontractors (Varian and Hughes). The devices to be produced by these subcontractors are vital components of several experiments in the U.S. Fusion Program. These experiments are to be performed on several magnetic fusion devices: EBT-S (see Sect. 1.1), EBT-P (see Sect. 1.2), ISX-B (see Sect. 1.3), Doublet III (D-III), Poloidal Divertor Experiment (PDX), Tandem Mirror Experiment (TMX), and Mirror Fusion Test Facility (MFTF-B). All have on-going or formally proposed experiments that *require* successful gyrotrons.

In 1963 ORNL FED began working with private industry in an effort to develop multikilowatt sources of microwave energy at subcentimeter wavelengths. The devices are for use in heating fusion plasmas by electron cyclotron coupling. In 1976 the development efforts turned to the gyrotron as the most promising means of developing high power at the required wavelengths. The gyrotron, sometimes called an electron cyclotron maser, is a vacuum tube that uses a rf circuit (cavity) with dimensions that are large compared to the wavelength of

the microwave energy. These large circuits allow the gyrotron to operate at power levels that are orders of magnitude greater than that of conventional tubes. The efforts in the mid-1960s produced marginally useful traveling wave tubes capable of 5 kW of cw power at 55-GHz (1/2 cm) the present gyrotron efforts have produced 200 kW of cw power at 28 GHz (1.1 cm). The 200-kW, cw, 60-GHz devices are being designed by our subcontractors, and higher power, higher frequency devices are the subject of research at MIT and Naval Research Laboratory (NRL).

In mid-1979, as a result of increased recognition of the importance to the U.S. Fusion Program of successful gyrotron tubes, all fusion program gyrotron research and development was transferred to the D&T Branch of the Office of Magnetic Fusion Energy (MFE). Shortly before the decision to consolidate gyrotron R&D, ORNL let development contracts to Hughes and Varian to produce 110-GHz, 200-kW, cw gyrotrons for EBT-P. In November 1979, as a result of a change in EBT-P plans, the contractors were instructed to change frequency from 110 GHz to 60 GHz. The 1980 gyrotron program effort has increased to 2.5 times the 1979 levels.

In December a 28-GHz tube (Serial No. 7M) was delivered to EBT-S. This tube was a "triple-miter" output configuration, typical

of all the cw tubes that had been built during the first three years of the program. To increase the probability of this tube being delivered, Varian, at our direction, limited its output to 60 kW of cw power during testing. This power is expected to be suitable for the needs of EBT-S at least through February 1980. In June 1979 initial tests were run on a cw tube (Serial No. 5) with an axisymmetric output configuration. This configuration brings the microwave energy out the end of the electron collector, rather than requiring that the microwave energy traverse three 90° bends inside the gyrotron. As a result of operating experiences, tube No. 5 was modified twice during 1979. It reached 60 kW of average power and 195 kW of peak power during tests in 1979. (In February 1980, No. 5 achieved its 200-kW, cw operating level, a world-record achievement.)

In 1980 we expect to essentially complete the 28-GHz development at Varian, build short pulse test models of the 60-GHz tubes, and start development at one or two higher frequencies. The choice of the higher frequency(s) will be based on the needs of several experimental programs (i.e., D-III, EBT-P, MFTF-B, PDX, and TMX) that are now in various stages of planning and startup.

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2. DIAGNOSTIC DEVELOPMENT

2. DIAGNOSTIC DEVELOPMENT

ABSTRACT. A new linear high-power laser oscillator has been constructed for the submillimeter-wave laser Thomson scattering experiment. Profiles of electron density and plasma current density have been simultaneously measured on the Impurity Study Experiment-B (ISX-B) with a four-channel, combined interferometer-polarimeter operating at a wavelength of ~ 0.4 mm. The measurements of the plasma current density profiles are the first to be made on a tokamak using Faraday rotation of submillimeter-wave-length radiation.

A low-energy neutral particle spectrometer is being constructed to measure the fluxes of hydrogen and of impurities such as carbon, oxygen, and iron from ISX-B. This spectrometer uses a cesium vapor cell to produce negative ions and will be calibrated using known neutral beam fluxes produced by laser photodetachment of negative ions with energies ranging from 10-500 eV.

A neutral beam analyzer for measuring the fractions of the neutral H^0 particles at full energy E (originally H^+), one-half energy $E/2$ (originally H_2^+), and one-third energy $E/3$ (originally H_3^+) has been calibrated.

2.1 MEASUREMENT OF PLASMA ION TEMPERATURE BY THOMSON SCATTERING

A high-power linear laser oscillator has been constructed during the past year for the Thomson scattering experiment. The oscillator has a gain length of 4 m and a diameter of 15 cm. A submillimeter grating is used as the back reflector both to purify the spectral emission and to improve the degree of polarization of the pulsed D_2O laser.

The Thomson scattering receiver, originally designed for an intermediate frequency (i.f.) of 8.4 GHz, has been redesigned for an intermediate frequency of 4.7 GHz. This design change was precipitated by our inability to obtain a low-noise i.f. amplifier at 8.4 GHz. Noise measurements on the new receiver configuration are in progress.

Studies are under way to determine the feasibility of using continuous-wave (cw), submillimeter-wavelength Thomson scattering to monitor the injected hot ion population in ISX-B by observing density fluctuations and pulsed Thomson scattering from nonthermal particles.

2.2 MEASUREMENT OF TOKAMAK CURRENT PROFILES BY SUBMILLIMETER-WAVE LASER FARADAY ROTATION

A four-channel heterodyne interferometer-polarimeter has been used to measure the

electron density and current density profiles in the ISX-B tokamak.¹ A schematic diagram of a single channel of the diagnostic is shown in Fig. 2.1. The system consists of a pair of submillimeter-wave (smw) lasers operating on the 0.381-mm line of deuterated formic acid (DCOOD). Each smw laser is optically pumped by a 35-W carbon dioxide laser joined with the smw laser in a single invar structure. The combined interferometer-polarimeter is constructed by adding a polarization (Faraday) modulator and analyzer to a dual-laser, phase-modulated interferometer. A Putley detector is used for both electron density and plasma current measurements.

During the first phase of diagnostic operation, density measurements were taken for experiments in which density increased rapidly (i.e., during solid H_2 pellet injection) or became too high for reliable operation of the 2-mm interferometer. An example of rapid density rise during pellet injection is shown in Fig. 2.2. Although the interferometer was designed for four channels, only two were used initially because of the unavailability of detectors. A detailed description of the interferometer has been given elsewhere.²

As the system was improved, it became possible to measure the combined electron density and Faraday rotation signals simultaneously. Studies were carried out at four

Fig. 2.1 A single channel of the four-channel submillimeter-wave laser interferometer-polarimeter.

Fig. 2.2 Line-averaged electron density vs time on ISX-B. The data were obtained using two channels of the submillimeter-wave laser interferometer. The electron density increases rapidly with injection of a solid H_2 pellet.

channels that corresponded to different values of the plasma minor radius r . Figure 2.3 shows the electron density, Faraday rotation signals, and plasma current versus time for a series of neutral-beam-heated discharges on

ISX-B. The Faraday rotation signal, which is proportional to the product of electron density and plasma-current-induced poloidal field, resembles the electron density variation during periods when the plasma current is relatively constant. In Fig. 2.4, the Faraday signal and electron density are plotted against radius for successive times during the discharge. The Faraday signal is seen to respond to a ramp in the total plasma current early in the discharge (beginning at $t \approx 40 \text{ ms}$) and to neutral beam injection later in the experiment (at $t \approx 80 \text{ ms}$). Since the electron density signal shows a relatively constant and smooth profile during the entire discharge, the changes in the Faraday signal can be attributed to fluctuations in the plasma current density.

Future plans for the diagnostic call for the construction of an eight-channel vertical interferometer-Faraday system and the development of a five-channel horizontal system for two-dimensional mapping of the electron and current density profiles.

Fig. 2.3 Temporal variations of (a) line-averaged electron density, (b) poloidal-field-induced Faraday rotation, and (c) plasma current in ISX-B discharge. The channel positions are defined as: channel 1, 0 r/a ; channel 2, 0.28 r/a ; channel 3, 0.56 r/a ; channel 4, 0.85 r/a .

2.3 LOW-ENERGY NEUTRAL PARTICLE SPECTROMETER AND CALIBRATION SYSTEM

Analysis of the flux of neutral particles (including hydrogen and impurities such as oxygen, carbon, and iron) from a tokamak yields valuable information concerning plasma parameters without disturbing the environment.³ To extend these measurements to the extreme low-energy range (10-500 eV), we are presently constructing for installation on ISX-B a four-channel neutral particle spectrometer that uses a cesium heat pipe as a charge exchange cell.⁴ Unlike conventional gas stripping cells, which produce positive ions, the cesium cell produces negative ions by electron attach-

Fig. 2.4 Plots of (a) Faraday rotation and (b) line-averaged electron density vs radius at successive times during an ISX-B discharge.

ment, providing a far greater conversion efficiency at low energies. A low-energy accelerator employing a Colutron ion source and a modified Van Zyl deceleration section⁵ has been constructed to serve as a calibration facility in the range of 10-2000 eV. Calibration of the spectrometer using a well-defined atomic hydrogen beam - produced by photodetachment of low-energy H^- ions by a neodymium laser⁶ - will allow us to make an absolute determination of the low-energy hydrogen flux. Further measurements will include mass resolution to give information on the neutral impurity fluxes of oxygen, carbon, and iron.

2.4 NEUTRAL BEAM ANALYZER

One of the difficulties encountered in H^0 neutral beam development has been the precise measurement of the fractional beam components that arise from neutralization of H^+ , H_2^+ , and H_3^+ . A neutral particle analyzer has been designed, fabricated, and calibrated to measure the H^0 components in the energy range of 10-50 keV. The analyzer comprises a nitrogen gas stripping cell, a 45° -parabolic-electrostatic energy analyzer, and three Faraday cups positioned to measure the three beam components simultaneously.

When taking quantitative measurements on intense beams, two problems are encountered.

1. The intense beam impinging on the target increases the local pressure, resulting in a change in the charge state of the sampling beam. (A small sampling beam is provided by an aperture in the target plate.)
2. The H^0 formed from H_2^+ and H_3^+ has an energy dispersion that results from the conversion of potential energy in the center-of-mass frame of reference to kinetic energy in the laboratory frame.

To overcome the first problem, the analyzer was calibrated at a target thickness of $2.2 \times 10^{-5} \text{ cm}^{-2}$ for the target within the nitrogen gas cell. This was sufficient to ensure charge equilibration at all energies less than 50 keV. Thus, the measured conversion

efficiency is independent of the charge state of the incident hydrogen beam. The energy dispersion problem was eliminated by making the analyzer resolution ($\Delta E/E$) 0.08, which ensured collection of all particles. For H^0 formed from 50-keV H^+ , H_2^+ , and H_3^+ beams, the analyzer conversion efficiency was 0.37, 0.22, and 0.14 for the E, E/2, and E/3 components, respectively. From previous measurements of equilibrium fractions, the expected conversion efficiency was 0.60, 0.42, and 0.30. Apparently, collisional scattering in the stripping cell results in a marked decrease in the measured conversion efficiency.

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3. PLASMA THEORY

R. A. Gory, Section Head
C. L. Hedrick, Associate Section Head
D. J. Sigmar, Associate Section Head

D. E. Amurios ¹	L. E. Deleanu ²	M. A. Iskra ^{2,3}	C. E. Parker
H. Ashida ³	J. Denavit ^{2,4}	E. F. Jaeger	Y.-K. M. Peng ^{2,5}
S. E. Attenberger ⁶	T. J. Dolan ^{2,6}	K. A. Jeffrey ^{2,7}	J. W. Reynolds ^{2,8}
M. A. Amini ⁹	O. C. Eldridge ^{2,9}	C. O. Kemper ²	J. A. Rome
D. B. Batchelor	G. A. Emmert ^{2,6}	A. H. Kritiz ^{2,6}	K. E. Rothe ²
C. D. Beasley, Jr.	R. H. Fowler ²	L. I. Lao	J. Smith ^{2,5}
J. Bell ²	J. E. Francis, Jr.	D. K. Lee ²	M. Soler ⁸
G. T. Blair ²	J. P. Freidberg ^{2,6}	S. J. Lynch ²	D. A. Spong
S. K. Borowski ^{2,5}	P. W. Gaffney ²	V. E. Lynch ²	W. M. Stacey ^{2,2}
T. Breaszeale ^{2,6}	R. C. Goldfinger ²	T. S. M. Sengill	J. S. Stanton ²
R. D. Burris ²	D. E. Greenwood ²	J. E. McCune ^{2,6}	C. R. Stewart, Jr.
J. D. Callen ⁷	C. E. Hammons	J. R. McNally, Jr.	K. A. Stewart
B. Carreras ^{2,7}	H. R. Hicks ²	H. K. Meier	D. J. Strickler ²
P. J. Catto ^{2,2}	S. P. Hirshman ^{2,7}	A. T. Mense ^{2,2}	J. S. Tolliver ²
L. A. Charlton ²	L. M. Hively ^{2,3}	K. Molvig ^{2,6}	K. T. Tsang
D. N. Clark ²	J. T. Hogan	J. K. Munro, Jr. ²	T. C. Tucker ²
W. A. Cooper ²	J. A. Holmes ²	D. B. Nelson ^{2,2}	N. A. Uckan
E. C. Crume, Jr.	W. A. Houlberg	D. R. Overbey	W. I. van Rij ²
J. N. Davidson ^{2,2}	H. C. Howe	L. W. Owen ²	H. Weitzner ^{2,6}
	J. C. Whitson ²	R. M. Wieland ²	

1. Also, Adjunct Professor Nuclear Engineering, Massachusetts Institute of Technology, Cambridge, Massachusetts
2. Computer Sciences Division
3. Visitor from Electrotechnical Laboratory, Tokyo, Japan
4. Visitor from JAERI, Tokai, Japan
5. Student, University of Michigan, Ann Arbor, Michigan
6. Y-12 Maintenance Division
7. Former Section Head; now Consultant, University of Wisconsin, M. Jison, Wisconsin
8. Visitor from Spanish Nuclear Energy Commission, Madrid, Spain
9. Present address: Institute for Advanced Study, Princeton, New Jersey
10. SAI, Boulder, Colorado
11. Consultant, Georgia Institute of Technology, Atlanta, Georgia
12. Northwestern University, Evanston, Illinois
13. Oak Ridge Associated Universities Faculty Research Participant, University of Missouri-Rolla, Missouri
14. Consultant, University of Tennessee, Knoxville; part-time with Experimental Confinement Section
15. University of Wisconsin, Madison, Wisconsin
16. Consultant, Massachusetts Institute of Technology, Cambridge, Massachusetts
17. ORNL Wigner Fellow
18. Student, University of Illinois, Champaign/Urbana, Illinois
19. EBT Research
20. Hunter College, New York, New York
21. Present address: Subcommittee on Energy Research and Production, House of Representatives, Washington, DC
22. Present address: Office of Fusion Energy, Department of Energy, Washington, DC
23. On leave to ETF Design Center
24. Instrumentation and Controls Division
25. Consultant, University of Tennessee, Knoxville, Tennessee
26. New York University, New York, New York

3. PLASMA THEORY

ABSTRACT. Extended summaries of the ELMO Bumpy Torus (EBT) theory, tokamak theory, and computing support efforts are given in Sects. 3.1, 3.2, and 3.3. Important new insights include: (1) an understanding of the role of the neutral particle boundary condition in determining thermal stability of low-collisionality transport studies in EBT, (2) formulation of a model for predicting energy loss by the EBT hot electron rings, (3) an awareness that the stability in EBT of the coupled plasma and ring system remains an area requiring improved analysis, (4) enhancement of the resistive magnetohydrodynamic (MHD) tokamak stability codes to increase the realism of the treatment for high-temperature discharges and for finite aspect ratio and beta parameters, (5) interpretation of enhanced tokamak electron thermal losses as a consequence of stochastic magnetic fields calculated with a strong turbulence theory of drift waves in sheared magnetic fields, (6) development of codes for analyzing and minimizing losses of energetic particles caused by field ripple and bundle divertors, and (7) identification of the conflicting indications from ideal MHD stability requirements and long-pulse transport phenomena on the one hand, and of the observed tendency of tokamaks to provide best confinement for rather peaked profiles on the other.

Major sets of theoretical studies were completed during 1979 in support of the proposal to build a new bumpy torus device, EBT-P, and in preparation for a proposal to revamp the Impurity Study Experiment (ISX) device into a divertor studies test bed called ISX-C.

3.1 EBT THEORY

C. L. Hedrick, Jr.

Radially resolved neoclassical transport calculations for the EBT have been the primary focus for comparison to experiment. A sequence of advances has occurred in this area in the past year. Earlier recognition that a small group of resonant ions contributes most to ion neoclassical transport losses made it possible to obtain ambipolar electric field solutions for the low electron collisionality observed experimentally; an artificial stabilization mechanism was still required, however, to prevent thermal runaway. Changing the neutral hydrogen boundary condition to more closely model the experiment allowed removal of the artificial stabilization terms so that thermally stable, purely neoclassical solutions to the radial transport equations became possible at low electron collisionality. The effects of magnetic geometry have now been included in the transport equations (flux surface averaging) as well as in calculations of the transport coefficients. Initial studies indicate that these geometric improvements only modestly change the gross particle and energy balance,

but do affect details of core plasma density and temperature profiles in the region occupied by the hot electron rings.

The transport coefficient calculations are based upon plateau transport coefficients for the ions in the EBT. Variational corrections to these coefficients were calculated during 1979 so that they could be extended more accurately to lower ion collisionality. The effects on thermal stability produced by changes in the neutral hydrogen boundary condition heightened interest in the interaction of the plasma and the wall. Some new techniques for evaluating the long-term behavior of the wall interior were developed for the EBT and should also be useful for longer-pulse tokamak experiments planned for the future.

Electron cyclotron heating (ECH) is essential to the production of the hot electron rings in the EBT. The diversity of investigation necessary to understand this topic was particularly evident in 1979 activities. For example, at the fundamental level, a new eikonal theory has been developed that permits ray tracing near cyclotron resonances. This theory is applicable for ion cyclotron resonance heating (ICRH) as well as for ECH and can be equally

well applied to EBT or tokamak devices. At the more device-specific level are models that can be used to calculate heat deposition in the core and cold edge plasmas in EBT-I and EBT-P. These models for power deposition were used in the EBT-P design study and influenced the basic magnetic design.

The hot electron rings are absolutely essential to the EBT. The principal loss mechanisms for the hot electron rings are Coulomb drag and synchrotron radiation. Power balance models for the rings have been compared with the experimental results from EBT-I (and from earlier mirror experiments, such as ELMO) and have been applied to EBT-P as well as to an EBT reactor. At present, the manner in which the volume scales is the element with the most uncertainty. Volume scaling in turn depends upon interactions of the hot electron rings with the toroidal core plasma and ECH microwave fields. Tractable models for the interaction of the ring and the core have been created, and initial numerical studies have been conducted. These have pointed out deficiencies in present theories of microwave heating of hot electron plasmas, and some promising new ideas have been identified for treating second-harmonic heating.

The sensitive and critical elements of physics and technology in the program leading to the EBT reactor are identified by the advanced system study and are used to assess theoretical and experimental priorities. In addition to the physics discussed in this section, an interesting new development is a technique for microwave coupling in the EBT reactor that is highly efficient and that reduces the thermal burden on the cryopumping system.

The sections that follow present the diverse activities of the Plasma Theory Section in its EBT work. In many instances, this presentation is in the form of republished abstracts from the literature. In such cases, reference citations direct the reader to the complete work in question for detailed information.

3.1.1 Field Design, Equilibria, and Orbits

Abstract of Vacuum field confinement in the EBT-P preliminary conceptual design¹

L. W. Owen; C. L. Hedrick, Jr.

The logic of the magnetics design resulting from the Oak Ridge National Laboratory (ORNL) EBT-P Phase I project definition study is presented. This preliminary conceptual design has evolved from the earlier EBT-II design² and reflects further optimization of plasma confinement based on single-particle drift orbit calculations and theoretical scaling laws. The basic design has a major radius of 5.6 m and 36 NbTi superconducting coils to provide the required resonant magnetic field of 3.9 T for electron cyclotron heating with microwaves at 110 GHz. It is possible to modify the aspect ratio and mirror ratio with aspect ratio enhancement (ARE), with trim coils, or with both without changing the dimensions of the basic device; effects on single-particle confinement of including these coils in addition to the toroidal mirror coils are examined. The design presented here has good single-particle confinement, satisfies proof-of-principle requirements, has a high degree of flexibility for plasma physics experiments and for later upgrade to a larger device with ARE coils, and represents a significant test of the technology that will be required in an EBT reactor magnetics system.

Magnetics system of the EBT-P proposed reference design

L. W. Owen; C. L. Hedrick, Jr.

The magnetics system of the proposed reference design for the EBT-P test device³ resulted from the extensive collaboration of ORNL, EBASCO, Grumman, McDonnell-Douglas, and Westinghouse during Phase I of the EBT-P project definition study. The reference design uses NbTi superconducting mirror coils mounted in a dewar and surrounded by tungsten shielding. The device has 36 sectors, a major

radius of 4.5 m, and mirror ratio m of 2.2. Single-particle drift orbits in the vacuum magnetic field, the effects of wedging the coils, and the effects of adding ARE coils have been examined and these conclusions have been drawn:

1. A simple mirror coil set appears to be the logical design for EBT-P.
2. Wedging, when subject to consistent constraints, does not significantly improve particle orbits, though modest wedging is not ruled out.
3. Additional ARE coils improve particle orbits and increase the effective aspect ratio, but can be operated in the enhancement mode only at the lower field corresponding to 60-GHz operation.

On the average, the magnetics system for the EBT-P proposed reference design is slightly less conservative at the highest field (frequency of ~ 90 GHz) than are those of previous designs. This is justified by recent advances in the state of the art. As a result, vacuum field confinement in the basic device (without ARE) is excellent, and at the same time, the design offers maximum flexibility for improvements and upgrades.

Abstract of Hot electron ring equilibria and effects on EBT transport⁴

L. W. Owen; E. F. Jaeger; C. L. Hedrick, Jr.

Averaging the EBT transport equations along magnetic field lines introduces geometric form factors into the one-dimensional (1-D) equations. These form factors are modified by changes in the magnetic field produced by the hot electron rings. The sensitivity of these form factors to the radial thickness and position, axial length, beta, and anisotropic pressure model of the annuli is determined by calculating a large number of two-dimensional (2-D) tensor pressure equilibria. The results show that for realistic annulus anisotropy, the behavior of the form factors is dominated

by mod 8 in the midplane. Field line averaging is seen to have a modest (10-20%) effect on gross particle and power balance and to affect details of the density and temperature profiles in the region occupied by the rings. For example, the geometric form factors introduce a term proportional to midplane values of dB/dr into the 1-D particle and heat flux equations. The sense of this type of additional term is to retard losses at the outer edge of the core plasma where dB/dr is positive and to decrease gradients of density and temperature at smaller radii where dB/dr is negative.

3.1.2 Transport

Abstract of Transport calculations for EBT-P design studies⁵

E. F. Jaeger; C. L. Hedrick, Jr.

Transport calculations for various EBT-P designs have been made from a 1-D solution to the magnetic field line-averaged transport equations in combination with a finite-beta magnetic equilibrium calculation in 2-D mirror geometry. In calculating the magnetic field, the hot electron annulus pressure is assumed isotropic and peaked off-axis, whereas the toroidal plasma pressure is neglected because of low toroidal beta. For a 36-sector, 60-GHz EBT-P device with major radius $R = 4.5$ m, the annulus is assumed to exist at a radius $a = 20$ cm in the midplane with a half-width of about 4 cm. With 500 kW of microwave power absorbed by the toroidal plasma electrons, results show $\bar{n}r_E \sim 0.3-0.6 \times 10^{12} \text{ cm}^{-3} \text{ s}$ and $T_i \sim 0.5-0.6 \text{ keV}$ for electron collisionalities $\nu/\Omega_{0,e} \sim 0.3-0.1$. With 400 kW of microwave power and 100 kW of direct ion heating, $\bar{n}r_E \sim 0.5-0.9 \times 10^{12} \text{ cm}^{-3} \text{ s}$ and $T_i \sim 0.8-1.1 \text{ keV}$ for the same electron collisionality range. Densities are in the range $n \sim 1-1.3 \times 10^{13} \text{ cm}^{-3}$, which is approximately one-quarter of the cutoff density.

Abstract of Radial transport in ELMO Bumpy Torus with constant edge neutral flux⁶

E. F. Jaeger; C. L. Hedrick, Jr.; K. B. Am^{*}

One-dimensional radial transport equations for the EBT are solved numerically assuming a constant flux of cold neutrals at the plasma edge. With this boundary condition, thermal stability of numerical solutions is achieved for the first time in the collisionless electron regime using a fully classical transport model. Results show a parametric dependence on edge neutral pressure in the collisionless regime that is similar to that observed in experiments.

Abstract of Radial transport in the ELMO Bumpy Torus in collisionless electron regimes⁷

E. F. Jaeger; C. L. Hedrick, Jr.; D. A. Spong

One important area of disagreement between radial transport theory and the EBT experiment has been the degree of collisionality of the toroidal plasma electrons. Experiment shows relatively warm electrons ($kT_e \approx 300-600$ eV) and collisionless scaling, that is, energy confinement increasing with temperature. But results of early 1-D neoclassical transport models with electric fields pointing radially inward are limited to relatively cool electrons ($kT_e \sim 100-200$ eV) and collisional scaling. These early results are extended to include lowest-order effects of ion diffusion in regions where poloidal drift frequencies are small. The effects of direct, or nondiffusive, losses in such regions are neglected along with the effects of finite radial electric fields on electron transport coefficients and of self-consistent poloidal electric fields on ion transport coefficients. Results show that solutions in the collisionless electron regime do exist. Furthermore, when the effects of finite electron ring beta on magnetic fields near the plasma edge are included, these solutions occur at power levels consistent with experiment.

^{*}McDonnell-Douglas Research Laboratories, St. Louis, Missouri

Abstract of Variational corrections to EBT neoclassical ion plateau transport⁸

D. A. Spong; C. L. Hedrick, Jr.

Low collisionality corrections to EBT ion plateau transport coefficients are calculated with a variational method. Using a bounce-averaged drift kinetic equation and a drift orbit model appropriate to collisionless EBT ion transport, a solution is obtained that is accurate to first order in $\epsilon(\Omega_0/\nu_0)^{2/3}$ (ϵ = inverse aspect ratio, ν_0 = 90° deflection frequency, Ω_0 = average poloidal gradient B drift frequency). This is substituted into a variational principle (valid over both the plateau and banana regimes) from which particle and energy fluxes are obtained. The sensitivity of the transport rates with respect to variations in the radial and poloidal components of the ambipolar electric field and with respect to variations in the poloidal drift frequency scale length is examined. The transition from plateau to crescent orbit scaling occurs at approximately $\nu_0/\Omega_0 = \epsilon^{3/2}$.

Lorentz model collision operator calculations of EBT transport coefficients

D. A. Spong, J. S. Tolliver

During 1979, numerical studies were made of EBT transport rates for electrons and ions using a Lorentz model collision term and an orbital model that truncated the step size in resonant regions to the width of a separatrix banana orbit. Such calculations allowed inclusion of both resonant and nonresonant components in ion and electron coefficients for all collisionality regimes. The effect of bounce-averaging the collision term for ions was examined, and the aspect ratio dependence of low collisionality ion rates was studied. Ion rates were observed to scale as A^{-2} (A = aspect ratio) in the collisional and plateau regimes and as A^{-1} in the banana regime. For $-3 \leq e\phi/kT < 0$, the ion transport coefficients indicated peaks in the collisional limit ($\nu/\Omega_0 \geq 1$), which then dropped into plateau scaling as ν/Ω_0 was reduced ($\nu/\Omega_0 \leq 1$); at still lower collisionalities ($\nu/\Omega_0 \leq A^{-3/2}$),

a smooth transition to bananalike (proportional to v) scaling was obtained. At $e\phi/kT \leq -4$, a collisionality scaling similar to the large-field limit was observed. The main effect on ion rates of bounce-averaging the collision term was to sharpen the peak in the collisional ($v/\Omega_0 \geq 1$) range and move it to somewhat lower collisionality. For electrons, transport coefficients using this model were qualitatively similar to those obtained using the Landau operator. It was observed that ion transport rates decreased and electron rates increased when E_x (poloidal component of ambipolar field) was introduced such that mod B and ϕ surfaces were aligned. The electron rates were found to be relatively more sensitive to this parameter than were low-collisionality ion rates. It was only analytically tractable to bounce-average the collision term in the deeply trapped ($v_{\parallel} \ll v_{\perp}$) limit. Such an ordering is appropriate to resonant ion transport, but has resulted in negative transport coefficients when applied to electrons. Apparently, a bounce-averaged collision operator that is valid over a greater range of pitch angles is required for electron transport.

Abstract of A numerical method for nonlinear Volterra equations with singular kernels⁹

T. J. Dolan

A change of variables is used to remove the singularity from a Volterra integral equation of the second kind, transforming it into a nonsingular Fredholm equation of the second kind. Numerical quadrature is done for the resulting nonuniform mesh by fitting parabolas to three points at a time. The method is applied to a test equation, and the resulting errors are computed. The method allows use of a variable time step and is fairly simple to program.

Abstract of Gas emission from chamber walls¹⁰

T. J. Dolan

The neutral gas concentration in the chamber wall of a plasma experiment is described by a

1-D diffusion equation in the wall with a nonlinear boundary condition relating to gas emission rates. The equation is transformed to an integral equation, which is solved analytically for short and long times as well as numerically. Chamber pressure rise during and after plasma decay is compared with experimental data from EBT-I.

3.1.3 Electron Cyclotron Heating

Abstract of Electron cyclotron heating in EBT-PI¹¹

D. B. Batchelor

In scaling the EBT concept to a proof-of-principle device, the plasma parameters change so as to modify the heating processes present in EBT-I/S. Increasing density, temperature, scale lengths, and microwave frequency all tend to increase local cyclotron absorption rates. In particular, assuming $n_e \approx 1.5 \times 10^{13}$, $T_e \approx 2.5$ keV, $\omega/2\pi = 60$ GHz, and a plasma radius at the midplane of 20 cm, one obtains fractional absorption rates of 50-80% at the fundamental resonance for ordinary mode waves launched near the midplane. This is advantageous since the ordinary mode is preferentially absorbed into the hot, dense core plasma rather than into the surface plasma. In addition, direct absorption of the ordinary mode reduces the production of extraordinary mode energy in the high-field region, which tends to be equally partitioned between core and surface plasma. Ordinary mode absorption at the second-harmonic resonance is still small (fractional absorption of $\leq 3\%$).

Because of the increased annulus density and temperature, the extraordinary mode is quite heavily damped at the second-harmonic resonance. The annulus should be virtually opaque to extraordinary mode radiation propagating perpendicular to the field lines.

Abstract of An eikonal expansion of the Vlasov-Maxwell equations valid near cyclotron resonance¹²

H. Weitzner, D. B. Batchelor

In the usual formulations of geometrical optics, the physics of the medium enters the

equations through a conductivity tensor operator $\underline{\sigma}$. An essential assumption in the subsequent expansion is that the magnitude of $\underline{\sigma}^H$, the Hermitian part of $\underline{\sigma}$, is much smaller than $\underline{\sigma}^A$, the anti-Hermitian part. In a finite-temperature plasma with $\omega_{pe} \sim |\Omega_e|$, this condition is always violated sufficiently close to cyclotron resonance, even though in many cases the waves are weakly damped and k is slowly varying. Simultaneously expanding the Vlasov equation and Maxwell equations and taking explicit account of the relative magnitude of the electric field components in the ordering scheme yields a formalism in terms of real rays, real eikonal function, and slowly varying amplitude that is valid at cyclotron resonance. It is assumed that $\omega_{pe} \sim |\Omega_e| \sim \omega$, that each is large, that $\omega \approx |\Omega_e|$, and that $v_e k / \omega$ is small. It is shown that when the waves are weakly damped at cyclotron resonance, the ray trajectories are to leading order exactly those of cold plasma theory. A Poynting theorem that can be put into the standard form is obtained by taking the expansion to first order. Care must be taken, however, to include finite temperature effects in calculating the zero-order electric field. The theory is also applied to ion cyclotron resonance.

Abstract of Ray tracing near the electron cyclotron frequency with application to EBT¹³

D. B. Batchelor, R. C. Goldfinger, H. Weitzner

The general problem of ray tracing in the electron cyclotron range of frequencies is addressed with the view of applying geometrical optics techniques to microwave heating in EBT. It is shown that the usual formulation of geometrical optics breaks down at cyclotron resonance because of the large anti-Hermitian part of the dispersion tensor. A consistent eikonal expansion is presented that is valid near cyclotron resonance for cases in which the waves are weakly damped. The geometrical optics code RAYS, which was developed for the EBT study, is described. Ray tracing in

plane-stratified plasma geometries is discussed and examples are given of the various types of ray paths that occur in plasmas stratified perpendicular to the magnetic field and along magnetic field lines. The application of ray tracing techniques to EBT is discussed, and typical results are presented of rays traced in a bumpy cylinder magnetic field configuration with plasma parameters characteristic of EBT-I.

Abstract of A theoretical study of electron cyclotron heating in ELMO Bumpy Torus¹⁴

D. B. Batchelor, R. C. Goldfinger

The basic features of microwave propagation and electron cyclotron damping in EBT devices are investigated to provide an understanding of the heating of the toroidal core plasma component. The characteristics of microwave propagation in EBT geometry, where there is strong variation of $|B|$ along field lines, are described and contrasted with the electron cyclotron heating of tokamaks. Linear cyclotron damping calculations are presented for plasmas having parameters typical of EBT-I and parameters projected for EBT-II. It is shown that the extraordinary mode is completely absorbed near the fundamental cyclotron resonance when propagating from the high-magnetic-field side (mirror throat). For EBT-I parameters, heating by the ordinary mode at the fundamental cyclotron resonance and by both modes at the second-harmonic resonance is negligible. For projected EBT-II parameters, the ordinary mode is heavily damped at the fundamental and the extraordinary mode is heavily damped at the second harmonic. Since the right-hand cutoff prevents injected extraordinary mode energy from propagating to the cyclotron resonance, a mechanism must exist outside of geometrical optics by which the extraordinary mode reaches resonance. It is argued that this mechanism is the conversion of ordinary mode energy to extraordinary mode energy in the high-field region upon wall reflection.

Abstract of Budden tunneling in parallel stratified plasmas¹⁵

D. B. Batchelor

The process of Budden tunneling of obliquely propagating extraordinary mode is investigated in plasmas whose parameters vary along the magnetic field (parallel stratification). The wave tunnels through the evanescent region separating the right-hand cutoff layer from the electron cyclotron resonance. Coupled-mode equations describing both ordinary and extraordinary waves are derived for arbitrary angle of incidence with respect to the magnetic field. Under appropriate conditions [n_x and $(d \ln B)/dx$ not too large] the coupling can be ignored, and the usual Whittaker equation is obtained for a linear magnetic field profile. It is shown that deviation from strictly parallel propagation ($n_x \neq 0$) has a very small effect on tunneling for a wide range of angles of incidence. The analytical results are verified by numerical integration of the field equations. The theory is applied to the propagation of extraordinary mode waves in the surface plasma of the EBT devices. Fractions of incident power absorbed and transmitted to the high-field region in the range of 2-15% occur for a broad spectrum of n_x .

Abstract of Conversion between cold plasma modes in an inhomogeneous plasma¹⁶

H. Weitzner, D. B. Batchelor

For a particular angle of incidence, it is possible for an ordinary mode electromagnetic wave incident on an inhomogeneous plasma slab to be totally converted to an extraordinary mode. A previous analysis of this process has been extended to permit an arbitrary angle of incidence and to include magnetic field gradients. Although this process has recently been proposed as a method by which toroidal plasmas might be electron cyclotron-heated from the outside (the low-magnetic-field region), it was found that for plasma parameters characteristic of modern tokamaks, significant mode conversion occurs only for waves incident

within a very narrow cone ($\delta\theta \sim 1^\circ$) about the critical angle. Also addressed is the question of the location at which the conversion from ordinary to extraordinary mode occurs.

3.1.4 Ring Physics

Abstract of A simple annulus power balance in EBT experiments¹⁷

S. K. Borowski, M. A. Uckan, E. F. Jaeger, C. Harris*

An essential feature of the EBT concept is the presence of a relativistic electron annulus in each of the toroidal mirror sectors. These high-beta annuli are formed and sustained by microwave heating and are of sufficient density and temperature that diamagnetic currents produce the necessary minimum in the magnetic field required for magnetohydrodynamic (MHD) stability of the toroidal core plasma. Since electron rings play an important role in confinement characteristics and performance of EBT, the tradeoff between the quality of the confinement afforded by the rings and the power required to sustain them represents an important consideration in a fusion reactor. A power balance for the rings that includes energy transfer between the hot annulus electrons and the core electrons via Coulomb collisions (drag cooling), in addition to radiation (synchrotron and bremsstrahlung) and pitch angle scattering of the ring electrons, indicates that drag dominates the annulus energy balance in EBT-I. Drag cooling of the relativistic annulus electrons on the toroidal core plasma appears to provide a reasonable explanation for the decrease in the annulus electron temperature in going from ELMO to EBT-I. Theoretical estimates of the microwave power required to sustain the annulus are found to be within a factor of two of the experimentally determined value. Scaling projections for both EBT-I and EBT-S are shown, enabling one to examine the sensitivity

*University of Michigan, Ann Arbor, Michigan

of the annulus electron temperature as a function of core plasma density for various microwave power levels. The results indicate that a range of annular beta values from 15-30% is achievable for projected EBT-S toroidal plasma parameters if the microwave power levels are in the range of ~10-15 kW. Further extrapolations to future planned experiments are also included. The sensitivity of these results to the details of the hot electron distribution function and geometric and scaling parameters is also discussed.

EBT ring physics workshop

N. A. Uckan

A workshop on EBT ring physics was held at Oak Ridge, Tennessee, on December 3-5, 1979, under the sponsorship of ORNL's Fusion Energy Division. About 70 U.S. and Japanese participants attended the workshop, representing various U.S. industries, universities, and national laboratories; the Japanese Ministry of Education; and Nagoya University.

The workshop organizing committee members were: N. A. Uckan (chairman), W. B. Ard, R. A. Dandl, A. Favale, G. E. Guest, G. R. Haste, C. L. Hedrick, and H. Ikegami. The purpose of the workshop was threefold: (1) to evaluate the current experimental and theoretical understanding of the properties of hot electron rings; (2) to identify dominant physical processes that influence ring formation, scaling, and optimal behavior; and (3) to determine critical areas for near-term research.

The first part of the workshop was devoted to the presentation of technical papers and reviews (5 experimental, 20 theoretical); each presentation was followed by discussion. The second part of the workshop was devoted to open forum discussions of instabilities in hot electron plasmas and ring power balance implications for reactors. Summary talks by the chairmen of the six sessions concluded the workshop.

The proceedings of the workshop has been published by ORNL.¹⁸ The following are

abstracts of papers presented by the Theory Section staff at the workshop.

Abstract of Review of EBT electron ring stability.¹⁹ (D. A. Spong) Energetic electron rings are an essential ingredient in maintaining stability of the EBT toroidal core plasma. By forming a magnetic well of sufficient depth at the plasma edge, they provide stabilization against finite and interchange modes; and, at higher toroidal core betas, enhance stability against ballooning modes. Since the rings themselves can be subject to various instabilities, it is important to understand their stability boundaries in defining stable operating regimes for the toroidal plasma. Ring stability depends both on an interaction between the ring and toroidal plasma as well as on certain features intrinsic to the rings (e.g., a large guiding center drift frequency). In addition, geometric parameters of the rings, such as length, can influence stability boundaries. Because of the simultaneous importance of kinetic, finite-geometry, and profile effects, the general problem of ring stability is complex and has only been examined in the past using certain simplified models. In this paper, we shall review previous work in this area.

Abstract of Numerical calculation of the electron distribution function for EBT electron annuli.²⁰ (J. S. Tolliver; E. P. Jaeger; C. L. Hedrick, Jr.) In the EBT, high-beta relativistic electron rings in each mirror sector provide the necessary minimum-B configuration for stability of the core plasma. Using a nonrelativistic, bounce-averaged kinetic equation, we solve for the electron distribution function, which is assumed to be isotropic in velocity space. The equation is solved in time and in two dimensions, including both velocity and radial derivatives. The Fokker-Planck collision operator includes Coulomb collisions and first- and second-harmonic microwave heating terms. Ionization of cold neutrals (assuming a cylindrical

neutral transport model) provides a source of cold electrons. Under the influence of microwave heating, these electrons diffuse upward in velocity space, and a high-energy tail, corresponding to the hot electron rings, is observed to form on the electron distribution function. A simulated pitch angle scattering term provides a sink of hot electrons from the high-energy tail. The annulus pressure is observed to qualitatively reproduce experimental results as the neutral density is varied.

Abstract of Microwave heating of the annulus in EBT.²¹ (D. B. Batchelor) Our current understanding of electron cyclotron heating in the EBT is reviewed, and the propagation characteristics of the different modes are described for EBT geometry. Wave absorption processes near the second-harmonic resonance, where the annulus is observed to form in EBT, are discussed. Calculations of single-particle energy gain are presented for finite-wavelength waves propagating in a magnetic mirror geometry. It is shown that the heating is highly dependent on pitch angle and that relativistic effects are quite important even at moderate energies. Calculations are presented which suggest that coherent wave-particle interactions may be important in producing the annulus from a population of particles having small initial perpendicular energy.

Abstract of Ring power balance.²² (S. K. Borowski, W. A. Ukan, E. P. Jaeger, T. Kamash*) Recent studies of the ring power balance in EBT experiments have indicated that the energy transfer between the relativistic ring electrons and the core electrons via Coulomb collisions (drag cooling) is the dominant loss mechanism. In this paper, we review the scaling for the microwave power

requirements to sustain the rings against drag cooling and other classical loss mechanisms. Because the stability of the rings is essential to the overall stability of the EBT toroidal core plasma, ring stability requirements also been integrated into our scaling projections for annulus power. Annulus power estimates for both EBT-1 and EBT-S are shown. Extrapolation to future devices is also included.

3.1.5 Advanced Systems

Abstract of Ring power balance optimization in an EBT fusion reactor.²³

(C. A. Ukan, S. K. Borowski, D. B. Batchelor, C. L. Hedrick, Jr., E. P. Jaeger)

The key element of the EBT concept is the presence of the relativistic hot electron rings (annuli) that are formed in each sector of the torus. If the plasma in a reactor is heated to ignition, the toroidal plasma heating will be turned off; the ring heating will, however, have to be sustained throughout the steady-state operation because these hot electron rings are essential for the stable operation of EBT. Therefore, the tradeoff between the quality of confinement afforded by the hot electron rings and the power required to sustain them represents an important consideration in a reactor. A simple power balance scheme for a reactor has been developed to determine the ring power losses on the reactor Q value. The results indicate that the drag cooling dominates the ring power balance for ring energies in the range of 100-600 keV. Although a broad minimum exists in the annulus power loss curves for optimization, the results are sensitive to the details of the scaling parameters assumed. Within the extreme cases of the assumptions, the overall system Q is in the range of 8-100. Detailed studies are under way to determine the dominant mechanisms that influence the ring performance.

*University of Michigan, Ann Arbor, Michigan

Abstract of Microwave coupling in EBT reactor.^{2b}

B. A. Uckan, T. Uckan, R. A. Dardel*

For a typically sized EBT reactor (~1000 MWe), required microwave frequencies lie in the range of 60-110 GHz at power levels of 50-75 MW. As the frequency rises, the quality factor Q for the unloaded cavity (i.e., without plasma) decreases. Because of the short wavelengths of microwave heating power and the large cavity dimensions of a reactor, it is possible to apply quasi-optical principles in the efficient coupling of power to the plasma. The use of a confocal Fabry-Perot resonator with spherical mirrors is discussed; these serve to confine the microwave power to the region occupied by the plasma. The potential advantages of these resonators include high-efficiency utilization of microwave power, minimal thermal burden on the cryopumping system, and significant benefit in preventing microwave leakage from the device. An estimation of the unloaded cavity quality factor Q and the design considerations of Fabry-Perot resonator are given.

Ring power balance implications for a reactor: summary of the open session

B. A. Uckan, W. B. Ard†

The consensus from the open session discussions at the EBT Ring Physics Workshop on ring power balance was that the identified energy loss processes (drag, scattering, and synchrotron radiation) and the estimates of energy loss as a function of hot electron temperature, magnetic field, and plasma parameters are probably correct. The microwave power required to sustain the rings in EBT-I, in the Nagoya Bumpy Torus (NBT), and recently in EBT-S was judged modest and reasonably well estimated by the theoretical models.

There was, however, considerable discussion about how the total power to sustain the rings

would scale to a reactor, or more particularly, how the ring volume would scale with magnetic field, core plasma radius, magnetic field gradients, and so on. The widest parameter range accessible in the past mirror, EBT, and NBT experiments, which all had similar device and plasma parameters, is not sufficiently broad to permit a credible extrapolation to future larger devices, especially to a reactor. When scaled up from EBT-I/S, the ring power requirements for a reactor can become favorable or unfavorable depending on the scaling assumptions made for the volume and on the completeness and self-consistency of the models used (i.e., stability, ring-core interaction, etc.). There was also concern that rings sufficiently thin to allow a favorable power balance in a reactor would require radial density gradients in the core plasma that were unrealistically steep. Transport models, however, predict rather steep density profiles; the validity of these models of course depends on scale lengths for the density gradients of at least several gyroradii. Because the ring electrons and core ions near the surface have comparable gyroradii (i.e., the relativistic electron gyroradius is about equal to the ion gyroradius near the rings), the minimum ring thickness of a few gyrodiameters and the minimum core density scale lengths are not incompatible.

Another possible limitation on the thinness of the ring could result from stability requirements on the ring or on the minimum ring dimensions required for core stability. A number of ring instabilities (Whistlers, mirror modes, etc.), however, are avoided through isotropization. Since the ring length is related to the size of $T_{\perp}/T_{\parallel}$, such requirements may lead to a lower bound on the finite ring length.

3.2 TOKAMAK THEORY

D. J. Sigmar

Our research can roughly be divided into these categories: MHD equilibrium and magnetics

*San Marcos, California

†McDonnell-Douglas Research Laboratories,
St. Louis, Missouri

(poloidal field design), ideal and resistive MHD instabilities in their linear and nonlinear phases (minor and major disruptions), kinetic theory (anomalous transport), particle dynamics (ripple orbit effects and energetic particle slowing down), neoclassical transport of impurities and ripple diffusion theory, and transport simulation of present and future tokamaks incorporating divertors and applications for the Engineering Test Facility (ETF) and the International Tokamak Reactor (ITER). The following sections summarize our work in these areas. As in preceding sections, some of the activities and accomplishments are presented in the form of republished abstracts from the literature. In these cases, reference citations direct the reader to the complete work in question for detailed information.

3.2.1 Plasma Magnetics and Equilibrium Modeling for the Impurity Study Experiment (ISX)

Methods of MHD equilibrium and poloidal field design²⁵

D. J. Strickler, Y.-K. M. Peng

An important problem encountered in calculating free-boundary MHD equilibria is that of locating a set of external conductors with specified currents that create the correct external poloidal field for a given plasma shape. In determining this poloidal field system, we first compute a continuous current density distribution J along a contour r where the windings can be located. This current distribution satisfies the integral equations

$$B_r^{\text{ext}}(z) = \int_0^L J(s) G_r(z, s) ds$$

$$B_z^{\text{ext}}(z) = \int_0^L J(s) G_z(z, s) ds$$

where $B_r^{\text{ext}}(z)$ and B_z^{ext} are the components of the ideal external field at the plasma boundary

given by an equilibrium solution with a prescribed shape and G_r and G_z measure B_r and B_z at a point on the plasma boundary attributable to a unit current at a point on r . We solve this problem by (1) modeling J with a cubic spline with variable knots t_i and (2) approximating the solution by minimizing

$$\sigma^2(\tilde{c}, \tilde{t}) = \|B_p^{\text{ext}} - G(\tilde{t})\tilde{c}\|^2,$$

where $\tilde{c}$ is the vector of coefficients in a B-spline representation of J . The problem is then a linear least-squares problem in the c_i and a nonlinear one in the parameters t_i .

The results have been verified by using the vacuum flux created by the computed current density in free-boundary equilibrium calculations. This has proven to be a flexible approach to the study of poloidal field coils for different plasma shapes (Fig. 3.1) and has been applied in The Next Step (TNS) and ETF designs.

Conversion and improvement of the Rutherford Laboratory's magnetostatic computer code

T. C. Tucker, Y.-K. M. Peng

An improved version of the Rutherford Laboratory's magnetostatic computer code GFLIN3D is now fully implemented on the CDC-7600 at the National Magnetic Fusion Energy Computer Center (NMFECC). The improvements include a new iteration technique that greatly increases the probability of convergence and reduces the computation time by about 30% for calculation with highly nonlinear, ferromagnetic materials. Various features of using this code on the NMFE network and interesting future work have been explored and documented. In addition, the user's guide has been revised, and sample calculations have been provided. This code will have important applications to design analyses of poloidal field with an iron transformer core and toroidal field coil ripple reduction by use of ferritic materials.²⁶

Fig. 3.1. Current density (top) on coil contour creating the equilibrium field (middle) required to produce elliptical (A), racetrack (B), and D-shaped (C) free boundary equilibria. The equilibrium fields approximate the ideal B_z^{ext} , B_z^{ext} fields shown by the dotted contours.

Modeling of MHD equilibrium in ISX-B

Abstract of MHD equilibrium interpretation for ISX-B plasmas.²⁷ (D. J. Strickler, Y.-K. M. Peng, T. C. Tucker, D. W. Swain) In understanding finite-beta plasmas in ISX-B, it is necessary to model the free-boundary plasma equilibria and reproduce the measured parameters within experimental errors. The parameters of interest here include B_z , I_p , I (coils), β_0 , $\bar{\beta}$, a , R_0 , and the approximate pressure profile. Fully accounting for the poloidal field attributable to the iron core, the current in the outer coils (and hence the vertical field) has to increase with decreasing q_0 (i.e., increasing I_p) when the remaining

parameters are held unchanged. By reproducing the measured coil currents, the equilibrium modeling thus provides an estimate of q_0 , a quantity difficult to measure directly. We found that at low $\bar{\beta}$ ($<1\%$) and β_p (<1), q_0 should lie between 0.9 and 1.0, whereas at higher $\bar{\beta}$ ($>1\%$) and β_p (>1), q_0 becomes progressively larger than 1.0. Implications of this result for the interpretations of the ballooning instability and the sawtooth oscillations will be discussed.

An approach to rapid plasma shape diagnostics in tokamaks. (D. K. Lee, Y.-K. M. Peng) A study was made of a deterministic approach to

estimating the plasma boundary shape and location based only on the signals of small $\vec{B}$ pickup loops near the plasma. The pickup loops provide information on $\partial B_R/\partial t$ and $\partial B_z/\partial t$, which can be processed to provide data on B_R , B_z , ψ , and $\partial\psi/\partial n$ (normal derivative), thus leading to a Cauchy boundary condition for solving $\Delta^*\psi = 0$ between the loop locations and the unknown plasma boundary flux surface. Numerical equilibria calculated for ISX-B with different shapes were used to simulate pickup loop signals. Random errors were added to the signals to test the efficacy of two different approaches: global fitting and local fitting. Reasonably accurate results were obtained by the method of global fitting of the boundary values, which is based on the expansion of ψ in terms of eigenfunctions of $\Delta^*\psi = 0$ in a toroidal ring coordinate system. This approach has been shown to permit a random error of nearly 100% in poloidal $\vec{B}$ field values at the loops and still reproduces the plasma shape and location within a few percent. This method appears to have the potential of displaying the plasma shape and location rapidly. Some results from the global fitting are shown in Fig. 3.2.

3.2.2 Linear MHD Stability

Stability and resistive diffusion of tokamak equilibria

L. A. Charlton, R. M. Dery, D. K. Lee,
D. E. Nelson, Y.-K. N. Peng, D. S. Scott,
D. J. Strickler, R. Gruber,* P. Trogon*

The feasibility of the long-term persistence of high-beta, stable tokamak equilibria has been demonstrated by following the resistive diffusion of the equilibria with a periodic reassessment of stability during the relaxation process.

Values of beta [$\beta = 2(\int p dr / \int B^2 dr)$] of at least 5% are required for an economically feasible tokamak fusion reactor. We have shown that:

1. stable plasmas exist with $\beta = 10\%$, even for an aspect ratio of 4.0 and a moderate elongation of 1.65, and
2. resistive diffusion over a time scale longer than the plasma skin time does not

*Centre de Recherches en Physique des Plasma,
École Polytechnique Fédérale de Lausanne,
Lausanne, Switzerland

Fig. 3.2. Global fitting results (solid lines) as compared with the exact plasma boundary (dotted lines). The maximum random error of 5% was added in the ψ data, and the resulting $\vec{B}$ field values have the maximum error of 142.5%. The number of terms in the eigenfunction expansion of ψ is 7 in (a) and 9 in (b); (c) is obtained from taking the average of (a) and (b).

change the stability properties for toroidal wave numbers that are physically relevant.

Small- n stability (n = toroidal wave number) was studied with the computer code ERATO,²⁸ which applies a linear ideal MHD model, using a finite-element energy principle approach. Large- n stability was investigated using a ballooning computer code.²⁹

The critical beta values (beta at marginal stability) were found to scale as

$$\bar{\beta}_c = U q_0^2 \left(\frac{\beta_p}{A} \right)^{3/2} \frac{1.0 - (\beta_p/A)}{q_s^2 A}$$

with U a function of elongation as shown in Fig. 3.3; this has a maximum at $\beta_p \approx 0.5 A$. A perfectly conducting wall with a radius 20% larger than the plasma radius was assumed to represent the effect of appropriately chosen first wall material around the plasma. (Removing the wall to infinity reduces the $\bar{\beta}_c$

Fig. 3.3. U as a function of the elongation σ [$\delta = 0.8(1 - \sigma)$].

implied by the above scaling by a factor of two.) Unstable n 's greater than ~ 100 were discounted because finite gyroradius, kinetic, viscous, and resistive effects should play an important role. Lower values of aspect ratio A yield higher beta values (scaling as A^{-1}). For practical engineering reasons, we have concentrated on cases with $A = 4$ because the resulting beta values appear to be adequately high, and this corresponds to INTOR and ETF values.

An elongation of $\sigma = 1.65$ and a D-shaped plasma with δ (inward shift of the D shape relative to the minor radius) = 0.5 were chosen. The poloidal beta [$\beta_p = 2(\int p dr / \int B_p^2 dr)$], the ratio q_s/q_0 (where q_s = the safety factor at the surface and q_0 = safety factor at the magnetic axis), and the aspect ratio A were varied. The values of β_p , q_s/q_0 , and A ranged from 1.0-3.5, 2.0-4.0, and 1.5-4.0, respectively. For $q_s/q_0 = 2.0$, $q_0 = 1.3$, and $\beta_p = 2.5$, the stable critical beta for an equilibrium of the flux-conserving type was 10%. The relatively low value $q_s/q_0 = 2$ corresponds to a compromise between broad current profiles to reduce ballooning effects and narrow profiles to control kink modes.

This partially optimized case was chosen for a long-term study using Nelson and Grad's transport theory,³⁰ in which the plasma evolves according to

$$\frac{dx}{dt} = \left[\frac{\eta(\alpha^2)}{a^2} (aKx)' \right]',$$

where $x = (q)^{-1}$, η = the parallel resistivity, $\alpha = Fa$ ($F = RB_T$, where B_T = the toroidal magnetic field and R = the distance from the major axis), $a = \langle R^{-2} \rangle$, and $\langle \cdot \rangle = \langle |\nabla|^2 / R^2 \rangle$ (V = the volume). In the above, primes denote toroidal flux derivatives and triangular brackets denote flux surface averages. It was assumed that both temperature and density could be controlled and fixed by external processes (energy and particle injection). The externally imposed shaping field and q at

the edge of the plasma were held constant. During the evolution the equilibrium was periodically recomputed. Spitzer resistivity was used with a peak temperature of 20 keV to provide a typical skin time of ~ 250 s. The evolution of the current density and safety factor profiles is shown in Fig. 3.4. All times scale as $T^{3/2}$ (T = the temperature). In steady state, $\beta = 9.3\%$.

The plasma generally becomes more stable in relation to ideal modes as the current profile evolves, except that localized internal modes appear if the q profile goes through a non-monotone phase with q near a ratio of small integers. At very long times, the profile is monotone and stable for n values up to ~ 25 .

Effect of cross-section shape and shear on the onset of high- n ballooning instabilities in axisymmetric plasmas

J. E. Mauro, Jr.; D. B. Nelson

We have investigated the stability of axisymmetric plasmas with respect to the onset of ballooning modes for different plasma cross-sectional shapes and safety factor profiles $q(\psi)$ for low values of shear. Equilibria were generated numerically with the free-boundary equilibrium code written by D. Stevens³¹ to solve the Grad-Shafranov equation.³² The flux on the boundary of the

computational domain was parameterized to give ordinary circular, D-, and inverse D-shaped plasmas. A quadratic was used as the functional form for $q(\psi)$ with $q_{\text{axis}} = 1$ and $q_{\text{wall}} = 2$ for three different profiles, characterized by the values of $q'_{\text{axis}} = 0, 1, 2$. The plasma aspect ratio was 3. The plasma pressure $p(\psi) = c_1 \psi$.

The ballooning stability analysis was done using a program obtained from R. W. Moore,³³ which solves the ballooning mode eigenvalue ordinary differential equation (ODE) derived by Dobrott et al.²⁹ The constant c_1 in the expression for $p(\psi)$ was varied to determine the value for which the plasma first became unstable on one of the flux surfaces. The volume-averaged beta ($\bar{\beta}_{\text{crit}}$) corresponding to each critical value of c_1 was then computed. Figure 3.5 shows the dependence of $\bar{\beta}_{\text{crit}}$ on plasma shape for the three different $q(\psi)$ profiles used. Figure 3.6 shows the stability boundary for the least stable flux surface, which was at the edge of the plasma in all cases studied here, in terms of the normalized shear as a function of $p'(\psi)$, both local quantities as represented in the figure. The ordinary D-shaped plasmas appear to exhibit very different behavior from the other shapes. The reasons for this are not presently understood.

Fig. 3.4. Safety factor (Q) and toroidal current density (J) profiles in the midplane as a function of time and distance from the major axis (R).

Fig. 3.5. Dependence of β_{crit} on plasma shape for three different $q(\psi)$ profiles.

Fig. 3.6. Stability boundaries for the least stable flux surface of different cross-sectional shapes in terms of the dependence of the local shear on p' .

3.2.3 Nonlinear Resistive MHD Instabilities

The understanding of tearing modes as a possible mechanism for major disruptions and other MHD activity in tokamaks has been further advanced. Because of the development of a new computer code, RSF, it became possible to extend major disruption calculations to lower, realistic levels of resistivity. The earlier result, that mode coupling leads to fast explosive behavior, still holds, even after a period of algebraic mode growth. The present calculations include electron temperature evolution. For realistic values of parallel and perpendicular electron thermal conductivity, the effects of temperature evolution are minor.

The inclusion of toroidal effects (with code LOBETO) tends to destabilize the tearing

modes, resulting in larger saturated islands or, in the case of mode coupling, in a stochastic region encompassing nearly the full plasma volume.

The MHD behavior in the Japanese Institute of Plasma Physics (JIPP) T-II tokamak has been carefully analyzed. Over a wide range of cases, including stable cases and cases in which voltage spikes are exhibited, the numerical calculations using the three-dimensional (3-D) tearing mode model agree with the subsequent plasma evolution. In disruptive cases, the magnitude and time scale of the voltage spike are consistent with the experiment. For cases with a Mirnov loop signal, the calculation accurately predicts the signal level.

Nonlinear coupling of tearing modes with self-consistent resistivity evolution in tokamaks

B. Carreras, H. R. Hicks, J. A. Holmes, B. V. Waddell

The nonlinear interaction of tearing modes of different helicity in tokamaks was studied to determine realistic values of resistivity and parallel heat conduction. The self-consistent evolution of the resistivity was taken into account through the electron heat conduction equation. For equilibrium q profiles inferred from electron temperature profiles measured before a tokamak disruption, the essential result is that the mode $m = 2$, $n = 1$ nonlinearly destabilizes other modes on a rapid time scale. Because of the development of magnetic islands of different helicity, the toroidal current density is severely deformed. These islands overlap and field lines become stochastic in a sizable plasma volume, flattening the temperature profile in this region through parallel heat transport. The deformation of the toroidal current produces a rapid decrease in the self-inductance of the plasma, and the voltage at the limiter decreases, becoming increasingly negative.

We have studied a wide spectrum of equilibrium q profiles to better understand the

characteristics of this nonlinear interaction. A very general result is that the strong destabilization of the 3/2 mode happens when

$$r_{21} - r_{32} < \frac{1}{2} (W_{21} + W_{32}) ,$$

where r_{mn} is the radial position of the singular surface for the m/n mode and W_{mn} is its single-helicity saturated width. For all the cases we considered, the rapid destabilization of the 3/2 mode by the 2/1 mode starts when both magnetic islands touch. Apparently, the usual resistive flow effects that reduce the mode growth from exponential to linear are not operative in this explosive growth phase because the magnetic flux surfaces are being destroyed.

These results confirm the ones obtained when resistivity evolution was not taken into account, except that here the disruption proceeds twice as fast. This time scale compares well with the experimentally observed time scale for the major disruption.

The effects of toroidicity on tearing modes

*B. Carreras, H. R. Hicks, J. A. Holmes,
D. K. Lee, S. J. Lynch, V. E. Lynch*

The time evolution of tearing modes in toroidal geometry was studied in the low-beta and large-aspect-ratio limit. In this study we used an initial-value 3-D computer code, LOBETO, which numerically advances the reduced set of resistive MHD equations. In general, toroidicity has a destabilizing effect on tearing modes in the limits mentioned.

A detailed analysis of the linear regime was also performed to investigate the toroidal effects on the linear growth rate and eigenfunctions of the tearing modes. We considered several safety factor profiles and different values for the aspect ratio. These results show that semianalytic Δ' calculations based on the coupling of only two modes give excellent agreement with the initial-value results. A scheme for performing such calculations has been developed.

The nonlinear phase of the evolution was also numerically followed. In the case of strong interaction of different helicities, a significantly larger region (Fig. 3.7) of stochastic magnetic field lines results than in the cylindrical geometry case. When the islands do not interact (Fig. 3.8), the saturated island widths are increased by toroidal effects.

The toroidal coupling between the 1/1 and 2/1 tearing modes during the nonlinear phase was also studied. We found that the 2/1 tearing mode can be destabilized by the 1/1 mode through the toroidal coupling.

Analysis of the disruptions in the JIPP T-II tokamak

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

Results from the JIPP T-II tokamak³⁴ offer some of the most complete data so far for testing the hypothesis that major disruptions can be understood in terms of a coupled-helicity tearing mode mechanism.³⁵ These data were obtained through a systematic study of the effects of controlling the toroidal current density profile by gas puffing, by programming the plasma current wave form, or by a combination of the two. Using the toroidal current profile inferred from the measured electron temperature profile and the central q value, we numerically analyzed the three types of discharge of Toi et al.³⁴ at several times and found agreement with the experiment. We studied stable cases — that is, cases that exhibit a large saturated $m/n = 2/1$ mode — and cases that are terminated by a major disruption. Toroidal effects as well as realistic levels of resistivity were included in the calculation.

The method of lines solution of the reduced resistive MHD equations

P. W. Gaffney, B. Carreras, H. R. Hicks

We have solved the reduced resistive MHD equations by the method of lines (MOL). In this method the spatial derivatives that

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Fig. 3.7. Time evolution of magnetic field lines in a toroidal plasma. This case shows that when the equilibrium is such that mode coupling dominates, the plasma evolves to a state in which nearly the entire field is stochastic.

Fig. 3.8. Tearing mode ($m = 2, n = 1$) saturation island width for cylindrical ($\epsilon = 0$) and toroidal ($\epsilon = 0.1$) cases as a function of λ , a parameter controlling the equilibrium profiles. For larger values of λ , mode coupling prevents saturation. The effects of toroidicity are to increase saturation widths and the probability of mode coupling.

appear in the equations are discretized, and the resultant system of coupled ODEs in time is then solved using an ODE integrator. The results indicate that the usual method of solving the ODEs, namely, a variant of Gear's method, is unsuitable for this problem. In using MOL, we were particularly interested in the nonlinear regime of the MHD equations. Preliminary results show that in this case MOL is approximately three times faster than our existing Fourier series code RSF.³⁶ Moreover, the solution obtained by MOL is entirely in agreement with that obtained by RSF.

Three-dimensional nonlinear calculations of resistive tearing modes

*H. R. Hicks, B. Carreras, J. A. Holmes,
D. K. Lee, B. V. Waddell*

Recent numerical calculations of the evolution of resistive tearing modes in cylindrical geometry have been central to the understanding of MHD activity and disruptions in tokamaks. The nonlinear, 3-D, initial-value computer code RSF has provided many of these results. This code uses a Fourier series representation in the two periodic coordinates and a finite-difference representation in the remaining direction. This choice makes RSF considerably more accurate and efficient than previous codes. The RSF code has furnished the basic numerical scheme for a subsequent toroidal code, LOBETO.

Central processing unit (CPU) times for nonlinear resistive MHD calculations are very strongly dependent on the input variables. Although some regions of parameter space can be studied very cheaply, the conditions that correspond to tokamak plasmas are extremely time-consuming to study. Unfortunately, it is not possible to extrapolate reliably from the easy cases to the interesting regime. In recent years, we have improved the efficiency of our codes by two orders of magnitude. As a result, with RSF it is now feasible for us to run some realistic tokamak cases on a CDC-7600. A reduction in run time of an additional

factor of four is obtained by going to the CRAY-1, where the most time-consuming routines are vectorized. This last factor makes feasible a program of systematic studies.

Multiple helicity tearing mode calculations: voltage spike

*B. Carreras, H. R. Hicks, J. A. Holmes,
B. V. Waddell*

For profiles observed prior to major disruptions,^{37,38} the $n = 2$, $n = 1$ and $m = 3$, $n = 2$ tearing modes are linearly unstable, where m and n are the poloidal and toroidal mode numbers, respectively. We have been able to follow their nonlinear evolution with RSF. The essential result is that the 3/2 mode and other modes are nonlinearly destabilized on a rapid time scale ($\tau^{-1} = \gamma_{21}^0$) by the 2/1 mode (Fig. 3.9). Because of the development of islands of different helicity, the toroidal current density is severely deformed. These islands overlap and field lines become stochastic in a sizable plasma volume, flattening the temperature profile in this region through parallel heat transport. The current deformation produces a rapid decrease in the self-inductance of the plasma, and the voltage at the limiter drops, becoming increasingly negative at the end of the run (Fig. 3.9), in agreement with the negative voltage spikes observed experimentally.

These results have been obtained for an ample range of parameters. The results shown in the figure are for the Princeton Large Torus (PLT) equilibrium q profile prior to a major disruption,³⁷ with $S = 10^6$, $\hat{\chi}_\parallel = 10^7$ ($\chi_\parallel = 10^{11}$ cm²/s for PLT parameters), and $\hat{\chi}_\parallel/\hat{\chi}_\perp = 2.7 \times 10^8$.

3.2.4 Kinetic Theory

Anomalous transport

A fundamentally new result regarding anomalous electron heat conduction has been found. Abstracts of papers published in 1979 concerning this topic are presented below.

Fig. 3.9. Radial overlap of islands (top) and the voltage of the limiter (bottom) in a cylindrical calculation. Overlap of the $m/n = 2/1$ and $m/n = 3/2$ magnetic islands leads to explosive growth.

Abstract of Finite- β_e universal mode turbulence and Alcator scaling.³⁹ (K. Molvig, S. P. Hirshman, J. C. Whitson) A self-consistent theory of finite- β_e universal mode turbulence is developed. Saturation results from resonance broadening of the electron response owing to magnetic shear. Electron diffusion for $\beta_e > m_e/m_i$ is due to the magnetic part of the fluctuations. The diffusion coefficient

$$D = 0.1 \left(\frac{T_e}{T_i + T_e} \right) \frac{m_e}{m_i \beta_e} \left(\frac{L_s}{L_n} \right) \frac{2v_{ti}^2}{L_n}$$

scales inversely with density, is independent of magnetic field, and is in excellent quantitative agreement with observations on the Alcator tokamak.

Abstract of Turbulent destabilization and saturation of the universal drift mode in a sheared magnetic field.⁴⁰ (S. P. Hirshman, K. Molvig) In a sheared magnetic field, turbulent diffusion of electrons in the vicinity of a mode rational surface can eliminate the stabilizing influence of non-resonant electrons and lead to an absolute instability at small but nonzero wave amplitudes. As the turbulence grows, the inverse electron Landau resonance is broadened in both velocity and configuration space, and the convective shear damping owing to ions is enhanced by turbulent spatial broadening of the mode until saturation occurs.

Abstract of Self-consistent spectrum of electrostatic drift wave fluctuations due to electron phase space correlations in a sheared magnetic field.⁴¹ (S. P. Hirshman, P. H. Diamond*) The spectrum of electrostatic universal mode fluctuations owing to electron phase space correlations (clumps) in a sheared magnetic field is self-consistently calculated. The pair correlation equation for electron in a sheared field is derived and renormalized and an approximate solution for the correlation function is obtained. Using the density correlation function as a source term in Poisson's equation, the renormalized dielectric operator is inverted to obtain the turbulent spectrum. Self-consistency is imposed by requiring that the diffusion coefficient derived from the calculated spectrum equal that used to obtain the spectrum originally. The diffusion coefficient obtained in this way is in close agreement with that

*Institute for Advanced Study, Princeton, New Jersey,

calculated from one-point turbulence theory. The calculated spectrum is rapidly convergent for large wave numbers and is an even function of the wave numbers, consistent with the quadratic dependence of the relative diffusion coefficient on the orbit separations for small clump dimensions.

Abstract of Two-dimensional transport of tokamak plasmas.⁴² (S. P. Hirshman, S. C. Jardin*) A reduced set of two-fluid transport equations is obtained from the conservation equations describing the time evolution of the differential particle number entropy, and magnetic fluxes in an axisymmetric toroidal plasma with nested magnetic surfaces. Expanding in the small ratio of perpendicular to parallel mobilities and thermal conductivities yields as solubility constraints 1-D equations for the surface-averaged thermodynamic variables and magnetic fluxes. Since Ohm's law $E + uB = R'$, where R' accounts for any nonideal effects, only determines the particle flow relative to the diffusing magnetic surfaces, it is necessary to solve a single 2-D generalized differential equation $(\partial/\partial t)\nabla\phi$. The equation $\nabla p - JB = 0$ is used to find the absolute velocity of a magnetic surface enclosing a fixed toroidal flux. This equation is linear but nonstandard in that it involves flux surface averages of the unknown velocity. Specification of R' and the cross-field ion and electron heat fluxes provides a closed system of equations. A time-dependent coordinate transformation is used to describe the diffusion of plasma quantities through magnetic surfaces of changing shape.

Linear theory of microinstabilities

Abstract of Velocity space instabilities of alpha particles in tokamak reactors.⁴³

(D. J. Sigmar) Of all species composing the fusion plasma, the alpha particles have the

strongest deviation from thermodynamic equilibrium, in configuration space as well as in velocity space. Under typical circumstances (main plasma density and temperature profile somewhat peaked and startup of the fusion burn not slower than an energy confinement time), the ensuing alpha profile gradients will lead to a low-frequency drift Alfvén wave instability, and the hollow alpha velocity distribution will lead to a high-frequency compressional Alfvén wave instability. This lecture will address the conditions for the linear theory and the anomalous relaxation of the velocity space instability of isotropic alpha distribution functions.

Low-mode-number shear Alfvén waves destabilized by alpha particles in a tokamak.

(K. T. Tsang, D. J. Sigmar, J. C. Whitson)

The finite Larmor radius corrections to the shear Alfvén eigenmode equation in an inhomogeneous (cylindrical) tokamak plasma have been derived. A quadratic form for this fourth-order eigenmode equation can be constructed to show that the eigenmodes are always neutrally stable in the absence of sources of free energy and dissipation. Alpha particle drifts in tokamak geometry destabilize these modes. Although the electron Landau resonance contributes to damping, modes with low radial mode numbers remain unstable for typical reactor parameters.

Ion drift resonance in tokamak. (K. T. Tsang) In ideal MHD, the toroidal curvature of a tokamak can drive ballooning instabilities. In microscopic stability theory, the effects of ion magnetic curvature have been discussed by a number of authors. In all these works, however, the ion drift frequency ω_d is assumed to be small compared with the mode frequency ω and is treated iteratively. This assumption is not always valid, especially for the ion temperature gradient drift mode. We have derived an eigenmode equation including the ion drift frequency exactly using the ballooning mode formulation. Important ion drift

*Plasma Physics Laboratory, Princeton, New Jersey.

resonance effects are retained without difficulty. Assuming adiabatic electron response, the preliminary numerical result showed the existence of an unstable ion drift resonance mode. Further investigation is in progress.

Finite-beta trapped electron instabilities.

(K. T. Tsang, E. J. Catto, M. H. Rosenbluth*)

The trapped electron distribution function in the presence of electromagnetic perturbation has been derived with a boundary layer analysis at the trapping boundary in the limit of very small collisionality. The trapped electrons are unaffected by the magnetic perturbation, and their distribution function is well behaved on the mode rational surface. This corrects the oversight in previous calculations.⁴⁵ The eigenmode equations obtained were solved numerically. The collisionless trapped electron drift mode was found to be insensitive to beta variation, and the shear Alfvén mode with $\omega \approx \omega_{ce}$ to be stable for normal cases.

Numerical computation of radial drift eigenmodes

Several novel developments occurred in the linear kinetic theory of tokamak confinement. Drift or universal instabilities have long been thought to play an important role in the determination of cross-field heat conduction in tokamaks. The standard method for studying these modes was to obtain unstable eigenmodes of the differential equation derived from the integral equation by assuming the radial wavelength was sufficiently long, that is, $k_{\perp} \rho_i \ll 1$. Although the answers so obtained were consistent with this assumption, our own results seemed to indicate that they were on the verge of breaking down. We therefore decided to solve the integral equation in order to check these results. The outcome of this work is reported elsewhere.^{45,46}

*Institute for Advanced Study, Princeton, New Jersey

Integral equation study of the temperature drift instability

C. O. Beasley, Jr.; J. Denavit; J. E. McCune; K. T. Tsang; W. I. van Rij

An integral equation was solved numerically to obtain the radial eigenmodes of unstable ion drift waves localized near mode-rational surfaces in toroidal plasmas. We used an ion gyrokinetic equation and an electron drift kinetic equation in a slab geometry to describe the behavior of the plasma. The usual approximation to the Bessel operations in x-space was avoided by performing these operations exactly in k_x -space. Likewise, the convective derivative in k_x -space was avoided by performing a scalar multiplication in x-space. We obtained eigenmodes by employing an initial-value approach and an efficient numerical algorithm for following the time response of the distribution function. Over a wide range of poloidal modes, shear, and temperature gradients, we found the mode structure from solutions of the integral equation sufficiently complex to render invalid results based on the assumption that the radial wavelength is much larger than an ion gyroradius. Our results permit clear identification of the mode as a resonant ion mode localized about the ion point, contrary to the nonresonant behavior of the differential approach.

3.2.5 Impurity Transport and Impurity Physics Packages

Neoclassical theory

Several important gaps remain in fundamental neoclassical theory concerning effects of neutral beams and impurities. The following describes work during 1979 that has brought us closer to a full understanding of this field.

Abstract of Response of a tokamak plasma to particle and momentum sources.⁴⁷

(W. M. Stacey, Jr.; D. J. Sigmar) The response of an axisymmetric toroidal tokamak plasma to first-order particle and momentum sources is investigated. The momentum sources drive coupled

poloidal and toroidal mass flows and electrostatic field evolution, which relax to asymptotic values on a time scale characteristic of the dominant viscous or external drag mechanism. The asymptotic steady-state momentum balance provides the necessary condition to completely determine the ambipolar potential, the particle fluxes, and the currents in the flux surfaces and, hence, to determine transport fluxes across flux surfaces. Transport fluxes are driven across flux surfaces either by the usual interspecies collisional momentum exchange or by momentum exchange between the plasma and external sources and/or drags. A generalized Ohm's law is obtained and used to determine how particle and momentum sources drive parallel currents and alter the evolution of the q profile. The theory is formulated for arbitrary plasma cross sections, beta, and collision regimes.

Abstract of Impurity control by neutral beam injection.⁴⁸ (W. M. Stacey, Jr.; D. J. Sigmar) A recently developed generalization of neoclassical theory is applied to study the use of neutral beam injection to drive impurities out of a tokamak plasma. The theory accounts for the direct momentum exchange between beam and plasma, which has been studied previously, and also for the effects of the beam in altering the first-order particle flows and the ambipolar potential. Conditions are established under which coinjection drives impurity fluxes outward. It is estimated that order-unity changes of the radial impurity flow should be found in current experiments and that modest levels of neutral beam injection should suffice for impurity control in reactors. In the course of this work, the general theory of particle transport in the presence of particle and momentum sources is evaluated for a two-

species plasma with arbitrary plasma geometry and beta.

Abstract of Low-Z impurity transport in tokamaks.⁴⁹ (R. J. Hawryluck,* S. Sukewer,* S. P. Hirshman) Low-Z impurity transport in tokamaks was simulated with a 1-D impurity transport model including both neoclassical and anomalous transports. The neoclassical fluxes are due to collisions between the background plasma and impurity ions as well as to collisions between the various ionization states. The evaluation of the neoclassical fluxes takes into account the different collisionality regimes of the background plasma and the impurity ions. A limiter scrapeoff model is used to define the boundary condition for the impurity ions in the plasma periphery. To account for the spectroscopic measurements of power radiated by the lower ionization states, fluxes attributable to anomalous transport are included. The sensitivities of the results to uncertainties in rate coefficients and plasma parameters in the periphery are investigated. The implications of the transport model for spectroscopic evaluation of impurity concentrations, impurity fluxes, and radiated power from line emission measurements are discussed.

Abstract of Neoclassical transport in plasmas containing ions with multiple charge states.⁵⁰ (C. D. Boley,† E. M. Gelbard,† S. P. Hirshman) It is shown that in the calculation of the Spitzer function and the Pfirsch-Schlüter transport coefficients for a

*Plasma Physics Laboratory, Princeton University, Princeton, New Jersey

†Applied Physics Division, Argonne National Laboratory, Argonne, Illinois

multispecies tokamak plasma, a considerable analytic simplification occurs for the case of ions with multiple charge states. The number of coupled integrodifferential equations to be solved can be reduced from the total number of species (each charge state constitutes a distinct species) to the number of ions with different nuclei. The solutions are expressed in terms of quantities characterizing a system of interacting electrons and ion aggregates.

Physics packages for impurity transport simulation

E. C. Crume, Jr.; D. E. Amurcius

The two vehicles for the numerical simulation of impurity transport in tokamaks, RECYCL⁵¹ and IMP TAR⁵² (impurity transport and recycling), have been upgraded. New modules have been devised for the calculation of impurity transport coefficients for use in IMP TAR and for the calculation of the corona model atomic physics processes of carbon, oxygen, argon, and iron for use in RECYCL, IMP TAR, and other ORNL tokamak simulation programs. The IMP TAR program was taken to the Soviet Union and was successfully run on the EC-1040/1 computer at the Kurchatov Institute in Moscow.

Impurity transport coefficients. The approximate particle transport coefficients of Isler, Crume, and Howe⁵¹ have been replaced with newly derived approximations, and their exact coefficients⁵¹ have been upgraded to better account for the plateau-Pfirsch-Schlüter transition.^{49,53} The "averaged-ion" method of calculating parallel friction coefficients in the Pfirsch-Schlüter regime⁴⁹ was checked against the exact method and was found to yield very good results. The averaged-ion method, however, does not include the effects of the plateau-Pfirsch-Schlüter transition, so this time-saving technique has not been

employed in the coefficient module devised for IMP TAR.

Corona model atomic physics. The corona model atomic physics of iron given by Breton, de Michelis, and Mattioli⁵⁴ has been put into a self-contained calculational module after a significant upgrading of some of the details of the treatment of dielectronic recombination. A similar module was prepared for argon to replace the generalized module used earlier in the analysis of T-4 experiments.⁵⁵

Analysis of T-4 and T-10 experiments in the Soviet Union. Special IMP TAR computing packages in IBM- and DEC-compatible forms were prepared for use in analyzing data from Soviet tokamak experiments at the Kurchatov Institute in Moscow. The IBM-compatible form could be run on the Soviet EC-1040/1 computer, which includes a version of the IBM 360 operating system. The length of the visit and computer time limitations allowed only preliminary analysis of some data from T-4 at the Kurchatov Institute; however, some new data were brought back for analysis at ORNL.

3.2.6 Particle Dynamics

Work in particle dynamics concerns rippled magnetic fields and bundle divertor field studies in relation to single-particle orbits, a Hamiltonian formulation of guiding center orbits, and the solution of the collisional Fokker-Planck equation for energetic particles.

Fokker-Planck calculations of fast particles

A Fokker-Planck code with particle trapping effects. (R. H. Fowler, J. Smith, J. A. Rome) A computer code has been developed to solve a bounce-averaged Fokker-Planck equation that includes particle trapping. Cordey⁵⁶ has previously derived such an equation; the equation we solve, however, was derived by

Rome and differs slightly from Cordey's results. Letting $x = v/v_0$, we solve

$$\begin{aligned} \frac{\partial f}{\partial t} = & -\frac{\tau_s}{\tau_{cs}(x)} f + \frac{1}{x^2} \frac{\partial}{\partial x} \left[\left(x^3 - 2Bx + x_c^2 + \frac{C}{x^2} \right) f \right] \\ & + \frac{1}{x^2} \frac{\partial^2}{\partial x^2} \left[\left(Bx^2 + \frac{C}{x} \right) f \right] + \frac{D}{x^2} \left(1 - \frac{n_1}{x^2} + D_2 x \right) \\ & \times \frac{1}{a(\theta)} \frac{\partial}{\partial \theta} \left[a(\theta) \frac{\partial f}{\partial \theta} \right] + E \left[-(\cos \theta) \frac{\partial f}{\partial x} + \frac{\sin \theta}{x} \frac{\partial f}{\partial z} \right] \\ & + \tau_s \sum_i n_{fi} S_i(x, \theta) . \end{aligned}$$

where

$$a(\theta) = \left\langle \frac{1}{x_1} \right\rangle \sin \theta \cos \theta ,$$

$$k(\theta) = \frac{\sin \theta}{\cos \theta} \left\langle \frac{x_1}{B} \right\rangle ,$$

$$\left\langle \frac{1}{x_1} \right\rangle = \begin{cases} \frac{4}{\cos \theta} K \left(\frac{2c \sin^2 \theta}{\cos^2 \theta} \right) & \text{(passing ions)} \\ \frac{4}{\sqrt{2c} \sin \theta} K \left(\frac{\cos^2 \theta}{2c \sin^2 \theta} \right) & \text{(trapped ions)} \end{cases}$$

$$\left\langle \frac{x_1}{B} \right\rangle = \begin{cases} 4 \cos \theta E \left(\frac{2c \sin^2 \theta}{\cos^2 \theta} \right) & \text{(passing ions)} \\ 4 \sqrt{2c} \sin \theta \left[E \left(\frac{\cos^2 \theta}{2c \sin^2 \theta} \right) - \left(1 - \frac{\cos^2 \theta}{2c \sin^2 \theta} \right) K \left(\frac{\cos^2 \theta}{2c \sin^2 \theta} \right) \right] & \text{(trapped ions)} \end{cases}$$

The functions K and E are the complete elliptic integrals of the first and second kind, and $c = r/R$. The other terms have previously been defined.^{5,7} We apply the usual boundary conditions in the passing particle region. The solution is required to be symmetric about $\theta = \pi/2$ in the trapped particle region. The conservation of flux of particles between the passing and trapped region leads to a jump condition at the boundary between trapped and passing particles. However, because of the required symmetry in the trapped particle region, it is inconsistent to include the usual loss region in this code.

The code has been used to study both coinjecting and counterinjecting ions slowing down in the plasma.

Heating by collisional slowing of fast ions in noncircular tokamaks. (L. M. Hively,

J. Rome, G. E. Miley*) Plasma heating by energetic, charged fusion products is a crucial issue for obtaining fusion reactor conditions. The evolution of alpha particle density, momentum, kinetic energy, and heat flux is important in the study of startup and ignition scenarios as well as in steady-state operation. Since fast ions have large banana-width guiding center orbits in tokamaks, the usual locally defined particle distribution function loses its usefulness. Accordingly, the problem of the collisional slowing down of fast ions has been formulated to describe these orbits in an axisymmetric, non-circular tokamak.

Three constants of motion (COM) are required to describe the guiding center orbit of a collisionless ion in an axisymmetric tokamak. The usual COM are the particle energy E , the magnetic moment μ , and the toroidal canonical angular momentum p_ϕ . However, this set is inconvenient for our problem. Following Rome and Peng,⁵⁸ we label the orbits by the ion speed v ; the maximum value of the poloidal flux function along the guiding center orbit ψ_x (where ψ is increasing from the magnetic axis); and the cosine of the angle between the parallel component of plasma current and the ion velocity evaluated at ψ_x , that is, $\zeta = \hat{J}_\parallel \cdot \hat{v} / (|\hat{J}_\parallel| |\hat{v}|)$. Using the COM (v, ζ, ψ_x) , the resulting orbit equation takes the form

$$B = B_0(\psi; v, \zeta, \psi_x) \quad (3.1)$$

where $\psi_n \leq \psi \leq \psi_x$ and ψ_n is the minimum value of the poloidal flux on the orbit. Assuming up-down symmetry, there is a one-to-one correspondence between the flux coordinates (B, ψ) , given by Eq. (3.1), and the spatial coordinates (R, z) , where R is the major radial position and z is the distance above the midplane. This correspondence is obtained

* University of Illinois, Urbana, Illinois

by locally inverting the tokamak equilibrium. The resulting orbit topology includes co- and countercirculating orbits as well as trapped orbits.

The Fokker-Planck equation is bounce-averaged after the variables have been transformed to the COM space described above. This is the only treatment known to us that accurately describes the fast ion distribution in realistic, high-beta, noncircular equilibria such as those of Callen and Dory.⁵⁹ In addition, this formulation can treat first-orbit losses, finite slowing down time and slowing down drifts. With Monte Carlo techniques (such as those of Mikkelsen and Post⁶⁰), it is relatively easy to obtain moments of the distribution function (i.e., alpha heating). However, large amounts of computer time are required to find the distribution function itself (for comparison with charge exchange data). Our method allows determination of analytical scalings as well as the distribution function and its moments.

Neglecting pitch angle scattering, speed diffusion, and charge exchange, the Fokker-Planck equation can be reduced to a first-order, 3-D partial differential equation. This problem has been solved using the method of characteristics, which also gives μ/E as a constant of the motion (μ = magnetic moment and E = ion energy). The resulting particle distribution function is

$$f = \frac{\bar{S}}{r v^2 (dv/dt)}, \quad (3.2)$$

where $\bar{S}$ is the bounce-averaged alpha source rate and dv/dt is the bounce-averaged slowing rate along the slowing down characteristic. By assuming that the slowing down time is fast compared with the equilibrium evolution, and using the invariant μ/E , the problem is reduced to the solution of a single first-order ODE for the ψ_x slowing down drift:

$$\frac{d\psi_x}{dv} = \frac{d\psi_x/dt}{dv/dt}. \quad (3.3)$$

This ψ_x drift is inward as fast ions approach some nominal flux surface $\psi_{\text{nom}} < \psi_x$ to form small banana-width (thermalized) orbits.

Moments of the distribution function yield the alpha particle density, current, and heating profiles. There is a net cogoing alpha current that is solely due to finite banana-width effects. The current contributions owing to co- and countergoing orbits tend to cancel, leaving mainly the trapped contribution. The path length along the cogoing piece of such a trapped orbit is longer than the countergoing path length and thus results in a net cocurrent, since the travel time goes as path length/speed. Bootstrap effects might enhance this current as much as tenfold, thereby reducing the volt-second requirements for a reactor. Results show that central ion heating by the alphas is 10-15% below in-situ deposition, indicating that startup and maintenance of ion temperature may be more difficult than anticipated. However, the lower central alpha density (compared with in-situ density) indicates a reduced alpha ash accumulation on axis, somewhat ameliorating the impurity removal problem.

The effects of local magnetic ripple on fast ions in tokamaks

J. A. Rome, J. P. Lyon,* E. H. Fowler,
V. E. Lynch

Local magnetic ripple is introduced into tokamaks by the addition of external coils for bundle divertors or for ripple injection. The destruction of axisymmetry weakens or eliminates the invariance of the toroidal canonical angular momentum P_ϕ . Unless P_ϕ is replaced by a new invariant, collisionless fast ions may leave the tokamak in times much shorter than the slowing down time τ_s .

From following numerous guiding center orbits of fast ions, we can state:

- Circulating fast ions are relatively unaffected by local ripple. Although P_ϕ is

* Experimental Confinement Section

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the toroidal drift of bananas to stagnate where $\nabla B = 0$ and increases their interaction time with the ripple.

On the basis of these observations, we propose magnetic field design criteria for bundle divertors. The field lines must go through the divertor in the desired manner, leaving room for shielding and without hitting the coils. The plasma flux surfaces must remain well formed to avoid losing the background plasma. Simultaneously, the B surfaces must not neck down too much in the divertor region to avoid mirroring the bananas in the divertor region.

We have applied these criteria to the comparison of the two-coil and four-coil bundle divertor configurations for INTOR/ETF as illustrated in Fig. 3.10. The two-coil divertor is clearly inadequate. The flux surfaces are ergodic, and the B surfaces shrink severely in cross section near the divertor and close off completely in the case shown. The particle is trapped between the divertor maxima and quickly leaves the center of the plasma. The four-coil divertor is significantly better. The flux surfaces are good, and the B surfaces do not change greatly in the divertor region. Not enough space is allowed, however, for shielding the inner legs of the coils; thus, more improvement is required.

Putting in more than one bundle divertor in symmetric locations can heal otherwise ergodic flux surfaces by cancelling the lowest Fourier harmonics of the perturbation. However, if the divertors introduce a local maximum, the fast ions may get stuck between adjacent divertors and leave the system more quickly than in the single-divertor case. Thus, it is preferable to have only one properly designed divertor.

Magnetic fields and guiding center orbits

$|B|$ surface studies. (V. E. Lynch, R. H. Fowler, J. A. Rome) In a tokamak, collisionless trapped particles always bounce on the same $|B|$ surface. In order to study these surfaces

in the full 3-D geometry, a new contour plotting program, MODB, was developed to illustrate changes in the $|B|$ surfaces near local magnetic ripples. It has been used for eyeglass ripple coils, the ISX-B bundle divertor, a hybrid divertor, and several bundle divertors for INTOR/ETF. The technique of overlaying several of these surfaces has been successfully used to show the effect of these magnetic perturbations on $|B|$. Along with the contours, the code can draw a torus wall, a plasma outline, divertor wires, or any combination of these.

This code uses the subroutines of the field-line orbit code (FLOC)⁶¹ to calculate the total $|B|$ and a 2-D contour plotter package written by B. Clark and R. Fowler. First, the contours are found in the R - z plane at several values of the toroidal angle ϕ near the divertor. All of these contours are plotted on one 2-D plot in R and z , with each contour labeled according to ϕ . The same process is repeated in the R and ϕ plane at several values of z . These contours are then projected into three dimensions. No assumptions of symmetry are made in finding the contours. For each divertor design, $|B|$ is calculated only once for each grid of R and z and R and ϕ and is stored. Then, any number of $|B|$ surfaces can be found. A typical $|B|$ surface for a four-coil INTOR bundle divertor is shown in Fig. 3.11.

Guiding center orbit theory. (H. K. Meier)

Besides the energy, a tokamak with axial symmetry has P_ϕ (the toroidal angular momentum) as a constant of the motion of a charged particle in a magnetic field. The existence of this constant greatly simplifies the computation of energetic particle orbits. The insertion of ripples or a bundle divertor destroys this symmetry, and with it the invariance of P_ϕ .

Determination of the particle motion, though now quite complex, is nevertheless an indispensable element of the energy confinement problem. To study such motion, a Hamiltonian

Fig. 3.11. Typical $|\vec{B}|$ surface for a four-coil INTOR bundle divertor ($|\vec{B}| = 4.6$ T).

formulation has been derived that is valid when the magnetic moment is a good adiabatic invariant. A Hamiltonian formalism facilitates not only the numerical constancy of the particle energy but the incompressibility of the motion in phase space. A computer program is being developed that numerically determines the Hamiltonian for an arbitrary magnetic field in a generalized toroidal coordinate system.

Analytic representations of tokamak equilibria of arbitrary cross sections

H. K. Meier

In a variety of applications - matching experimentally observed equilibria, working with neoclassical and anomalous transport theory, and studying energetic particle orbit - a rapidly convergent equilibrium representation of the form

$$R(\psi, \theta) = \sum_{n=0}^N R_n(\psi) \cos n \theta,$$

$$Z(\psi, \theta) = \sum_{n=1}^N Z_n(\psi) \sin n \theta$$

is highly desirable.

It has been found that for a particular choice of metric (poloidal angle variable), the expansion can be made to take the form

$$R(r, \theta) = D_0(\psi) + D_1(\psi) \cos \theta + \sum_{n=2}^N D_n(\psi) \cos n \theta,$$

$$Z(\psi, \theta) = E(\psi) \left[D_1(\psi) \sin \theta - \sum_{n=2}^N D_n(\psi) \sin n \theta \right]$$

and that the D_n amplitudes converge quickly. It is possible to fit a wide class of experimentally relevant and numerically exact equilibria with $N < 3$. It has also been found that the values of D_n and E can be accurately

represented with three terms of the series

$$D_n(\psi) = \psi^{n/2} \sum_{m=0}^n D_{nm} p^{[n-(1/2), -1/2]} \frac{\psi}{\psi_{\max}}$$

where $p^{[n-(1/2), -1/2]}$ is a Jacobi polynomial.

3.2.7 Transport Simulation

High beta and plasma-wall interaction

Abstract of The accessibility of high-beta tokamak states.⁶² (J. T. Hogan) Encouraging results with neutral beam heating and adiabatic compression of tokamak plasmas have prompted new experiments that will study the approach to high-beta states. As projected tokamak beta values become nonnegligible (average beta of 4% is the goal), the models previously used for transport calculations will become inadequate. The models will then be required to account for the evolution of the magnetic geometry as well as for the change in plasma parameters. An axisymmetric transport model is presented that should be useful for studying the approach to higher beta values in tokamak experiments. Results from transport calculations with this model allow an analysis of the limits to beta in two seemingly unrelated experiments: (1) electron heating by neutral injection in the Oak Ridge Tokamak (ORTOK) and (2) adiabatic compression in the adiabatic toroidal compression (ATC) experiment. By extrapolating empirical models for plasma transport, it is found that flux-conserving equilibria ought to be accessible by neutral beam heating. Finally, it is found that the nature of cross-field transport may be expected to change as significant beta values are reached. Enhanced transport from ballooning instabilities is likely to play a role as important as that now played by sawtooth ($m = 1$) and saturated ($m = 2$) instabilities. New techniques for describing this transport are required.

Abstract of Beta limits in beam-heated tokamaks.⁶³ (J. T. Hogan) Consideration is given to some practical features associated

with beta optimization in tokamaks employing neutral beam heating. Previous ideal MHD stability calculations of beta limits have used ad hoc pressure and current density profiles. We use a coupled 1½-D transport ballooning eigenmode model to simulate the beam deposition and heating process, including neutral gas and impurity effects. With this model we find that:

1. beta values exceeding the ideal MHD limits have been obtained,
2. strong effects of direct beam contributions to the pressure or indirect effects of strong local beam heating can outweigh the relatively remote vacuum field shaping in well-penetrated discharges, and
3. beam-induced currents can produce significant changes in transport by changing the location and width of islands produced by saturated resistive tearing modes.

Abstract of General principles of magnetic fusion confinement.⁶⁴ (J. T. Hogan) In these lectures, we describe a few of the areas in which there is close interaction between magnetic fusion physics and atomic and molecular physics. The comparisons between predictions of neoclassical transport theory and experimental results depend on knowledge of ionization and recombination rate coefficients. Modeling of divertor-scrapeoff plasmas requires better low-energy charge exchange cross sections for $H + A^{n+}$ collisions. The range of validity of neutral beam trapping cross sections must be broadened, both to encompass the energies typical of present injection experiments and to deal with the problem of prompt trapping of highly excited beam atoms at high energy. Plasma fueling models present certain anomalies that could be resolved by calculation and measurement of low-energy (<1-keV) charge exchange cross sections.

Abstract of Models for impurity effects in tokamaks.⁶⁵ (J. T. Hogan) Impurity effects will, along with limits to attainable β , fix the plasma parameters of tokamak reactors. There has been recent progress in calculating

the effects of impurities on tokamak plasmas, so that detailed needs for more accurate modeling have become apparent. Many experiments have demonstrated impurity transport with neoclassical features. Competing classical processes could account for the remaining discrepancies with theory; such processes include charge exchange of multiply charged impurities with hydrogen and MHD transport owing to resistive modes (e.g., reconnection associated with sawteeth or electrostatic barriers created by large-scale islands).

Models for the sources of impurities are still sketchy. Physical sputtering of wall-limiters by hydrogenic neutrals can be calculated; the uncertainty in the sputtering yields of typical materials under the low-dose, prompt-response conditions of present high-power neutral beam experiments limits the usefulness of such calculations. Arcing is suspected to play an important role; there are some theoretical and experimental reasons to believe that discharge initiation produces most such activity. Quantitative models for arcing effects have begun to be used.

Gradual accumulation of impurities also diminishes the useful pulse length in a high-beta plasma by enhancing the resistive decay of flux-conserving tokamak (FCT) equilibria. As the q profile evolves from its FCT value, transport at the sound speed to walls-limiter-divertor reduces the plasma β to pre-FCT values.

Summary of conference paper: Atomic and molecular physics processes in magnetic fusion research.⁶⁶ (K. Dolder,* J. T. Hogan) A NATO advanced study institute was held recently at Chateau Bonas, France, on the topic "Atomic and Molecular Processes in Controlled Fusion Research." Under the direction of Profs. M. R. C. McDowell (London) and A. M. Ferendeci (Istanbul), the institute provided an opportunity for physicists working

in magnetic fusion research (chiefly with tokamak devices) to survey the state of a long-standing collaboration of atomic and molecular physicists.

Atomic and molecular processes have always played an important role in magnetic fusion plasma research. As hotter and denser plasmas have been produced, the low-temperature region typically governed by atomic and molecular processes has been steadily pushed toward the exterior and away from the hot core of the plasma. Somewhat paradoxically, however, the importance of these processes has grown. There are several reasons for this.

1. The edge region, where such processes always dominate the transport, must be controlled in order to direct the large power flow in modern devices toward a manageable disposal point — the divertor in toroidal systems or a direct convertor in a mirror. In addition, the use of cold gas to fuel long-pulse discharges presents anomalies: electron density profiles measured during gas puffing show qualitatively different shapes than do the inverted profiles predicted by standard models. Study of the governing atomic and molecular processes in the plasma corona has shown that charge transfer reactions of the form $H^0 + A^{n+} \rightarrow H^+ + A^{(n-1)+}$ have quite large cross sections in the energy range 10 eV–1 keV. If the cross sections are found to be as large as simple estimates and extrapolations suggest, cold gas puffing should not produce as large an electron density increase as predicted; rather, it should depress the ionization stage of such ubiquitous impurities as oxygen, leading to enhanced radiative loss and eliminating the inverted profile anomaly.

2. The feeling has grown that beam heating is a well-understood and predictable process, in sharp contrast to the complexity and perplexity of many wave heating schemes. This feeling has given way to some anxiety recently, however, as cross-section measurement and calculation have disclosed an important interaction between fast injected hydrogenic neutral atoms ($E_0 \sim 20\text{--}60$ keV/amu) and the

* University of Newcastle, United Kingdom

multiply charged, incompletely stripped ions found in the plasma. It has been shown that the processes of electron capture from an impact ionization of H^0 by multiply charged ions are just as effective in trapping the injected beam as are the reactions considered earlier, namely, $H_{fast}^0 + H_{thermal}^+ \rightarrow H_{thermal}^0 + H_{fast}^+$ and $H_{fast}^+ + H_{thermal}^+ \rightarrow e$. Not only does this impurity reaction stop the beam in unpredictable ways, but the electron transfer by this reaction is a very powerful recombination mechanism, and the beam impurity recombination process dominates by orders of magnitude the dielectronic and radiative recombination processes operative in high-temperature plasmas. Adding more beam power depresses the ionization stage of the impurities even further and leads to higher radiative losses, thus tending to defeat the already relatively weak electron heating provided by neutral beams.

3. Experiments in a variety of tokamaks (PLT, T-10, ORMAK, TFR, Alcator) have shown that the high-temperature core plasma region can be dominated by heavy-metal impurities. Limiter elements, such as tungsten or molybdenum, in trace amounts ($\sim 0.1\%$ of plasma densities) can directly radiate away all the input power. These radiative losses have produced hollow temperature profiles in ohmic heating experiments and have reduced the electron heating in high-power neutral beam experiments.

Simulation of tokamak experiments

Abstract of Comparison of ISX-B neutral beam heating experiments with theory.⁶⁷ (J. T. Hogan; W. A. Cooper; J. K. Munro, Jr.) Injection of up to 1 MW of neutral beam power into an approximately circular ISX-B plasma has produced values of $\beta^* [\equiv 2\mu_0 (\int dV p^2)^{1/2} / B_z^2]$ up to 3%. Results have been obtained in the density range $1.5 \times 10^{13} < \bar{n}_e \text{ (cm}^{-3}\text{)} < 4.9 \times 10^{13}$. At the higher densities, the beam pressure is $\sim 30\%$ of that attributable to the thermal background, but it can be double the thermal level in the low-

density cases. The β^* in the high-density cases exceeds the β_{crit}^* predicted for low mode numbers ($n = 1, 3$) by the PEST code, whereas the low-density cases show a broad region of instability to high- n modes in the plasma interior. These latter cases are calculated with a tensor pressure equilibrium code and a high- n ballooning eigenmode code developed by R. G. Bateman. The zone of central instability grows with increasing beam power, first arising at $P_b \sim 300$ kW. At high density, the high- n unstable region occurs on the plasma exterior and is a result of broadened beam deposition profiles.

Abstract of High-beta tokamak transport modeling studies.⁶⁸ (J. T. Hogan) The attainment and maintenance of a high-beta plasma ($\beta > 4\%$) by neutral beam injection is an important goal of current tokamak experiments. We discuss work in progress on a number of modeling problems suggested by such experiments, including optimization of the configuration by beam deposition and current programming, the role of neutral-beam-induced currents in determining the shear profile, possible use of Fisch-Bers lower-hybrid current drive for tailoring profiles,⁶⁹ and properties of the final stage of FCT evolution.

As part of this study we have modified our 1½-D transport code to:

- calculate high-beta limits self-consistently by using the Bateman-Nelson ballooning code⁷⁰ within the transport calculation;
- calculate lower-hybrid power dissipation and current generation using a module developed by D. Ehst;⁶⁹ and
- calculate magnetic reconnection of nonmonotone current profiles and magnetic island effects owing to saturated tearing instabilities using criteria developed by Waddell, Carreras, and Hicks.

As before, beam deposition is calculated with the Fowler-Rome noncircular version of the FREYA module developed by D. Post. We find transport-generated D-shaped plasmas with $\beta > 10\%$ that are self-consistent and stable (stable has been taken to mean linearly unstable over less than 20% of

the plasma volume). We also find that:

- The TOSCA results do not violate the ballooning criterion.
- Benefits for ballooning stability from force-free currents (from external voltage programming or lower-hybrid current drive) may be counterbalanced by tearing-mode-induced enhanced transport.
- The Zakharov-Shafranov catastrophe⁷¹ (loss of equilibrium on the post-FCT time scale) should not be accessible in the near future.
- Neutral-beam-induced currents completely determine the shear profile at low density for coinjected plasmas and thus remove some of the profile tailoring capability. Perpendicular injection restores it, but a study of anisotropic pressure equilibrium and stability is needed for this case.⁷²

Hydrogen recycle modeling

H. Howe

A realistic hydrogen recycling model may be used with a 1-D transport code to estimate fueling requirements for long-pulse experiments, to calculate tritium buildup in the wall of devices such as the ETF and the Tokamak Fusion Test Reactor (TFTR), and to model isotopic changeover experiments. In the model, the charge exchange neutrals from the plasma that are not reflected by the wall are deposited in the wall with a depth distribution dependent on incident energy. These particles then diffuse according to Fick's law and may reach the surface, where they recombine and desorb. An example of this process is shown in Figs. 3.12 and 3.13, where the source term in the wall and the evolution of the wall concentration are depicted for an ISX-C simulation. The neutral beam is turned on at 500 ms. The increase of T_i owing to injection results in a hardening of the charge exchange spectrum at the wall and in deeper penetration of the neutrals into the walls. The desorption rate decreases relative to the deposition rate because of the deeper neutral penetration (Fig. 3.14). Thus, injection heating of the plasma lowers the recycle

Fig. 3.12. Rate of deposition in the wall of charge exchange neutrals from a simulated ISX-C plasma. Beam heating begins at 500 ms.

Fig. 3.13. Evolution of wall concentration.

Fig. 3.14. Ratio of desorption to deposition rate decreases owing to plasma heating.

rate and increases the fueling rate required to maintain the density. This effect may be involved in the density clamping observed in ISX-B. In ISX-C simulations, we find that several seconds are needed for the wall to approach saturation. This, one of several interesting effects that will be studied in future long-pulse tokamaks.

Abstract of A guide to making time-lapse graphics using the facilities of the National Magnetic Fusion Energy Computing Center⁷³

J. K. Munro, Jr.

The advent of large, fast computers has opened the way to modeling more complex physical processes and to handling very large quantities of experimental data. The amount of information that can be processed in a short period of time is so great that use of graphical displays assumes greater importance as a means of dis-

playing this information. Information from dynamical processes can be displayed conveniently using animated graphics. This guide presents the basic techniques for generating black and white animated graphics, considering the aesthetic, mechanical, and computational problems. The guide is intended for use by someone who wants to make movies on the NMFECC CDC-7600. Problems encountered by a geographically remote user are given particular attention. Detailed information is given that will allow a remote user to do some file checking and diagnosis before giving graphics files to the system for processing into film in order to spot problems without having to wait for film to be delivered. Source listings of some useful software are given in appendices along with descriptions of how to use it.

Abstract of HLYWD: A program for postprocessing data files to generate selected plots or time-lapse graphics⁷⁴

J. K. Munro, Jr.

The program HLYWD is a postprocessor of output files generated by large plasma simulation computations or of data files containing a time sequence of plasma diagnostics. It is intended to be used in a production mode for either type of application; that is, it allows one to generate along with the graphics sequence (1) segments containing title, (2) credits to those who performed the work, (3) text to describe the graphics, and (4) acknowledgement of funding agency. The current version is designed to generate 3-D plots and allows one to select type of display (linear or semilog scales), choice of normalization of function values for display purposes, and viewing perspective. It also offers an option to allow continuous rotations of surfaces. This program was developed to be relatively easy to use and reasonably flexible and to require a minimum investment of the user's time. It uses the TV80 library of graphics software and ORDERLIB system software on the CDC-7600 at the NMFECC at Lawrence Livermore Laboratory in California.

3.2.8 Plasma Systems Analysis and Divertor Studies

The plasma systems activities primarily involve development of physics models for tokamak reactors and evaluation of these models in terms of technological requirements. The areas of emphasis this past year included (1) development of divertor physics models and assessment of the impact on pellet and gas fueling requirements in a reactor, (2) investigation of the physics of burn control and associated instrumentation needs, and (3) a physics evaluation of large-aspect-ratio tokamaks for achieving operation characterized by a very long pulse.

Abstract of The electric sheath at a limiter or divertor collector.⁷⁵ (J. W. Davidson, G. A. Emmert, A. T. Mense, R. M. Wieland) The plasma sheath equation for a collisionless plasma with arbitrary ion temperature in plane geometry is formulated. Outside the sheath, this equation is approximated by the plasma equation, for which an analytic solution for the electrostatic potential is obtained. In addition, the ion distribution function, the wall potential, and the ion energy and particle flux into the sheath are explicitly calculated. The plasma sheath equation is also solved numerically with no approximation of the Debye length. The numerical results compare well with the analytical results when the Debye length is small.

Abstract of Computer model for the scrapeoff layer of a bundle divertor.⁷⁶ (A. T. Mense, A. Nicolai) A set of six equations is presented governing plasma parameters (hydrogen ion temperature) within the burial chamber of a bundle divertor and the potential steps in the divertor throat and in front of the divertor plate. Secondary electron emission at the divertor plate, the particle and energy flow reduction owing to the mirror, and the backflow of impurities into the scrapeoff plasma are taken into account. The modeling is based mainly on the assumption that strong relaxation processes

resulting from the microinstabilities arising because of the mirror and the secondary emission decrease the deviations from a Maxwellian distribution function to a negligible level in both the scrapeoff region and the divertor chamber. Particle sink and energy loss terms attributable to the divertor are derived for 1-D transport code calculations. In this way, the plasma parameters in the scrapeoff region can be calculated consistently with those of the central core and the divertor chamber. Therefore, one can account for the influence of additional heating, gas puffing, pellet injection, and so forth on the plasma parameters in the scrapeoff region as well as in the divertor chamber.

Abstract of Deuterium and tritium fueling in a reactor with divertor.⁷⁷ (S. E. Attenberger, W. A. Boulberg, H. C. Howe) Fueling by pellets and neutral gas in the presence of a divertor is examined with a 1-D multispecies transport code. Deuterium, tritium, and alpha particles are treated as independent thermal species. With an efficiently operating divertor, it becomes impossible to maintain high plasma density ($\bar{n} \sim 10^{14} \text{ cm}^{-3}$) with neutral gas fueling alone because of the high probability of the gas being ionized in the scrapeoff layer. Pellet fueling significantly reduces the particle load on the divertor and, with feedback control, can maintain the plasma density at the desired level. A low level of deuterium gas fueling can then be used to maintain the density of the scrapeoff layer and increase shielding against sputtered impurities. Even with an effective shielding divertor, the energy and particle fluxes to the first wall from both charged and neutral particles may be significant. The fluctuations at the plasma edge and in the scrapeoff layer introduced by the pellets cause the particle and energy fluxes to the divertor and the first wall to fluctuate. This makes simulation more difficult but may be used to experimentally determine radial and parallel transport properties in the scrapeoff layer.

Abstract of Physics of burn control in tokamak reactors.⁷⁸ (S. E. Attenderger, W. A. Houlberg) We use a 1-D multifluid transport code to investigate some of the critical physics considerations for long burn times in a tokamak fusion reactor. Among these are the effects of helium ash buildup, toroidal field ripple, and fueling in the presence of a divertor. Particle and energy transport models are based on extrapolations from present day devices.

Figure 3.15 shows a typical startup and burn for a reactor with a rather small ripple. A somewhat larger ripple may be tolerated by increasing the density. This reduces the ignition temperature, thereby reducing the ripple diffusion at ignition. The machine parameters used are those of INTOR, namely, major radius = 4.8 m, minor radius = 1.2 m, elongation = 1.5 m, toroidal field on axis = 5.0 T, plasma current = 4.0 MA. A 150-keV deuterium beam was injected at 12° from perpendicular.

Figure 3.16 shows four possible burns corresponding to different assumptions about

Fig. 3.15. INTOR base model plus toroidal field ripple.

Fig. 3.16. INTOR helium and buildup.

alpha containment. In all cases the electron density $\langle n_e \rangle$ is held constant by feedback control. For $\langle n_e \rangle = 10^{14} \text{ cm}^{-3}$ and no thermal alphas, a steady state is reached. However, if no thermal alphas escape [$\Gamma_a(a) = 0$] the plasma reactivity is reduced as n_a increases, and the burn quenches within about 20 s. The plasma may also quench when the alphas are allowed to diffuse from the system, as shown by the curve corresponding to a gradient boundary condition on the alpha density. Increasing the plasma density to $\langle n_e \rangle = 1.2 \times 10^{14}$ provides a greater margin for ash buildup, and a steady state may again be reached as shown by the solid curve.

Abstract of Controls and instrumentation of a power-producing tokamak.⁷⁹ (W. A. Houlberg) Plasma physics studies indicate that feedback control of plasma density can be accomplished primarily by pellet injection rather than by

gas puffing. Progress continues to be made toward understanding the thermal consequences of a major plasma disruption with respect to the first wall. In addition, the parameters used in the feedback control system of ISX-B have been more clearly defined. Discussions are presented concerning (1) diagnostics suitable to ETF and (2) future research and development needs. Emphasized are some preliminary ideas about the measurement of the deuterium-tritium ratio in the plasma, synchrotron radiation, and divertor control. Finally, the thermal consequences to the first wall from a major plasma disruption are documented.

Large-aspect-ratio tokamaks. (J. A. Holmes, W. A. Boulberg, Y.-K. M. Peng, D. J. Strickler) An aspect ratio of 8 was chosen as representative of large-aspect-ratio tokamaks (LARTs) for more detailed plasma physics and configurational analysis.⁶⁰ This design represents a reasonable compromise between increased burn time and increased cost as aspect ratio is increased at a constant power level. The selected configuration achieves a burn time of 3.5 n, which reduces the number of burn cycles by at least an order of magnitude compared with a TNS type reactor but at an increase of approximately 30% in unit capital cost.

Neutral beam heating was applied for 4 s. The neutral beam power was varied to determine the minimum requirements for ignition, and after ignition was achieved, the steady-state burn parameters were determined as a function of plasma density. It was found that the ripple losses in LARTs were large enough to prevent ignition at acceptable values of beta, at a plasma density $< 10^{14} \text{ cm}^{-3}$, and at a beam power of 100 MW. A 25% reduction was made in the ripple (from 1.12 to 0.85% at the plasma edge) so that suitable burn regimes could be found. The reduction in ripple for LARTs would, in practice, require modifications to the toroidal magnetic field design; even further reduction may be required if other ripple phenomena prove to be significant.

Figure 3.17 shows fusion power, volume-averaged

Fig. 3.17. Steady-state parameters for a large-aspect-ratio tokamak as a function of plasma density.

beta, and average electron and ion temperature as a function of plasma density for steady-state burn.

Tokamak plasma variations under adiabatic compression to small aspect ratios. (Y.-K. M. Peng, J. A. Holmes, D. J. Strickler, S. J. Lynch*) The change in tokamak plasmas undergoing large adiabatic compression in major radius have been examined numerically over the range of aspect ratios: $1.5 \leq A \leq 3$ in circular, elliptic, and D-shaped cross sections. The numerical approach combines the computation of fixed boundary FCT equilibria and single-fluid, flux-surface-averaged energy and particle balance equations. The fixed boundary approach allows a precise prescription of the plasma boundary position and shape. During compression, the minor radius (a) is adjusted

* 166 Lincoln Ave., Haverhill, Massachusetts

iteratively to ensure the invariance of $\psi_{\text{edge}} - \psi_{\text{axis}}$, $q(\psi)$, and $F(\psi_{\text{edge}})$. It is found that the dependences of I_p (plasma current) and β_p (poloidal beta) on C (the compression ratio) differ significantly from those proposed by Furth and Yoshikawa,⁸¹ whereas the dependences of α , $\bar{\beta}_t$ (averaged toroidal beta), and P (pressure) show a milder difference.

The present interpretation is that compression to small A dramatically increases the plasma current, which lowers β_p and makes the plasma more paramagnetic. Despite the large $\bar{\beta}_t$ values ($\geq 30\%$ with $q_{\text{axis}} \approx 1$, $q_{\text{edge}} \approx 3$), this tends to concentrate more toroidal flux toward the magnetic axis, which requires reduced minor radius to preserve the continuity of I at the plasma edge. For D-shaped plasmas with mild elongation (1.6), the averaged vertical field decay index $[(\partial \ln B_z)/(\partial \ln R)]$ on the midplane changes from -0.7 to +0.2 as A is reduced from 3 to 1.5. This indicates that the vertical stability of the D-shaped plasma column is enhanced with decreasing A .

Abstract of Plasma heating to control magnetic flux diffusion.⁸² (S. J. Lynch,* J. A. Holmes, Y.-K. M. Peng) In the operation of a tokamak fusion reactor, it will be necessary to control the evolution of the safety factor profile over a pulse duration to ensure MHD stability. In an ETF a pulse lasts hundreds of seconds, a length similar to or larger than the resistive time scale that governs the evolution of the q profile. We study numerically the evolution of the q profile on the resistive time scale using a 1½-D single-fluid transport code. In particular, we examine this evolution as a function of the plasma heating deposition profile. A Monte Carlo neutral beam calculation is used to obtain the deposition profile, which can be varied by changing the neutral beam energy, the plasma density, or both. It is shown that hollow (or centralized) heating tends to increase (or

decrease) the q value at the plasma center and that the evolution of the q profile can be controlled by controlling the plasma heating deposition profile.

Abstract of Low-density buildup ignition scenarios in tokamak reactors using neutral injection heating.⁸³ (J. A. Holmes, J. A. Rome, Y.-K. M. Peng, W. A. Boulberg, S. J. Lynch*) Plasma heating and ignition by neutral injection have been studied using a Monte Carlo neutral injection computer code coupled to a single-fluid, 1-D transport code and a 2-D, flux-conserving equilibrium code. It is shown that by taking advantage of central alpha heating, profile effects, and flux surface shifts in elongated plasmas, it is possible to ignite a modeled, prototypical reactor plasma using 45-30 MW of 100-150-keV (D^+) neutral beams. To do this, the plasma is started at full bore but low density. The density is then increased by peripheral fueling so that the central core begins to ignite before the neutral beams no longer penetrate this region. The fusion alpha particles take over the heating requirements in the core region. Because of the decreasing beam line efficiency with increasing energy, it is found that a nearly constant extracted power of about 85-95 MW is needed for ignition in the range studied. Thus, there is little economic difference in this energy range. Higher energy beams (around 150 keV), however, imply fewer injectors and perhaps lower impurity production rates during heating to ignition.

Abstract of Evolution of tokamak equilibria with preprogrammed cross section.⁸⁴ (J. A. Holmes, Y.-K. M. Peng, S. J. Lynch*) The evolution of MHD equilibria toward high beta is modeled by magnetic flux conservation with a given $q(\psi)$ and by particle and energy balances that determine $p(\psi, t)$. One-dimensional, single-fluid transport equations, written with the magnetic flux ψ as the independent variable,

*166 Lincoln Ave., Haverhill, Massachusetts

*166 Lincoln Ave., Haverhill, Massachusetts

are coupled to the 2-D axisymmetric MHD equilibrium and flux conservation equations through ψ and $p(\psi, t)$. In moving boundary studies (plasma compression), the resulting system of equations is advanced in time from an initial state by a procedure that uses two nested predictor-corrector loops together with an implicit time-stepping technique. The inner predictor-corrector loop advances the transport equations according to a given equilibrium configuration, and the outer loop evolves the equilibrium using a flux-conserving code with fixed plasma boundaries. For fixed boundaries, this procedure is modified for greater computational speed. Our results show satisfactory quality in numerical convergence. The use of the fixed boundary equilibrium technique allows the evolution of the plasma cross section to be programmed. This method can be applied to the study of flux-conserving evolution of equilibria involving dramatic changes of plasma position, shape, and profiles while prescribing the evolution of the plasma boundary. As an example, the compressional scaling laws of Furth and Yoshikawa are modified for small aspect ratio.

3.2.9 Advanced Fuels

Abstract of Fusion reactivity graphs and tables for charged particle reactions⁸⁵

J. Rand McNally, Jr.; K. E. Rothe; R. D. Sharp

Graphs and tables are presented for 31 light isotope fusion reaction parameters: $\langle\sigma v\rangle$, $n\langle\sigma v\rangle$, $\langle\sigma v\rangle Q_+$, $n\langle\sigma v\rangle Q_+$ ($n = 10^{20}$ fuel ion species per m^3 and Q_+ = energy release in charged particles) in the kinetic temperature range 1-1000 keV.

Abstract of Some thoughts on the $p(^6\text{Li}, \alpha)$ $^3\text{He}(^6\text{Li}, p)$ ^8Be chain reaction⁸⁶ (J. Rand McNally, Jr.; K. E. Rothe) Recently, R. W. Conn (University of Wisconsin) has revived⁸⁷ the prospect of a charged particle fusion chain reaction involving ^6LiH fuel,⁸⁸ which might be suitable for use in multipole configurations or other low central magnetic field devices. A completely catalyzed reaction ($p + ^6\text{Li} \rightarrow \alpha + ^3\text{He} + 4.0 \text{ MeV}$) + $^6\text{Li} \rightarrow \alpha + p + ^8\text{Be} + 20.9 \text{ MeV}$

would promote fast or suprathermal ^3He and p ions, which would react in propagation chain fashion⁸⁷ with ^6Li fuel supplemented by chain branching reactions involving $p + ^6\text{Li}$, $^6\text{Li} + ^6\text{Li}$, $^3\text{He} + ^6\text{Li}$, and so on. The two primary mechanisms for propagation chain reaction release of energy give power production (the $^6\text{Li}^6\text{Li}$ reactions involve six exothermic channels)

$$p_{16} = n_1 n_6 \langle\sigma v\rangle_{16} \frac{Q_{16} + p_{36} Q_{36}}{1 - p_{16} p_{36}}$$

and

$$p_{36} = n_3 n_6 \langle\sigma v\rangle_{36} \frac{Q_{36} + p_{16} Q_{16}}{1 - p_{36} p_{16}}$$

where the subscripts 1, 3, and 6 denote p , ^3He , and ^6Li ; $Q_{16} = 4.9 \text{ MeV}$; $Q_{36} = 16.9 \text{ MeV}$; and p_{16} and p_{36} are the reaction probabilities of the fast ions with ^6Li as they slow down on plasma ions and electrons. Neglecting nuclear elastic collisions (which promote suprathermal ions), the probabilities p_{ij} can be written (provided $\tau_s \ll$ the particle confinement time)

$$p'_{i6} = \int_{E_i}^{(3/2)T_i} n_6(\sigma v)_{i6} dt$$

$$= \int_{E_i}^{(3/2)T_i} n_6(\sigma v)_{i6} \frac{dE}{E} dt$$

where

$$p_{i6} = 1 - e^{-p'_{i6}}$$

and

$$\frac{dE}{dt} = - \frac{2[E - (3/2)T_e]}{\tau_s} - \frac{2[E - (3/2)T_i]}{\tau_s} \left(\frac{E_c}{E}\right)^{3/2}$$

where

$$\tau_s = \frac{1.2 \times 10^{12} T^{3/2} A_i}{n_e Z_i^2}$$

and

$$E_c = 14.8 T_e A_i \left[\frac{\sum (n_{ions} Z_{ions}^2 / A_{ions})}{n_e} \right]^{2/3}$$

Synchrotron-free burn calculations have been performed, and ignition does not occur. Figure 3.18 illustrates the importance of the fast proton reactions ${}^6\text{Li}(p, {}^3\text{He}) \alpha + 4.0 \text{ MeV}$ and ${}^6\text{Li}(p, p') {}^6\text{Li}^*(2.2 \text{ MeV}) \rightarrow d + \alpha - 1.5 \text{ MeV}$ for several electron temperatures as well as the σv parameters versus proton energy E_p . We do not

find ignition of $p{}^6\text{Li}$, contrary to the findings of Shuy and Conn.⁸⁷ Also, we find a significant neutron production rate from ${}^6\text{Li}({}^6\text{Li}, n)\alpha + {}^7\text{Be}$.

3.3 COMPUTER SUPPORT

C. E. Harmons

The computer support group provides the Fusion Energy Division with computing resources in two areas: operation of the User Service Center and development of the experimental data acquisition network. The User Service Center

Fig. 3.18. Fast proton reactions with ${}^6\text{Li}$.

provides (1) access to the large computing systems at the NMFEC in Livermore and (2) immediate analysis and storage of experimental data. The data handling network supports the growing needs of the Division's experimental efforts by providing a rapid, state-of-the-art means of acquiring and storing experimental data.

3.3.1 User Service Center

*C. E. Barrera, D. W. Clark, G. S. Macrengill,
C. E. Parker*

The latest version of the DECsystem 10 operating system was merged with NMFEC software and made operational. This has improved the reliability of the operating system in the User Service Center (USC). The ac power line conditioners from Deltec were installed in March and have provided relative immunity from pulses created by the Division's experiments, which have been a problem in the past. The isolation provided, plus the conditioned, regulated power, has further enhanced the reliability of the USC.

The 256K core Ampex memory, procured through competitive bid, was installed and made fully compatible with the KL10 memory bus. This has provided some relief from excessive swapping, but the user community is growing at roughly the same rate as the capabilities, so we continue nearly at capacity. During 1979, we also procured several terminals to provide the Division staff with better access to the USC, and an RPO6 disk system was added to provide data storage for modest user growth and to accommodate the new requirements of the beam development group.

The KL10 processor was upgraded to the model B processor. This allows a larger amount of memory to be added as needed, and it increases the processor power by 20-30%. Future releases of the KL10 microcode will be usable as a result of this upgrade.

The PDP 11/45 software was made fully operational as a front-end processor for the USC. The PDP 11/45 is connected via its UNIBUS to the DECsystem 10, which provides a

one-megabyte data path between the systems. This allows the PDP 11/45 to accept spooled data from the USC for printing and plotting while routing data from acquisition computers attached to the Division's experiments.

A Qume printer with a double print wheel was added to the USC. The unit was assembled from the basic printer and was connected to a microprocessor, which in turn was connected to a standard communications port. This was accomplished through cooperation with personnel at the Los Alamos Scientific Laboratory (LASL) USC.

3.3.2 Data Handling

EBT data

*R. D. Burris, J. E. Francis, R. A. Geoffroy,
D. E. Greenwood, J. S. Stanton*

Considerable improvements to EBT computing support were made in 1979 in all areas, including data acquisition, experiment control, computer-to-computer communications, and the usability and availability of the computers.

Improvements in data acquisition and experiment control have resulted from the decision to support all future diagnostics with CAMAC hardware and FORTRAN software and to convert existing diagnostics as time and personnel permit.

A Nuclear Data 6600 computer-based analyzer to be used to monitor soft x-ray signals has been purchased and installed. Hardware and software to permit transfer of data from that computer to others in the Division is now under development.

A plan to upgrade computer-to-computer communications for EBT has nearly been completed with the purchase and installation of a new computer — a PDP 11/34. The chief purpose of this computer is to permit very fast, low-overhead, easily accessible data transfer from EBT to the Division network.

The availability and usability of the PDP 8 computers have been improved through the purchase

of a DECTape for the PDP 8/a, permitting the convenient physical interchange of files between the two PDP 8s, and through the addition of CAMAC support to both computers.

ISX data

*D. R. Overbay; J. E. Francis, Jr.; K. A. Stewart;
R. J. Burris; E. T. Blair*

The ISX PDP 11/34 was expanded by the addition of 128 ADC CAMAC data channels, a one-megabit communications link to the central PDP 11/45 and a color acquisition computer status monitor.

Velocity filter data acquisition routines were developed and installed on an ISX PDP 8. The Thomson scattering support programs were enhanced by the addition of temperature and density profile outputs. Similarly, new programs were developed for the infrared camera diagnostic, which allows for 74 scan lines and 110 data points per scan.

Data acquisition routines for the PIN and MHD monitors were developed for a PDP 11/34 system. Selected shots are saved and analyzed further on the USC DEC 10. Development work has begun on the Interdata 8/32 computer system to support Thomson scattering and PIN-MHD monitors. Support was initiated for the new coated limiter experiment.

Medium energy test facility and high-power test facility data

C. R. Stewart; J. E. Francis, Jr.

Most of the initial goals for neutral beam data acquisition and analysis were achieved. These include simultaneous acquisition, display, and printing of data on the Medium Energy Test Facility (METF) PDP 11/40. Transmission capability to the USC DEC 10 was installed, comprising up to 32 calorimeter channels and 14 channels of waveform data per shot. This capability is provided as a switch-selectable option for the experimenter. A data base is maintained on the USC DEC 10 for further analysis. Older data are stored on magnetic tape as an archive. Analysis programs for the neutral beam data stored on

the USC DEC 10 include those for calculating plot perveance, reliability, and power loading.

All the METF routines were also written to support data acquisition on the High-Power Test Facility (HPTF). A room shielded from radio frequency (rf) emissions is being acquired for the HPTF data acquisition computer to protect the system from the rf noise generated by the beam source.

3.3.3 Data Communications

C. O. Kemper, R. D. Burris, D. R. Overbay

The following enhancements were made to the data acquisition network:

- Printer-plotter front-end support was added to the PDP 11/45 to off-load the USC DEC 10 device support.
- Five one-megabit data links between the PDP 11/45 and the data acquisition computers were added (LAB, MET, ISB, ICL, and SCM).
- The old DIVA disk system on the PDP 11/45 was replaced by a larger and faster RP04. The additional space is used for data spooling.
- A dual RK05 cartridge disk system was added to the system (PDP 11/45) to enhance data transport, backup, and system support capabilities.
- The system's communications software has been extensively modified in the interests of speed, flexibility, and transportability.

3.3.4 Graphics

W. H. Gray, R. D. Burris, K. A. Stewart,
D. H. Clark*

The new PLOT command was implemented, giving USC users graphics capability on all graphics devices at ORNL. A new intermediate graphics file format called TKF was implemented into the PLOT command. This format is an ASCII image of the character stream used to control a Tektronix terminal. Several improvements were made to the system software that

*Magnetics and Superconductivity Section

performs the vector-to-raster conversion for the USC's Versatec printer-plotter. This new software executes three times as fast and has a sharable high segment. The TCSTEK software library (the basic Tektronix graphics library) was enhanced to automatically access the terminal's transmission speed code. Also, a new synchronizer for terminal input/output that improves page erasure and hard-copy operation was replaced. The FICHE command interface was written to spool both ASCII and graphics files for output to the FR80 at the Oak Ridge Gaseous Diffusion Plant. This relieved data traffic on the link.

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4. ATOMIC, MOLECULAR, AND NUCLEAR PHYSICS

C. F. Barnett ¹	H. J. Kim ¹
C. Bottcher ¹	M. I. Kirkpatrick ¹
D. H. Crandall ¹	E. W. McDaniel ⁵
B. J. Farmer ¹	R. H. McKnight ⁶
H. B. Gilbody ²	F. W. Meyer ¹
D. C. Gregory ³	T. J. Morgan ⁷
S. W. Hathorne ¹	R. A. Phaneuf ¹
R. H. Hughes ⁴	E. W. Thomas ⁵

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1. Physics Division
 2. Consultant, Queen's University of Belfast, Belfast, Northern Ireland
 3. Participant from the Joint Institute for Laboratory Astrophysics, Boulder, Colorado
 4. Participant from the University of Arkansas, Fayetteville, Arkansas
 5. Consultant, Georgia Institute of Technology, Atlanta, Georgia
 6. Consultant, California State Polytechnic University, Pomona, California/National Bureau of Standards, Gaithersburg, Maryland
 7. Consultant, Wesleyan University, Middletown, Connecticut

4. ATOMIC, MOLECULAR, AND NUCLEAR PHYSICS

ABSTRACT. Atomic physics activities encompass (1) experimental studies of basic atomic collision processes of interest to fusion research, (2) some theoretical studies of these processes, and (3) categorization and compilation by the data center of fusion-related atomic data from the literature.

During 1979, we continued cross-section measurements using crossed ion and atom beams to determine total electron loss from collisions of 40-keV H^0 with N^{3+} ions. The apparent cross section is dominated by the small fraction of reactant H^0 atoms formed in highly excited Rydberg states. Our experimental study of electron impact ionization of multicharged ions has been extended by converting the measured cross sections to Maxwellian rate coefficients. A simple form of the Lotz formula has been found to be in reasonable agreement with our measurements. We have studied formation and destruction of H^- in cesium. The "slow-encounter" theoretical model has been shown to predict electron detachment cross sections for collisions of H^- with cesium in reasonable accord with experiment. Electron capture and detachment cross sections for hydrogen projectiles in cesium vapor have been measured down to an energy of 100 eV/amu and join smoothly with previous measurements. The H^- equilibrium yields deduced from these cross sections agree, within uncertainties, with previous Oak Ridge National Laboratory (ORNL) measurements, having a peak value of 26% at 250 eV/amu. Low-energy electron capture measurements for Ar^{+q} and Xe^{+q} ions in atomic hydrogen suggest a cross-section variation with charge q for $q > 9$ that is less than the linear scaling predicted by generalized theories. For ions of a given charge q at fixed velocity, a dependence on the number of electrons remaining on the ion was also observed, with the more highly stripped ions having smaller cross sections. The laser ion source has been used for measuring electron capture cross sections for C^{+q} ions ($q = 3, 4, 5$) in H_2 at energies of a few hundred electron volts per charge. In this region of very low energy, the cross sections do not scale with charge q , and all exhibit an increase with decreasing velocity.

In the Controlled Fusion Atomic Data Center, data compilation and preparation of indexed bibliographic files have been streamlined by the newly installed computer system. The data compilations on sputtering and ion reflection from surfaces were updated and published, and a volume of tabulated data is being prepared concerning atomic collision cross sections for impurity ions in tokamak plasmas.

4.1 EXCITED-STATE CONTRIBUTION TO THE ELECTRON LOSS COLLISIONS OF A FAST H^0 BEAM WITH N^{3+} IONS

During 1979, we continued crossed-beam experiments to measure cross sections for total electron loss by multicharged ions at neutral beam injection velocities.¹ Observed cross sections are dominated by ionization of the small fraction (0.3%) of H^0 atoms formed in highly excited (Rydberg) states during the neutralization of H^+ in water vapor. The higher excited states of principal quantum number $n > N$ can be ionized and removed from the beam by applying an electric field ϵ , where $N \propto \epsilon^{-1/4}$.

Figure 4.1 shows the apparent cross section for electron loss by hydrogen atoms in collisions with N^{3+} ions as a function of the ionizer field strength ϵ (which defines N). This

apparent cross section is a weighted sum of the electron loss cross section for atoms with a given principal quantum number n (α_n) times the fractional population (P_n). Using $P_n \propto n^{-3}$, as determined in an auxiliary experiment,² our data are consistent with the scaling $\sigma_n = 1.19 \cdot 10^{-16} n^{3.12} \text{ cm}^2$ ($9 < n < 24$). This is in substantial disagreement with the n^2 increase that is predicted by both classical^{3,4} and quantum⁵ theories and that is represented by the dashed curve in Fig. 4.1. We conclude that the classical as well as the first-order Born theory may be inadequate to describe ionization of loosely bound electrons in Rydberg states by multiply charged ion impact.

Electron loss cross sections for ground-state ($n = 1$) hydrogen atoms will be measured either by producing H^0 by electron detachment

Fig. 4.1. The population-weighted cross section σ_n as a function of N , which is the upper cutoff of n selected by the field ionizer. Round points were obtained by electrostatic field ionization, and square points by magnetic field ionization. The theoretical results based on n^2 scaling of the cross section σ_n are normalized to the data at $N = 14$ and are shown by the dashed line. The solid line is a similarly normalized least-squares result. The collision energy is 40 keV/amu.

from H^- in a thin gas target or by crossing a fast multicharged ion beam with a thermally dissociated ground-state H^0 beam from an effusive tungsten oven.

4.2 ELECTRON IMPACT IONIZATION OF MULTICHARGED IONS

Measured cross sections for electron impact ionization of multicharged ions have been obtained from experiments employing crossed electron and ion beams. Some of the measurements were reported in last year's annual report⁶ and in the open literature.^{7,8} All of our data are now compiled in an ORNL report⁹ and have been compared with available theoretical estimates. In addition, all of the measurements

have been used to derive ionization rate coefficients⁹ for the ions B^{3+} , B^{4+} , C^{3+} , C^{4+} , N^{3+} , H^{4+} , N^{5+} , O^{3+} , O^{4+} , O^{5+} , and Ar^{4+} in thin plasmas with Maxwellian electron temperature. Figure 4.2 shows the derived ionization rate for N^{4+} and compares it with plasma-observed rates¹⁰⁻¹² and with theory.¹³⁻¹⁵ In general, a simple version of the Lotz formula¹⁴ has been found to agree reasonably well with our measurements.

Fig. 4.2. Ionization rates for ground-state N^{4+} as a function of plasma electron temperature: solid curve, our result; open circle, plasma-observed rate of Kunze; solid circle, plasma-observed rate of Källne and Jones; open triangle, plasma-observed rate of Rowan and Roberts; dashed curve, Golden and Sampson's scaled Coulomb Born; dot-and-dashed curve, Lotz's formula; dotted curve, ECIP calculation of Summers. The error bar on the solid curve at 200 eV is the cross-section uncertainty at high confidence (equivalent to a 90% confidence level).

All our cross-section measurements agree with the Lotz formula within $\pm 50\%$, and generally within $\pm 20\%$. Figure 4.3 shows our cross-section data for Ar^{4+} , the only heavy ion investigated, as well as the data of Becker et al.¹⁶ and the theoretical predictions of Lotz¹⁴ and Salop.¹⁷

Fig. 4.3. Cross section for electron impact ionization of Ar^{4+} : solid points, our data (error bars represent a 90% confidence level for counting statistics); open points, data of Becker et al.; dot-and-dashed curve, Lotz's formula; solid curve, classical theory calculated by Salop.

The structures in the measured cross section at 140 eV and 260 eV are due to excitation of an inner-shell electron followed by autoionization. The Lotz formula does not specifically include this process, but would overestimate the cross section if the excitation-autoionization mechanism did not contribute to the measured cross sections. Thus, agreement with the Lotz formula may be somewhat fortuitous. Additional heavy ions will be investigated for which excitation-autoionization may be significant.

4.3 THEORY OF ELECTRON DETACHMENT IN COLLISIONS

In our study of collisions of interest to ion source development, we are investigating the process described by the reaction

over the range of energies from threshold to several hundred kiloelectron volts. Apart from its practical importance, this reaction serves as a test for theoretical methods, since several quite plausible theories fail badly. Our aim

is to derive the cross section from well-established data on electron scattering from cesium.

The simplest theory in reasonable accord with experiments is the slow-encounter model,¹⁶ compared in Fig. 4.4 with the measurement of Meyer, as reported in Sect. 4.4. This theory uses only the total $\text{e} + \text{Cs}$ cross section for electron scattering from cesium over the range 0-20 eV.¹⁹ Work is in progress (in collaboration with V. Sidis, Université de Paris Sud, Orsay, France) on a more elaborate theory in which the electron-cesium data are fitted by separable potentials of a type long used in nuclear physics. Some interesting predictions can be made — for instance, that excitation and ionization of the cesium become significant for hydrogen energies above 10-20 keV. The calculations are easily repeated for targets other than cesium.

Fig. 4.4. Comparison of theoretical and experimental cross sections for the collisional electron detachment process $\text{Cs} + \text{H}^- \rightarrow \text{Cs} + \text{H} + \text{e}$. The solid curve represents our current theory, which does not allow for excitation of the cesium; the square point accounts for excitation. Circles represent the experimental data of Meyer.

4.4 LOW-ENERGY ELECTRON CAPTURE AND DETACHMENT CROSS SECTIONS FOR HYDROGEN PROJECTILES IN CESIUM VAPOR

Using a low-energy accelerator employing a Calutron ion source and a modified Van Zyl deceleration section,²⁰ we have measured electron capture cross sections for $\text{H}^+ (\text{D}^+)$ and H^0

(D^0) projectiles, as well as electron detachment cross sections for H^- (D^-) projectiles incident on cesium vapor in the energy range 100-2000 eV/amu. These measurements are part of a study of collisional formation and destruction of H^- at energies of interest to negative ion source development programs.

The cross sections are obtained under single-collision conditions from the slopes of the growth curves for the collision product ion or atom of interest as a function of the cesium target thickness. Usually, the cesium target thickness is varied by changing the cesium density inside the target while maintaining the target length. In our experiment, the cesium target thickness was varied by changing the length of the target cell using precision micrometers while maintaining a constant cesium density inside the target. Using this technique, only the incremental change of the target length must be determined, instead of the total effective length of the target — that is, the geometrical length plus an end correction, a quantity that can at times be difficult to assess. The cesium density inside the target was measured using a surface ionization detector. The neutral component of the projectile beam emerging from the cesium target was measured using a secondary electron emission detector calibrated as previously described.²¹ The charged components of the projectile beam were measured using suppressed Faraday cups. Both the Faraday cups and the neutral detectors had angular acceptances of 2.2° , and the projectile beams were collimated upstream from the cesium target to an angular divergence of $\sim 1/2^\circ$.

Figure 4.5 shows our measured electron capture cross sections σ_{+0} and σ_{0-} and detachment cross sections σ_{-0} for H^+ , H^0 , and H^- projectiles incident on cesium in the energy range 100-2000 eV/amu. All three cross sections join smoothly with the existing higher-energy measurements of Meyer;²² Leslie, Sarver, and Anderson;²³ and Nagata.²⁴ (The results of Leslie et al.¹⁶ have been renormalized upward by a factor of 2.1 by using the correct value of the cross section σ_{+0} at 15 keV, to which

Fig. 4.5. Projectile energy dependences of experimental electron capture and detachment cross sections for hydrogen and deuterium projectiles incident upon cesium vapor. Open circles, squares, and triangles represent σ_{+0} , σ_{0-} , and σ_{-0} , respectively, for hydrogen projectiles; closed circles, squares, and triangles represent these same cross sections for deuterium projectiles. Other symbols are defined as: pluses, σ_{+0} as measured by Meyer; diamonds, σ_{0-} as measured by Leslie, Sarver, and Anderson; inverted triangles, σ_{-0} as measured by Nagata.

they normalized.) Using our measured cross sections σ_{0-} and σ_{-0} , we have calculated the H^- equilibrium fraction F_-^e in the two-state approximation $F_-^e = \sigma_{0-}/(\sigma_{0-} + \sigma_{-0})$. Our calculated equilibrium fractions are shown in Fig. 4.6, where they are compared with our previously measured yields²¹ and with those of Schlachter et al.²⁵ The calculated equilibrium fractions peak at an energy of ~ 250 eV/amu with a value of $\sim 26\%$. Although these are slightly higher than our previously measured yields, the two results agree within the stated experimental uncertainties. We have also measured F_-^e directly using this apparatus at an energy of 300 eV/amu and find excellent agreement with the value calculated from the cross section measurements at that energy.

Fig. 4.6. The H^- (D^-) equilibrium fraction F^e for hydrogen (deuterium) projectiles incident on a thick cesium target vs projectile energy.

4.5 ELECTRON CAPTURE FROM ATOMIC HYDROGEN BY LOW-ENERGY IONS

Measurements of cross sections for electron capture by multicharged ions from atomic and molecular hydrogen have been extended to new ionic species. Most of the new measurements are for heavy multicharged ions²⁶ of $Xe^{(2-12)+}$, $Ar^{(2-9)+}$, and $Fe^{(5,6)+}$ produced by the ORNL PIG source. Figure 4.7 shows capture by $Xe^{(3-12)+}$ and $Ar^{(3-9)+}$ in atomic hydrogen at a fixed collision velocity. The scaling of cross section with initial ionic charge q follows quite well the generalized theories of Grodzanov and Janev²⁷ and of Olson and Salop²⁸ up to $q = 9$, but for higher q , the data suggest a variation of cross section with q that is less than the predicted linear scaling. Measurements at still higher q are needed to test the predicted scaling. In addition, the measured cross sections are uniformly lower in magnitude than those predicted by the generalized theories.

Fig. 4.7. Variation with ionic charge q of cross sections for electron capture by multicharged ions from atomic hydrogen at a fixed velocity: solid circles, Xe^{q+} ions ($3 \leq q \leq 12$); open circles, Ar^{q+} ions ($3 \leq q \leq 9$). The tunneling theory of Grodzanov and Janev and the absorbing-sphere model of Olson and Salop are shown for comparison.

Another interesting feature of cross sections for heavy, partially stripped ions is illustrated in Fig. 4.8. For multicharged ions at a given velocity, the strongest variation of the cross sections is with ionic charge q . However, for a fixed low velocity and fixed q , Fig. 4.8 illustrates an additional dependence on the number of electrons remaining on the partially stripped ion. Ions that initially have several electrons remaining are observed to have larger cross sections than do nearly fully stripped ions (ions with fewer than five remaining electrons) of the same ionic charge q . (In the figure, measurements for nearly fully stripped ions are those of Crandall, Phaneuf, and Meyer,²⁹ as are the measurements for argon, iron, and xenon ions.²⁶)

Fig. 4.8. Cross sections at a fixed velocity for electron capture by multicharged ions from atomic hydrogen for initial ionic charges $q = 4, 5, 6$ as a function of the number of electrons remaining on the incident ion ($Z-q$). Number, on the experimental points denote q .

4.6 ELECTRON CAPTURE MEASUREMENTS WITH THE LASER ION SOURCE

The transient plasma produced when a pulse of carbon dioxide laser radiation is focused in vacuum to a power density of $3 \times 10^{10} \text{ W/cm}^2$ onto a graphite target has been used as a source of highly stripped carbon ions for measurements of charge exchange cross sections. The time-of-flight collision apparatus is shown in Fig. 4.9. Apertures collimate an ion beam from the plasma blowoff, and an electrostatic analyzer selects ions with a fixed energy per charge. The ion beam is directed through a gas target cell, is charge-analyzed once more by deceleration, and is detected by a channel-

Fig. 4.9. Pulsed-laser ion source and time-of-flight electron capture collision apparatus.

plate electron multiplier. Charge exchange cross sections are determined by measuring the fraction of ions that capture an electron as the ions pass through gas of a known density in the target cell. Cross sections are plotted in Fig. 4.10 for C^{+q} ions ($q = 3, 4, 5$) colliding with H_2 . Also shown are cross-section measurements at higher velocities made at ORNL by Phaneuf, Meyer, and McKnight³⁰ and by Crandall et al.^{31,32} using other sources of highly charged ions. It is significant that C^{4+} has the largest cross section at these low velocities, counterindicating a simple scaling with ionic charge, and that all the cross sections increase with decreasing velocity. Measurements will also be made for O^{+q} and Fe^{+q} ions, and the ORNL atomic hydrogen oven³⁰ will be used to provide H^0 data for direct comparison with emerging models of charge exchange in the difficult low-velocity region.³³

4.7 CONTROLLED FUSION ATOMIC DATA CENTER

In a continuing cooperative effort with the Atomic Transition Probabilities Data Center of the National Bureau of Standards, the fifth volume of the *Atomic Data for Fusion* newsletter was compiled and published. An author and subject index for this volume was prepared as a supplement to the bimonthly issues. Newsletter distribution has been streamlined through the use of data processing equipment; mailing lists and labels are now produced by means of the remote computer terminal and high-speed digital printer recently installed for input and output of our indexed bibliographies.

We conducted a thorough evaluation of the 1978 issues of our newsletter to compare their content with the material presented in the International Atomic Energy Agency (IAEA) bulletin for the same period. This study led

Fig. 4.10. Experimental electron capture cross sections vs ion velocity for multiply charged C^{+q} for H_2 : points with error flags at lowest velocities, measurements using laser ion source; circled points, the data of Phaneuf, Meyer, and McKnight; points without flags, the data of Crandall et al.

to a decision to continue publishing our bulletin until 1981, when further evaluation will be made.

The "Surfaces" section of our two-volume data compilation *Atomic Data for Controlled Fusion Research* underwent major revisions. Data sets in the sections "Sputtering" and "Ion Reflection from Surfaces" were updated and supplemented, and a new section was added on particle trapping. These revisions, totaling 81 data sets, were published in November 1979 as an ORNL report³⁴ designed to be integrated with the second volume of the parent compilation. The Controlled Fusion Atomic Data Center has continued to receive requests for this series since its first publication in 1977. A third volume is being prepared that tabulates atomic collision cross sections involving impurity ions in tokamak-produced plasmas. Experimental data have been compiled on charge exchange, electron excitation, and electron ionization collisions.

Our indexed bibliography of atomic and molecular processes for 1976-77 was edited, reproduced, and distributed to selected laboratories and universities in the United States and abroad, as well as to the IAEA for its use in publishing a comprehensive bibliography of data provided by institutions in several countries. Data entries for our 1978 indexed bibliography were input, manipulated, and formatted under our newly developed computer code. This bibliography has been edited and prepared for publication.

Journal searches were performed on a monthly basis, which allowed us to update our computer files according to journal publication schedules. Consequently, all 1979 bibliographical data have been entered and are awaiting final formatting. The savings in time and effort gained by our new computer system has allowed us to monitor our files more closely; professional staff check preliminary printouts for incorrect reactants and miscategorized or irrelevant entries.

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5. PLASMA HEATING AND FUELING

5. PLASMA HEATING AND FUELING

ABSTRACT. The Plasma Technology Section is principally concerned with (1) development of multi-megawatt neutral beam lines, including electrical systems and vacuum pumping, for plasma heating and (2) development of techniques to fuel the plasma using high-velocity pellets.

In the neutral beam development program, work proceeded on the ion source for the Poloidal Divertor Experiment (PDX) at the Princeton Plasma Physics Laboratory (PPPL). This is a 100-A source with a grid 30 cm in diameter. A variety of aperture shapes for the multiaperture plasma grid was tested to determine the geometry that yielded the best beam optics, which was judged by the maximum beam power transmission down the beam line and the minimum thermal loading on the grids. An improvement in the efficiency of the beam by 10% or more over that of the injectors used in the Princeton Large Torus (PLT) or Impurity Study Experiment (ISX) is due to use of a "mutual exclusion" electrode shape predicted by theoretical studies. The PDX ion source has delivered 1.6 MW of 50-keV neutral particles to the torus target for pulses 100 ms long. The goal is to extend these power levels to 500-ms pulses.

In the Advanced Positive Ion Source program a rectangular ion source was fabricated for the development of source components, such as the electron feed and accelerator grids that can withstand the thermal loading associated with long-pulse beams. Hollow cathodes were shown to be capable of 10-s arc pulses, and a 6-s beam pulse (at 40 kV and 143 mA/cm²) was limited in pulse length only by the thermal capabilities of the beam target. Additional computer analysis of grid deformation caused by thermal loading was carried out in support of this long-pulse study.

Experiments were undertaken to assess the optical properties of high-energy, multiaperture ion sources with large-gap, single-stage acceleration. With a 1.2-cm gap, current densities of 300 mA/cm² at 80 kV were achieved. Although the beamlet divergence angles are higher than for two-stage accelerators, the beam power transmission is comparable.

Two-stage, single-beamlet optics were studied over a range of aperture configurations. Calorimetric measurements over a wide angular range confirmed the presence of large-angle wings ($>2^\circ$), which was predicted theoretically. The ion optics algorithm used for electrode design was made more than ten times faster and has been extended to three dimensions.

Two new versions of the energy recovery experiment based on magnetic electron suppression yielded recovery efficiencies of greater than 60%.

Because of the reliable operation of the parallel tube modulator at the 50-kV, 100-A level, a dual-tube, dual-deck modulator using four vacuum tubes for 150-kV, 100-A operation is being designed. In addition to the electrical support given to novel experiments, the electrical technology group is investigating the use of zinc oxide surge arrestors in forcing high-voltage sharing.

In the pellet fueling program, 1-mm-diam hydrogen pellets fired into ISX-B at 1 km/s gave massive density rises ($\Delta n_e / n_e \sim 3$) without adversely affecting the confinement and gross stability properties of the discharge. Plasma densities measuring 1×10^{19} cm⁻³ over a 1- to 1.5-mm region around the pellet were observed with a ruby laser holographic interferometer developed in collaboration with the Massachusetts Institute of Technology (MIT).

A 2-m² cryosorption pump using type Na-Y molecular sieve has been completed and is ready to install on the Medium Energy Test Facility (METF). Hydrogen ion beams of 11.5 keV, 1.3 A, and 100-300 ms have been fired into a small cryosorption pump with no ill effects.

Work on beam line pumping optimization proceeded with the Monte Carlo simulation of neutral particle transport in the PDX beam line.

5.1 PDX BEAM LINE DEVELOPMENT^{1,2}

The neutral beam heating systems for the PDX tokamak at PPPL must be capable of delivering 1.5 MW of 50-keV H^0 to the plasma for a minimum pulse duration of 500 ms. In addition, the ion source that produces these beams must have an extracted atomic ion fraction in excess of 80% and be able to sustain these parameters with at least a 90% reliability. To meet these demands it was decided to develop a scaled-up 100-A version of the modified duoPigatron source that has successfully met the injection demands of both the PLT and ISX-B tokamaks.

Of prime importance in the development of this source is the optical quality of the beam. The cross-sectional shape of the apertures of the first grid (next to the source plasma) plays the most important role in determining the quality of beam optics, since it provides the coupling interface between the source plasma and accelerating field. Figure 5.1 shows the types of grid aperture shaping used to compare beam optics. These choices were

Fig. 5.1. Cross-sectional view of four plasma grid aperture shapes that were tested on the PDX ion source. The source plasma is at the top of each.

based on previous experience and theoretical considerations.^{3,4} The best figure of merit we have for determining optical quality is the beam transmission efficiency (i.e., the transmitted power normalized to extracted power). Experimental results for these aperture shapes as a function of beam perveance (perveance = $1/V^{3/2}$) are shown in Fig. 5.2. The term *precel* in these results to the technique of biasing the source plasma to preaccelerate ions,⁵⁻⁷ which has the effect of improving beam optics for most aperture geometries. Power loading to the accelerator grids occasionally provides another indication of optical quality, that is, the lower the grid power loadings the better the optics. A comparison of the above aperture shapes has shown that Type A not only had the lowest peak transmission efficiency (54%), but

Fig. 5.2. Ion beam power transmitted to the torus target (expressed as a percentage of the extracted $I \times V$ beam power) vs beam perveance ($1/V^{3/2}$) for a number of plasma grid aperture shapes.

also the highest total grid loading (4.7% of the extracted power at the same perveance), whereas type D⁸ had a peak transmission efficiency of 67% and a total grid loading of 3.2%. The PDX beam line (Fig. 5.3) operating at 50 kV and 100 A for 100-ms pulses transmitted 1.6 MW of neutrals to the torus target; at 52 kV, 87.5 A, and 500 ms, the beam line transmitted 1.4 MW to the same target.

5.2 THEORETICAL ION OPTICS³⁻²⁷

A new algorithm for two-dimensional (2-D) ion optics has been written that typically is ten times faster than the previous version.^{3,4} It also has been extended to do slot geometry as well as cylindrical geometry. In every case examined, slot geometry has yielded beam root-mean-square (rms) divergence in the transverse direction, which is over twice that of cylinder geometry. For a typical straight-bore triode, the minimum rms angle is 3.1° for slots and 1.8° for cylinders.^{20,21} For a Pierce-shaped electrode, the rms angle is 1.4° for slots and 0.62° for cylinders. This is the first time a

systematic and reliable study has been done on this important question in the neutral beam program. Slot steering as a function of displacement and perveance was also considered.²²

Other work completed during 1979 included:

- The three-dimensional (3-D) code was advanced²³ over the preliminary version,²⁴ and for the first time a study of slot ends was made.²⁵
- The prediction that large-radius apertures yield high-quality optical beams for tetrodes²⁶ was substantiated.²⁷
- A new electrode shape⁸ was found to inhibit beam aberrations by making the regions occupied by the beam and aberrations mutually exclusive. This concept has been employed on the PDX ion source and has enabled an increase of power efficiency on target by 10%. An additional increase of 6% was obtainable by applying a preacceleration voltage as previously described.⁵⁻⁷
- A theoretical study of neutralizer gas evolution was conducted.⁹

Fig. 5.3. Power deposition along the PDX beam line. Starting at the ion source, power is monitored for the source itself (backstreaming electrons), the grids, the gas cell, the ion dump, the defining plates, the first drift tube, the drift aperture, and the torus target. Unmonitored power losses (unlabeled arrows) occur at the entrance to the gas cell, in the bending magnet region, and in the second drift tube.

5.3 HIGH-ENERGY SOURCES

Work on the high-energy (>50-keV) ion source was directed to (1) investigation of the conventional triode accel-decel arrangement at higher voltages and (2) beamlet optics studies using a two-stage accelerating scheme.

5.3.1 High-Energy Triodes

It is generally believed that as the beam energy exceeds about 60 kV, the conventional triode accel-decel scheme suffers from serious loss in current density.^{26,28,29} A two-stage acceleration scheme has been shown to overcome this problem.^{29,30} The limitations of the triode, however, have not been carefully scrutinized. The simplicity of triodes as opposed to tetrodes motivated an examination of the characteristics of long-gap (1- to 2-cm) triodes. The major results are as follows:

1. Even though the half-width half-maximum (HWHM) divergence angles of triodes are almost a factor of two higher than those of the tetrodes, the power transmission efficiency is only marginally lower. Figure 5.4 illustrates power deposition data for a triode with a 1.2-cm gap. Similar curves for two-stage sources are available elsewhere.³⁰
2. The primary problem with triode operation at long gaps is the direct interception of the beamlet wings on the second and third grid. This not only increases the grid loading, but also affects the operational stability of the source.
3. The decrease in current density with an increase in gap length is not as drastic as is commonly believed, and current densities in the neighborhood of 0.3 A/cm² have been achieved for sources operating at 80 kV.
4. The dominant causes of grid loadings have been identified. Interception of the wings of the beamlets is responsible for the ground grid loading. The loading on

Fig. 5.4. Power deposition for a 1.2-cm-gap triode.

the middle grid is partly by the interception of the beamlet wings and partly by the interception of ions from the gas cell plasma. The plasma grid loading, excluding the contribution from the plasma, is due to the interception of secondary electrons produced at the second grid. The experiments suggest that the effect of volume processes is relatively minor, particularly at large gaps. It has also been revealed that grid loadings acceptable for long-pulse operation (≥ 1 s) can be attained by appropriate shaping of the apertures.

5.3.2 Beamlet Optics Studies

The power density profile of a beamlet, accelerated by employing a two-stage scheme, was measured calorimetrically over a range of

aperture configurations. The power density profiles were measured over a wide angular range ($\pm 11.5^\circ$) to study the behavior of the core as well as of the wings of the beamlet. The principal results are:

1. The intensity distribution is Gaussian only in the central core of the beam, and large-angle wings ($>2^\circ$) are present that represent a serious loss in beam power and a source of grid loading (Fig. 5.5).
2. Shaping the plasma grid aperture reduces the wings and improves the transmission efficiency even though it does not significantly alter the HWHM divergence angle of the beamlet intensity profile.
3. Increasing the voltage distribution ratio between the accel gap and the extraction gap reduces the beamlet wings and affects the divergence angle.
4. Ion beams extracted by using large apertures (≥ 1 cm in diameter and with

aspect ratios ≥ 1) show current densities and beam divergences comparable to those for beams extracted by using small apertures (Fig. 5.6).

5. For large aspect ratios (≥ 1), the beam current obeys the well-known three-halves power law $V^{3/2}/d^2$, where V is the total voltage and d is the sum of the accel and extraction gaps. With small apertures, the extraction gap decides the current.²²⁻²⁴

5.4 LONG-PULSE OPERATION

5.4.1 Advanced Positive Ion Source

The next generation of fusion experiments calls for high-energy (~ 100 -keV/nucleon), high-current (~ 100 -A), long-pulse (~ 10 -s) ion sources for neutral beam heating applications. The energy involved in these long-pulse, high-power beams poses thermal problems for the ion source's electron feed and accelerator structure. A rectangular-cross-section plasma generator 25×35 cm has been fabricated to address these problems.^{31,32} This design uses a 10×25 -cm multiaperture grid pattern for the accelerator electrodes, with the water cooling channels running across the short dimension. The increased cooling — water lines have been placed between each row of apertures — and the shorter path lengths should give the grids greater mechanical stability from thermal deformation. Improvements in beamlet optics are being sought to reduce the thermal loading attributable to beam interception by the grids. To improve the long-pulse capability of the electron feed, hollow cathodes³³ and ac-heated oxide filaments were studied for use as electron emitters. Each of these cathodes requires a different pressure regime for proper operation. Thus, changes in the gas conductance of the intermediate electrode channel become necessary. The intermediate electrode geometry is being studied and evaluated with respect to gas efficiency, arc power efficiency, and arc discharge reliability. Although the ion source has been designed for two-stage acceleration

Fig. 5.5. Intensity profile of a beamlet formed by using straight-bore apertures in two-stage configuration. The dashed curve is Gaussian.

Fig. 5.6. Beam current and current density variation with energy for different aspect ratios (S) for a notch electrode shape.

(tetrode grid structure), operation up to now has used a single-stage, triode grid arrangement with a 1.2-cm gap. For 0.1-s pulses, a 73-kV, 285-mA/cm² (24-A) beam has been extracted. Long pulses (6 s) have been run at 40 kV and 143 mA/cm² (12 A) using a La₂O₃-doped molybdenum hollow cathode as the electron emitter.

5.4.2 Grid Deformation Study

The water-cooled grids of ion sources can receive a thermal load of about 2% of the extracted ion beam power. It has long been a concern that this level of thermal loading may

mechanically deform the grids and thus degrade the injector performance. A thermal deformation analysis of the plasma grid of the ion accelerator has been performed for the PDX injector development. By employing a 3-D finite element code,^{34,35} it was found that a water-cooled copper grid with a loading of 250 W/cm² reaches a maximum steady-state temperature of 200°C. This yields a maximum vertical displacement of 0.25 cm and a maximum radial displacement of 0.013 cm, which could significantly degrade the beam transmission. The importance or validity of this study has not yet been established.

It was suggested that a high bellows kink outside the aperture pattern and movable grid edges could reduce the grid's thermal deformation.

The idea of a movable grid edge was adopted for the study of an ion source compatible with the Tokamak Fusion Test Reactor (TFTR) (having a 13×43 -cm grid aperture pattern, with water cooling across the narrow dimension). The results of this study indicated that the thermal deformation would be negligible.³⁶

5.5 ENERGY RECOVERY EXPERIMENT

The initial test of direct energy recovery of the unneutralized, full-energy ion component showed that the transverse magnetic field blocking of gas cell electrons worked and that energy recovery appeared to be achieved under some operating conditions.³⁷ However, an undesirable energy drain was observed that was due to fractional-energy ions terminating on the exterior surface of the gas cell. When this happens, secondary electrons emitted immediately proceed to ground potential surfaces, thereby producing a large power drain. Two versions of the experiment have been developed to eliminate the secondary electron power drain problem, each having recovery efficiencies ranging from 60 to 80%.

In the first version,³⁸ all ions are permitted to deflect in the magnetic field before experiencing the retarding effect of the electric field. This is accomplished by extending the walls of the gas cell in order to exclude the electric field. This version is called the loosely coupled experiment. Energy recovery efficiencies for 15-A to 20-A beams at ~ 30 kV were based primarily on current measurements, with some support from calorimetric data.

In the second version, called the closely coupled experiment, the gas cell exit is constructed so that the ions experience magnetic deflection and electrostatic retardation simultaneously. Preliminary analysis of both current and calorimetry data taken at 40 kV and

20 A indicates a recovery efficiency comparable to that of the loosely coupled experiment. It has been pointed out³⁹ that the magnetic and electric field coupling in the two versions may be quite similar because of the buildup of a plasma at ground potential, which may determine the position of the sheath over which the ions are retarded.²⁹

5.6 NEGATIVE ION SOURCES

During 1979, a low-level effort was maintained on both the modified duoPIGatron⁴⁰ and the modified calutron sources.⁴¹ The beam output (at 1 kV and 0.1 A) of the duoPIGatron was analyzed with a Wein filter. Using peak amplitudes, results indicate $\sim 12\%$ of the output is electron current and $\sim 16\%$ is impurity.

The beam impurity of the calutron was analyzed magnetically.⁴² The impurity content is very high ($\sim 10\%$) with a fresh charge of cesium dichromate. However, the level reduces to less than 1% as the source is conditioned. An all-metal ion source has been prepared to eliminate the carbon contamination of the cesiated conversion surface.

5.7 ELECTRICAL TECHNOLOGY DEVELOPMENT

The parallel tube modulator installed at the METF has been operated routinely at 100 A during 1979. Because of the modulator's good performance, work has begun on a dual-tube, dual-deck modulator using four higher-voltage vacuum tubes. The device will be installed about mid-year in 1980 and will replace a 60-A three-deck modulator. Current capability will be increased to 100 A, and voltage capability will be increased from 120 to 150 kV.

A multi-channel telemetry system was developed during the year. This is an expanded version of an earlier system developed several years ago. The telemetry system operates and monitors the isolated power supplies and transmits data across the high-voltage gradient through fiber optics 10 ft long and 1/8 in. in diameter, with one end at high voltage and the other end at

ground. Analog signals are digitized, transmitted through the optical system, and demodulated at ground. There are ten down links, four analog up links, and four pulsed up links.

In support of energy recovery research, the power output capability of a commercial decel grid modulator was upgraded. The output current was increased by a factor of two and voltage regulation was improved. Procurement delays and fund expenditures were avoided for at least the present power level by this change.

There is an inherent development challenge in designs using series connected switching components. Unless equal voltage sharing can be accomplished for all transient and steady-state conditions, the desired reliable operation cannot be achieved. Forced voltage sharing has previously depended on a compensated voltage divider technique that uses combinations of resistors and capacitors. During 1979, we gained experience with new zinc oxide surge

arrestors developed by General Electric Company. Reliable operation has been achieved at 45 kV dc per unit with two units in series.

Ion source filaments have traditionally been operated from dc power supplies using three-phase, full-wave bridge rectifiers. During 1979, an experiment was conducted to determine if ac filament heating would give satisfactory results. No ill effects were observed, so two test stands were changed from dc to ac filament heating. As a result, equipment has been simplified and arc current is distributed more uniformly through the filaments.

5.8 PELLET FUELING

As in previous years, the pellet fueling program during 1979 included pellet injector development, tokamak experiments (Figs. 5.7 and 5.8), and pellet-plasma diagnostic development. Details of the experiments performed on

Fig. 5.7. Schematic of pneumatic pellet injector installation on ISX-B.

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Fig. 5.8. Time exposure showing trail of luminous gas left in the wake of a pellet as it traverses the entire plasma column from left to right and impacts the inner liner wall. The vertical luminous area to the left is the plasma interacting with the material limiter.

ISX-B in phases of operation employing ohmic and neutral beam (~ 1 MW) heating have been discussed elsewhere (see Sect. 1). The principal result from this experiment was the successful demonstration of massive density rise ($\Delta n_e / n_e \sim 3$ in ~ 250 μ s) from single hydrogen pellet injection (1-mm diam at 1 km/s) without adversely affecting the confinement and gross stability properties of the discharge. This finding has a strong impact on the pellet injector development program in that it appears to allow the use of large pellets for future refueling needs. As in the ISX-A experiments, fueling by pellet injection was found to be more efficient than edge refueling by gas puffing.

In collaboration with MIT, a ruby laser holographic interferometer was constructed and applied to the ISX-B device to obtain in situ measurements of the plasma density in the region surrounding the pellet. A schematic of the experimental arrangement is shown in Fig. 5.9. Plasma densities measuring 1×10^{19} cm^{-3} were observed in partly ionized gas over a 1- to 1.5-mm region surrounding the pellet.

No evidence was found for a large neutral gas cloud directly exposed to hot plasma ions that might serve as a source of large charge exchange losses and require higher pellet feed rates to sustain plasma density.

The pneumatic pellet injector used in the ISX-B experiment was removed from ISX-B and installed on the PDX device at PPPL. Joint ORNL/PPPL experiments may be performed early in 1980 to demonstrate single pellet injection on a divertor-equipped tokamak and to test the concept of plasma refueling beyond the divertor scrape-off region.

The applicability of single pellet injection results to multiple pellet operation will be determined in future experiments on ISX-B using repetitive pellet injectors of the mechanical and pneumatic types. We are currently working to improve a unique centrifugal mechanical injector with design pellet velocities of 1-2 km/s (Fig. 5.10).^{4,5}

A modified pneumatic injector capable of delivering four pellets sequentially at variable timing intervals has been built for use on the ISX-B divertor modification. This will provide

Fig. 5.9. Schematic of ruby laser interferometer experiment on ISX-B.

Fig. 5.10. Schematic of mechanical frozen hydrogen pellet injector.

a quasi-steady fueling mode of operation in the high-velocity regime of 1-2 km/s with 1-mm pellets.

5.9 VACUUM COMPONENTS DEVELOPMENT

5.9.1 Plasma Edge Technology

During 1979, the focus of program activities shifted from vacuum technology to problems in handling very high heat and particle fluxes such as would occur in the Engineering Test Facility (ETF) or in a fusion reactor. The principal area of concern was the magnetic divertor. Heat fluxes as high as 20 kW cm^{-2} and particle loads up to 10^{23} s^{-1} were considered. It was found that conventional water cooling methods could handle the total expected power of 100 kW with 10 m^2 of target. Proposals were made for work in enhanced heat transfer, liquid metal getters, and high-pressure cryopumping. This work has been documented elsewhere.⁴⁴

5.9.2 Beam Line Cryosorption Pump

Fabrication was completed on the pump, which is similar to the type C cryocondensation pumps originally supplied by Arnold Engineering Development Center (AEDC) for the METF target chamber. The cryopanel has a total area of 2 m^2 and is covered with type Na-Y molecular sieve. This sieve has been shown⁴⁵ to give roughly ten times the pumping speed of the type 5-A material that has been customarily used.⁴⁴ Sieve bakeout is accomplished with radiant heaters mounted on the chevron baffles next to the cryopanel. The baffles are supplied with liquid nitrogen from an overhead 100-liter dewar. The panels are cooled by a continuous trickle flow of liquid or gaseous helium. The pump will be mounted in the METF

target chamber next to the existing AEDC type C pump. Direct comparison tests between the pumps are planned.

5.9.3 Energetic Particle Pumping

Further tests were carried out with a 10-cm ion source injecting directly into a small cryosorption pump borrowed from Los Alamos Scientific Laboratory (LASL) in an effort to assess the ability of a cryosorption pump to handle high-energy particles.⁴⁶ Hydrogen ion beams with energies up to 11.5 keV, currents up to 1.3 A, and pulse lengths of 100-300 ms were fired directly into the pump. The only observable effect of the ion beam was a slight increase in peak pressure; no evidence of pump instability was seen. Calorimetric measurements showed that about half the $I \times V$ power reached the pump. The results indicate that cryosorption pumps can handle energetic particle fluxes of at least $10^{16} \text{ cm}^{-2} \text{ s}^{-1}$ without difficulty.

5.9.4 Monte Carlo Simulation of Neutral Particle Transport

Calculations with the modified MORSE code⁴⁷ were extended to simulate neutral particle transport in a fully detailed PDX beam line. The geometry faithfully renders all essential features; for example, even the individual cryopump chevrons are modeled.

Figure 5.11 shows neutral pressures and densities per particle along the beam line axis for a cosine-distributed source of particles located at the ion source grids. The pressure in the drift region is reduced by nearly a factor of three when a drift tube cryopump is added in the drift chamber.⁴⁸ Other calculations of pressures at actual locations of ion gauges on the existing beam line agreed within 25% with the pressures observed on the gauges.

Fig. 5.11. Neutral density and pressure along beam centerline as calculated by the Monte Carlo particle transport code.

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6. SUPERCONDUCTING MAGNET DEVELOPMENT

P. N. Haubenreich, Manager
Large Coil Program

M. S. Lubell, Manager
Magnetics and Superconductivity Section

L. Alley, Jr.	W. J. Kenney	G. A. Nardone ⁶
J. K. Ballou	K. L. Kilby ²	L. W. Nelms ²
R. L. Brown	J. P. Kois ⁴	B. E. Nelson ³
L. Dresner	M. H. Kunselman ³	J. C. Parks ²
J. F. Ellis	P. S. Litherland ³	C. C. Queen ³
W. A. Fietz	C. J. Long ⁵	J. P. Rudd
C. M. Fitzpatrick	J. W. Lue	T. L. Ryan ³
W. M. Fletcher ¹	J. N. Luton	S. S. Shen
J. S. Goddard ¹	J. R. May ⁶	R. E. Stamps ³
W. H. Gray	R. A. Michelotti ³	W. C. T. Stoddart ⁴
C. G. Hudson ²	J. R. Miller	P. B. Thompson ³
R. O. Hussung ³	J. L. Murphy ¹	W. H. Wagner
	P. L. Walstrom	H. T. Yeh ⁷

1. Fusion Energy Electrical and Instrument Engineering Department
2. ORNL (Y-12) Project Engineering Department, UCC-ND Engineering
3. Research Engineering Department, UCC-ND Engineering
4. UCC-ND Engineering
5. Metals and Ceramics Division
6. Grumman Aerospace Corporation
7. Present address, Bell Telephone Laboratories, Illinois

6. SUPERCONDUCTING MAGNET DEVELOPMENT

ABSTRACT. With management, technical direction, and research and development support by Oak Ridge National Laboratory (ORNL), the U.S. Large Coil Program (LCP) proceeded with design and preparations for manufacture of three superconducting toroidal field coils and with design and construction of the Large Coil Test Facility (LCTF) at Oak Ridge. The International Energy Agency (IEA) fusion magnet development program continued, with EURATOM, Japan, and Switzerland engaged in various stages of design of one coil each. These coils are to be tested with the U.S. coils in the LCTF. Preliminary tests in the LCTF with two coils are scheduled for 1981, and tests in the full six-coil array are to begin in 1982.

Work in the Magnetism and Superconductivity Section fell into four categories: (1) research and development in support of the LCP, (2) two 12-T coil projects involving industries, (3) advanced conductor development, and (4) specific magnet designs. Notable progress was made in each of these areas.

6.1 INTRODUCTION

ORNL continues to play a leading role in the development of superconducting magnets for toroidal fusion reactors. The principal effort in 1979 was on the Large Coil Program (LCP), which entails designing, building, and testing several different 8-T toroidal field coils about one-third the size of those for a tokamak Engineering Test Facility (ETF) reactor. ORNL is guiding and monitoring General Dynamics-Convair, General Electric, and Westinghouse Electric in the design and manufacture of one coil each. In addition, through an International Energy Agency agreement, EURATOM, Japan, and Switzerland are designing and will build one coil each to LCP specifications for testing at Oak Ridge along with the U.S. coils in the LCTF. ORNL is designing and managing the construction of the LCTF and will be responsible for operating the facility and testing the coils. Supporting research and development for the LCP is carried out in the Magnetism and Superconductivity Section.

In addition to supporting the LCP, the Magnetism and Superconductivity Section has a major role in the U.S. 12-T coil program, advanced conductor development, and magnet design for specific projects. The main emphasis has been and continues to be in support of LCP in numerous ways. The research and development activities in support of LCP include work on instrumentation and diagnostics, conductor verification tests, and coil winding technology.

The Section also supplies structural analysis in support of the LCTF test stand design and the technical support necessary to monitor the coil contracts.

The 12-T coil program was initiated in 1978 to supplement the goals of the LCP. In this program, ORNL is directing two projects involving major industrial subcontractors [General Dynamics with Air Reduction Company and General Electric with Intermagnetics General Corporation (IGC)] in the design and development of large conductor suitable for tokamak toroidal field coils operable at fields to 12 T. Each project will fabricate a 1-m-diam coil to test its conductor by operation in the High-Field Test Facility at Lawrence Livermore Laboratory.

ORNL continues to advance magnet technology in areas specific to fusion applications by developing better analysis techniques, generating basic design information, and evaluating small quantities of experimental conductor embodying advanced concepts. The advanced conductor development program in 1979 continued to concentrate on developing multifilamentary Nb_3Sn , understanding force-cooled magnets, and gaining more perspective on ac losses, all closely tied to LCP and 12-T coil activities. This work is, therefore, reported below in the sections on the LCP and the 12-T program rather than in a separate conductor development section.

The main magnet design project of the Magnetism and Superconductivity Section was the reference design for ELMO Bumpy Torus Proof-of-Principle (EBT-P). A development program has

been formulated for the prototype coils and is being carried out.

6.2 LARGE COIL PROGRAM

LCP activities in 1979 consisted of design and preparations for manufacture of test coils, LCTF design and construction, and supporting research and development. All activities progressed on schedule toward the goals of partial testing of two coils in 1981 and initiation of tests of a full six-coil array in 1982.

6.2.1 Program Management

After review and approval of their final designs, General Dynamics and General Electric were authorized to proceed with coil manufacture. Cost estimates based on final design were higher than earlier estimates, reflecting rising prices of materials and changes found to be necessary as a result of verification tests and detailed analyses. To avoid some of the additional costs, program requirements for coil instrumentation and documentation of design and verification test results were relaxed.

The Department of Energy Oak Ridge Operations (DOE-ORO) issued a directive for construction of the complete six-coil test facility at a total estimated cost of \$26,400,000. Subsequently, the LCTF became a line item construction project by Congressional action.

Substantial progress was made on the implementation of the international Large Coil Task. Information exchange took place through reports, meetings, and workshops in the United States and Switzerland and through visits by the participants to each others' laboratories and manufacturing facilities. Two scientific staff members from the Japan Atomic Energy Research Institute were assigned to the LCTF site for periods of several months to facilitate communication between the design teams for the LCTF and the Japanese test coil.

6.2.2 Test Facility

Design and procurement for the LCTF proceeded according to a schedule that will enable the

testing of two coils in the summer of 1981 and of all six coils in the summer of 1982.

The 11-m-diam vacuum vessel was constructed by Pittsburgh-Des Moines Steel Company, and its acceptance tests were initiated in December. Fabrication of the pedestals for the test stand structure proceeded after some initial difficulties. Japan Steel Works (JSW) forged and machined the central component of the test stand (the bucking post). Under an extension of their contract, JSW proceeded with fabrication of the helium piping and the lower collar that will be installed on the bucking post before its shipment to the LCTF site in mid-1980. Chicago Bridge and Iron Company began fabrication of the test stand base (spider frame) in Memphis, Tennessee. The other major structural components of the test stand, that is, the torque ring and shims, were redesigned to reduce costs. A contract was awarded for a new 55-ton crane to be installed in the LCTF building.

Construction was completed on the helium refrigerator. Helix Process Systems began detailed engineering and procurement required for modification of the refrigerator to provide the capacity needed for high-duty factor test operation with six coils. Design of the liquid nitrogen system was completed by Planning Research Corporation, and procurement of the cold wall for the large vacuum vessel was initiated.

Specifications for procurement of the power supplies for the test coil were prepared. A decision was reached on switches for rapid discharge of the coils and cooperative development began at Los Alamos and Oak Ridge. Detailed design of the pulse coil system was initiated.

6.2.3 U.S. Test Coils

General Dynamics-Convair

During 1979 Convair completed Phase II (verification tests and detailed design) of the LCP coil contract except for the scalability report. Measurements of heat transfer to boiling helium in a series of tests simulating a section of winding showed decreased values in

long vertical sections, but changes in the configuration of the interturn insulation resulted in heat transfer adequate to ensure stability. Phase III (procurement, tooling, and manufacture) proceeded on schedule toward delivery of the General Dynamics coil in the spring of 1981. Convair set up equipment in their San Diego plant for assembling conductor from superconducting cable and copper bars and for winding their coil. Stainless steel for the coil case was procured from Eastern Stainless Steel Company, with the exception of the heavy forging for the nose piece, which was procured in Japan.

General Electric

General Electric finished Phase II with minor exceptions. Chicago Bridge and Iron began coil case construction at their Memphis plant. After review of the Phase II cost estimate, LCP management and General Electric worked out some measures to reduce costs, principally by reductions in instrumentation and in formal reports of verification tests. Production of superconductor by IGC proceeded through the stage of extrusion of composite billets. Delivery of the copper for the conductor assembly was delayed; the revised schedule shows delivery of the General Electric coil in the spring of 1981.

Westinghouse

Westinghouse completed most of Phase II. Remaining work included stress analysis, conclusive tests on sheath material, and documentation. The material and thickness of the conductor sheath were changed to resist higher compressive loadings revealed by detailed analyses. Preliminary tests indicated that Nitronic 40, the new sheath material, would probably have acceptable low-temperature properties after treatment. (Sensitization of the sheath material during heat treatment to produce Nb₃Sn in the contained cable was recognized as a possible problem.) Air Reduction Company successfully demonstrated the welding of heavy-wall sheath in production machinery. Investiga-

tion of aluminum alloy plate deformation during machining was completed. Conductor samples showed critical current and stability well in excess of specification requirements, allowing an acceptable margin for possible degradation of the Nb₃Sn performance from strains during winding. The Phase III schedule was laid out to deliver the Westinghouse coil in the spring of 1982.

6.2.4 Research and Development Activities

Conductor measurements

Stability of cable-in-conduit conductors.¹

Pressure surges of several atmospheres accompanied by fluid flow velocities of several meters per second are induced in initially stationary helium by energy transfers typical of conductor recovery (about a few hundred mJ/cm³ in a few tens of ms), according to calculations based on the linearized equations of compressible flow. These strong induced flows are responsible for the recovery of cable-in-conduit conductors cooled by initially stagnant helium.

Pressure surges of the expected magnitude were observed in studies of the stability margin of a single triplex of NbTi/Cu strands soldered around a heater wire, sheathed in a stainless steel tube, and cooled by pressurized helium. In these studies, the stability margin was found to be multivalued for certain combinations of transport current, ambient helium pressure, and externally imposed helium flow (see Fig. 6.1). A qualitative model based on the idea that good recovery in stagnant or slowly flowing helium is due to heating-induced flow accounts well for the complex, multivalued behavior of the stability margin. Based on the model, a scaling rule has been proposed for the limiting transport current below which the stability margin is single valued. With the single-triplex data as a jumping-off point, the scaling rule indicates that the Westinghouse cable-in-conduit conductor should not exhibit multiple stability at the LCP operating conditions. Stability measurements carried out on a one-third scale model of the Westinghouse conductor confirm this expectation.

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Fig. 6.1. The figure shows a three-dimensional sketch of the stability margin of cable-in-conduit conductors as a function of transport current and helium flow. The fold in the surface is associated with the multiple stability discussed in "Stability of Cable-in-Conduit Conductors" in Sect. 6.2.4.

Stability measurements were also carried out on a single triplex of the Westinghouse conductor and a full-size, insulated strand prototype. We were not able to quench the full-size conductor at operating conditions of 20 kA transport current in an 8-T background field, although 1.35-J heat energy was injected per cm^3 of conductor volume. Extrapolations from both single-triplex and one-third scale data corroborate this result.

To perform the stability measurements on the Nb_3Sn conductors, a pulsed heating technique was developed based on induction coils installed outside the conductor sheath and driven by a resonating capacitive discharge power supply. The power supply comprised four modules capable of being fired simultaneously in any combination of one to four. In the subsize test, peak powers injected into the conductor on the order of 170 kW for 3 ms were attained. Finally, measurements and calculations of the pressure rise during the quench of cable-in-conduit conductors have been started.

Abstract of Heating-induced flows in cable-in-conduit conductors.² The flow and pressure transients induced in stagnant supercritical helium in a pipe by energy transfers typical of those associated with the recovery of cable-in-conduit conductors ($\sim 50\text{--}200 \text{ mJ/cm}^3$ in about 10–20 ms) are calculated. Pressure surges of several atmospheres accompanied by fluid flow velocities of several meters per second are possible. These strong induced flows are probably the mechanism of recovery of cable-in-conduit conductors cooled by initially stationary helium.

Abstract of Normal zone propagation in thermally insulated superconductors.³ Scherer and Turowski⁴ recently reported measurements of normal zone propagation velocities in thermally insulated superconductors. After comparing their results with theory, they concluded that it was "difficult to find agreement." The lack of agreement is caused by their use of theoretical formulas that neglect current sharing. In this paper, theoretical formulas are derived that take into account both current sharing and the temperature variation of the thermophysical properties. These formulas are used to analyze Scherer and Turowski's data as well as some data of others.

Abstract of Stability measurements of a large Nb_3Sn force-cooled conductor.⁵ The Westinghouse coil for the LCP will use a cable-in-conduit conductor made of copper-stabilized, multi-filamentary Nb_3Sn strands enclosed in a stainless steel jacket. The operating current will be 16 kA with an 8-T maximum field. The stainless steel jacket provides a channel around the conductor to allow forced cooling by supercritical helium. In this study we investigate the performance of a subsize conductor similar in construction, but with only one-third as many active strands in the cable.

Abstract of Eddy current heating of irregularly shaped plates by slow ramped fields.⁶ Complementary variational principles (i.e., a maximum principle and a minimum principle) are given for the eddy current power in a thin conducting plate of arbitrary shape. These

principles are used to derive upper and lower bounds for the eddy current power. The bounds are of two types. The first type compares by inequalities eddy current powers in plates with different but related shapes. The shapes may be related by inclusion or by Steiner symmetrization. The second type of bound relates the eddy current power to certain geometric quantities dependent on the shape of the plate. Exact results are quoted for the circle, ellipse, equilateral triangle, and rectangle. Approximate results are given for the rectangle, isosceles triangle, sector of a circular annulus, rhombus, and L-shaped plate. A detailed comparison is made with the work of Sikora et al.⁷

Abstract of Lectures on the practical solution of differential equations.⁸ This report comprises lectures on the practical solution of ordinary and partial differential equations given in the In-Hours Continuing Education Program for Scientific and Technical Personnel at Oak Ridge National Laboratory.

Abstract of Development of a pulsed heating technique for stability testing in the Westinghouse LCP coil.⁹ In the Westinghouse LCP coil, the superconducting composite cable is enclosed in a stainless steel sheath, making the task of simulating transient heating for stability testing a difficult one. The technique described here uses heating by eddy currents in the copper matrix of the composite conductor induced by coils wound on the outside of the conductor sheath. The induction coils are driven by a special pulsed power supply designed to give the experimenter control over the amount of energy deposited. The frequency of oscillation of the induction coils can be chosen for most efficient transfer of energy to the conductor. Details of the power supply design and heating technique are described as well as experimental results from simulated operating conditions.

Abstract of Nb₃Sn cable-in-conduit conductor tests.¹⁰ Two tests were performed on the stability margin of cable-in-conduit conductor.

One was done on a 6 x 3" Nb₃Sn cable identical to the conductor to be used by Westinghouse in one of the LCP coils. The roughly 4-m-long conductor sample was tested by bending it into a paper clip shape, allowing two straight legs of the conductor to be located in the 15-cm high-field region of an 8-T split solenoid pair. Pulsed heating loads were imposed by oscillating magnetic fields from pulse coils wound around the outside of the jacketed conductor. At 20 kA of transport current (the limit of our vapor-cooled leads), no quenching of the conductor was observed with the capacitor bank discharged into the pulse coils from its maximum of 1.9 kJ. Another test was done on a single triplex that was the same size as the triplexes of the cable described above. Its stability was measured by overdriving the conductor transport current beyond its critical value by a pulse current superimposed on the steady-state current. The entire 3-m-long sample was coiled up and placed in a background field of 7 T. The stability margin was measured with steady-state transport current up to 94% of its critical value. Stability margins equal to or higher than the constant-pressure helium enthalpy difference between current sharing temperature and bath temperature were observed.

Transient properties of superconductors (ac losses)

We have carried out both theoretical and experimental studies of ac-loss properties of all six LCP coils. The results were also applied to the conductors being designed in the 12-T coil program, with the ETF plasma disruption condition taken into account. Major progress can be summarized as follows.

Losses in General Electric LCP cable conductor.

An experimental study was carried out to determine to what degree the superconducting strands can be coupled through the copper core in General Electric LCP conductor. The results, which agree well with our early analysis, indicated a fully coupled effect that exhibited

transient losses 50 times larger than those of a steel core. Final reports presenting the details of the measurement results as well as discussions with respect to the coil performance and its scalability to the ETF machine have been distributed to all LCP participants.

Abstract of AC losses in rectangular composites.¹¹ Detailed magnetization measurements are performed and analyzed for typical cryostable NbTi composite conductors (1 mm in diameter and with a copper-to-superconductor ratio of 6.5) with filaments nonuniformly distributed in both circular and rectangular cross sections (aspect ratio ~2:1). As a result of the excellent agreement between the measurements and a previously reported transformer model, both transient magnetization and losses incurred in each sample can be described by functions of a unique time constant τ_0 , which depends strongly on the shape of the cross section. Other prominent features of the measurement results, such as the deduction of values of J_c and matrix resistivity, are demonstrated. In an exponentially decaying field from 2 T to 0, J_c is 4.5×10^3 A/m² at 1.5 T, and the average matrix resistivity is 2.73×10^{-10} Ω -m. Saturation effects were also observed for field decaying time constant τ comparable with τ_0 .

Abstract of Paramagnetic properties of mixed-matrix NbTi conductors.¹² Magnetization measurements are performed up to 4 T on a series of samples with different compositions of copper, CuNi, and superconductors. It is observed that most samples exhibit a strong paramagnetic moment that is in contrast to the inherent hysteretic magnetization associated with superconducting filaments. Chemical analysis results of nickel contents and iron impurities are also reported and correlated with the measured paramagnetic properties. For samples with high contents of nickel (Cu-30%Ni) and iron impurities (>500 ppm), the saturated paramagnetic magnetization field can be as high as 0.02 T, and the initial susceptibility reaches $\chi = 2.50 \times 10^{-2}$. (In mks units, $\chi_g = 2.24 \times 10^{-4}$ emu/gm.)

Inductive voltage compensation in superconducting magnet systems.¹³ Several techniques of inductive voltage compensation are presented that have been developed for quench detection in superconducting magnet systems with multiple coils and power supplies, with particular application for the LCTF. Sources of noise, their magnitudes, and the sensitivity required for normal zone detection to avoid damage to the magnets are discussed. Two passive compensation schemes (second difference and central difference) are introduced and illustrated by parameters of LCTF; these take advantage of coil symmetries and other system characteristics. An active compensation scheme based on current rate input from pickup coils and using theory on ac loss voltage for calibration was tested, and the experimental setup and test results are discussed.

Instrumentation and diagnostics

The bulk of the work performed during 1979 concerning instrumentation and diagnostics was in support of LCP. Activities included development of specialized instrumentation for LCP coil diagnostics, technical guidance of the LCP instrumentation system design, and performance of verification tests in support of LCP coil contractor designs. Other work included strain gage measurements on the Impurity Study Experiment (ISX) and on the Coil Winding and Test Experiment (CWTE), a superconducting magnet for the Superconducting Magnet Development Program. The following lists some of the specific activities:

- Construction of an apparatus for testing cryogenic pressure transducers.
- Test of the long-term stability of Validyne CD-90 carrier amplifiers, which are being evaluated for use as a signal-conditioning unit for LCP strain gage measurements.
- Design and prototype fabrication of a strain-gage signal-conditioning transformer isolation system for use with carrier amplifiers in LCP. The system is capable of measuring signals of a few microvolts in wide-band noise

of a few volts and isolating the strain gages on a substrate (e.g., the coil conductor) at high voltage (up to 5000 V above ground). The unit was described at the Eighth Symposium on Engineering Problems of Fusion Research held in San Francisco, November 1979.

- Design and fabrication of a prototype non-isolated carrier amplifier system for use on grounded structures, such as the coil bobbins.

- Design and fabrication of a prototype moving-slug displacement transducer (MSDT) and associated signal conditioning electronics. The transducer will be used to measure displacement of the coil winding pack with respect to the bobbin.

- Design and fabrication by BLH of blanks for the thin film carbon thermometers. The carbon films will be applied at ORNL, and the completed thermometers sent to the coil manufacturers for installation.

- Design of cryogenic heat sinks for LCP with high-voltage standoff capability. Prototypes were tested to 2500 V after thermal cycling.

- Tests of strain gages selected for use in the LCP by the coil contractors. These tests included measurement of apparent strain versus temperature in the range of 300 K to 4 K and measurement of magnetic field effects from 0 to 8 T. A test was also performed to determine the effect of temperature changes on magnetoresistance near 4.2 K in Karma alloy foil gages.

- A series of steady-state and transient heat transfer tests for a simulated General Electric LCP winding configuration. This work was reported at the Cryogenic Engineering Conference held in Wisconsin, August 1979.

- Pulse test of a sample of the General Electric heater for LCP on a length of LCP preprototype conductor.

- Instrumentation of the ISX finger joints with strain gages and use of our isolated carrier system to measure the strains. This work was also reported at the Eighth Symposium on Engineering Problems of Fusion Research.

- Instrumentation of CMTX with strain gages and displacement transducers.

- Test of the thermal wedge for LCP. The test wedge was instrumented with thermocouples, resistance thermometers, strain gages, and displacement transducers, and thermally cycled in the constrained and unconstrained condition. Data were taken by an automatic digital system and recorded on disc for further analysis.

Abstract of Application of microcomputers to superconducting magnet experiments.¹⁴

Three types of microcomputer applications have been implemented for superconducting magnet experiments: (1) a dedicated programmable waveform generator, (2) a dedicated remote data acquisition terminal, and (3) a stand-alone data processing unit for experiments. We report here our experiences with these implementations and the rationale behind the choice of hardware and software in each type of application. In particular, details are given on implementing an Intel SBC 80/10 microcomputer system as a remote data acquisition terminal of a PDP 10 and as an on-line signal conditioning device.

Abstract of Heat transfer and helium replenishment in cabled conductor cooling channels.¹⁵

In the design of the conducting element of the General Electric LCP test coil, it is necessary to know the heat transfer capabilities of the conductor as a function of orientation (location along the coil perimeter) in the local hydraulic environment, that is, with helium flow passages in the conductor and turn-to-turn insulation. In the present work, a sample winding stack of 30 prototype conductor elements similar to the LCP design was assembled by General Electric and sent to ORNL. There, it was instrumented with heaters and thermocouples and then was subjected to heat pulses while immersed in a bath of liquid helium. Data were taken by a fast digital recording system. The data, when extrapolated to the final LCP conductor configuration, indicates the coil will be stable at the design current.

Abstract of Analysis and test to predict the fatigue life of the ISX-B toroidal field coils' finger joints.¹⁶ A new and more rigorous structural evaluation of the ISX toroidal field (TF) finger joints was undertaken to assess the effects of high-beta operation of ISX-B. A new poloidal field coil set that allows high-beta operation and produces larger out-of-plane loads on the TF coils was installed as part of the change to ISX-B. It was determined that the iron core significantly affects the out-of-plane load distribution, and forces were calculated using the GFUN3D code, which considers three-dimensional (3-D) iron core effects. These loads were applied to a half-symmetric, finite-element NASTRAN code model in which the TF coils were modeled as a string of beam elements. A more detailed local model of the finger joint region, which included modeling of the joint's bolts, was used to determine the detailed internal load distribution. Strain gages were applied directly to the copper surface of two TF coils, and the results were correlated with analysis. The strain gage technique, which requires voltage isolation and signal noise reduction, was developed at ORNL, and this represents the first major use of this technique. The TF coil chock block supports permitted direct measurement of out-of-plane reactions by use of piezoelectric load transducers. Forces were directly recorded, with dynamic effects readily apparent. Dynamic magnification factors were significant, especially during a plasma disruption, and are compared with theoretical values. Stress magnitudes using stress concentration factors were calculated. Fatigue life was predicted for a desired spectrum using a strain life fatigue code. Safe operating conditions were established that will ensure sufficient fatigue life for ISX-B, including the future use of a bundle divertor.

Abstract of The large coil test facility instrumentation system design.¹⁷ The design of the instrumentation system for the LCTF is described. Sensors are divided into two

categories: coil diagnostic sensors and facility sensors. The coil sensors are installed in the coils and lead wires brought out to interface connectors by the coil contractors. Signals from the interface connectors either are taken directly to the control room or pass through one or more stages of patching before going into the signal conditioning. For sensors that may "float" at the coil dump potential, isolation amplifiers or transformers are provided. Overvoltage protection is provided for ground-referenced sensors inside the coil case. After signal conditioning, data from fast channels (e.g., voltage taps) are multiplexed, digitized, and stored in up to four microcomputer systems programmed to be used in a ring buffer mode in order to record data before and after receipt of a random trigger from the normal zone detection circuitry. The microcomputer systems are interfaced to a central minicomputer system with cathode ray tube (CRT) display, magnetic disc and tape storage, and hard copy graphics output. Slow channels are multiplexed and digitized by a 1000-channel scanner also interfaced to the central minicomputer. Redundant panel meter displays and chart recorders are provided for critical channels. Facility variables are digitized by a separate scanning system that can be patched into the coil diagnostic data system if needed. Selected facility variables are also displayed continuously on panel meters. Finally, a number of unassigned panel meters and chart recorders into which variables of interest can be patched are provided. Critical fault variables are entered into a programmable logic controller for automatic coil dump in case of certain fault conditions. One of these variables is the output of hardware quench detection units, which declare a quench on the basis of threshold criteria for magnitude and duration of inductively compensated voltage tap signals.

Abstract of Thermally actuated shims provide passive clamping device.¹⁸ Thermally actuated shims designed for the LCTF provide passive

clamping capability for inaccessible structural interfaces. These shims were designed to hold the test coils tightly in place against the bucking post. This particular interface is relatively inaccessible, tends to loosen as the structure cools down, and experiences high loads owing to magnetic interaction of the test coils. As designed, the shims are loose at assembly and tighten as the temperature of the structure decreases. These devices utilize the difference in thermal coefficient of expansion of materials to drive a double-acting wedge, which causes an increase in shim thickness as the temperature is decreased. A prototype shim has been instrumented and tested to verify calculated performance.

Abstract 6- A strain-gage signal-conditioning system for use in the LCP.¹⁹ A strain-gage signal-conditioning system providing wide-band noise rejection and isolation from high voltages occurring during emergency coil discharges has been developed and tested. The multichannel system combines double-shielded transformers, neutralizing networks, and bandpass filters with commercial 3-kHz carrier amplifier modules to isolate the strain gages to 5000 V, eliminate thermoelectric effects, and provide a signal bandwidth of 200 Hz. Common mode interference occurs primarily as a result of "beat note" effects between the carrier and the superimposed noise at frequencies near the odd harmonics of the carrier. The common mode rejection of the test circuit was measured to be 120 dB for noise at 2750 and 3250 Hz, 135 dB at 3 kHz, and 135 dB and better at the odd harmonics of 9 kHz and above. The system has been successfully used in strain measurements on the toroidal field coils of the ISX-B tokamak and will be used in the LCTF to monitor strains in the energized coil conductors.

LCTF structural analysis

The LCTF test stand is designed to provide restraint for a variety of both in-plane and out-of-plane loading conditions while supporting one to six dissimilar coils. The six toroidal

coil designs have different structural stiffnesses and current densities that reflect different preferences in the satisfaction of the LCP design specification. Evolving competitive designs are not necessarily available for free exchange between the coil manufacturers. Considering the influence of other coil manufacturer's designs, this problem of design and analysis was resolved by the establishment of a scheme in FY 1978 that enables the coil designers to design their coils with a limited amount of knowledge regarding the LCTF. This, in turn, allows ORNL to design the LCTF without detailed characterization of the various coils. The procedure involves the determination of the LCTF structural behavior at its interface with any selected test coil; for the purposes of our analysis, the coils were assumed to be identical. [Under the direction of ORNL, Science Applications Incorporated (SAI) has performed the overall LCTF analysis.] The resulting interface response is then available to the coil designers to represent the behavior of the remaining LCTF structure in their detailed coil designs. Similarly, the facility side of the LCTF interface is then available to ORNL for determination of input forces and deflections to more refined models of LCTF components. Three major LCTF components were selected for a more refined structural analysis in FY 1979. They were the spider base, bucking post, and torque rings.

Spider base structural analysis. Several finite-element models, each with progressive refinements, were used in the spider base analysis. The models were created with plate and boundary spring elements. Loads were obtained from the SAI LCTF structural model; however, the worst-case load was that of a 12-T coil's weight (an extended design consideration). The appropriate simulation of pinned joints and rollers was considered, and the symmetry conditions were used to model the boundary conditions.

This model was refined to analyze the stress concentration attributable to the bolts

that penetrate the top surface and attach the bucking post collar to the spider base. These models were used to size plate thicknesses for the spider base to maintain a working equivalent stress of 115 MPa at room temperature and of 138 MPa at 4 K.

Bucking post structural analysis. Because of the complexity of the applied loads (again, these loads were obtained from the SAI LCTF finite-element model) and the difficulty in understanding the bucking post response to these loads, a simplified procedure evolved. All the known loads were divided into their simplest components. These load components were individually applied to several finite-element models. The resulting stresses were determined per unit of applied load component. This procedure yielded scaling factors that can be used to analyze the stress conditions for any combination of the applied load components. The results can then be superimposed to find the total stress state in any region of the bucking post. After using the scaling procedure, the worst-case equivalent stress for a loading conditions was found to be 208 MPa, existing in the root of the bucking post groove adjacent to a zero current coil.

Torque ring structural analysis. The LCTF torque ring consists of three identical segments, each of which spans two coils. A series of finite-element models, each more complex than the last, was used to reflect design considerations as the analysis progressed. The models were generated using plate and beam elements and with spring elements representing boundary conditions. Worst-case loads occurred in analyzing the two-coil test — a proposed test of just two coils for shakedown of the LCTF. These loads produce a torsional deflection of the torque ring. Symmetry was considered in the two-coil case, allowing a one-sixth symmetric model. Results were used for the sizing of the pipe members and plate thicknesses throughout the torque ring. From this analysis it has been

learned that the torque ring design is extremely sensitive to changes in interface loads. Because of this, a new LCTF finite-element model is in the process of being created.

The new LCTF structural model. With the experience gained from the orderly progression of structural analysis of the individual components of the LCTF, a new structural model of the individual components of the LCTF is being created. This analysis will proceed into FY 1980. Contained within this model are the four major components of the LCTF: the spider base, the torque rings, the bucking post, and the coils. Because of several unresolved concerns with the behavior of the torque ring, a simulation of the two-coil test has been decided upon for the first phase of the new structural analysis task. Therefore, only two coils and a 120° segment of the upper and lower torque rings are modeled, along with a complete spider frame and bucking post.

The upper and lower torque rings are discretized with plate, beam, and rod elements. The spider base is modeled with plate and beam elements. Since the mechanical response of the spider base is so beam-like, a simplistic model can be used to capture the correct behavior.

At present, two coils are modeled for the beginning shakedown of the model. They are identical and reflect General Dynamics' current toroidal field coil design. A single rod is used to model the conductor winding pack, and this rod is connected to the coil case via two radial and two axial rods. The coil case is modeled with plate elements of the appropriate thicknesses.

Other LCTF structural analysis. The lifting frame for the LCTF vacuum tank top was analyzed using the finite-element method. An estimated acceleration load was applied to the finite-element model of the frame to simulate a worst-case condition. It was determined from the analysis that to maintain reasonable deflections between the frame legs another connection was required. The frame was built,

modified, and successfully used to lift the vacuum tank top.

Other support activities

LCP contract monitoring. A significant portion of time has been devoted to reviewing the U.S. LCP coil contractors' test coil designs. Meetings have been held both at ORNL and design team sites to present the latest contractor design to GRNL. Recommendations have been made for both improvements in hardware design and analysis techniques.

Maintenance of structural analysis programs.

In order to respond rapidly to design changes in LCTF hardware, the structural analysis computer codes GIFTS4 and SAP5 have been maintained for ORNL by the Magnetics and Superconductivity Section. These codes allow structural analyses to be performed interactively, thereby reducing the turnaround time for an analysis. Also, to take advantage of the National Magnetic Fusion Energy Computer Center (NMFECC), the GIFTS code was modified to execute on the NMFECC CARY-1 computer at Livermore. This code will be made available to the entire structural analysis community as soon as this version stabilizes.

Maintenance of computer graphic software.

At the core of all good structural analysis programs is software that will plot various important structural parameters. These plotting packages are invaluable during the model generation, solution, and results phase of a structural analysis. Because these plotting packages usually require nonstandard interfaces to plotting hardware, numerous interfaces of this type have been created and are maintained by this group. These interfaces have been provided to the entire Fusion Energy Division to prevent duplication of effort.

CWTFX structural analysis. Several parts of the CWTFX coil have been structurally analyzed using the finite-element method (for fault conditions) and STANSOL (for normal operating conditions). Also, STANSOL was used to calcu-

late the buildup of compressive stress in the CWTFX bobbin during winding. Unfortunately, the Nomex lacing tape used to insulate the windings exhibited mechanical creep during the winding process. This produced a winding that was not as tight as was hoped for. The comparative evaluation between experimental bobbin winding strain data and the output of STANSOL was extremely poor because of STANSOL's inability to model creep.

Other finite-element models examined the response of the end plates during a hypothesized fault condition and determined the stress concentration at the lip of the sample hole between the two coil halves. The fault condition required a rather thick end plate for the coils. The stress concentration at the lip was not excessive and did not require redesign.

ISX-B magnetic field analysis. In order to investigate the influence of ISX-B's iron core upon the magnetic flux lines around a proposed bundle divertor, a computer program was written to draw flux lines in a 3-D space. This program used magnetic fields calculated from the GFUN3D computer program and isoparametric finite elements to follow flux lines within a region of interest. The impact of the iron core is still being assessed.

EBT-P structural analysis. During the proposal phase, several structural models of EBT-P components have been analyzed. STANSOL was run to determine assumed axisymmetric stress conditions within the windings of the magnetics. Also, an axisymmetric finite-element model of the ORNL split-bobbin design was made to size the wall thicknesses. Future work requires full 3-D models of the coil and its support structure (which is not axisymmetric).

Abstract of A derivation of bending-free toroidal shell shapes for tokamak fusion reactors.²⁰ This paper describes an extension of mathematical solutions to bending-free shapes of toroidal pressure vessels. Previous work on this subject derived a pure tension

shape of a toroidal membrane that was subjected to a uniform internal pressure. This research was directed toward finding a pure tension shape of a toroidal membrane that is subjected to a pressure inversely proportional to the square of the toroidal radius. Such a relationship is possible when electromechanical forces are considered. This family of shapes, which are nearly circular for aspect ratios greater than 2.5, is derived by solving the equilibrium, constitutive, and kinematic relationships for a uniform toroidal shell using the linear membrane theory. Serious consideration should be given to these toroidal shapes, since they provide alternative methods of design for tokamak fusion reactors.

Abstract of A user's guide to program MAD: a computer program for the organization and manipulation of magnetic tape directories.²¹ MAD is a computer program for the organization and manipulation of the information contained in magnetic tape directories. Program MAD creates, updates, and interrogates a set of four random access files collectively called the MAD unified data base. Although program MAD was originally intended as an information compression mechanism, it has evolved into an organization system with the added feature of an approximately 60% reduction in the space required to store the data. This program is easy to use, relatively fast, efficient in its use of disk space, and available to all users of the Fusion Energy Division DECsystem 10.

Abstract of The structural design of superconducting magnets for the large coil program.²² This paper presents a review of the pre-Phase II status of the structural designs, analysis methods, and verification tests being performed by the participating LCP design teams in the United States, Switzerland, Japan, and the Federal Republic of Germany. The significant structural mechanics concerns that are being investigated with the LCP are presented.

Abstract of User and system considerations for the TCSTEK software library.²³ This report documents the idiosyncrasies of the Tektronix

PLOT 10 Terminal Control System level 3.3 software as it currently exists on the ORNL Fusion Energy Division DECsystem 10 computer. It is intended to serve as a reference for future Terminal Control System updates in order that continuity between releases of Terminal Control System PLOT 10 software may be maintained.

Abstract of Status of the structural design of superconducting magnets for the large coil program.²⁴ This paper presents a review of the status as of October 1979 of the structural designs, analysis methods, and verification tests being performed by the participating LCP design teams in the United States, Switzerland, Japan, and the Federal Republic of Germany.

Abstract of The structural analysis of magnetic fusion energy systems in a combined interactive/batch computer environment.²⁵ A system of computer programs has been developed to aid in the preparation of input data for and the evaluation of output data from finite-element structural analyses of magnetic fusion energy devices. The system uses the NASTRAN structural analysis computer program and a special set of interactive pre- and postprocessor computer programs and has been designed for use in an environment wherein a time-share computer system is linked to a batch computer system. In such an environment, the analyst must only enter, review, and/or manipulate data through interactive terminals linked to the time-share computer system. The primary pre-processor programs include NASDAT, NASERR, and TORMAC. NASDAT and TORMAC are used to generate NASTRAN input data. NASEKR performs routine error checks on these data. The NASTRAN program is run on a batch computer system using data generated by NASDAT and TORMAC. The primary postprocessing programs include NASCMP and NASPOP. NASCMP is used to compress the data initially stored on magnetic tape by NASTRAN so as to facilitate interactive use of the data. NASPOP reads the data stored by NASCMP and reproduces NASTRAN output for selected grid points, elements, or data types. This reproduced data can either be evaluated directly or used as input to a variety of special-purpose

programs that can tabulate, manipulate, and graphically display results. The structural analysis system has been used to evaluate tokamak devices such as ISX-B and LCP. Results are described from a sample problem that consists of two LCP type coils and their attaching structure.

Abstract of On maximum attainable magnetic fields in toroids and solenoids subject to strain limitations.²⁶ Several simple scaling laws are derived for the maximum attainable magnetic field in superconducting solenoids and constant tension D-shaped toroids. These scaling laws are based on magnet geometry, current, material properties, and allowable strain.

Abstract of Large coil test facility test stand design analysis.²⁷ The LCTF test stand (a component that supports the large coils) is designed to provide restraint for a variety of both in-plane and out-of-plane loading conditions while supporting one to six dissimilar coils. The current design analysis of the LCTF test stand is documented by this paper.

Abstract of System software for the ORNL version of GIFTS 4B.²⁸ ORNL maintains a version of the GIFTS system structural analysis computer program modules. Presented herein are the local software enhancements for the ORNL version of GIFTS as well as a description of the ORNL implementation of the system-dependent portions of the GIFTS software library.

Coil fabrication technology

The coil winding facility has been used for practice winding of the type of conductors that will be employed by industrial participants in the LCP. The practice winding was done on a D-shaped bobbin that is approximately 3×5 m and with conductor tensions up to 8500 N. The most extensive work was layer-winding a solid copper conductor (9×31 mm) on edge. This simulates the superconductor that General Dynamics will use (see Fig. 6.2). It was found that most of the winding problems result from

using a nonround bobbin. Some of the more important parameters bearing on this problem are conductor tension, precision of the winding surface, and clamping forces. Other important things were learned about routing the leads, installing insulations, and electrical test during winding.

CWIX fabrication. Winding techniques and equipment were developed and built for winding the CWIX coils. (CWIX is described below.) A knurling device that applies a high-friction surface to the superconductor and a spiral wrapping insulating device were added to the existing conductor tensioner. Equipment for manufacturing the insulating fillers was built. A technique for making in-line bumpless superconducting splices was developed. Lastly, an involved winding method was used to take advantage of a unique coil winding having both layers and a pancake. This type of winding brings both coil leads to the outside, which ensures that a lead will not penetrate the bobbin.

Facilities

CWIX magnet system. This year the winding and assembly of the CWIX were completed. The system consists of a concentric pair of solenoids: a low-field (8-T) background coil and an insert coil that provides a higher field (12 T). Respective diameters are 38 and 20 cm. Radial access is provided at the midplane by two mutually perpendicular cylindrical ports 6.7 cm in diameter extending through both coils (see Fig. 6.3). The pair of magnets may be operated with the central field either horizontal or vertical. Horizontal operation permits testing of short samples in a vertical orientation, allowing them to be inserted and removed from one of the access ports without warming the magnet. To test the sample in a horizontal orientation the other access port may be used, but the magnet must be warmed to change the sample. The magnets can also be mounted so that the field vector is vertical for testing solenoidal coils in a background

Fig. 6.2. Practice winding of General Dynamics LCP conductor was carried out at ORNL using a half-size bobbin.

Fig. 6.3. The CWTX solenoid produces 8 T in a radial access port of 67 mm in diameter. The magnet is shown here instrumented in preparation for the initial test.

field or conductor samples that can be wound in the space available. This entire range of operations can be used with the pair of magnets (12 T) or with only the background coil (8 T).

The system was conceived and designed in 1978 to develop winding and testing procedures that might be useful for fusion facility operation. Additionally this system would provide a facility for testing large conductor specimens at full rated current in high magnetic fields.

Winding and assembly of the 8-T coil were completed in early fall of 1979, but difficulties with the dewar have resulted in postponing the test until early 1980. Most likely this test will be coordinated with the LCP refrigerator to reduce helium costs.

Parameters of the system are given in Table 6.1.

Table 6.1. Parameters of the CWTX magnet system

Low-field section	
Bore	380 mm
Central field	8 T
Radial access ports	66-mm diam
Outside diameter	1 m
Length	0.8 m
Conductor	10 × 3-mm compacted cable
Operating current	2200 A
Number of turns	2850
Stored energy	6 MJ
Average current density	5100 A/cm ²
Total weight	1900 kg
High-field section ^a	
Bore	200 mm
Central field	12 T
Radial access ports	66-mm diam

^aOrdered from IGC; delivery expected June 1980.

6.3 12-T COIL DEVELOPMENT PROGRAM

6.3.1 12-T Program

The General Electric-IGC team finished its ETF Scoping Study and issued its final report

on November 2, 1979. The conductor concept they chose on the basis of this study is a graded bath-cooled Rutherford cable of 16 triplets of 3-mm Nb₃Sn-bronze-copper strands (12-T grade) in a U-shaped stainless steel channel. The conceptual design of a model coil pancake wound with this conductor is finished; the final design review was scheduled for January 3, 1980. ORNL is currently studying a proposal from General Electric for the detailed design and construction of the model coil.

The IGC Material Development Program on the external bronze process is divided into four subcontracts: (1) improvement of the plating, diffusion, and reaction steps; (2) improvement of the high-field critical current by adding tantalum to the niobium; (3) improvement of the high-field critical current by adding a third element (indium, gallium) to the tin plate; and (4) production of copper-niobium multifilamentary wire. Among the important achievements of this program this year are (1) successful production of 0.3-mm solid strands (filaments all the way through) and 0.4-mm strands with a 0.2-mm, tantalum-protected copper core; (2) demonstration of a current density at 12 T in excess of 5×10^4 A/cm² over the bronze-niobium area of these strands; and (3) substantial increases (about two to three times) in high-field critical current density resulting from the addition of tantalum to the niobium.

6.3.2 12-T Program

This program was initiated by signing contracts with Airco, Inc. and General Dynamics-Convair for conductor development and 12-T coil designs.

Airco, Inc. has two development contracts, one for a new technique for producing Nb₃Sn multifilamentary conductor and a second to start production of conductor for a model coil to be tested at a later date. Both contracts are about half completed and are

progressing satisfactorily. The new manufacturing technique is based on the manufacture of high-purity bronze tubing. This part has been successfully demonstrated, resulting in the establishment of a commercial source for the tubing. It is anticipated that a considerable cost reduction for certain conductors will result.

The contract with General Dynamics consisted of two phases, a scoping study of a 12-T tokamak machine based on ETF parameters and conceptual design of the model coil to test the conductor chosen in the scoping study. The scoping study was completed in 1979, and a monolithic pool-boiling Nb_3Sn conductor was shown to meet the requirements of ETF. Specifications for coil parameters and structure were selected and used to evaluate the performance of the conductor in a tokamak device.

6.4 MAGNET DESIGN -- EBT-P PROTOTYPE COILS

During the preliminary conceptual design phase of EBT-P, it was recognized that a magnet development program was needed to reduce the risk of manufacturing the full complement of coils. Because it was to be about a year before the major industrial participant and his coil manufacturer were selected and because of the short construction schedule for EBT-P, this development program was begun at ORNL.

Before the preliminary designs of the four industrial groups and the preliminary design by ORNL were brought together and the proposed reference design (PRD) chosen, two magnet types were included in the program. One was a NbTi metastable pool-boiling coil like the one described in the ORNL Preliminary Conceptual Design Report²⁹ and the other was a forced-flow Nb_3Sn coil. The main line effort was the NbTi coil. The Nb_3Sn coil was to be developed as a backup option.

When the five preliminary designs were compared and discussed in a series of meetings at ORNL in October and November and the PRD

was chosen by ORNL,³ the design of the development coil was changed to the PRD magnet, and the Nb_3Sn effort was stopped.

The specific objectives of the Magnet Development Program are (1) to demonstrate design features and fabrication techniques that will produce the required field accuracy and eliminate systematic (cumulative) field errors; (2) to solve the problems of structural containment and of maintaining proper alignment in the dewar after cooldown and energizing; (3) to evaluate thermal design features and determine heat removal capability; and (4) to develop techniques, facilities, and criteria for acceptance testing of the coils manufactured for EBT-P.

Magnet development as presently planned consists of designing, fabricating, and testing two instrumented test magnets of the NbTi pool-boiling type as proposed for the EBT-P reference design. The coil tests will include simulating the heat input from gamma rays and other sources. The magnets will first be tested separately and then in a two-magnet array. This task is to be completed by November 1981.

The NbTi superconductor for the test magnets was specified in FY 1979 based on ORNL's proposed design for EBT-P, and sufficient superconductor was ordered for two test magnets. Delivery is expected in July 1980.

The development schedule as now proposed is intended to make significant test results available before ordering the superconductor for all magnets and early enough (about April 1981) to impact the detailed design for EBT-P. Essentially complete test results will be available prior to the start of magnet manufacture (about April 1982). The industrial participant who will direct the final design and manufacture of the mirror magnets will be selected and will interact with the development activity, assist in determining appropriate acceptance tests, monitor or participate in performing the tests, and apply the results and experience to work on the EBT-P magnets.

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7. ENGINEERING TEST FACILITY DESIGN CENTER

7. THE ENGINEERING TEST FACILITY

ABSTRACT. The vehicle by which the fusion program would move into the engineering testing phase of fusion power development is designated the Engineering Test Facility (ETF). The ETF would provide a test bed for reactor components in the fusion environment. In order to initiate preliminary planning for the ETF decision, the Office of Fusion Energy established the ETF Design Center activity to prepare the design of the ETF. This section describes the status of this design.

7.1. INTRODUCTION

In September 1978 the U.S. Department of Energy (DOE) issued a policy statement for fusion energy.¹ This policy statement encompassed the efforts of both the magnetic confinement and the inertial confinement approaches. In order to develop fusion energy as an economically attractive and environmentally acceptable energy option, the statement outlined a three-phase strategy. Sequentially, these phases focus on scientific feasibility, engineering testing, and reactor demonstration.

It is anticipated that scientific feasibility will be achieved within the 1980s for both the magnetic confinement approach and the inertial confinement approach. Following the achievement of scientific feasibility, each of the two approaches would move from the applied research phase into an engineering testing phase. The vehicle by which the fusion program would move into this phase of development is designated the Engineering Test Facility (ETF). The ETF would provide a test bed for reactor components in a fusion environment. These components would be the essential building blocks of the facilities constructed and operated during the reactor demonstration phase. These reactor demonstration phase facilities are designated the Engineering Prototype Reactor (EP2) and the Commercial Demonstration Reactor (DEMO).

In order to initiate preliminary planning for the ETF decision in the magnetic confinement approach, the Office of Fusion Energy (OFE) established the ETF Design Center activity at Oak Ridge National Laboratory (ORNL) to prepare the design of a tokamak ETF. At the

same time, a mirror ETF design task was established at Lawrence Livermore Laboratory. ETF design tasks will be considered for other confinement concepts as their physics performance is brought up to the level that has been achieved by the tokamak and mirror experiments. Currently, the tokamak ETF effort at the ETF Design Center is receiving the highest level of support because tokamaks have achieved the highest level of plasma performance. The emphasized effort directed toward the tokamak will be of general value to the magnetic fusion program. It is noted that many of the key technological and engineering problems of magnetic confinement fusion concepts are generic in character. Such generic issues are highlighted by the needs in materials, magnets, plasma heating by beams and by radio frequency (rf) energy, impurity control and ash removal, tritium handling, maintenance, and availability/reliability.

The ETF Design Center activities are divided into five areas as indicated in Fig. 7.1. This figure also indicates the scope of responsibility under each area. As shown in Fig. 7.2, the Design Center organization represents a unique laboratory/industry partnership. The participation is split 60%/40% between industry and the laboratory, respectively. Approximately 70% of the work is carried out at the Design Center, and approximately 30% is carried out at the home organizations.

To initiate the ETF Design Center activities, it was deemed essential to establish a statement on the mission of the ETF in the overall fusion program strategy. In the TNS (The Next Step) activities of FY 1978,²⁻⁴ the various design teams addressed the issue of the mission

ORNL-DWG 80-2207R FED

Fig. 7.1. ETF Design Center areas of activity.

ORNL-DWG 80-2352 FED

A UNIQUE LABORATORY/INDUSTRY PARTNERSHIP HAS BEEN IMPLEMENTED
WITH THE ESTABLISHMENT OF AN ON-SITE DESIGN TEAM

Fig. 7.2. Laboratory/industry partnership.

for the TNS. The OFE used the mission statement results of last year's TNS activities to develop a preliminary mission statement for the ETF. In order to develop a more detailed and complete mission statement for the ETF, the ETF Design Center organized and conducted an ETF Mission Workshop held in Knoxville, Tennessee, February 13-15, 1979. The Mission Workshop brought together a broad range of expertise from within the fusion community and involved representatives from both the fusion research centers and industry. Approximately 125 participants representing 25 organizations attended the workshop. The ETF Design Center used the results of the workshop as a major input in developing the Mission Statement. The deliberations of the workshop and the resulting Mission Statement have been docu-

mented.^{5,6} The Mission Statement Document represents an initial effort and will be reviewed at least once a year for possible revisions. The essential elements of the ETF mission are summarized below.

1. The ETF should serve as a focus for developing and testing reactor-relevant technologies and components.
2. The ETF should demonstrate the operation of a reactor-relevant fusion core. This demonstration consists of the control and reproducibility of plasma parameters at fusion power densities of several MW/m³.
3. The ETF should demonstrate engineering operations in a reactor-relevant environment. This demonstration includes maintenance operations (both remote and contact),

safety procedures, and environmental monitoring.

4. The ETF design philosophy should allow for flexibility in the testing program. This flexibility applies both to the testing of components and to possible machine upgrades (e.g., electricity generation).

7.2 ETF DESIGN STATUS

To provide a point of departure for the ETF design effort, the ORNL TNS design⁷ (Figs. 7.3 and 7.4) was adopted as the ETF Baseline Design. This device was designed as an ignited, deuterium-tritium (D-T) burning, long-pulse, air-core tokamak. It has a D-shaped plasma with a major radius of 5 m, a minor radius of 1.2 m, and an elongation ratio of 1.6 (see Table 7.1). The fusion power is 1140 MW during burn.

The toroidal plasma chamber is formed by the assembly of 16 shield sectors. The bulk shield sector comprises a stainless steel tank loaded with stainless steel balls and cooled

by borated water. Water-cooled tubular panels are installed on the plasma side of the shield sector to provide a first wall structure.

The toroidal field of 5.3 T on axis is provided by 12 Nb₃Sn superconducting coils. These toroidal field (TF) coils are nested around a central coil support cylinder that accommodates the centering loads.

The poloidal field (PF) system, which consists of a superconducting ohmic heating (OH) solenoid and a combination of superconducting and copper equilibrium field (EF) coils, induces the plasma current and provides plasma position control. The EF coils located outside of the TF coils are superconducting; those trapped inside the TF coils are copper so that they can be disassembled and replaced.

Impurities are removed from the plasma chamber by means of a bundle divertor. An array of lithium jets is used to intercept the helium and impurity ions magnetically diverted from the periphery of the plasma. Pellet injection is used for fueling.

Fig. 7.3. ETF Baseline Design elevation view.

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Fig. 7.4. ETF Baseline Design plan view.

Table 7.1. ETF Baseline Design parameters

Plasma major radius, R (m)	5.0
Plasma elongation, δ	1.6
Plasma minor radius, a (m)	1.2
Plasma volume, V_p (m ³)	230
Plasma current, I_p (MA)	5.0
Neutron wall loading, L_w (MW/m ²)	2.4
Total fusion power, P_{tot} (MW)	1140
Fusion power density, n (MW/m ³)	5.0
Number of TF coils	12
TF coil vertical bore (m)	9.9
TF coil horizontal bore (m)	6.2
Field at TF coil, B_m (T)	10.9
Field on axis, B_T	5.3
Steady-state burn time (s)	500
Total cycle time (s)	560
Total Wh	83
Neutral beam power, P_{inj} (MW)	50
Neutral beam energy, E_{inj} (keV)	150
Injection time, τ_{inj} (s)	6.0
Microwave power (MW)	1.0
Microwave frequency (GHz)	120

Plasma startup is assisted by the injection of 1 MW of power at 120 GHz, and bulk heating is accomplished by the injection of 50 MW of 150-keV neutral beams.

The major thrust of the ETF Design Center activities during FY 1979 has been directed toward project definition. Starting in FY 1980 and continuing for approximately 18 months, the activities will be directed toward preconceptual design. During the Preconceptual Design Phase there will be two formal review points, the first to take place approximately nine months after the initiation of the Preconceptual Design Phase and the second to take place approximately nine months later. These two formal reviews will be supplemented by an ongoing interaction between the Design Center and the fusion community.

The preconceptual design effort will center around two design envelopes departing from the baseline design. The first design envelope will include a bundle divertor, a conventional TF coil system (i.e., a TF coil system with

associated ripple within the ripple limits), and both an OH system startup and an assisted startup scenario (e.g., rf preheating). The second design envelope will include a poloidal divertor, a TF coil system incorporating a minimum number of coils with means for compensating for the associated high ripple, and an OH system startup. The goal of the Preconceptual Design Phase is to identify a single design that would be the basis for the Conceptual Design Phase. Some specific design issues are considered below.

7.2.1 First Wall

The first wall is being designed to accommodate the energy associated with the alpha particles not removed by the divertor system. Current estimates of this wall load range from 50% to ~70% of the total alpha particle energy, which constitutes 20% of the total plasma energy produced.

A water-cooled panel system has been proposed for the first wall in the ORNL TNS studies.⁸ In this design 2.5-cm stainless steel tubes are welded to headers and brazed to stainless steel sheets to form individual panels. Each of the faceted inner surfaces of the plasma chamber is lined with an independently cooled panel. Because the ORNL TNS torus has 16 segments and each segment has 6 facets, a total of 96 panels is installed.

Studies have indicated that the effects on the plasma of impurity sputtering at the plasma chamber walls can be substantially mitigated by coating the chamber wall with a material of low atomic number such as carbon.⁹ A carbon coating ~25 μm thick should prevent sputtering of the nickel and iron in the basic panel material.

In addition to accommodating the alpha particle heat load and limiting the effects due to sputtering, the first wall must withstand the deposition of energy due to plasma disruptions. This poses a severe problem because it is expected that disruptions will occur on a time scale that will result in vaporization of the first wall material,

suggesting the need for a thicker section for the first wall. Thicker sections, however, increase the thermal stresses during normal operation and reduce the wall lifetime.

One approach to accommodating both the normal thermal loading and some disruptions is to install radiation-cooled tiles at the face of a convectively cooled structure.¹⁰ In order to limit the impurity effects of the tiled wall sputtering on the plasma, these tiles should be made of a low-Z material such as graphite. However, above ~1500°C, hot graphite combines with atomic hydrogen to form acetylene. It is therefore necessary to make the tiles relatively thin; consequently, they will be able to withstand only a very limited number of disruptions.

However, if insignificant disruptions can be sensed and if they can be controlled to impact on a preferred surface, only one wall need be protected. In this case the tiles can be cooled by radiation to the opposite wall as well as to the wall upon which they are mounted, and they can be made much thicker. Studies indicate that a 2-cm tile mounted on the inboard wall can be adequately cooled during normal plasma operation and survive ~500 full-power disruptions, if the energy load from the plasma is uniformly distributed over the inboard wall and the carbon is vaporized as monatomic carbon. Substantial R&D is required to establish whether the direction of the plasma impact can be controlled and to determine the distribution of the heat load upon impact. The distribution among the carbon species upon vaporization also has to be determined.

7.2.2 Shield

A major concern in the ETF program is to provide high availability for test operations. Previous studies have concluded that provisions should be made to maximize contact maintenance outside the reactor shield.^{11,12} Accordingly, the outboard shield is being designed to limit the personnel dose rate to 1 mrem/h 24 h after reactor shutdown. This results in a shield thickness of 1.2 m for the torus bulk shield.

The neutral beam injectors, plasma chamber vacuum pumps, divertors, and the ducts connecting this ancillary equipment to the reactor also have to be shielded. These shields range from 0.5 m to ~1.0 m, depending upon the composition selected. In some cases it may be desirable to use rather sophisticated designs so that the shield can be easily removed for access. At other locations it may be desirable to use thicker designs of less effective material to minimize the shield cost.

Another design consideration affected by the availability requirement is the vacuum topology. The current plasma chamber design concept employs a segmented shield so that the first wall and shield can be removed in large sections. The vacuum boundary is located at the inside surface of the shield segment, and vacuum seals between segments are provided at the outer surface of the segments.

A novel approach for welding and cutting the seal between adjacent segments without the necessity of gaining access to the very constricted poloidal bore is being considered. With this design (Fig. 7.5), a permanently installed cylinder forming the inboard vacuum boundary is located inboard of the shield segments. The seal flanges are abutted to this cylinder in such a way that they can be

Fig. 7.5. Welded torus design

severed and a shield segment removed without disturbing the central cylinder.

Test stations will be provided for testing materials, first wall systems, blankets, and shield systems for future fusion power reactors.⁶ It is planned to design the shield with penetrations to accommodate 30-40 test modules ~10 cm-1 m in diameter (Fig. 7.6). Provisions also have to be made to accommodate diagnostics as well as the neutral beam injectors and plasma chamber vacuum pumping ducts.

7.2.3 TF Coils

The current design, based on the ORNL TNS study, has a toroidal field of 5.3 T on axis produced by a superconducting coil with a maximum field of 10.9 T at the coil. A design employing an insulated, cabled conductor using Nb₃Sn filaments in a copper stabilizing matrix is being considered. The cables are encapsulated by a thin stainless steel jacket and cooled by forced-flow supercritical helium.

Fig. 7.6. Test module arrangement.

The conductors are pancake wound around a stainless steel coil bobbin.

Superconductors made of NbTi are also being considered. This type of superconductor has the advantage of being a ductile material, but its performance is marginal at 11 T. Current research being carried out on the Large Coil Project (LCP) and the 12-T Program should provide better information for making the final selection.

Another major issue in the TF coil design is the number of coils. Previous studies^{13,14} indicate that the optimum number of TF coils, accounting for capital cost only, is typically 16-20. With a modest cost penalty, the number can be reduced to 12, and a substantial advantage can be gained in terms of access for maintenance operations. Any further reduction, however, tends to be very costly and must be carefully justified on the basis of increased availability as the result of any improvement in maintenance operations.

7.2.4 PF Coils

The PF system includes the OH and EF coils required to induce a current in the plasma, maintain the plasma in equilibrium both horizontally and vertically, and protect the TF coils from magnetic flux changes.

The OH coil is a NbTi superconducting coil with a capability of 50 Wb and a maximum rate of flux change of 1.3 T/s. Operating in a half-biased mode, the solenoid coil has a maximum flux of +8 T.

For the EF coil system a hybrid arrangement is being considered. With this arrangement, the higher response coils, which must be close to the plasma and are therefore located inside the toroidal bore of the TF coil system, are normal-conducting copper so that they can be segmented and removed if necessary. The lower response coils are located outside the toroidal coil envelope and can be made superconducting to minimize the power loss.

7.2.5 Plasma Heating

A combined rf preheating and neutral beam bulk heating system is being considered to raise the plasma temperature to ignition conditions.

The rf system employs five 200-kW gyro-klystrons that deliver 1 MW to the plasma at 120 GHz for approximately 2 s. If this system can be used to preheat the plasma electrons, it should substantially reduce the voltage requirements for plasma breakdown.

After preionization by means of rf heating and induction of the plasma current through flux swing with the OH system, the bulk heating can be accomplished by means of neutral beams. This requires 50 MW of injected power with a beam energy of 150 keV for a period of ~6 s.

The beam energy requirement is very sensitive to the startup scenario used. At plasma burn conditions the energy requirement for penetration to the center of the plasma is 300-500 keV. The efficiencies of such high-energy beam lines become very low, requiring exorbitant levels of input power. Negative ion sources, which have the potential for higher efficiencies, require substantial R&D and therefore are encumbered with uncertainties with respect to development risk. By using a low-density startup, it is believed that moderate energy, positive ion beams at ~150 keV can be utilized.

7.2.6 Impurity Control

In order to achieve the long burn pulses envisioned for the ETF, helium ash removal and impurity control are required. Although the application of a suitable low-Z coating on the plasma chamber wall may provide at least partial passive control of sputtered material, ash removal probably requires an active system.

The principal alternative being considered for the impurity control/ash removal system are the bundle divertor and the poloidal

divertor. Limited research is being conducted on both schemes: the bundle divertor has been used on the Divertor and Injection Tokamak Experiment (DITE) and will be tested on the Impurity Study Experiment (ISX-B); poloidal divertor physics will be explored on the Poloidal Divertor Experiment (PDX).

Unfortunately, both schemes may present major obstacles. The bundle divertor, while being the most palatable in terms of installation considerations, presents potential difficulties in the plasma behavior because of the magnetic field anomalies introduced by the toroidally discrete locations of the divertor coils. The poloidal divertor, on the other hand, avoids the problem of toroidal anomalies but presents very difficult installation problems and the possibility of very high cost penalties.

Of major concern in either system is design particle load. If the divertor is designed to accommodate the total impurity particle load, it will also divert and process a very large throughput of tritium, presenting a problem in terms of tritium system design and tritium inventory. The divertor design capacity becomes, therefore, a problem of determining the minimum particle load that will permit adequate removal of impurities and helium ash.

7.2.7 Vacuum Pumping

It is expected that compound cryosorption pumps will most likely be used for removing spent gases at the end of each burn. To reduce the pressure from 10^{-4} to 10^{-6} torr at the end of the burn, four pumps, each with a specific speed of 2×10^5 liters/s, are required.

Among the areas of concern involving the cryosorption pumps are (1) the adequacy of the duct arrangement and shielding provisions to prevent excessive nuclear heating of the cryopanel and (2) the implications of storing large quantities of helium in the pumps between regeneration cycles.

The large ducts required for the vacuum pumps present a potential problem in terms of

nuclear heating because of both neutron streaming during burn and activation of pump components as a result of the neutron streaming. It should be possible to mitigate these effects by installing a shutter shield that would be closed during the burn and by designing the pumping ducts in such a way as to reduce neutron streaming. If the pumping ducts are integrated with the neutral beam ducts, it may be possible to use the same shutter shield for both the neutral beams and the vacuum pumps.

Two approaches are available for coping with the potential safety problem of a large holdup inventory of tritium on the cryopanel. One approach, the possibility of regenerating the pumps at relatively short intervals, has the disadvantage of requiring a large rate of cryogen use to cool the panels at the completion of each regeneration cycle. Another possibility is to locate the pumps in a vault so that if a release is encountered, confinement can be effected without contaminating the entire reactor building.¹⁵ This approach, however, does require the installation of fast-acting penetration valves to prevent the loss of tritium into the reactor building in the event the vacuum boundary at the reactor torus or any of the ancillary equipment is penetrated.

7.2.8 Fuel Handling

The fuel-handling system receives, processes, stores, and delivers the tritium and deuterium used by the ETF. The spent plasma gases and contaminated deuterium from the neutral beam injectors are processed by a combination of filtration, gettering, cryogenic trapping, cryogenic distillation, and catalytic cracking oxidation. If lithium is used in the divertor collector, the tritium will be reclaimed either by sorption on solid yttrium or by a permeation diffusion process using niobium.

A primary area of concern is tritium confinement. All tritium process equipment will be double enclosed, and the volume between the two enclosures will be continuously flushed by a circulating inert buffer gas. To maintain a

low tritium level, the buffer gas will be processed by a cleanup system.

A cleanup system will be provided to remove the tritium from the reactor building in the event of a release into this space. Because of the large volume, a large cleanup system is required to reduce the concentration in the space to man-access levels within a few days. Tradeoffs will be generated to obtain a design point for this system.

7.2.9 Power Supply

Power supply systems must be provided for the neutral beam injectors, the TF and PF coil systems, and the rf heating system.

The coil systems impose heavy demands in terms of both power and energy storage on the power supply system. It is expected that the ETF will be located where a tie-in can be made to a utility with 500-kV lines so that the load can be drawn directly from the line if necessary. For normal operations, however, flywheel motor-generators will be installed for energy storage. Assuming 500-kV line backup, two motor-generators, each with a capacity of, for example, 300 MVA, would be capable of supplying the 600 MVA required by the TF and PF coil systems for the rf pre-heating scenario. In the event utility backup is not available for the pulse power requirements, two motor-generators, each with the capacity to supply the total power requirements, would be installed.

7.3 CONCLUDING REMARKS

In anticipation of the success of the near-term scientific and technology programs associated with the tokamak concept, an industry/laboratory team has been assembled at ORNL. This team, designated the ETF Design Center, has as its objective to define a tokamak-based ETF and to prepare its design, schedule, and cost in appropriate detail to support the initiation of the ETF project at the earliest possible date. In order to guide the Design Center, an ETF Mission Statement based on

community-wide input was developed. As a point of departure for the design studies, a baseline design was adopted based on the TNS studies of 1977 and 1978. The Preconceptual Design Phase will center around two designs representing options relative to the ETF Baseline Design. As part of the Design Center activities, an ETF R&D assessment is being developed. The purpose of this effort is to identify those technical areas that (1) are essential to the ETF and (2) are not adequately covered in the current fusion programs. Periodically the ETF design and required R&D will be reviewed relative to the ETF mission requirements. These reviews may indicate the need to reevaluate and revise the ETF mission. The first such review is scheduled to take place in the summer of 1980. Through such activities we intend to involve the fusion community in the evolution of the ETF.

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8. MATERIALS

R. G. Alsmiller, Jr.¹

J. M. Barnes²

R. Behrisch³

J. T. Bell⁴

B. L. Bishop²

E. E. Bloom⁵

R. R. Colman, Jr.⁶

J. H. DeVan⁷

H. L. Dodds⁷

D. P. Edmonds⁵

K. Farrell⁸

T. A. Gabriel¹

M. L. Grossbeck⁵

T. J. Hoffman²

J. A. Horak⁵

M. Hou⁸

J. R. Keiser⁵

R. H. Kernohan⁹

J. F. King⁵

C. E. Kitzbunde⁶

C. T. Liu⁵

K. C. Liu⁵

C. J. Long⁵

G. Maderlechner³

P. J. Maziasz⁵

F. V. Nolfi¹⁰

J. D. Redman¹¹

M. T. Robinson⁶

T. K. Roche⁵

R. T. Santoro¹

B. M. U. Scherzer³

J. L. Scott⁵

J. E. Selle⁵

J. A. Spitznagel¹²

J. O. Stiegler⁵

P. F. Tortorelli⁵

F. W. Wiffen⁵

J. W. Woods¹³

1. Engineering Physics Division
2. Computer Sciences Division
3. Max-Planck Institut für Plasmaphysik, Association EURATOM-IPP
4. Chemical Technology Division
5. Metals and Ceramics Division
6. Solid State Division
7. The University of Tennessee, Knoxville, Tennessee
8. Université Libre de Bruxelles
9. Retired from Solid State Division
10. Argonne National Laboratory, Argonne, Illinois
11. Chemistry Division
12. Westinghouse Research and Development Center, Pittsburgh, Pennsylvania
13. Engineering Technology Division

8. MATERIALS

ABSTRACT. The primary purpose of the materials work for fusion at Oak Ridge National Laboratory (ORNL) is to develop a suitable near-term structural alloy for the Engineering Test Facility (ETF) and more advanced alloys for commercial fusion reactors. Two classes of alloys, austenitic stainless steels and ferritic steels, are being evaluated for the ETF. For commercial fusion reactors, these same classes of alloys are being considered as are the higher-strength iron-nickel-chromium alloys; vanadium, niobium, and titanium alloys; and the long-range-ordered alloys based on the system $(\text{Fe,Ni})_3\text{V}$.

The alloys selected for the ETF are already being manufactured commercially in large tonnages, and a large design data base of unirradiated properties exists. Consequently, our work with these alloys has emphasized the effects of irradiation on material properties. Wherever possible, the irradiation tests are done so as to produce high concentrations of helium in the lattice to simulate the fusion reactor environment. Because there is no large data base of unirradiated properties for the more advanced alloys being studied for commercial reactor applications, the near-term goal of work on refractory metals and on long-range-ordered alloys is to develop this information. At the same time, samples are included in irradiation tests whenever possible to begin the process of obtaining data on irradiated samples. We are also assessing the corrosion rates of all important alloys in liquid lithium, in molten salts, and in contact with solid lithium compounds, and are studying the diffusion of tritium through commercial steam generator materials.

Neutronic calculations are an important adjunct to the fusion materials development program. These calculations are performed on the blanket structural materials for candidate fusion reactors to define the atomic displacement levels and hydrogen and helium production rates in actual operation. Analyses are also performed on available irradiation facilities such as the High-Flux Isotope Reactor (HFIR), the Oak Ridge Research Reactor (ORR), and the Experimental Breeder Reactor II (EBR-II) to determine displacement and gas production levels in experimental tests. In the ORR, the neutron flux can be adjusted to match the reactor environment by varying the moderator and fuel compositions near experiment positions in the core.

In the area of damage analysis, the application of numerical transport theory to particle range distribution in solids has continued. The calculational methods have also been applied to sputtering in the near-surface region. A good correlation has been found between the calculated sputtering yields and those measured experimentally.

Irradiation tests on organic insulations for use in superconducting magnets have been taken to a total dose of 100 MGy (10^{10} rad). The mechanical properties of many candidate materials are severely degraded under these conditions.

8.1 ALLOY DEVELOPMENT FOR IRRADIATION PERFORMANCE

The following subsections describe alloy development activities during 1979 in the area of irradiation performance. Many of these activities are presented in the form of republished abstracts from the literature, as are a number of the activities in Sects. 8.2-8.5. The reference number following an abstract title refers the reader to the complete work for details.

8.1.1 Abstract of The Change of Material Properties During Irradiation in Fusion Reactor Service¹

P. W. Wiffen

Design of a fusion power reactor and analysis of its performance and life time of operation require knowledge of a number of material properties. This knowledge is required even in conceptual design, since these properties set temperature, pressure, and lifetime limitations on the reactor structural components. Some of

the material properties change in response to the rate of neutron bombardment, whereas others change in response to the integrated neutron bombardment. Stress and stress distribution within the structure will be modified by irradiation-produced swelling as a stress source and irradiation creep as a stress relief and redistribution mechanism. The ability of the structure to accommodate these and other operating stresses will be influenced by the effects of neutron irradiation on the mechanical properties. The most limiting *identified* property degradation is the significant reduction in ductility produced by several irradiation-related mechanisms. Less extensively studied, but potentially as important, are property changes resulting from the pulsed stresses of cyclic operating conditions. The growth of preexisting flaws may be the most limiting of these changes, but fatigue and creep fatigue interactions may also be important.

Our available data base for predicting these properties in fusion systems comes mainly from fission reactor programs. The two most extensively studied properties of irradiated metals are swelling and postirradiation tensile properties. Both of these reveal important differences between the behavior in fission reactor systems and the predicted behavior in fusion reactor systems.

Swelling is dependent on both material and irradiation parameters. Swelling rate differences of more than an order of magnitude have been encountered in alloys within the Fe-Ni-Cr system, and these differences can be produced by changes in either major or minor alloy constituents. In some of the alloys that show appreciable swelling in the annealed condition, modification of the microstructure by cold working can further reduce the swelling rate. Under conditions of fission breeder reactor irradiation, swelling occurs over the temperature range $0.3 < T/T_m < 0.6$, with maximum swelling occurring at an intermediate temperature. The most important difference in going from fission to fusion irradiation is the large increase in the rate of helium production

in transmutation reactions. Examination of swelling for helium production rates in a range spanning the fusion case has shown that the helium can influence the swelling rate. Neutron irradiation experiments with helium production rates higher than expected in fusion result in swelling that is nearly independent of temperature across most of the potential application range, followed by considerably higher swelling rates for temperatures above $\sim 0.5 T_m$. Ion bombardment simulation of the neutron irradiation has shown that the dependence of swelling on the helium introduction rate may be determined by temperature and by the ratio of damage to helium introduction rates.

Postirradiation tensile properties also show a sensitivity to the irradiation-produced helium. Although the trends in changes of the tensile properties with irradiation are approximately similar for fission- or fusion-simulated conditions, the magnitude of the changes can be different. Of these differences, the most serious is the loss of ductility at the higher temperatures. Tensile elongations near zero result from the helium-triggered grain boundary separation. This fracture mechanism shifts to lower temperatures as helium contents increase.

The data base produced by the fission reactor programs will be examined for its application, limitation, and restriction for use in fusion reactor design. Current experimental programs directed at complementing and extending the available data for fusion system design will also be outlined.

8.1.2 Abstract of Mechanical Properties of Materials in Fusion Reactor First-Wall and Blanket Systems²

E. E. Bloom

With respect to the effects of irradiation on mechanical properties, the most significant difference between fast fission and fusion reactor spectra is the relatively large amount of helium produced by (n, α) transmutations in the latter. Relevant information on the effects of large amounts of helium (with concomitant

displacement damage) comes from irradiation of alloys containing nickel in mixed spectrum reactors. At helium levels of interest for fusion reactor development, properties are degraded to unacceptable levels above $0.5 T_m$. Below this temperature, strength and ductility are retained, and fractures remain transgranular. Importantly, the properties remain sensitive to composition and structure. A comparison of the response of refractory alloys to that of stainless steel at equivalent damage levels shows the same general trends in properties with homologous temperature. The refractory alloys do offer potential for higher-temperature applications because of their higher melting temperatures.

8.1.3 Abstract of Swelling, Mechanical Properties, and Microstructure of Type 316 Stainless Steel at Fusion Reactor Damage Levels³

J. A. Borak, E. E. Bloom, M. L. Grossbeck, P. J. Maziasz, J. O. Stiegler, P. W. Wiffen

Alloys such as type 316 stainless steel exhibit more swelling and larger decreases in ductility when irradiated to produce fusion reactor helium and displacement-per-atom (dpa) levels rather than fast reactor helium and dpa levels. For $T \geq 0.5 T_m$, swelling is decreased and ductility is increased by cold work, by small additions of titanium, or both. Structural components of fusion reactors may have to operate below 550°C to ensure adequate ductility for long-term service.

8.1.4 Abstract of Microstructures Developed in Simulated Fusion Irradiations⁴

J. A. Spitznagel, P. W. Wiffen, P. V. Holfi

Microstructural information is reviewed for irradiation conditions that simulate one or more salient features of the irradiation environment anticipated at first wall and blanket positions in a fusion reactor. The effects of high-energy neutron damage from $T(d, n)$, $\text{Be}(d, n)$ and 16-MeV proton irradiations and the effects

of simultaneous atom displacement damage and helium and hydrogen production from mixed spectrum fission reactor and dual-ion irradiation experiments are summarized for a variety of metals. It is shown that observations of microstructural sensitivity to damage rate, irradiation temperature, and helium are providing important clues for developing damage-resistant structural alloys for fusion reactor applications.

8.1.5 Abstract of Tensile Properties of Type 316 Stainless Steel Irradiated in a Simulated Fusion Reactor Environment⁵

M. L. Grossbeck, P. J. Maziasz

Tensile and fracture properties of type 316 stainless steel in the annealed and 20%-cold-worked conditions were investigated after the steel had been exposed to neutron fluences producing 16 dpa and 1000 appm He resulting from irradiation in the HFIR. Tensile elongations remained above 6% at 350° and 450°C and 4% at 575°C . Fracture behavior has been examined in the 20%-cold-worked material. At 350°C the fracture mode is unchanged by irradiation and is characterized by ductile tearing. At 450°C irradiation causes the fracture mode to change from ductile tearing to crystallographic fracture, which may occur along the slip bands introduced by the cold working. At 575°C the fracture is changed from ductile transgranular to intergranular. The tensile properties observed are believed to be adequate for fusion reactor service within the values of damage parameters investigated.

8.1.6 Abstract of The Precipitation Response of 20%-Cold-Worked Type 316 Stainless Steel to Simulated Fusion Irradiation⁶

P. J. Maziasz

The precipitation response of 20%-cold-worked type 316 stainless steel has been examined (1) after irradiation in the HFIR at 380° - 600°C , (2) after irradiation in the Experimental Breeder Reactor II (EBR-II) at 500°C , and (3) after thermal aging at 600° - 750°C . Eta phase forms during exposure to all environments. It

constitutes a major portion of the precipitation response and after thermal aging is richer in nickel, silicon, and molybdenum than the $M_{23}C_6$. It is not normally reported in 20%-cold-worked type 316 stainless steel. The eta, $M_{23}C_6$, Laves, sigma, and chi precipitate phases appear at similar temperatures after irradiation in the HFIR and EBR-II or after thermal exposure. There are, however, some differences in relative amounts, size, and distribution of phases among the various environments. Eta phase is the only carbide-type phase observed after irradiation in HFIR from 380°-550°C. The large cavities associated with it at 380°C contribute significantly to swelling. Re-solution of fine $M_{23}C_6$, eta, and Laves particles and reprecipitation of massive particles of sigma, $M_{23}C_6$, and chi are observed after recrystallization in HFIR.

8.1.7 Abstract of The Formation of Diamond-Cubic Eta (η) Phase in Type 316 Stainless Steel Exposed to Thermal Aging or Irradiation Environments⁷

P. J. Maziasz

Precipitation in type 316 stainless steel in thermal aging, fast reactor, or mixed spectrum reactor environments has been examined by many workers. Many properties such as strength, corrosion resistance, or swelling resistance can be sensitive either to precipitation directly or to changes in the composition of the matrix induced by the precipitation. After irradiation in EBR-II or the HFIR or after thermal aging, eta phase is found in solution-annealed 316 (1 hour at 1050°C) and 20%-cold-worked 316. Based upon data reported in the literature, the widespread appearance and dominance of the diamond-cubic eta phase in type 316 is unexpected. That this phase may exist undetected in other heats of type 316 already investigated, whether thermally aged or irradiated, should at least be considered a distinct possibility because the phase has been found in this heat of type 316 and because it can be difficult to detect. The direct relation

between alloy composition, microstructure, and properties means that characterizing and understanding precipitation behavior is important to improving performance at elevated temperatures and to supporting predictions of properties.

8.1.8 Abstract of The Influence of Irradiation on the Tensile Properties of Austenitic Stainless Steel Weldments²

F. W. Wiffen, J. A. Horak, D. P. Edmonis, J. F. King

Weldments in the first wall and front sections of the blanket of a fusion reactor will be exposed to approximately the same operating conditions as will the base metal. Thus, the irradiation response of weld metal, of the weld-heat-affected zones in the base metal, and of the base metal are all of equal concern. Austenitic stainless steels will most likely be joined by a gas tungsten arc welding process. Welds were made by this process between sections of 6-mm-thick base plate of type 316 stainless steel in the 20%-cold-worked condition, with either type 316 or 16-8-2 stainless steel filler metal. Rod tensile specimens were cut through the welds, containing weld metal in the central gage portion.

Weld-containing tensile specimens were irradiated in the HFIR at 55°C and throughout the temperature range 280°-620°C. The neutron fluences ranged from 0.5 to 1.6×10^{26} n/m² (>0.1 MeV). The corresponding displacement damage levels ranged from 4.5 to 12.1 dpa, and the helium generation from the thermal neutron captures in nickel resulted in 100-550 appm He.

Tensile tests at temperatures near the irradiation temperature resulted in fractures in the weldments, not in the weld-heat-affected zone. These tests showed that after irradiation and test temperatures up to 375°C, irradiated weldments were stronger than identical unirradiated weldments. For temperatures of 475°-620°C, the strength of irradiated weldments was equivalent to or lower than the strength of control material. After irradiation and throughout the temperature range, the strength of both weld materials was

lower than the strength of irradiated 20%-cold-worked type 316 stainless steel. Total elongation of irradiated weldments showed a broad minimum in the range 300°-400°C at approximately 4% elongation. Elongation of the base metal at 350°C is approximately 8% for similar irradiation conditions. (Because the gage length of the specimen contained weldment, weld-heat-affected material, and base metal, the measured elongation includes contributions from each of these. It is not truly a measure of the weldment ductility.) Fractography showed that all material tested at temperatures near the irradiation temperature had failed in a ductile shear mode. Only a single sample irradiated at 55°C and then tested at 700°C failed by a grain boundary separation mode.

These tensile properties indicate that for operating temperatures below 620°C and operating times of 1 to 2 MMyr/m², conventional welding techniques used with 316 stainless steels would not impose additional operating limitations on the reactor. Although these data will guide material selection for near-term fusion power reactors, higher-fluence evaluation will be required for commercial reactor design.

8.1.9 Abstract of Lifetime Estimates of a Fusion Reactor First Wall by Linear Damage Summation and Strain Range Partitioning Methods⁹

K. C. Liu, M. L. Grossbeck

A generalized model of a first wall made of 20%-cold-worked steel was examined for neutron wall loadings ranging from 2-5 MM/m². A spectrum of simplified on-off duty cycles was assumed with a 95% burn time. Independent evaluations of cyclic lifetimes were based on two methods: (1) linear damage summation, which is currently being used in ASME high-temperature design code Case N-47 and (2) strain range partitioning, which is being studied for inclusion in the design code. An important point is that the latter method can incorporate a known decrease in ductility for materials subject to irradiation as a parameter, so low-cycle fatigue

behavior can be estimated for irradiated material. Lifetimes predicted by the two methods agree reasonably well despite their diversity in concept. Lack of high-cycle fatigue data for the material tested at temperatures within the range of our interest precludes making conclusions on the accuracy of the predicted results, but such data are forthcoming. The analysis includes stress relaxation owing to thermal and irradiation-induced creep. Reduced ductility values from irradiations that simulate the environment of the first wall of a fusion reactor were used to estimate the lifetime of the first wall under irradiation. These results indicate that 20%-cold-worked type 316 stainless steel could be used as a first-wall material meeting a lifetime goal of 8-10 MMyr/m² for a neutron wall loading of about 2 MM/m² and a maximum temperature of about 500°C.

8.1.10 Abstract of The Tensile Properties of High-Oxide Sintered Aluminum Product Containing Helium and Tritium¹⁰

F. J. Maziasz, K. Farrell

Up to 1700 appm each of ⁴He and ³H were created in sintered aluminum product (SAP) alloys of 17-25 wt % oxide by doping the alloys with ⁶Li and irradiating them in a thermal reactor at ambient temperature. To simulate the distribution of these gases at a proposed fusion reactor first-wall operating temperature of about 670 K, the irradiated SAP was annealed for 1000 h in the range 573-773 K (0.61-0.83 T_m). Immersion density measurements indicated no discernible swelling from the gases, a portion of which was shown to be trapped in tiny bubbles at the Al₂O₃-Al interfaces. Tensile tests performed in the temperature range 310-773 K revealed no significant effects from the gases on strength, ductility, or fracture mode. However, the ductility in the unirradiated condition was low (less than 0.5% at 673 K). These data suggest that although a fine distribution of stable second-phase particles in first-wall alloys might

provide a means of ameliorating gas swelling and embrittlement, there may be a high cost in terms of fabrication problems and original ductility.

8.1.11 Material Procurement for the National Fusion Materials Inventory

T. K. Roche

Procurement activities during the past year for the various classes of alloys being evaluated for irradiation performance in fusion reactors are summarized below.

Path A. Austenitic Alloys

Carpenter Technology Corporation has processed approximately 30.5 m of seamless tubing from drilled bar stock of Path A prime candidate alloy, a titanium-modified stainless steel. The tubing, which has an outside diameter of 4.57×10^{-3} m and a wall thickness of 2.54×10^{-4} m, is presently undergoing nondestructive inspection for qualification as specimen stock for irradiation experiments.

Path C. Reactive and Refractory Metals

A contract with Westinghouse Electric Corporation for the production of three vanadium- and two niobium-base alloys was completed. These alloys are now available from inventory in sheets 7.62×10^{-4} , 1.52×10^{-3} , and 2.54×10^{-3} m thick and in rods 6.35×10^{-3} m in diameter.

Path D. Innovative Materials Concepts

The long-range-ordered alloys Fe-Ni-V and Fe-Co-Ni-V developed at ORNL show attractive properties for advanced energy systems. Specifications have been prepared for the semiproduction scale-up of two alloys by an industrial source yet to be selected. Sheet stock will be processed for neutron irradiation, mechanical property, and compatibility tests. This scale-up is expected to be completed during 1980.

Path E. Ferritic Stainless Steels

Seven small heats of nine Cr-1 Mo and 12 Cr-1 Mo ferritic steels with nickel additions from 0-2% were procured from Combustion Engineering, Inc. The alloys have been processed into sheet stock 7.62×10^{-4} and 2.54×10^{-3} m thick for irradiation specimen fabrication.

8.1.12 Current Irradiation Experiments in the Magnetic Fusion Energy Materials Program

M. L. Grossbeck, J. W. Woods

Magnetic energy (MFE) irradiation experiments at ORNL are all neutron irradiations in fission reactors. The HFIR, the Oak Ridge Research Reactor (ORR), and EBR-II are the reactors now being used. The HFIR and ORR are mixed spectrum reactors used because of their ability to form helium through thermal neutron absorption in nickel. The HFIR is used primarily to study the effects of high helium concentrations; the ORR is used for instrumented irradiation under conditions more closely simulating the fusion reactor environment. The neutron spectrum in the ORR is tailored to match the helium-to-damage ratio of a fusion reactor. EBR-II is being used to achieve high damage levels with little helium production and, in conjunction with cyclotron helium injection and tritium decay, to correlate such helium doping techniques with continuous helium production in a mixed spectrum reactor.

Specimen types include tensile specimens, pressurized tube irradiation creep specimens, electron microscope specimens, fatigue specimens, and auger fracture specimens.

8.1.13 Corrosion Inhibition in Systems of Lithium with Nickel-Bearing Alloys

P. P. Tortorelli, J. H. DeVan, J. E. Sella

Molten lithium, which is being considered for use as a tritium-bearing fluid and as a coolant in fusion reactors, could dissolve enough of the first wall that loss of structural integrity and flow restrictions by mass transfer

deposits could become serious problems during the reactor lifetime. It was found that adding about 5 wt % Al to the lithium before testing dramatically reduced the corrosion. The aluminum in the lithium reacted with the nickel in the alloys to form a surface intermetallic layer that was significantly more corrosion resistant than the original material. Such a procedure decreased the maximum corrosion rate of type 316 stainless steel at 600°C in flowing lithium by a factor of five and also inhibited the dissolution of alloy 600 in static lithium between 500° and 700°C. Tests with type 316 stainless steel in static lithium defined the dependence of the aluminum reaction on time and temperature. In addition, silicon and possibly manganese could be used as corrosion-reducing additives. This method of corrosion inhibition is particularly attractive because of its ease of implementation, since the reaction layers can be produced *in situ*.¹¹

8.1.14 Corrosion Tests of Higher-Nickel Alloys in Static Lithium

P. F. Tortorelli, J. H. DeVan

The compatibility of pure, static lithium with alloy 800 (46 wt % Fe, 32 wt % Ni, 21 wt % Cr) and alloy 600 (75 wt % Ni, 15 wt % Cr, 8 wt % Fe) was investigated. Both alloys had corroded significantly between 500° and 700°C after 1000 h. Their corrosion rates were much greater (by one to two orders of magnitude) than those of similarly exposed type 316 stainless steel. Further experiments with alloy 600 showed that its corrosion could be effectively inhibited when 5 wt % Al was added to the lithium.^{12,13}

8.1.15 Corrosion of a Long-Range-Ordered Alloy (Fe-Ni-V) in Static Lithium

P. F. Tortorelli, C. T. Liu, J. H. DeVan

Coupons of the long-range-ordered (LRO-35) alloy (45 wt % Fe, 32 wt % Ni, 23 wt % V) were exposed to pure, static lithium by placing tensile specimens in sealed LRO-35 alloy cylinders,

which in turn were placed in type 316 stainless steel capsules. Experiments were performed for 2000 h at 500° and 600°C with resulting average weight losses of 0.4 g/m² at both temperatures. Alloy 800 (46 wt % Fe, 32 wt % Ni, 21 wt % Cr) that was exposed to static lithium for 1000 h at 500° and 600°C experienced average weight losses of 16.8 g/m² and 61.8 g/m², respectively. Additionally, exposure to the static lithium did not affect the tensile ductility of the LRO-35 alloy at room and elevated temperatures.¹²

8.1.16 Compatibility of Type 316 Stainless Steel with LiF-LiCl-LiBr^{14,15}

J. R. Keiser, J. H. DeVan, P. F. Tortorelli

A LiF-LiCl-LiBr salt has been proposed for use as a means to extract tritium from a lithium blanket. The corrosion of type 316 stainless steel by LiF-LiCl-LiBr flowing at 15 mm/s has been studied to determine whether the structural material would be compatible with such a system. Preliminary analysis of the data showed that after 8000 h at a maximum temperature of 540°C, the corrosion rate of type 316 stainless steel in LiF-LiCl-LiBr was low (<1 mg/m²-h).^{14,15}

8.2 PERMEATION STUDIES

J. D. Redman, J. T. Bell

Knowledge of tritium permeation through steam generator materials exposed to high-pressure steam is needed to evaluate various tritium management schemes for experiments involving large quantities of tritium and, ultimately, for fusion reactor power stations. Hydrogen permeation through ceramic and glass materials is known to be as much as ten orders of magnitude lower than that through metallic construction materials. Consequently, the formation of oxide layers on steam generator materials may effectively reduce tritium permeation into the steam system to an acceptable level. The objective of this program is to

determine tritium permeabilities through candidate steam generator materials under realistic temperature and dynamic steam pressure conditions.

During the past year the apparatus required for the permeation measurements was designed and constructed. Tritium permeation rates can be measured while a metal or alloy is undergoing oxidation by steam at pressures as high as about 10 MPa and temperatures up to about 800 K. The first measurements will be made with Incoloy 800 and Inconel 601.

8.3 NEUTRONICS ANALYSIS

8.3.1 *Abstract of Calculated Irradiation Response of Materials Using Fission Reactor (HFIR, ORR, and EBR-II) Neutron Spectra*¹⁶

T. A. Gabriel, B. L. Bishop, F. W. Wiffen

To plan radiation damage experiments using fission reactors for fusion reactor applications, it is necessary to know fission reactor dpa and gas production rates for many materials. This report supplies this data for the elemental constituents of alloys of interest to the U.S. fusion reactor alloy development program. The calculations are presented for positions of interest in the HFIR, ORR, and EBR-II. The dpa and gas production rates in alloys of interest can be synthesized from these results.

8.3.2 *Abstract of Sensitivity of Primary Knock-On Atom Spectra and Displacement-Per-Atom Cross Sections to Different Secondary Neutron Energy and Angular Distributions and In-Group Weighting Schemes*¹⁷

T. A. Gabriel, B. L. Bishop

The sensitivity of primary knock-on atom (PKA) spectra and dpa cross sections to different secondary neutron energy and angular distributions and "in-group" weighting schemes is investigated. It is shown that the sensitivity of the PKA spectra and dpa cross sections for the (n, n' unresolved) and (n, 2n) reactions in iron to different angular distributions and the

same secondary neutron spectrum is reasonably large (~15%). For grossly different secondary neutron spectra and the same angular distribution, the change in the dpa cross section is smaller than one would initially expect. It is also shown that for aluminum, the sensitivity of dpa cross sections to different in-group weighting schemes is, for the most part, small.

8.3.3 *Abstract of The Effects of Fe-Ni-Co-V Structural Alloys on Fusion Reactor Neutronic Performance*¹⁸

J. M. Barnes, B. L. Bishop, R. T. Santoro, T. A. Gabriel

Calculations have been carried out to determine the effect of Fe-Ni-Co-V alloys on fusion reactor neutronic performance. The radiation damage in the first structural wall and the tritium breeding ratio were calculated as a function of alloy composition using the tritium breeding blanket module of the Experimental Power Reactor (EPR) as the reference design. The atomic displacement rate was found to be approximately constant for all alloy combinations. However, the gas production and tritium breeding exhibit reasonably large variations (~20%) as the cobalt concentration is increased in the alloy.

8.3.4 *Abstract of The Sensitivity of the First Wall Radiation Damage to Fusion Reactor Blanket Composition*¹⁹

J. M. Barnes, R. T. Santoro, T. A. Gabriel

The atomic displacement and hydrogen and helium gas production rates in a 0.01-m-thick type 316 stainless steel first wall have been calculated as a function of the composition of a 0.5-m-thick fusion reactor blanket. The atomic displacement rates range from a low value of 8.4 dpa/yr for an empty blanket to a high value of 22.6 dpa/yr for a lead-filled blanket. Hydrogen gas production rates vary from 423 to 536 appm/yr, and the helium gas production rates vary from 134 to 159 appm/yr for the empty and carbon-filled blankets, respectively.

8.3.5 Abstract of Calculated Atom Displacement and Gas Production Rates of Materials Using a Fusion Reactor First-Wall Neutron Spectrum²⁰

T. A. Gabriel, B. L. Bishop, P. W. Wiffen

The dpa and gas production rates have been calculated for a number of alloys and elements using a design neutron spectrum at the first wall of a fusion reactor. These rates can be combined for most alloys to yield the defect production rates, the parameters currently used to extrapolate available irradiation effects data to fusion reactor conditions. Calculated rates of atom displacement and hydrogen generation in stainless steels are relatively insensitive to recent changes in the nuclear data files and to neutron spectrum differences produced by slight reactor design changes. In contrast, the helium production rate is sensitive to these changes and to the exact alloy composition. Composition variation within the specification range for type 316 stainless steel can produce variations of $\pm 9\%$ in the helium generation rate.

8.4 DAMAGE ANALYSIS

8.4.1 Abstract of Computer Simulation of Low-Energy Sputtering in the Binary Collision Approximation²¹

M. Hou,* M. T. Robinson

The sputtering of amorphous copper targets by low-energy atoms has been investigated in the binary collision approximation using the computer program MARLOWE. Particular attention was given to the influence of the surface binding model on the results. Calculations were made of the dependence of the sputtering yield on the incident particle direction, energy, and mass. Angular, energy, and yield distributions of the ejected atoms were evaluated. Comparisons with experimental results on polycrystalline targets show that the planar

surface binding model is to be preferred over the isotropic surface binding model, especially with regard to the angular and energy distributions. Calculated yields are in reasonable agreement with experiment at energies below about 1 keV, but deviate at higher energies, apparently because of crystal correlation effects that are neglected in the amorphous model.

8.4.2 Abstract of The Sputtering Mechanism for Low-Energy Light Ions²²

R. Behrisch, G. Maderlechner, B. M. U. Scherzer,* M. T. Robinson

The computer simulation program MARLOWE, which follows the trajectories of energetic ions and recoiling target atoms in solids, has been used to calculate sputtering yields for low-energy (0.1-10 keV) light ions (hydrogen, deuterium, tritium, ⁴He). Recoil energy densities were calculated for comparison with analytical theories. The sputtering yields obtained for amorphous iron agree within a factor of two with experimentally measured values for polycrystalline stainless steel, whereas the calculated yields for protons on amorphous molybdenum are more than twice the experimental values on polycrystalline material. The calculations show that in the parameter range investigated, ions backscattered in the solid contribute a major part to sputtering. This result confirms earlier calculations of the threshold energy for sputtering that are in agreement with recent measurements.

8.4.3 Abstract of Range Calculations Using Multigroup Transport Methods²³

T. J. Hoffman; M. T. Robinson; H. L. Dodds, Jr.†

Several aspects of radiation damage effects in fusion reactor neutron and ion irradiation environments are amenable to treatment by

*Université Libre de Bruxelles.

*Max-Planck-Institute für Plasmaphysik, Association EURATOM-IPP.

†University of Tennessee, Knoxville.

transport theory methods. In this paper, multigroup transport techniques are developed for the calculation of particle range distributions. These techniques are illustrated by analysis of ^{196}Au atoms recoiling from $(n, 2n)$ reactions with gold. The results of these calculations agree very well with range calculations performed with the atomistic code MARLOWE. Although some detail of the atomistic model is lost in the multigroup transport calculations, the improved computational speed should prove useful in the solution of fusion material design problems.

8.5 ABSTRACT OF RADIATION EFFECTS ON ORGANIC INSULATORS FOR SUPERCONDUCTING MAGNETS²⁴

C. J. Long; R. R. Colman, Jr.; C. E. Kitchende; R. B. Kernohan

During 1979, the Oak Ridge Low-temperature Irradiation Facility (LTIF) was revised to increase the gamma radiation dose rate and the neutron-to-gamma ratio. The higher gamma dose rate allowed exposure of insulators to a total dose of 100 MGy; the new neutron-to-gamma ratio gave a better simulation of service conditions. Materials, including two castable epoxy formulations, three glass-epoxy laminates, a polyamide paper, and a polyimide film, were irradiated at 5 K to gamma doses of 24 and 100 MGy, then tested mechanically at 77 K and electrically at room temperature. Electrical properties generally were degraded, especially at 100 MGy, but will not likely constrain magnet designs. Mechanical properties, on the other hand, reflected damage serious enough to require revision of designs made in the absence of these data. Radiation damage to organic insulation systems evidently becomes a serious design problem at doses of about 10 MGy (1×10^9 rad), if conventional materials are used.

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9. NEUTRON TRANSPORT

K. G. Alsmiller, Jr.¹

V. C. Baker²

J. Barish³

J. M. Barnes³

B. L. Bishop³

G. T. Chapman¹

T. A. Gabriel¹

R. A. Lillie¹

B. F. Maskewitz¹

G. L. Morgan⁴

E. M. Oblow¹

R. H. Peelle¹

M. M. H. Ragheb⁵

R. W. Roussin¹

R. T. Santoro¹

D. K. Trubey¹

1. Engineering Physics Division

2. Present address: General Dynamics, Pomona, California

3. Computer Sciences Division

4. Los Alamos Scientific Laboratory, Los Alamos, New Mexico

5. University of Illinois, Nuclear Engineering Program, Urbana, Illinois

9. NEUTRON TRANSPORT

ABSTRACT. The neutron transport program includes both experimental and analytic phases. The experimental program is designed to provide (1) data necessary for verifying the analytic methods and (2) cross-section data used at the Oak Ridge National Laboratory (ORNL) and throughout the United States for fusion reactor neutronics design calculations. Experiments are being carried out to determine the neutron 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. Neutronics calculations have also been carried out in support of the design of the Tokamak Fusion Test Reactor (TFTR) being built at Princeton University. The Radiation Shielding Information Center (RSIC) has continued to supply a broad range of services to the fusion energy research community.

9.1 CONFERENCE PAPER: ANALYSIS OF ORNL FUSION REACTOR SHIELD INTEGRAL EXPERIMENTS¹

R. T. Santoro; R. G. Altmiller, Jr.; J. M. Barnes; G. T. Chapman; G. L. Morgan; E. M. Oblow; R. W. Puelle

The nuclear performance of the blanket and shield in a fusion reactor will have a substantial impact on the overall operation and capital cost of the reactor.² It is necessary, therefore, to have experimental verification of the nuclear data and radiation transport methods that will be used to carry out the nuclear design calculations for these assemblies. Integral experiments are being performed at ORNL to provide the experimental data needed for this verification. The verification will be made by extensive comparisons between calculated results and these experimental data. Measurements of the transport of 14-MeV neutrons through laminated shields containing materials that are anticipated for use in fusion applications and the effect of introducing penetrations in the shields will be made. This paper summarizes the results of measurements and calculations to determine neutron and photon spectra behind laminated shields of varying composition and thickness.³

The experiments are carried out using 14-MeV neutrons having a source strength of $\sim 10^4$ n/s produced in the interactions of 200-keV deuterons with tritium saturated in titanium target. The target is enclosed in an iron can, which has the purpose of tailoring the neutron distribution to

correspond to that incident on the first wall of a fusion reactor. The target, the iron can, and the shield assembly are supported in a concrete superstructure. Shields of various thicknesses are constructed by arranging slabs of type 304 stainless steel (SS-304), borated polyethylene, and heavymet (tungsten alloy) in selected sequences; each slab measures 152 cm square and 5.08 cm thick. Neutron and gamma-ray spectra are measured at various positions relative to the deuteron target axis and as a function of shield thickness using an NE-213 liquid scintillator. Neutron and gamma-ray pulse height spectra are simultaneously obtained using pulse shape discrimination methods, and unfolding techniques are used to resolve the energy spectra from the pulse height data.

The calculated neutron and gamma-ray energy spectra are obtained using two-dimensional (2-D) radiation transport methods. The experimental configuration is represented in r-z geometry with cylindrical symmetry about the axis of the deuteron beam using 92 axial and 76 radial mesh intervals to describe the geometry. The neutron source used in the calculations takes into account the angle-energy relationship of the neutrons emitted from the deuterium-tritium (D-T) reaction and takes into account deuteron slowing down in the target. Separate calculations are performed to determine the first collision and uncollided flux distributions in the geometry meshing using black absorbers about the neutron source to define the angular interval into which neutrons in a given energy

interval are emitted. These data are then combined to form a single source term for input into the discrete ordinates code, DOT,⁴ which completes the radiation transport. The transport calculations are carried out using an S_{12} angular quadrature and a P_3 Legendre expansion of the transport cross sections. A 53-neutron, 21-gamma-ray multigroup cross section was obtained by collapsing the 171-neutron, 36-gamma-ray VITAMIN C⁵ data set using the ANISN code.⁶

Neutron and gamma-ray energy spectra have been measured and calculated for these shield configurations:

1. No shield
2. Three SS-304 slabs (15.24 cm thick)
3. Six SS-304 slabs (30.48 cm thick)
4. Configuration 3 plus one SS-304 and one borated polyethylene slab (40.64 cm thick)
5. Configuration 4 plus one SS-304 slab (45.72 cm thick)
6. Configuration 5 plus one borated polyethylene slab (50.80 cm thick)
7. Configuration 6 plus one SS-304 slab (55.88 cm thick)
8. Configuration 7 plus one heavymet slab (60.96 cm thick)

The neutron energy spectra measured and calculated behind shield configuration 4 are compared in Fig. 9.1 for the cases in which the detector is both on and off the axis of symmetry. The axis of symmetry is the deuterium-tritium target axis. The solid lines are the experimental data, the two lines for each spectrum represent the statistical uncertainty band in the unfolded neutron data, and the dots are the calculated results. The calculated data were obtained by smoothing the flux per unit energy in each multigroup energy interval with an energy-dependent Gaussian function that characterizes the neutron response of the detector.

The agreement between the calculated and experimental neutron and gamma-ray energy spectra for all of the shield configurations is very encouraging. In general, the neutron data agree within ~10% over the energy range from ~800 keV to 14 MeV, and the gamma-ray spectra agree within ~20% over the energy range from ~750 keV to 14 MeV.

Fig. 9.1. Measured and calculated neutron energy spectra as a function of position behind a shield configuration seven SS-304 slabs and one BP slab.

9.2 ABSTRACT OF INTEGRAL EXPERIMENTS FOR FUSION REACTOR DESIGN: EXPERIMENTATION⁷

G. T. Chapman, G. L. Morgan

Integral experiments that measure the energy spectra of neutrons and gamma rays due to the transport of ~14-MeV T(d, n) ⁴He neutrons through thick steel and borated polyethylene shield configurations have been performed at ORNL. Descriptions of the facility and experimental techniques are given, along with curves showing data representative of the results.

9.3 ABSTRACT OF INTEGRAL EXPERIMENTS FOR FUSION REACTOR DESIGN: ANALYSIS⁸

R. T. Santoro; R. G. Altmiller, Jr.; J. M. Barnes; E. M. Obloy

Integral experiments that measure the energy spectra of neutrons and gamma rays due to the transport of ~14-MeV D-T neutrons through laminated SS-304 and borated polyethylene shield assemblies have been performed. Measured and calculated energy spectra and integrated flux distributions are compared for a typical shield assembly as a function of detector location.

9.4 ABSTRACT OF SHIELDING OF FUSION REACTORS⁹

R. G. Alamiiller, Jr.

The blanket and shield assemblies of fusion reactors will contain a significant number of very sizable penetrations (neutral beam injection ducts, pumping ports, etc.). The combination of high-energy neutrons and large penetrations will introduce severe design problems that are quite different from those encountered previously. Fusion reactors with their penetrations are very complex geometric structures, and in calculating nuclear effects (heating, activation, etc.), tradeoffs must be made between computing efficiency and accuracy in the geometric modeling. The types of problems that arise because of large penetrations will be illustrated by the calculations that have been carried out to aid in the design of the shielding for the neutral beam injectors of the TFTR being built at Princeton University.

9.5 ABSTRACT OF SHIELDING CALCULATIONS FOR THE TFTR NEUTRAL BEAM INJECTORS¹⁰

R. T. Santoro; R. A. Lillie; R. G. Alamiiller, Jr.; J. M. Barnes

Two-dimensional discrete-ordinates calculations have been performed to determine the location and thickness of concrete shielding around the TFTR neutral beam injectors. Two sets of calculations were performed, one to determine the dose equivalent rate on the roof and wall of the test cell building when no injectors are present and one to determine the contribution to the dose equivalent rate at these locations from radiation streaming through the injection duct. Shielding the side and rear of the neutral beam injector with 0.305 and 0.61 m of concrete, respectively, and lining the inside of the test cell wall with an additional layer of concrete having a thickness of 0.305 m and a height above the axis of deuteron injection of 3.10 m is sufficient to maintain the biological dose equivalent rate outside the test cell to ~ 1 mrem per D-T pulse.

9.6 DOSE RATES FROM INDUCED ACTIVITY IN THE TFTR TEST CELL¹¹

R. G. Alamiiller, Jr.; J. Barish; R. T. Santoro; R. A. Lillie; J. M. Barnes; M. M. H. Ragheb

Calculated dose rates in the TFTR test cell from induced activity have been obtained. The dose rates for the case of no neutral beam injector and of neutral beam injector (with the resultant large penetration through the primary shield) are given separately. The photon transport calculations are carried out using Monte Carlo techniques, since this greatly facilitates the consideration of different pulse sequences and times after reactor shutdown. Dose rates are given for several different D-T pulse sequences and for a variety of times after the reactor is turned off. At a few positions in the test cell, the contribution to the dose rates from individual residual nuclei is also presented. In general, it is found that the presence of the shielded neutral beam injector does not have a large effect on the dose rates in the test cell.

9.7 ABSTRACT OF MAGIK - A MONTE CARLO SYSTEM FOR COMPUTING INDUCED RESIDUAL ACTIVATION DOSE RATES¹²

J. Barish; T. A. Gabriel; R. G. Alamiiller, Jr.

The photon dose rate from the induced activity produced by sustained bombardment of materials by neutrons and charged particles may present a significant radiation hazard. To minimize this hazard, the material configuration must be so designed that the photon dose rate decays to an acceptable level soon after the source beam is turned off. MAGIK calculates the time-independent photon dose rates that result from activities produced by nucleon-nucleus and meson-nucleus collisions over a wide range of energies. The system has been used both for high-energy accelerator studies and for fusion reactor studies. In the MAGIK system, the lengthy photon transport calculations are carried out independent of time, and the time dependence is introduced in the final

program, thereby permitting study of various operating scenarios with a minimum computing cost.

9.8 CONFERENCE PAPER: ONE-DIMENSIONAL GAMMA-RAY SHIELDING CALCULATIONS FOR THE EBT-2¹³

R. A. Lillie, T. A. Gabriel, B. L. Bishop,
V. C. Baker

Because of the expected large loss of electrons on the magnetic coil assemblies and other components of the proposed TLMO Bumpy Torus Proof-of-Principle (EBT-P) plasma device, the induced gamma-ray field from bremsstrahlung will be very intense. These gamma rays will yield increased heating loads on the superconducting cryogenic coils, as well as produce a significant biological hazard. In this summary, the results from a series of one-dimensional (1-D) calculations in which the gamma-ray heating loads were determined are presented for several coil shield configurations. Also, 1-D results are given that indicate the level of biological dose for the varying thicknesses of either lead or concrete that have been proposed for the primary shielding.

The transport of electrons, positrons, and gamma rays through the magnet assembly was carried out using the EGS-PEGS code system.¹⁴ This Monte Carlo code package takes into account all important physical processes in the energy range of interest.

The EGS-PEGS code system could have been used for the entire set of calculations were it not for the large statistical uncertainties associated with the calculated results at large shielded distances from the radiation source. To circumvent this problem, the EGS-PEGS code system was only used to generate gamma-ray spectra. These spectra were then input to the 1-D discrete-ordinates transport code ANISN.⁶ This code has been used extensively for 1-D shielding calculations and is very applicable to deep penetration problems.

The 1-D geometries employed in EGS-PEGS and ANISN, and in both the coil heating and biological dose calculations, consisted of a series of

infinitely long concentric cylinders as shown in Fig. 9.2. In the heating calculations (Fig. 9.2a), electrons were assumed to be isotropically incident on the front face of the 0.01-m-thick aluminum liner for the transport using the EGS-PEGS code system. Although the bremsstrahlung gamma rays produced by the electrons were somewhat smeared radially throughout the aluminum, the gamma-ray source was placed at a radius of 0.213 m for the ANISN calculations.

Fig. 9.2. One-dimensional geometric models.

The biological dose calculations were performed using the geometry illustrated in Fig. 9.2b. In these calculations, the biological dose rates for up to 0.305 m of lead and 1.83 m of concrete were obtained with and without 0.0318 m of ²³⁸U coil shielding. The bremsstrahlung spectra used in the heating calculations were again used for these calculations.

The rate at which electrons are lost from the plasma was assumed to be proportional to a Maxwellian distribution weighted by $E^{-3/2}$,

$$\text{loss} = \frac{A \times \text{EXP}(-E/kT)}{E},$$

where E represents the electron energy and T represents the plasma temperature. This distribution was normalized to a maximum power loss of 1 MW, yielding a normalization constant of $A = 6.242 \times 10^{18}/kT$.

The cumulative bremsstrahlung spectra for the various kT values were obtained from the expression

$$h(E_Y, kT) = \frac{\int_{0.239}^{10 \text{ MeV}} g(E_Y, E_e) \text{EXP}(-E_e/kT)/E_e dE_e}{\int_{0.239}^{10 \text{ MeV}} \text{EXP}(-E_e/kT)/E_e dE_e},$$

where $g(E_Y, E_e)$ represents the bremsstrahlung spectrum produced from electrons of energy E_e . (Since very-low-energy electrons do not contribute significantly to either the gamma-ray heating or the biological dose, only electrons with energies above 0.239 MeV were considered.) The integral was also extended to 20 MeV for a kT of 2.0 MeV, assuming the bremsstrahlung spectra from electrons with energies greater than 10-MeV electrons. This spectrum will be denoted by the use of an asterisk (*) above the kT value.

The total energy deposition rates obtained are presented in Fig. 9.3 for electron temperatures of 1.0, 1.5, 2.0, and 2.0* MeV as a function of coil shield thickness. The energy deposition rate for the tungsten shield was only calculated using the $kT = 2.0^*$ -MeV spectrum, since this spectrum yields the greatest heating.

The calculated biological dose rates as a function of lead shield thicknesses are presented in Fig. 9.4. These results were obtained with and without 0.0318 m of ^{238}U coil shielding for the electron temperatures of interest. The

Fig. 9.3. Total energy deposition rate in copper coils vs coil shield thickness for several electron temperatures.

isotropic gamma-ray sources used in these calculations were the same as those used in the heating calculations. For a kT value of 2.0* MeV in the absence of any ^{238}U coil shielding, 0.318 m of lead is required to reduce the biological dose rate to 2.5 mrem/h. (The corresponding amount of concrete is 1.92 m.) Since no reduction of the gamma-ray spectrum was assumed to occur because of shielding by the components of the machine, these results should represent the maximum required shielding to obtain a radiation level of 2.5 mrem/h.

9.9 RADIATION SHIELDING INFORMATION CENTER

R. W. Roussein, B. P. Maskewitz, D. K. Trubay

The Radiation Shielding Information Center (RSIC), established in 1962, serves its user community by collecting, organizing, processing,

Fig. 9.4. Biological dose rate vs lead shield thickness with and without 0.0318 m of ^{238}U coil shield for several electron temperatures.

evaluating, packaging, and disseminating information related primarily to nuclear reactor and weapons radiation. The scope includes the physics of interaction of radiation with matter, radiation production and transport, radiation detectors and measurements, engineering design techniques, shielding materials properties, computer codes useful in research and design, and nuclear data compilations. Originally established to support research related to fission, RSIC now supports fusion reactor technology. The major activities include (1) operating a computer-based information system

and answering inquiries and (2) collecting, testing, packaging, and distributing computer codes and processed nuclear data libraries.

RSIC activities are directed toward developing a radiation transport technology base and providing needed technology to the user community. The following are examples of how this is accomplished. The general purpose fine-group cross-section library, VITAMIN C, developed under RSIC guidance for use in the fusion neutronics community, was improved and updated to include data for 66 materials. Other representative contributions to the RSIC technology base include CLC-359/MAGIK¹² (see Sect. 9.7) from ORNL, the DLC-71/GAMMON¹⁵ activation cross-section and decay gamma spectrum library from Los Alamos Scientific Laboratory, and the DLC-72/MONTUK¹⁶ activation and transmutation library from the Atomic Energy Research Establishment, Harwell, England.

The computer-based information system was upgraded to utilize ORNL ORCHIS systems for file maintenance and search, which will result in files that are more up-to-date and in faster turnaround on searches. A minicomputer was installed, allowing more efficient and less costly information processing.

The above represents a small portion of the total activity for the reporting period.

The RSIC technology base increased in each subtask during the year:

1. Literature - 9400 citations¹⁷ may be accessed on DOE/RECON, the Department of Energy (DOE) data base management system; 43 RSIC reports were issued with several volumes in series.
2. Codes collection - 365 complex shielding code packages¹⁸ and 145 auxiliary data processing packages.¹⁹
3. Data collection - 72 data library packages²⁰ and additional working cross-section libraries are now maintained for the DOE Office of Fusion Energy and the Defense Nuclear Agency.
4. RSIC Newsletter - distribution is about 1640.

Latest user statistics (FY 1979) for RSIC indicate that 3450 separate letters and telephone calls (14 each work day) were processed during the year, and 96 visitors were received. The fusion research community accounted for about 21% of the total number of requests.

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10. MANAGEMENT SERVICES

V. C. Kruzic, Section Head

S. K. Adkins	D. A. Hambright ¹	J. C. Neeley ¹
D. R. Alford	J. R. Hickey ²	P. L. Ray
A. B. Barnette	J. M. Holladay	E. M. Ruckart
P. G. Beasley	P. J. Hudson ³	W. K. Russell
S. S. Bell ¹	H. W. Jernigan	R. C. Satterfield
D. P. Brooks	J. R. Jernigan ³	S. R. Schwartz ¹
J. I. Burke	R. P. Jernigan	C. M. Sekula ¹
E. L. Cagle ¹	B. L. Johnson ¹	D. G. Sharp
C. J. Chamberlain	C. H. Johnson ¹	J. G. Sharp ¹
C. H. Cox	D. Y. Johnson ¹	B. L. Straine
K. M. Dobbs ¹	M. N. Johnson ³	C. K. Thomas
R. S. Edwards	J. K. Lovin	E. L. Watkin ¹
J. C. Ezeli	J. B. Martin ^{1,3}	E. E. Webster ¹
M. L. Fullard ¹	B. J. McClure ¹	W. H. West ¹

E. R. Wells

T. G. Yow¹

-
1. Information Division
 2. Finance and Materials Division
 3. Part-time

10. MANAGEMENT SERVICES

ABSTRACT. The Management Services Section has now completed three years of operation in essentially its original organizational form. Services are provided in general administration, personnel, financial management, communications (including text and graphics generation), management information, library safety, quality assurance, and nonprogrammatic engineering services. The value of centralizing management functions within a section has been clearly demonstrated, and the general methodology of the functioning of the section within the Division has been established. Thus, more attention is now turning to improvements in relationships with external organizations and to procedural improvements. In this report, highlights of specific accomplishments are presented.

10.1 FINANCE OFFICE

The funding trend for the Fusion Energy Division (FED) and the Fusion Energy Program continues to exhibit substantial growth, as illustrated in Table 10.1 and Fig. 10.1. This trend creates a related need for growth and improvement in specific areas of financial control. In order to meet the research objectives of the program, increased emphasis must be placed on financial analysis, variance reporting, and accurate cost projections.

During the past year the Finance Office continued to provide monthly cost reports for the various work breakdowns, to work with research personnel in properly planning expenditures within budget constraints, to operate the coordinated work order control system, and to perform many other functions associated with Division financial and accounting transactions.

Fig. 10.1. Fusion Energy expense funding.

10.2 OFFICE OF THE ADMINISTRATOR

This office is responsible for most of the general administrative functions of the Division.

10.2.1 Visitor Control

Arrangements were made during the past year for 1596 visitors to the Division, including 220 noncitizens.

10.2.2 Personnel Functions

During the past year arrangements were made for interviews with 30 prospective employees,

19 of whom accepted employment. In addition, seven new employees transferred from other ORNL organizations. Sixty guest assignments were processed, by which nonemployees participate in Division programs.

10.2.3 Subcontracts

In 1979 industrial involvement through subcontracts exceeded \$9 million. Major contracts were for the Large Coil Program (LCP) prototype fusion reactor coils and gyrotron development for the ELMO Bumpy Torus (EBT) and Impurity Study Experiment (ISX) programs.

Table 10.1. Fusion energy program expense funding

Activity	Dollars in thousands					
	FY 1977 actual cost	FY 1978 actual cost	FY 1979 actual cost	FY 1980 funding as of February 1980	FY 1981 budget submission	FY 1982 budget submission
Applied plasma physics						
Fusion plasma theory	\$ 2,063	\$ 2,196	\$ 2,335	\$ 2,290	\$ 3,330	\$ 4,239
Experimental plasma research	843	898	894	970	1,170	1,280
National MFE computer network	271	189	270	290	450	600
Total applied plasma physics	\$ 3,177	\$ 3,283	\$ 3,499	\$ 3,550	\$ 4,950	\$ 6,119
Confinement systems						
Tokamak systems						
Research operations	\$ 6,811	\$ 5,973	\$ 7,481	\$ 7,705	\$ 9,100	\$ 9,600
Major device fabrication	4,574	4,709	4,044	2,700	8,530	13,700
Magnetic mirror systems						
Research operations	2,147	2,500	3,705	5,400	7,950	12,150
Major device fabrication	0	0	0	2,000	1,590	1,800
Total confinement systems	\$13,532	\$13,182	\$15,230	\$17,805	\$27,080	\$37,250
Development and technology						
Magnetic systems	\$ 5,691	\$ 7,593	\$11,432	\$ 9,840	\$ 7,605	\$ 5,800
Plasma engineering	6,063	5,442	6,442	5,045	13,400	16,950
Fusion reactor materials	1,645	2,796	2,998	3,515	4,425	5,993
Fusion systems engineering	2,535	1,954	2,025	4,825	7,715	11,172
Environment and safety	0	55	254	600	500	220
Total development and technology	\$15,934	\$17,840	\$23,151	\$23,525	\$33,645	\$40,135
Reactor projects - Tokamak Fusion Test Reactor (TFTR)	\$ 321	0	0	0	0	0
Total ORNL funding	\$32,964	\$34,505	\$41,880	\$44,880	\$65,675	\$83,504

10.2.4 International Agreements

The administrator's office was heavily involved in the legal and financial details of implementing agreements signed by the Department of Energy (DOE) and foreign countries. Among the agreements were those between the United States and the Soviet Union and the United States and Japan, the Large Coil IAE agreements, and special Oak Ridge National Laboratory (ORNL) agreements with Spain, Germany, England, and France.

Also handled by this office were arrangements for personnel from other national laboratories and industrial firms to participate in the Engineering Test Facility (ETF) Design Center. The participants in the ETF Design Center were chosen for their expertise

in designing systems for reactors (magnetic, etc.) and their familiarity with fusion requirements.

10.2.5 Communications Center

The FED Communications Center encompasses a wide range of publishing services - editing, word processing, graphics, photography, and reproduction. In 1979, over 500 publications were processed for the Division. These included 250 reports, journal articles, and conference papers and 260 abstracts. The graphics group prepared approximately 10,000 prints and vugraphs and 200 posters, covers, and signs.

During the year, reproduction services were improved by the acquisition of a Xerox 9400

copier, which enables simultaneous two-sided printing. In addition, plans were made to update the Communications Center word processing equipment; alternatives to the IBM Mag Card II typewriters now in use are being investigated, and a new system will be selected in 1980.

Late in the year, one of the editorial staff was designated Communications Center Liaison for the newly funded ETF Design Center. The Communications Center staff will be augmented as needed to keep pace with the growing ETF publication demands.

10.3 MANAGEMENT INFORMATION SYSTEM

The purpose of the FED Management Information System (MIS) is to provide the Division's management staff with timely and relevant information required to control costs and schedules. Progress was made in several areas during 1979, although efforts were not as intense as in the previous year. During this reporting period, FED MIS personnel were also responsible for the progress made on the ORNL MIS. The Laboratory effort reduced the progress made on the FED MIS, but will benefit the Division in the future. Most of the effort during this reporting period was to keep existing systems operational and to modify them to respond to the changing information needs of management.

The interaction PERT program and store room inventory module were kept operational. The capabilities of the Procurement Information System were expanded and the accuracy of its data improved. The procurement system now uses a data base management system that reduces the run time. The major change in this system is that the data base is extracted from a data base for the Laboratory, which saves the cost of inputting the Division data. The Laboratory system is based on the MIS work accomplished in the FED.

Significant improvements to the Personnel Information System enable a quicker response to user questions. Using mathematical and logical operators on any of the data elements,

the program provides the user with almost unlimited flexibility in structuring information requests. Because the system includes a user-controlled report generator and can plot any two data elements, the user is able to ask unstructured questions. This software is also being incorporated in other FED MIS modules.

Problems in obtaining accurate labor data prevented the FED Labor Information System from becoming operational this year. Scheduled software development was accomplished. Cross-checks of labor data revealed discrepancies in the data which are being resolved.

Work has begun on a new system that will provide information on project equipment and materials. Scheduled delivery and installation dates, receiving and storage location, and drawing and bill-of-material numbers are available. In addition to allowing data structuring under the accounting system, the system incorporates a work breakdown structure (WBS).

10.4 ENGINEERING SERVICES

Engineering Services provides all of the nonprogrammatic engineering work and coordination for the Division. This includes coordination of machine shop work, maintenance craft work, inventory and storage of equipment and materials, telecommunications coordination, and the coordination of Division requests for general plant projects (GPP), general plant equipment, and line item facility improvements. The group is also responsible for maintaining buildings, facilities, and equipment and for planning future facilities. Some of the highlights of this activity in the past year include the following:

- Completion of 15 expense and 5 capital equipment funded projects.
- Initiation of several new expense equipment and GPP Projects, as well as new engineering studies in building and facility upgrading.

- Coordination of 12 man-years of machine shop work, comprising 161 jobs at the X-10 plant shops. Shop work at the Y-12 plant comprised 485 jobs and 3.1 man-years. The total value of these jobs was \$2,194,000.
- Completion of the offices and the surface physics and laser laboratories in Building 9201-2. In addition, two large motor-generator sets in the building were reworked, and construction was started on a new rest room.
- Installation of a demineralized water system and continued construction on the electrical power upgrade in Building 9204-1.

10.4.1 Procurement Expediting

The Procurement Expediting Group is functioning well. During 1979, this group processed 2059 requisitions for materials, equipment, and services representing a volume of over \$15,000,000. Continuing efforts to improve delivery of ordered items are showing success for certain commodities. The group now consists of four people, and the addition of a part-time purchasing representative on site has proven to be very valuable to the Division.

10.4.2 Safety and Emergency Planning

The Division Safety Program includes monthly safety inspections by individual sections, quarterly inspections by an independent inspection committee comprising the division safety officer, a Y-12 safety representative, and a Y-12 maintenance supervisor. Monthly fire inspections are performed by professional fire inspectors. Six safety training sessions are conducted each year, which all Division personnel must attend. The building Emergency Squad continued its cardio-pulmonary resuscitation (CPR) training program as well as response training for building emergencies. Fifty-two people were qualified in CPR (27 recertified, 25 new). The FED Emergency Procedures was updated and reissued.

10.4.3 Quality Assurance

The Quality Assurance (QA) Program for the Division continues to improve. A DOE Oak Ridge Operations audit of one program noted no major deficiencies. The QA Program was expanded to include imposing QA requirements in subcontracts for major external work for the Division. The Program continues to be effective in ensuring reliable and cost-effective experimental equipment, both fabricated and purchased. A Special Power Supply Committee, chaired by the Quality Assurance Coordinator, was established to improve the quality of purchased power supplies.

A complete revision of the Division QA manual was completed and issued to comply with new requirements of ORO IMD 02xx, UCC-ND SPP D2-16, and the ORNL QA manual.

10.5 LIBRARY

The Library provides library and information services to the Division, ORNL, Union Carbide Corporation Nuclear Division (UCC-ND), and DOE. A collection of books, technical reports, and scientific journals concerning plasma physics and fusion energy technology is maintained. Computerized information retrieval services provide access to many data bases (DOE RECON, SDC ORBIT, LOCKHEED DIALOG, and OCLC). Access to bibliographic citations is unlimited in these data bases. For example, the Energy Information data base of DOE RECON alone had over one million computer searchable items as of January 1, 1978. The INSPEC file, one of approximately 100 computer searchable data files in the LOCKHEED DIALOG system, has 800,000 citations dating from 1969 to September 1976, with approximately 12,500 citations added monthly. The OCLC data base, which incorporated approximately 21 networks, provides access to the library holdings of major North American libraries.

FUSION ENERGY DIVISION

MAY 1980

D. B. WONG
Z. S. RICHARDSON
R. A. NEEDY
R. H. BISHOP

1. DUAL CAPACITY
2. FED TECHNOLOGY UTILIZATION OFFICER
3. PART TIME
4. FINANCE AND MATERIALS DIVISION
5. INFORMATION DIVISION
6. INSTRUMENTATION AND CONTROLS DIVISION
7. RENSSELAER POLYTECHNIC INSTITUTE
8. COMPUTER SCIENCES DIVISION
9. UNIVERSITY OF TENNESSEE
10. JAPAN ATOMIC ENERGY RESEARCH INSTITUTE
11. SPANISH NUCLEAR ENERGY COMMISSION
12. MASACHUSETTS INSTITUTE OF TECHNOLOGY
13. NEW YORK UNIVERSITY
14. ELECTROTECHNICAL LABORATORY, JAPAN
15. UNIVERSITY OF WISCONSIN
16. INSTITUTE FOR ADVANCED STUDY
17. GEORGIA INSTITUTE OF TECHNOLOGY
18. GENERAL ELECTRIC COMPANY
19. SIGMA ELECTRONICS COMPANY
20. STANFORD UNIVERSITY
21. INSTITUT FÜR PLASMAPHYSIK, KFA
22. METALS AND CERAMICS DIVISION
23. GRUMMAN AEROSPACE CORPORATION
24. UCC NO. ENGINEERING
25. R. B. JACOBS ASSOCIATES
26. WESTINGHOUSE ELECTRIC CORPORATION
27. GENERAL ATOMIC COMPANY
28. McDONNELL DOUGLAS AERONAUTICS COMPANY
29. PRINCETON PLASMA PHYSICS LABORATORY
30. BURNS & ROE, INCORPORATED
31. HANFORD ENGINEERING DEVELOPMENT LABORATORY
32. IDAHO NATIONAL ENGINEERING LABORATORY
33. BECHTEL NATIONAL, INCORPORATED

AFFIRMATIVE ACTION REPRESENTATIVE: C. E. BUSH (1181)
SAFETY OFFICER: R. S. EDWARDS (10120)

LIST OF ABBREVIATIONS

AEDC	Arnold Engineering Development Center
ARE	aspect ratio enhancement
ATC	adiabatic toroidal compression
COM	constants of motion
CPR	cardiopulmonary resuscitation
CPU	central processing unit
CRT	cathode ray tube
CW	continuous wave
CMTX	Coil Winding Test Experiment
DEMO	Commercial Demonstration Reactor
D-III	Doublet-III
DITE	Divertor Injector Tokamak Experiment
DOE	Department of Energy
dpa	displacement per atom
D-T	deuterium-tritium
EBR-II	Experimental Breeder Reactor II
EBT	ELMO Bumpy Torus
EBT-P	ELMO Bumpy Torus Proof of Principle
ECH	electron cyclotron heating
EF	equilibrium field
EPR	Experimental Power Reactor
ETF	Engineering Test Facility
FCT	flux-conserving tokamak
FED	Fusion Energy Division
FIR	far-infrared
FM	frequency modulation
FWHM	full width at half maximum
GA	General Atomic
GPP	general plant projects
HFIR	High Flux Isotope Reactor
HPTF	High Power Test Facility
IAEA	International Atomic Energy Agency
ICRH	ion cyclotron resonance heating
IEA	International Energy Agency
i.f.	intermediate frequency
IGC	Intermagetics General Corporation
INTOR	International Tokamak Reactor
IP	industrial participant
IPP	Institute of Plasma Physics (Garching)
ISX	Impurity Study Experiment
JIPP	Japanese Institute of Plasma Physics
JSW	Japan Steel Works
LART	Large Aspect Ratio Tokamak
LASL	Los Alamos Scientific Laboratory
LCP	Large Coil Program
LCTF	Large Coil Test Facility
LIFT	laser-induced fluorescence technique
LTIF	Low Temperature Irradiation Facility
MEA	momentum energy analyzer
METF	Medium Energy Test Facility
MFE	magnetic fusion energy
MFTF	Mirror Fusion Test Facility
MHD	magnetohydrodynamics
MIS	Management Information System
MIT	Massachusetts Institute of Technology
MOA	Memorandum of Agreement
MOL	Method of lines
MSDT	moving-slug displacement transducer
NBI	neutral beam injector
NBT	Nagoya Bumpy Torus
NMFECC	National Magnetic Fusion Energy Computer Center
NRL	Naval Research Laboratory

ODE	ordinary differential equations
OFE	Office of Fusion Energy
OH	ohmic heating
ORMAK	Oak Ridge Tokamak
ORNL	Oak Ridge National Laboratory
ORO	Oak Ridge Operations
ORR	Oak Ridge Research Reactor
PDX	Poloidal Divertor Experiment
PF	poloidal field
PKA	primary knock-on atom
PLT	Princeton Large Torus
PPPL	Princeton Plasma Physics Laboratory
PRD	proposed reference design
QA	quality assurance
R&D	research and development
rf	radio frequency
rms	root mean square
RSIC	Radiation Shielding Information Center
SAI	Science Applications, Incorporated
SAP	sintered aluminum product
SCR	silicon-controlled rectifier
SEM	scanning electron microscopy
SIMS	secondary ion mass spectroscopy
SMW	submillimeter wave
T-12	tokamak at Kurchatov Institute
TF	toroidal field
TFTK	Tokamak Fusion Test Reactor
TMX	Tandem Mirror Experiment
TNS	The Next Step
UCC-ND	Union Carbide Corporation-Nuclear Division
USC	User Service Center
VFA	velocity filter analyzer
VUV	vacuum ultraviolet
1-D	one-dimensional
2-D	two-dimensional
3-D	three-dimensional

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M. J. Saltmarsh, W. R. Grimes, and R. T. Santoro, "An Optimization of the Fission-Fusion Hybrid Concept."

1st Topical Meeting on Fusion Reactor Materials, Miami Beach, Florida, January 29-31, 1979

R. J. Colchin, "Materials Experience with Tokamak Plasmas."

R. A. Langley, "Interaction of Implanted Deuterium and Helium with Beryllium: Radiation Enhanced Oxidation."

R. A. Langley, R. J. Colchin, R. C. Isler, M. Murakami, J. E. Simpkins, J. L. Cecchi, V. L. Corso, H. F. Dylla, R. A. Ellis, Jr., and M. Nishi, "The ISX-A Graphite Limiter Experiment."

B. M. V. Scherzer, R. S. Blewer, R. Behrisch, R. Schulz, J. Roth, J. Borders, and R. A. Langley, "Radiation Induced Detrapping of Implanted Deuterium in BEO by High Energy HE-3 and Proton Irradiation."

J. E. Simpkins and R. J. Colchin, "Time Dependence of Gases from Plasma-Wall Interactions in ISX-A."

Seminar at North Carolina State University, Raleigh, North Carolina, February 12, 1979

F. W. Baity, Jr., "The EBT Experimental Program at ORNL."

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D. Steiner, T. E. Shannon, J. M. Rawls, and P. Reardon, "Presentations to the Plenary Session, ETF Mission Workshop."

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B. Carreras, H. R. Hicks, J. A. Holmes, and B. V. Waddell, "Multiple Helicity Tearing Mode Calculations: Major Disruptions."

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J. L. Dunlap, R. D. Burris, J. H. Harris, A. P. Navarro, and V. K. Paré, "Survey of Modes and Their Effects in ORMAK, ISX-A, and ISX-B."

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R. H. Dilworth, "Energy Alternatives for the Future."

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J. Sheffield, "Experimental Confinement Program."

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C. H. An and G. Bateman, "Computational and Analytic Study of Ballooning Modes in Highly Elongated Tokamaks."

M. Azumi and D. B. Nelson, "FCT Heating of Free Boundary Equilibria."

D. B. Batchelor and R. C. Goldfinger, "Finite Temperature Effects on Microwave Propagation in EBT."

C. O. Beasley, Jr., W. I. van Rij, and J. Denavit, "TED1 - A Numerical Simulation of the Time Evolution of Drift Waves."

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- T. Uckan, "Wall Effects on the Absorption of Electron Cyclotron Waves in ELMO Bumpy Torus (EBT) Plasma."
- J. H. Whealton, E. F. Jaeger, and R. W. McGaffey, "Comparison of Slot and Cylinder Optics."
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- R. A. Langley and W. P. Eatherly, "Graphite Application in Tokamak Fusion Reactors."

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- H. T. Yeh, "Transient 3D Eddy Current in Multiple Conductors."

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- W. H. Gray, C. J. Long, and W. C. T. Stoddart, "The Structural Design of Superconducting Magnets for the Large Coil Program."

NATO Advanced Study Institute on Atomic/Molecular Physics and Fusion, Toulouse, France, August 13-24, 1979

- J. T. Hogan, "General Principles of Magnetic Fusion Confinement."

1979 International Cryogenics Materials Conference, Madison, Wisconsin, August 21-24, 1979

- L. Dresner, "Heating-Induced Flows in Cable-in-Conduit Conductors."
- L. Dresner, "Propagation of Normal Zones in Thermally Insulated Superconductors."
- W. A. Fietz, "Conductor Requirements and Design Considerations for Superconducting Mirror and Tokamak Fusion Machines."
- J. W. Lue, J. R. Miller, and L. Dresner, "Stability of Cable-in-Conduit Conductors."
- F. N. Mazandarany, D. M. Parker, R. F. Koenig, and D. T. Read, "A Nitrogen-Strengthened Austenitic Stainless Steel for Cryogenic Magnet Structures."
- P. F. Michaelson, R. Quay, R. F. Koenig, P. L. Walstrom, and J. S. Goddard, "Heat Transfer and Helium Replenishment in Cabled Conductor Cooling Channels."
- J. R. Miller, J. W. Lue, and S. S. Shen, "Stability Measurements of a Large Niobium/Tin Force-Cooled Conductor."
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- W. A. Houlberg, H. C. Howe, and S. E. Attenberger, "Deuterium and Tritium Pellet Fueling in a Diverted ETF/INTOR Plasma."

Istituto di Fisico, Varenna, Italy, September 5-7, 1979

- H. H. Haselton, "Neutral Beam Injectors - The Oak Ridge Concept."

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- H. T. Yeh, "Transient Eddy Current in Magnet Structure Members."

Symposium on Atomic Spectroscopy, Tucson, Arizona, September 10-14, 1979

- R. C. Isler, "Spectral Features of Tokamak Produced Plasmas."

Conference on Impact of Advanced Systems on Scientific Computing, Livermore, California, September 12-13, 1979

- H. R. Hicks and B. Carreras, "Realistic 3D Resistive MHD Calculations on the CRAY-1."

INTOR Meeting, Atlanta, Georgia, September 12-14, 1979

- J. Sheffield, "A Bundle Divertor for INTOR."

5th European Conference on Controlled Fusion and Plasma Physics, Oxford, England, September 17-21, 1979

- F. W. Baity, K. H. Carpenter, J. A. Cobble, R. J. Colchin, R. A. Dandl, H. O. Eason, J. C. Glowienka, G. R. Haste, S. Hiroe, N. H. Lazar, B. H. Quon, T. Uckan, and T. L. White, "Experimental Scaling Results in ELMO Bumpy Torus (EBT-S)."
- F. W. Baity, K. H. Carpenter, J. A. Cobble, R. J. Colchin, H. O. Eason, J. C. Glowienka, G. R. Haste, C. L. Hedrick, E. F. Jaeger, R. K. Richards, T. Uckan, T. L. White, F. M. Bieniosek, K. A. Connor, R. L. Hickok, S. P. Kuo, R. A. Dandl, S. Hiroe, N. H. Lazar, and B. H. Quon, "Scaling in ELMO Bumpy Torus (EBT)."
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- D. Steiner, "Fusion May Provide Us with a Long-Term Energy Option in the Next Century."

2nd Annual Symposium on Small Systems, Dallas, Texas, October 2-3, 1979

H. T. Yeh and S. S. Shen, "Application of Microcomputers to Superconducting Magnet Experiments."

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26th National Symposium of the American Vacuum Society, New York, October 2-5, 1979

M. J. Saltmarsh, "A Review of the ISX-B Experimental Tokamak Program."

S. W. Schwenkerly, W. K. Dagenhart, P. M. Ryan, and C. C. Tsai, "Application of Cryopumping Techniques to Fusion Devices."

Joint Power Generation Conference, Charlotte, North Carolina, October 8-10, 1979

D. Steiner, "Fusion Power Development: Status and Prospects."

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J. R. McNally, "Fusion Energy - Alternative for the Future."

*American Nuclear Society 1979 Winter Meeting, San Francisco, California, November 11-16, 1979
(proceedings published in Trans. Am. Nucl. Soc., 1979)*

S. E. Attenberger and W. A. Houlberg, "Physics of Burn Control in Tokamak Reactors."

W. R. Becraft, "ETF and Its Mission."

W. R. Becraft, R. J. Onega, and C. Kukiela, "Thermal Consequences to the Tokamak First Wall During a Disruption for Spatial Non-Uniform Energy Deposition."

P. H. Edmonds, "The ISX Experiment - Facilities and Future."

W. M. Fletcher, J. S. Goddard, J. L. Murphy, R. W. Napier, and P. L. Walstrom, "The Large Coil Test Facility Instrumentation System Design."

W. L. Gardner, "The ORNL Prototype PDX Neutral Beam Injection System."

T. C. Jernigan, "Objectives of the Tokamak Magnetic Fusion Program."

R. A. Langley, "ISX-B Impurity Study Program."

J. T. Mihalcz, "ISX Objectives and Physics."

A. Nagy and R. L. Reid, "Evaluation of a Committed Site for Fusion Reactor Development."

Meeting of the American Physical Society - Division of Plasma Physics, Boston, Massachusetts, November 12-16, 1979 (proceedings published in Bull. Am. Phys. Soc., 1979)

D. B. Batchelor and H. Weitzner, "An Eikonal Expansion of the Vlasov-Maxwell Equations Valid Near Cyclotron Resonance."

S. C. Bates, M. J. Saltmarsh, and D. W. Swain, "Plasma Shaping Experiments on ISX-B."

C. O. Beasley, W. T. van Rij, J. Denavit, and J. E. McCune, "A Numerical Study of Non-Local Drift Modes Driven by an Ion Temperature Gradient."

F. M. Bieniossek, K. A. Connor, and R. L. Hickok, "A Study of EBT Plasmas with a Heavy Ion Beam Probe."

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C. E. Bush, R. C. Isler, and R. E. Worsham, "Power Losses from the ISX-B Tokamak."

J. D. Callen, "Resistive-G Modes in the Edge of a Tokamak Plasma."

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L. A. Charlton, D. B. Nelson, and R. A. Dory, "Resistive Diffusion of a Stable High FCT Equilibrium."

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Fusion Energy Division Staff

N. N. Brevnov, R. Clausing,
A. A. Kazachkov, and N. V.
Kuznetsov

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