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Abstract

FY 1980 CTR Plasma Engineering Studies performed at the University of Illinois are reported. Current studies concentrated on four major areas including i) field-reversed mirror (FRM) and related compact tori, ii) dynamic behavior of the reversed-field pinch (RFP) including transport and stability and start-up, iii) plasma buildup in small mirrors by including finite, ion orbit effects as well as the drift cyclotron loss-cone velocity-space diffusion, and iv) high-energy fusion product transport in non-circular and high- β tokamaks, alpha ash buildup and possible control in tokamaks. Various computer packages have been produced for FRM, FRMP, RFP, small compact tori and non-circular tokamaks. Basic models and the code packages developed and tested with available experiments are of vital information that can be used in conceptual reactor studies for the scaling and prediction of plasma behavior in near-term reactors.

Table of Contents

	Page No.
Abstract.	i
I. Introduction	1
II. Field-Reversed Mirror (FRM) and Related Compact Tori	3
II.1 FRM Global Model and Alpha Heating Studies	3
II.2 FRM Start-up and Buildup Code.	11
II.3 Heat Conduction Effects in a Field-Reversed Plasma	24
II.4 Spheromak Transport and Reactor Viability Studies.	30
II.5 Rotational Model for Field-Reversed Theta Pinch (FR Θ P)	33
II.6 Plasma Systems Evaluation of Moving Plasmoid Heater. (MPH).	38
III. Dynamic Behavior of Reversed-Field Pinch (RFP)	39
III.1 One-Dimensional Reversed-Field Pinch (RFP) Studies.	39
III.2 Reversed-Field Pinch (RFP) Start-up	39
IV. Plasma Buildup in Small Mirror Devices	44
V. Fusion Products Effects in Tokamaks.	45
V.1 Transport Model for Non-circular and High- β Tokamaks	45
V.2 Alpha Ash Buildup in Tokamaks.	46
VI. Description of Computer Codes.	49
VI.1 Fusion Product Transport Codes	49
VI.2 Reversed Field Configurations.	50
VI.3 Cyclotron Radiation Code	51
VII. References	52
VIII. Staff Contributors and Acknowledgements.	55
IX. Reports, Theses, and Publications.	56
X. Distribution List.	59

I. Introduction

The effort this year in developing the plasma systems data base necessary for realistic conceptual fusion reactor studies is continued from last year⁽¹⁾ and has largely concentrated on the development of a field-reversed mirror (FRM) start-up code and the comprehensive plasma rotation model for field-reversed theta pinch (FR θ P), on the refinement of 1-D dynamic MHD code of reversed-field pinch (RFP), and on the completion of the fusion product transport extended to the non-circular and high- β tokamaks. As described in Section II, both Monte Carlo and globally averaged, two-fluid (ion and electron), point model codes have been completed for alpha particle heating studies for FRM. As much as 50% of the fusion product energy (and the resulting thermal fusion products) are found to be deposited in the closed field region of the reference SAFFIRE reactor designs.

A one-dimensional radial code has been developed to model the start-up of an FRM. The spatial effects of neutral-beam ionization and the ability to compute plasma quantities in a self consistent manner are of particular significance. Detailed description of the results is included in Section II.2. Global decay and one-dimensional profile evolution plasma models have been developed for Spheromak transport and reactor viability. This global-decay code will provide a general reactor viability study in support of the PPPL S-1 Spheromak and also will provide a foundation for a crude conceptual design of a linear Spheromak fusion reactor.

Plasma rotation model along with associated instability and anomalous transport for FR θ P was studied (Section II.5). Supported by encouraging experiments at LASL, the FR θ P along with a Moving Plasmoid Heater Reactor (MPHR) may become another candidate for compact torus reactor.

One-dimensional dynamic MHD code and the start-up of the reversed field pinch (RFP) were made fully operational (Section III). A study of a small compact RFP reactor (CRFP) was also done. Refueling the CRFP was particularly successful due to its small size. The rapid penetration of the fuel and the extended burn have doubled the energy multiplication factor, Q . The wall loading was found less peaked as illustrated in Section III.1. To explore the equilibrium state associated with the initial phases of RFP start-up, further development associated with 1 1/2-dimensional code is discussed.

In Section IV, studies of plasma buildup in small mirror devices are described. The drift cyclotron loss-cone (DCLC) velocity-space diffusion was incorporated into the current mirror buildup model developed at Illinois. This modification allows the treatment of stream-stabilized plasmas such as 2XII-B, MFTF, and the TMX end cells. Simulations done for 2XII-B appear to be in good agreement with global simulations done at LLL earlier. Transport model for non-circular and high- β tokamaks was completed. As for applications to ETF and Doublet experiments whose plasmas are non-circular, the current study finds the start-up and maintenance of ignition more difficult due to reduced flux-surface-averaged ion heating rate. Alpha ash buildup has received continued effort to investigate the effects of radial electric fields on transport in tokamaks with finite cross-field conductivity. It was found, although radial electric fields large enough to give essentially infinite fuel confinement times are not likely to be practical, much smaller fields can significantly improve the performance of an ash-fraction-limited tokamak.

Finally, various computer codes completed during this year have been briefly described in Section VI. Persons interested in detailed information on these codes are suggested to contact the principal investigator.

II. Field-Reversed Mirror (FRM) and Related Compact Tori

II.1. FRM Global Model and Alpha Heating Studies

The effect of fusion products (fps) on Field-Reversed Mirror (FRM) reactor concepts has been evaluated through the development of two new computer models, MCFRM and FRMOD⁽²⁾. The first code (MCFRM) treats fps as test particles in a fixed background plasma, which is represented as a fluid. MCFRM includes a Monte Carlo treatment of Coulomb scattering and thus provides an accurate treatment of fp behavior even at lower energies where pitch-angle scattering becomes important. The second code (FRMOD) is a steady-state, globally averaged, two-fluid (ion and electron), point model of the FRM plasma that incorporates fp heating and ash buildup values which are consistent with the MCFRM calculations. These values are obtained by expressing the fp energy and particle retention in the closed field region in terms of the fraction of marginally confined fps (i.e., those fps that can interact with both the open and closed field plasma but still do not have a "loss cone"). Both MCFRM and FRMOD employ an approximate representation of the field-reversed equilibrium based on Hill's vortex model.

These codes have been used extensively in the development of both D-T and advanced-fuel FRM reactor design (SAFFIRE)⁽³⁾. These studies show that a significant amount of fp energy (and associated ash) deposition occurs in the closed field region of the FRM, despite the relatively small size of the plasma (radius equal ~ 5 to 15 ion gyroradii). As much as 50% of the fp energy (and the resulting thermal fps) are deposited in the closed field region of the reference SAFFIRE designs. This is particularly important from the standpoint of advanced-fuel operation, where attractive ($Q > 10$) systems are found within the current "stable" limits on the plasma size, together with possibility of "ignited" operation if the present stability limits are extended by 50%. These results are illustrated in Fig. 1

FSL-79-227

Figure 1. Energy Multiplication Factor and Net Power Output vs the Plasma Ion Temperature for D-³He Systems With "Near-Classical" Loss Rates.

and in Table 1. Typical parameters for a D-³He type SAFFIRE plant cell, as illustrated in Table 2, are: 2.3 MW from a 3/1 elongated, 21.5-cm mirror radius plasma, with ion and electron temperatures of 80 and 75 keV and a plasma ion density of $4.4 \times 10^{14} \text{cm}^{-3}$. Larger powers can be obtained by stacking cells.

A Catalyzed-D version of SAFFIRE has also been studied, and some typical results are illustrated in Fig. 2 and in Table 3. An investigation of the steady-state energy distribution of fps in the FRM was also carried out to check for the possibility that instabilities might develop. Some typical results are shown in Fig. 3. Included in Table 3 is an elongated Cat-D case similar to the D-³He case also in that table.

A brief summary of the unique features of the present analysis and some of the characteristics of the FRM system which have, for the first time, been identified includes:

1. The application of a Monte Carlo treatment of Coulomb scattering to fp motion in the FRM; thus allowing both fp energy and particle deposition to be evaluated.
2. The use of a modified Hill's vortex model of the FRM equilibrium (that allows for possible elongation) which not only provides a straightforward means of coupling the results of the Monte Carlo calculation into the plasma burn code, but also provides an improved basis for reducing the three-dimensional balance equations for particles and energy to zero-dimensional form.
3. The subsequent identification of a significant amount of fp heating and ash buildup in the FRM plasma (a result which was not previously expected due to the small size of the FRM).
4. A simplified means of expressing the absolute and closed-field confinement limits for fps in the FRM; which allows these limits to be determined for any charged fp in any FRM plasma by specifying only the vacuum field strength and the plasma size.
5. A corresponding means of correlating the MCFRM results for the actual fp energy and particle deposition in the closed field region of the FRM over a wide range of system parameters.
6. A model of heat conduction losses to the cold, flowing plasma on the open field lines that assumes the development of a low density "boundary layer" which helps to "insulate" the closed field region.
7. A calculation of fp energy distributions in the steady-state FRM, and a discussion of their trends.

Table I

Principle Plasma Parameters for Two Reference SAFFIRE Designs with
D-T and D-³He Fuel Cycles

	<u>D-T[†]</u>	<u>D-³He</u>
Stability Factor, $S \equiv \frac{R_{HV}}{3\rho_i}$	10	10
Vacuum Field (kG)	60	60
Elongation Factor, κ	1	3
Plasma Radius, Volume (cm, L)	26.0, 74	21.5, 125
Ion, Electron Temp. (keV)	60, 53	80, 75
Ion, Electron Density ($\times 10^{14} \text{ cm}^{-3}$)	7.4, 8.1	4.4, 6.6
Particle, Energy Confinement Times (sec)	0.58, 0.32	11.6, 4.3
Retained fp Energy (%)	79	49
Fractional Ash Buildup (%)	9	18
Gross Power (MW)	29	2.3
Radiation (%)	1	29
Leakage (%)	16	64
Conduction (%)	-	4
Neutrons (%)	79	3
Energy Multiplication, Q	57	16
Fusion, Net Electric Power (MW)	29, 12	2.2, 1.1
Overall Efficiency* (%)	43	48

*Assumes thermal, direct, injection, and ICRH efficiencies of 40, 60, 80, and 80% respectively.

[†]Assumes anomalous transport losses.

Table 2

Reference Case Comparison of Principle Plasma Parameters for Two
Elongated SAFFIRE Designs with D-³He and Cat-D Fuel Cycles

	<u>D-³He</u>	<u>Cat-D</u>
Stability Factor, S	10	10
Vacuum Field (kG)	60	60
Elongation Factor,	3	3
Plasma Radius, Volume (cm, ℓ)	21.5, 125	22.8, 150
Ion, Electron Temp. (keV)	80, 75	60, 57
Ion, Electron Density ($\times 10^{14} \text{cm}^{-3}$)	4.4, 6.6	7.2, 8.1
Particle, Energy Confinement Times (sec)	11.6, 4.3	12.6, 5.5
Retained fp Energy (%)	49	67
Fractional Ash/Buildup (%)	18	14
Gross Power (MW)	2.3	2.9
Radiation (%)	29	25
Leakage (%)	64	36
Conduction (%)	4	2
Neutrons (%)	3	37
Energy Multiplication, Q	16	37
Fusion, Net Electric Power (MW)	2.2, 1.1	2.9, 1.3
Overall Efficiency* (%)	48	45

*Assumes thermal, direct, injection, and ICRH efficiencies of 40, 60, 80, and 80% respectively.

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Figure 2. Energy Multiplication Factor and Net Power Output vs the Plasma Ion Temperature for Cat-D Systems With "Near-classical" Loss Rates.

Table 3

Reference Case Comparison of Principle Plasma Parameters for
Three Cat-D SAFFIRE Designs With Different Stability Factors

	<u>Case A</u>	<u>Case B</u>	<u>Case C</u>
Stability Factor, S	5	10	15
Vacuum Field (kG)	60	60	60
Elongation Factor*,	1	1	1
Plasma Radius, Volume (cm, ℓ)	15.2, 15	23.8, 56	27.7, 89
Ion, Electron Temp. (keV)	100, 75	65, 62	40, 39
Ion, Electron Density ($\times 10^{14} \text{cm}^{-3}$)	5.9, 6.4	8.0, 9.1	13.1, 15.0
Particle, Energy Confinement Times (sec)	6.0, 2.6	10.3, 4.6	6.9, 3.5
Retained fp Energy (%)	37	69	80
Fractional Ash Buildup (%)	7	14	10
Gross Power (MW)	0.50	1.5	3.5
Radiation (%)	14	24	32
Leakage (%)	53	36	29
Conduction (%)	3	2	1
Neutrons (%)	30	38	38
Energy Multiplication, Q	3.4	59	119
Fusion, Net Electric Power (MW)	0.39, 0.11	1.5, 0.67	3.4, 1.6
Overall Efficiency† (%)	29	46	45

*Larger powers may be obtained by using 1.

†Assumes thermal, direct, injection, and ICRH efficiencies of 40, 60, 80, and 80% respectively.

Figure 3. Effect of Plasma Size on the Photon Energy Distribution in the FRM. Since the ion temperature is 120 keV for both cases, this corresponds to varying the stability factor.

II.2 FRM Start-up and Buildup Code

To model the start-up of an FRM a one-dimensional radial code has been developed. This code allows for the inclusion of radial profiles for densities, currents, temperatures and fields. Also important to this model are the spatial effects of neutral-beam ionization and the ability to compute plasma quantities in a self-consistent manner. Future possibilities include the modelling of a radial electric field which could drive currents in the background plasma that oppose the ring-ion current and also the modelling of radial transport of background particles which could enhance the radial electric field.

The one-dimensional code employs a hybrid model. Low temperature background ions and electrons are modelled using two-fluid equations and the higher energy ions are treated by a kinetic model. Each of these models has been developed separately and together will form a complete description of FRM start-up.

An initial version of the background fluid code has become operational.⁽⁴⁾ Since the code is not yet merged with the ion ring kinetic model, some functional dependence of the ring current on r and t must be assumed. Refueling sources are assumed to be directly proportional to the ion density. To model the field-null region, the magnetic field there is assumed to be linear in r . The pressure balance equation $\nabla [p + \frac{B^2}{2\mu_0}] = -j_{ring}B$ can then be integrated to give a condition on the total plasma pressure over the field null region.

Results from a test run of the background plasma code are shown in Figures 4 and 5. The magnetic field and the background densities and temperatures are plotted as a function of r for times 0, 2 and 5 msec. Values at certain mesh points in these figures are indicated by numbered crosses. We started with an already reversed background plasma

Figure 4 Profiles for the Magnetic Field and the Assumed Ring Current at 0.0, 0.2, and 0.5 msec.

Figure 5 Profiles for the Background Electron Temperature, Ion Temperature and Ion Density at 0.0, 0.2 and 0.5 msec.

($B(r=0)/B_0 = -0.75$). Initially the ion and electron temperatures are equal in the closed field region, varying from 5 keV at the separatrix (92 cm) to 10 keV near the field null. The ion temperature in the open-field region is uniform at 2.5 keV except for a sharp drop to zero near the vacuum region (from 97 cm to wall). The open field electron temperature is fixed at the stream stabilization temperature of 100 eV. Ion and electron densities are initially uniform at 10^{14} cm, except for a sharp drop, mentioned above, at the edge of the plasma. The densities, temperatures and magnetic fields are all discontinuous at the separatrix but the plasma pressure plus the magnetic pressure is continuous there. The refueling rate is equal to the ion density divided by 1 msec. The refueling energy is 10.0 keV for ions while the associated electrons are cold. The ring current shape is assumed to be parabolic in r , lying between zero and 50 cm. The amplitude of the current rises linearly with time to a maximum of -8.7×10^2 A/cm² at 1 msec. As the ring current increases, the centerline magnetic field becomes further reversed. However, the plasma diamagnetic current partially opposes the increasing ring current. If the change in the magnetic field were determined by the ring current alone, $B(r=0)$ would be -4.55 kG instead of only -2.8 kG. Since the greatest fractional increase in B occurs near the centerline, the plasma there and the plasma just inside the separatrix on the same flux surface are compressed more than elsewhere. This accounts for the positive gradient in density as one moves outward from the field null region to the centerline or separatrix.

Because there are no diffusive losses in the closed-field region, refueling causes a density buildup. The refueling energy (10 keV) for ions is close to the ion temperature (5 to 10 keV) so there is no corresponding change in ion temperature. However, since the refueling electrons are cold, the electron temperature falls.

The pressure balance constraint remains well satisfied throughout the run which is a verification of the accuracy of the code. At $t = 0.5$ msec the total pressure is uniform to within 1.3% from the wall to the outer edge of the null region. In the inner part of the closed-field region, the gradient of the total pressure equals $-j_{\text{ring}}B$ to within 1.5%.

The fluid code converges well with decreasing timestep and mesh spacing. Additionally, two tests have been performed to establish the numerical accuracy of the fluid code. In the first test, the refueling beam intensity was doubled from its steady state value. No ring particles were present, so the total plasma pressure should have remained uniform throughout the plasma. Indeed, in the new plasma equilibrium (≈ 1.5 msec later) the total plasma pressure was uniform to within 1.3%.

The second test verifies the accuracy of the code in the presence of an (ad hoc) ion ring. A sharply peaked ring current, which grew with time, was used. As the flux surfaces passed through the ring, the adiabatic pressure law,

$$\left(\frac{p_{\text{final}}}{p_{\text{initial}}} = \frac{B_{\text{final}}}{B_{\text{initial}}} \right)^{\gamma},$$

was obeyed to within .5%. Figure 6 is a graph for this ring current and the total plasma pressure after 1.5 msec. The quasi-static pressure balance, $\frac{\partial}{\partial r} \left(\frac{B^2}{2\mu_0} + p_e + p_i \right) = -j_{\text{ring}}B$, was obeyed in the ring.

A new version of the high energy ion code is also operational. The aberrations in the total high energy ion current, as shown in Fig. 11 have been nearly eliminated by 1) a more accurate algorithm for updating the magnetic field, and 2) a numerical averaging technique which smoothes the sharp peaks at the edge of the current and density profiles characteristic to each $\varepsilon - P_{\theta}$ bin. Tests for the convergence of this code with decreasing timestep and mesh spacing are underway.

Figure 6. Results of pressure balance test. The total plasma pressure was constant to within 2%.

As mentioned previously in Section II.1, the FRM is envisioned to be a small device and during the initial phase of start-up, ions are injected to form a ring current on the order of an ion gyroradius. Thus the ratio of the ion gyroradius to the radius of the device is relatively large, making a fluid description of the ring ions invalid. Instead, a kinetic treatment of the ring ions is used based upon the characteristic motions of classes of particles.

Because of the assumed axisymmetry, there are two constants, or near constants, of the motion which completely characterize particle orbits. These two constants are the canonical angular momentum in the azimuthal direction, P_θ , and the total energy of the particle, E . In the model, a particle deposition routine computes a volumetric source of neutral-beam injected ions and converts this into an appropriate distribution in E - P_θ phase space. During a timestep, each location in E - P_θ space is then reconstructed into radial density and current profiles based upon particle orbits. Two assumptions are involved in this process, the first being that the total energy of the particles is taken to be constant during a timestep in order to solve the equations of motion. The second is that the number of ions located in a radial annular shell is directly proportional to the time to cross the shell. This second assumption is analogous to the "superparticle" approximation in SUPERLAYER⁽⁵⁾ with the difference being that this model makes an additional assumption about the radial distribution of particles rather than following the individual motions of a large number of particles. Between timesteps, the new magnetic field and possible radial electric field profiles are computed and the ring ions are slowed down in accordance with the energy lost due to the induced azimuthal electric field and also due to binary Coulombic collisions with the background plasma. The induced electric fields leave the canonical angular

momentum constant and it is also assumed that the collisions do the same because these are mainly dominated by small-angle scatterings. Thus, ring ions are allowed to slow down to thermal energies. The combined energy loss rate due to both of these processes determines the width of a timestep in order to satisfy the assumption of the constancy of the energy during a timestep.

Some results using only the dynamics of the ring-ions with an exception of particle balances are shown in Figures 7-12. The initial conditions consist of an 80-keV, 30-A neutral beam injected 10-cm off-axis through a 10-cm beamport into a 30-cm plasma. The background plasma profiles are originally cubic with an average ion and electron density of $2.5 \times 10^{13} \text{ cm}^{-3}$. The ion temperature is 5 keV and the electron temperature is 100 eV.

Reversal is achieved and maintained after .33 msec. The total ring current at this time is -6.16 kA. Ring ions make up 16.5% of plasma while background ions and electrons make up 32.2% and 51.3%, respectively. This buildup of the electron density is due to prompt losses of ring-ions during the ionization processes. Approximately 7.5% of the ions that are deposited in the plasma have orbits that intersect the wall. This also shows that a plasma potential should be an important concern. Still, here the potential has been arbitrarily constrained to be zero because the lack of background dynamics leads to unrealistic potential profiles.

The simulation shown here terminated at .444 msec when the magnetic field was 20% past the reversal. The total ring current was -7.47 kA and the total number of ring ions was 2.63×10^{16} forming 19.1% of the plasma. These results are quite optimistic but are expected to change dramatically when coupled with the background dynamics.

DENSITY PROFILES

Figure 7(a)

CURRENT PROFILE

Figure 7(b)

MAGNETIC FLUX

Figure 7(c)

Figure 7. Density, current and magnetic flux profiles for the ring-ion model. Figure 7(a) shows both the electron profile (dashed bars) and the background ion profile (solid bars) as well as the ring-ion profile (shaded bars). The magnetic field was 50% reversed.

DENSITY PROFILES

Figure 8(a)

CURRENT PROFILE

Figure 8(b)

MAGNETIC FLUX

Figure 8(c)

Figure 8. Density, current and magnetic flux profiles just after incipient reversal. Total current at this time was -6.3 kA with a reversal factor of -0.035.

DENSITY PROFILES

Figure 9(a)

CURRENT PROFILE

Figure 9(b)

MAGNETIC FLUX

Figure 9(c)

Figure 9. Density, current and magnetic flux profiles at the end of the simulation. Total current was -7.47 kA and the reversal factor was -0.215.

REVERSAL FACTOR

Figure 10. The time history of the reversal factor. T scale measures the timestep in terms of 100 vacuum gyroperiods ($= 3.48 \mu\text{s}$). Conave measures the validity of the assumption of constant energy during a timestep.

RING PARTICLE AND CURRENT EVOLUTION

Figure 11. Time evolution of the ring current and total number of ring particles.

PLASMA COMPOSITION

Figure 12. Fractional plasma composition plotted against time showing effects of charge exchange processes.

As mentioned previously the fluid and ring models presently exist as separate codes, are both operational and are soon to be merged. This merger is scheduled to occur before the end of the present contract period. Coupling between the fluid background and the high energy ring will be provided by a quasi-static pressure balance which treats both fluid and ring contribution to the plasma current:

$$\frac{\partial}{\partial \psi} \left(\frac{B^2}{2\mu_0} \right) + P_{\text{fluid}} = - j_{\text{ring}}/r$$

A flow diagram of the merged codes is shown in Figure 13. Field null current models will also be investigated before the start of the next contract period. Subsequent work will concentrate on the inclusion of radial electric field and transport effects into the background plasma model. With a radial electric field, the restriction that the ring-ion density be much smaller than the background plasma density can be lifted. With the inclusion of transport effects, the applicability of the code can be extended beyond start-up time scales.

II.3. Heat Conduction Effects in a Field-Reversed Plasma

Heat conduction is one physical effect which has not received adequate consideration for high- B , compact tori confinement schemes. The high temperature and the small plasma size result in very large temperature gradients and large surface-to-volume ratios. Thus, since heat conduction is a surface loss mechanism, this compact tori will exhibit larger relative conduction effects than other magnetic fusion concepts.

Examining Fourier's law,

$$\Delta T = - \frac{q_{\perp}''}{k_{\perp}} \equiv \frac{\text{perpendicular heat flux}}{\text{perpendicular conductivity}},$$

one sees that to attain large temperatures, either a large heat flux, small conductivity, or some combination of the two is required. For low fusion

Figure 13. Proposed flowchart for the merger of the background fluid code with the high energy ion model.

power density and large radiation losses, little energy is left for conduction and $q_{\perp}''$ is small. So, very small conductivities are needed to permit large temperatures.

The classical thermal conductivity, $k_{\perp} \propto \frac{n^2}{B^2 \sqrt{T}}$, is not necessarily small enough to support large temperature gradients. For example, the University of Illinois' D-³He SAFFIRE (FRM) design with $B \sim 60$ KG, $n \sim 5 \times 10^{14} \text{ cm}^{-3}$, and $T \sim 80 \text{ keV}$ will not support large temperature gradients near the center of the plasma. However, near the edge where the density drops significantly, large ΔT 's are permitted. Thus, for field reversed plasmas of comparable or larger $k_{\perp}$, the ability to support high temperatures can be considered as a function of the plasma edge conditions (temperature and density at the separatrix), which are partially determined by open field plasma and divertor characteristics. Thus, a critical conductivity, k_c exists for a given plasma (defined by Fourier's law) which allows high plasma temperatures. For a fusion plasma to exist, the actual edge conductivity must be less than or equal to the critical value.

The above example assumed classical thermal conduction. No experimental results have suggested classical behavior, so it cannot be assumed, especially for the small size (finite Larmor effects) and the large magnetic field curvature of compact tori. For example, tokamak experiments show the transport coefficients of 10^2 - 10^3 times classical values for compact tori, then it will be the dominant energy loss mechanism and it may even prohibit small fusion plasmas.

A steady state, MHD, field-reversed, radial transport model was developed to describe a fusioning plasma. The model consists of electron and ion power balances, particle continuity, and a static radial momentum balance. The resultant set of three second order, highly nonlinear,

O.D.E.'s has been implemented into a finite difference computer code which will soon be operational. The model uses a Braginskii(6) formulation including the thermal force term and other thermoelectric effects. The model describes radial transport in both the open and closed fields and also includes, phenomenologically, the effect of axial transport along open field lines. (See Figures 14 and 15) It also accounts for the coupling of ψ surfaces (surfaces of constant pressure) in the close field region. A description of the output and uses for this code follows.

Code Results

- Output:
1. Ion and electron temperatures, pressure and magnetic field profiles.
 2. Relative magnitudes of the different source and loss mechanisms.
- Uses:
1. Calculate reactor parameters and check zero-dimensional code and results.
 - 2a. Study the effects of regions with large temperature and density gradients on transport.
 - 2b. Study how the plasma size affects the plasma temperature. (Small plasmas have large gradients and thus large conduction losses).
 3. Study the possibility of using some type of particle or energy end stoppering technique on open field lines to affect the closed field plasma edge conditions and improve reactor performance.
 4. Study the effect of different transport scaling laws on reactor parameters.

This code package can be used for exploring the effects of transport processes, refueling profiles and open field plasma properties on the required plasma size, in both conventional and advanced-fuel, FRM and FRØP reactors.

GEOMETRY

Figure 14. Geometrical Illustration of Radial and Axial Transport.

MODEL

Six Equations : Ion continuity, Ion + Electron Energy Conservation and Heat Flux, Diffusion Velocity (Ohm's Law)
 Six Unknowns : Pressure, Ion and Electron Temperature and Heat Flux, Diffusion Velocity

EFFECTS INCLUDED:

METHOD OF SOLUTION:

1. Guess T_i, T_e at $r = R$.
2. Use Runge Kutta and solve radially outward from $r = R$.
3. Use Newton-Raphson iteration to make a better guess of T_i, T_e at R . Repeat till solution converges.

II.4. Spheromak Transport and Reactor Viability Studies

Global decay and one-dimensional profile evolution plasma models have been developed in our studies of Spheromak transport and reactor viability. The primary objectives of the global-decay code were to provide a general reactor viability study in support of the PPPL S-1 Spheromak proposal⁽⁷⁾ and to provide a foundation for a crude conceptual design of a linear Spheromak fusion reactor⁽⁸⁾ in which Spheromak plasmoids are continuously created and moved down a linear drift tube (See Fig. 16).

The global decay model⁽⁹⁾ is based upon an empirical fitting of two-dimensional study (using a modification of the Princeton Equilibrium Code⁽¹⁰⁾) of Spheromak MHD equilibrium and stability⁽¹¹⁾ and assumes a turbulent decay at marginal beta of a Spheromak plasma as a function of its geometry, flux-surface averaged radial plasma profiles, and external field strength.

Some typical results from this time-dependent, global-decay code are illustrated in Figure 17, in which the dependence of grid power (P_{NET}) on the maximum poloidal field strength at the plasma edge (B_e), and the plasma minor radius (a) are shown. According to this model, it appears that $P_{NET} \sim 500$ MW is readily obtained with a 0.5 to 0.75-m plasma and a field strength in the 4-7 Tesla range. However, we have also plotted limiting thermal wall loading curves at 2 MW/m^2 and 4 MW/m^2 for $n = 1$ and 2, where the wall radius r_w is defined by

$$r_w = n(R + a) .$$

In practice, the actual location of the first wall may be determined by plasma stability, for, if values of $n > 2$ are used, the wall surface area is increased to the point that wall loading is no longer a major problem. These calculations assume an overall plant efficiency of 35%, plasmoid

Figure 16. Basic Layout of the Linear Spheromak Fusion Reactor.

Figure 17. Grid electric power as a function of B_e for various sized plasmas and thermal wall loadings.

creation efficiency of 50%, and a 17 MW.yr.m^{-2} neutron damage first wall lifetime. Further results and details of these global-decay calculations are given in References 7 and 12.

In more recent work, the study has been extended to include spacial effects in order to investigate Spheromak plasma profile evolution, and more specifically, the possibility that local violation of a Mercier-type criterion in the central, low-shear region of the Spheromak could lead to a grossly-stable, turbulently-decaying Spheromak. For preliminary studies, a Spheromak modification of the one-dimensional RFPBRN code⁽¹³⁾ has been developed. The study is in progress to include the important 2-D Spheromak effects through the use of a 1 1/2-D code which is being developed at Illinois.

II.5. Rotational Model for Field-Reversed Theta Pinch (FR θ P)

The field reversed theta pinch (FR θ P) has been considered to be a potential candidate for a compact tori device.⁽¹⁴⁻¹⁶⁾ However, due to the $m = 2$ rotational instability, the plasma lifetime has been limited to be about 50 μs .⁽¹⁴⁾ The mechanism of plasma (ions) azimuthal acceleration is still not well understood. Several models have been proposed to explain this mechanism. However, none has been satisfactory enough according to the experimental data.⁽¹⁴⁾ A understanding of the mechanism is essential to control the $m = 2$ rotational instability and to extend the plasma lifetime.

A new model has been proposed and has provided good qualitative agreement with the FRX-A experimental data. This model assumes a two-fluid system (ion and electron) and a quasi-rigid-rotor equilibrium; that is, a plasma is in rigid-rotor equilibrium and a radial particle leakage is allowed at the same time. The model involves conservation of canonical angular momentum, conservation of number of particles and energy.

Three major sources accounting for the azimuthal acceleration have been identified: (a) the $\vec{J}_{\perp\alpha} \times \vec{B}$ bulk force, where $\vec{J}_{\perp\alpha}$ is the radial particle leakage current of ions ($\alpha = i$) or electrons ($\alpha = e$), (b) a decrease in the moment of inertia, which is due to change in the equilibrium profiles, and (c) the temperature decay, which is due to energy losses. The former two are easy to understand. The latter results in a decay in the total magnetic flux; consequently, according to the conservation of canonical angular momentum, the rotational speed $|\omega|$ has to increase. In summary, the plasma (or more specifically the ion) spin-up is due to both the particle and energy losses.

This model requires a knowledge of the particle and energy transport scalings. However, these scalings are not well known. What is currently being assumed in the computation for plasma spin-up is either a classical diffusion or the one corresponding to the lower-hybrid-drift (LHD) instability. Since both of these scalings give confinement time much larger than that from experimental measurement, a factor of 15-40 is multiplied to roughly fit the experimental values.

Typical results are shown in Figs. 18-20. Figure 18 illustrates the normalized ion rotation frequencies versus time. Curve 1 corresponds to the classical scaling multiplied by a factor of 14; curve 2 corresponds to the LHD scaling multiplied by a factor of 40 while the dashed curve corresponds to the experimental measurement from the FRX-A. Those numerical factors are chosen such that they give an initial particle confinement time of about 50 μ s. The discrepancies come mainly from the uncertainty of transport scalings. A better agreement would require a rapidly increasing particle and energy confinement time. Figure 19 shows the time evolution of the plasma major radius R , and the reversal parameter κ using LHD scaling. The major radius and the reversal parameter κ tend to increase. Figure 20 shows the total energy and the total number of particles inside the closed field region and the peak density versus

Figure 18. Normalized ion rotation frequencies versus time comparison between results of this model (curve 1 and 2) and FRX-A experimental data (curve 3).

Figure 19. Time evolution of the reversal parameter κ and the major radius R for the case of using LHD scaling.

FSL-80-202

Figure 20. Total energy E_T and total number of particles inside closed field region and the peak density n_m versus time for the case of using LHD scaling.

time for the case of using LHD scaling; these are qualitatively consistent with the experiment.

A refinement of this model lies in using a more exact and consistent transport model; this is not well understood at present. On the other hand, the understanding of the transport scaling is essential in studying scale-up experiment or reactor design. The loss-cone-like scattering, i.e., the particle loss due to the collision-induced diffusion in E - P_θ phase space, is now being considered as possibly a major mechanism for the observed particle confinement times, which are smaller than classical transport prediction by a factor of ≈ 15 . Based on the conditions for ion-confinement, that are $P_\theta < 0$ and $E < -\frac{qB_0 P_\theta}{mc}$, a transport model in (r, E, P_θ) space is being developed in which binary-collision approximation, for simplicity, is used for modelling particle collisions along with multigroups in E and P_θ .

II.6 Plasma Systems Evaluation of Moving Plasmoid Heater (MPH)

Investigation of a finite Larmour radius (FLR) compact tori plasma as a candidate for a Moving Plasmoid Heater Reactor (MPHR)⁽¹⁷⁾ plasma has continued. The scheme examined involves heating compact tori plasmas with neutral beams as they traverse from their formation chamber down the device to the burn chamber and ultimately a direct convertor⁽¹⁸⁾. Classical transport scaling has been assumed.

For a 100 MW_e D-T system about 5-6 plasmoids ($R = 20$ cm, $L = 160$ cm, $n \sim 2 \times 10^{15} \text{ cm}^{-3}$, $T_i \sim T_e = 20$ keV) would be needed. The recirculating energy fraction is $\sim 30\%$, however, the wall maximum loading from these small plasmoids ($R_{\text{wall}}/R_{\text{plasma}} = 2$) is $\sim 10 \text{ MW/m}^2$.

Remaining issues to be resolved are plasma stability and transport scaling. If transport scaling is much faster than classical, it is unlikely that net power production is possible.

III. Dynamic Behavior of Reversed-Field Pinch (RFP)

III.1. One-dimensional Reversed-Field Pinch (RFP) Studies

During the past year RFPBRN⁽¹⁹⁾, a one-dimensional Lagrangian mesh transport and stability code, was made fully operational and used to simulate start-up and burn of a reversed field pinch reactor (RFPR).

The goal of these studies is to consider past designs based on global models^(20,21) with a more physical 1-D model and to consider the coupled profile dependent problems of transport, stability, start-up and refueling. Initial studies compared RFPBRN with the global code developed at LASL.^(22,23) These studies also included Suydam induced anomalous transport and an optimization of RFPR, the LASL reference reactor design. The simulations generally behave in a manner similar to the global model with some improvement due to profile effects. Optimized RFPR parameters are shown in Table 4.

A study of a small compact RFP reactor (CRFP)⁽²⁴⁾ was also done. The impetus of this work was to test the feasibility of low capital cost near term device. Although performance was not as good as for the larger machine, it was acceptable as shown in Table 4.

Previous work⁽²⁵⁾ has indicated that burn times can be extended by refueling resulting an increase in the plasma Q. Results with RFPBRN show the same trends. Refueling the CRFP is particularly successful. This is due to its small size and this allows for rapid penetration of the fuel. By, thus, extending the burn, the Q is doubled (See Table 4). Also we find that the wall loading is less peaked, as shown in Fig. 21. These studies were done for both gas puffing and solid pellet refueling.

III.2 Reversed-Field Pinch (RFP) Start-up

More recent work has centered about start-up simulations. Using a convective start-up technique⁽²⁶⁾, we have found a scenario which

Table 4

SUMMARY DESCRIPTION OF DESIGN PARAMETERS

	<u>RFPR</u>	<u>Batch Burn CRFP</u>	<u>Gas Refueled CRFP</u>	<u>Pellet Refueled CRFP</u>
First wall radius, $r_w(m)$	1.5	.75	.75	.75
Major radius, $R(m)$	12.7	6.35	6.35	6.35
Burn time, $t_B(s)$	21.6	10.3	21.0	18.1
Cycle time, $t_C(s)$	26.6	15.3	26.0	23.1
Toroidal field on centerline $B_\phi(T)$	6.93	8.5	8.4	8.4
Toroidal current, $I_\phi(MA)$	20	12.0	13.2	13.2
$I/N, (A-m)$	$1.8(10^{-14})$	$2.0(10^{-14})$	$2.8(10^{-14})$	$2.8(10^{-14})$
Average fuel burnup, f_B	0.5	.23	0.3	0.3
Initial average plasma density, $n(1/m^3)$	$2.0(10^{20})$	$3.8(10^{20})$	$2.7(10^{20})$	$2.7(10^{20})$
14.1-Mev neutron current at first wall, (MW/m^2)	2.7	2.4	2.7	2.7
Total thermal power, $P_{TH} (MW_t)$	3000	628	820	706
Net electrical power, $P_E (Mwe)$	750	100	240	160
$Q_p = (\text{fusion energy})/(\text{ohmic} + \text{field energy})$	14.4	9.68	17.2	13.2

Batch Burn
CRFP

Gas Refueled
CRFP

Pellet Refueled
CRFP

Figure 2b. Wall loading for CRFP with refueling.

theoretically maintains ideal MHD stability to global and local modes during the start-up, heating and burn phases of the RFP discharge.^(27,28) The field programming necessary to do this is shown in Figure 22. Results indicate that in order to maintain stability it is necessary to peak the toroidal current off-axis initially and to peak the density off axis during the burn phase. This is the first time an ideal MHD start-up and burn scenario has been shown.

Although the 1-D simulations have been shown that ideal MHD stable start-up and burn is feasible in principle, it may be difficult to apply in practice. In order to maintain stability the one-dimensional scenario requires the plasma to extend to a conducting wall in the pre-reversed state. If there is a vacuum or a current-free region between the plasma and the wall, the plasma is unstable due to a pitch minimum. Instabilities can also occur if the toroidal current diffuses too rapidly into the central plasma region. Thus, one-dimensional stable start-up requires keeping the plasma against the wall and starting up the reactor faster than current can diffuse into the center.

Work has commenced on an effort to extend the 1-D start-up work done on RFPBRN⁽²⁴⁾ to include 2-D effects. The concept which is currently being explored is to use toroidal coils to provide poloidal barrier fields. This is analogous to moving the off-axis toroidal plasma current into the external conductor which should stabilize the current driven modes as the toroidal field is reversed. The off-axis peaking of toroidal current needed in the 1-D start-up simulations, is an encouraging indication that this technique should be successful.

A 2-D equilibrium solver (P.E.S.T.) has been used to explore the equilibrium state associated with the initial phases of RFP start-up. These have shown that equilibrium solutions exist that are similar to those we expect to see.

Figure 22. Magnetic field and gas feed rates at the wall necessary to maintain MHD stability during startup of an RFP.

The two dimensional nature of this work is clear. Development of a "1 1/2-D" code has begun. In this scheme the pressure balance (Grad-Shafranov) is solved in 2-D. Transport coefficients and various geometry coefficients are then averaged over poloidal flux surfaces. These are then used in a 1-D transport sub code. The new pressure and toroidal B fields are then used to find a new equilibrium pressure balance. This iteration is continued throughout the evolution computations. This scheme provides a method of solving 2-D quasi equilibrium problems in about the same time as a 1-D problem. Our approach is to use RFPBRN for the transport stage and P.E.S.T. for the equilibrium section. Upon completion this code will be available for use in a variety of toroidal high-beta devices.

IV. Plasma Buildup in Small Mirror Devices

Plasma buildup, both in density and temperature during neutral beam injection, remains a key aspect of the development of future mirror experiments like the end-cell sections of TMX and the MFTF device. In a small mirror plasma ($R_p/p_i \leq 40$) ion finite gyro-radius (F.G.R.) effects become important, particularly at the surface near the vacuum interface. This is because charge-exchange of incoming cold neutrals (from the injector and recycling off the first wall) occurs primarily in a surface layer only a few centimeters thick, causing plasma erosion.⁽²⁹⁾

Previous attempts at treating F.G.R. effects in mirrors have been restricted to one or two ion energies and a coarsely-averaged beam deposition.^(29,30) A new 2 1/2-D (1-energy, 1 1/2 spatial) dynamic model has been developed (called FOREMD for Finite Orbital Radial, Energy and Magnetic-field Dependent) which extends the F.G.R. treatment to a distribution of energies and also includes local magnetic field effects and more accurately calculated beam deposition. This model generates not only a more accurate density profile but also the F.G.R. radial temperature profile, which earlier models could not.

Details of the model and some preliminary results for buildup in a 2XII-B-like plasma have already been published^(31,32) and are also presented in our previous progress reports.

The FOREMD mirror buildup model has been recently modified to include DCLC velocity-space diffusion. This modification allows the treatment of stream-stabilized plasmas such as 2XII-B, MFTF and the TMX end cells. Simulations done for 2XII-B appear to be in good agreement with global simulations done by Berk⁽³³⁾ earlier. Results are presently being obtained for TMX end cells and MFTF. Runs already done for an MFTF type plasma indicate that part of the injected neutral current will have to be directed through the low density edge region to allow a uniform density buildup in the interior. This conclusion agrees with that of DeVoto at LLL⁽³⁰⁾.

Case studies have also been done for the Twin-Beam Mirror (TBM)⁽³⁴⁾ which show the development of an off-center density peak but a fairly uniform pressure profile. This effect increases the neutron flux output over a conventional (uniform) plasma for the same injection power. Additionally, results of a study done in 1979 on the effect of beam injection angle in this device have recently appeared in Nuclear Fusion.⁽³⁵⁾ We are now preparing a study of plasma buildup in the TMX end cells for publication.

V. Fusion Products Effects to Tokamaks

V.1. Transport Model for Non-Circular and High- β Tokamaks

Modeling of fast alpha slowing in non-circular tokamaks was completed. Numerical convergence problems in the 3-D phase-space integrals were resolved. Details of the model, numerical methods, the computer code, and the results are fully documented in Lee Hively's thesis⁽³⁶⁾. The results of this work are summarized as follows:

Fusion product (fp) thermalization and heating are calculated assuming classical slowing down. The present analytical model describes fast ion orbits and their distribution function in realistic, high- β , non-circular tokamak equilibria. First-orbit losses, trapping effects, and slowing-down drifts are also treated. By solving a 3-D (+ time) PDE, it is possible to obtain an invariant of the slowing down process: $\mu/E = (\text{magnetic moment})/\text{energy} = \text{constant}$, and explicit expressions for the slowing-down drifts. Large banana-width effects give rise to a net co-going alpha particle current. The large banana-width orbits smear the energy deposition over large regions of the plasma. This causes the flux-surface-averaged ion heating rates to be 10-20% below in-situ rates on axis, but enhances the edge heating ≥ 10 -fold over in-situ deposition. While this result implies reduced alpha "ash" accumulation on axis, the reduced heating rate makes start-up and maintenance of ignition more difficult.

V.2. Alpha Ash Buildup in Tokamaks

Work has continued this contract period on the effects of radial electric fields on transport in tokamaks with finite cross-field conductivity. Conditions necessary for control of alpha ash buildup have been investigated. It was found that, although radial electric fields large enough to give essentially infinite fuel confinement times are not likely to be practical, much smaller fields can significantly improve the performance of an ash-fraction-limited tokamak.

The cross-field conductivity used comes from Boozer(37):

$$\sigma = (\rho c^2 / B_p) \nu_N ,$$

where ρ is the mass density, B_p the poloidal magnetic field, and ν_N the phenomenological toroidal damping frequency. The density and damping must be evaluated for each species to give the cross-field conductivity applicable to that species. This is important, as different damping

frequency mechanisms will dominate different species in a tokamak. In particular, for a tokamak with edge neutral density, charge-exchange will dominate the conductivity for $Z = 1$ fuel, while charged-particle neutral elastic scattering will dominate for $Z > 1$ impurities and electrons. The effect of this ordering is to give a larger conductivity to fuel ions than to alpha ash or electrons in the plasma.

Applying this to a TFTR-sized plasma operating in the neoclassical regime, and assuming fixed (in time) profiles for the neutral density ($10^{20}(r/a)\text{cm}^{-3}$) and the temperature ($8(1-(r/a)^2)\text{keV}$), the effects on ash-buildup-limited burns of a known, constant (in time) radial electric field have been studied. The electric field is assumed to have a maximum strength on the order of that anticipated for such a physical setup (according to Petrie⁽³⁸⁾), and the shape, direction, and magnitude of the field were varied.

The results of this study⁽³⁹⁻⁴¹⁾ may be summarized by Figure 23. The energy output of an ash-fraction-limited burn (starting from a fuel density profile of $5 \times 10^{13}(1-(r/a)^2)\text{cm}^{-3}$) is shown as a function of the ash fraction allowed. Energy output is normalized to the zero-field case. The outward-directed (positive) electric fields result in decreased energy output and hence are not as interesting as the inward-directed (negative) fields where enhancement is obtained. The shape of the time isograms for these fields shows that as electric field strength increases from zero, the ash fraction at a given time increases, reflecting additional fuel loss, driven by the field. At longer times, this trend reverses at higher electric fields, as alpha production drops with lower fuel densities, giving a lower ash fraction and lower power output.

Inward-directed electric fields, on the other hand, enhance energy output for ash-fraction-limited pulses, since fuel losses are slowed by the inward flow caused by the field. Alpha ash is less affected, having a lower conductivity. Thus the fusion rate is higher (so the time isogram slopes

Figure 23. Energy output enhancement (energy output/energy output with no radial electric field) for ash-fraction-limited burn. Electric field $E_r(r) = E_{ra}(r/a)^3$. Inward-directed electric fields are negative. Values are for a TFTR-sized plasma with no refuelling during burn.

downward in this region as the field magnitude increases), and the energy output per pulse increases with the higher fusion rate and longer time required to buildup to the limiting ash fraction. Indeed, imposition of a $-5\text{kV/m } (r/a)^3$ electric field enhances energy output by 10% for a burn limited to only 1% ash, while a burn limited to 10% is enhanced by over 30% (see Fig. 23).

The studies reported here have assumed electric fields fixed in time. In practice these fields will have to be continuously produced, as the resulting current flow seeks to decrease the field. Prompt loss of fusion-product alphas in a fusing tokamak will create an inward-directed electric field. Other mechanisms, including charged-particle injection, fusion-product-loss enhancement, and unbalanced (toroidally) neutral-beam injection, are currently under investigation, with the development of a code to follow the evolution of a self-consistent electric field and the resulting impact on alpha ash buildup.

VI. Description of Computer Codes

A brief description of major codes completed during the current year is prepared. Potential users who need detailed information on the various code packages are suggested to contact the principal investigator.

VI.1. Fusion Product Transport Codes

1) SLWDN9 Code⁽³⁶⁾:

This code calculates flux-surfaced-averaged values of alpha density, current, and electron/ion heating profiles in realistic, non-circular tokamak plasmas. The input data file describes the tokamak equilibrium with various plasma average betas, major radii, and the elongation factors. The code is written in FORTRAN and executes on the CRAY-1 machine at the National Magnetic Fusion Energy Computer Center (NMFECC) located at Livermore, California. This code requires ~1200K actual words of memory.

2) TRAKEM Code⁽⁴²⁾:

The evolution of the particle orbit was studied by solving the equations for $\dot{\theta}$, $\dot{r}$, $\dot{\phi}$, $\dot{E}$, and $\dot{P}_{\phi}$ using this code, as the particle energy was reduced by slowing down in the background plasma. The reactor field, wall and plasma parameters, and the initial conditions with which the particle starts are read in as input data. The program is designed to stop if the energy of the particle falls below E_{final} during the slowing down or if the particle hits the reactor wall. This code is currently stored in IBM.

VI.2 Reversed Field Configurations

1) FRMOD Code⁽²⁾:

The zero-dimensional FRM plasma burn code, FRMOD, is written in the FORTRAN and calculates ion/electron temperatures, peak background ion/electron densities, particle/energy confinement times, neutron flux at the plasma surface, etc. for the study of fusion product effects in field-reversed mirrors. Typical input variables are magnetic field, plasma volume, elongation factor, and stability factor. The code is currently available on the A-machine at the NMFECC.

2) MCFRM Code⁽²⁾:

This is a Monte Carlo particle code written in also FORTRAN is available on either the A-machine or the C-machine at the NMFECC. The MCFRM calculates the average energy of the remaining test particles, the average fractional change in the particles' velocity due to drag and pitch angle scattering, average slowing down time for the confined fusion-products, confinement time, etc. based on the input data (e.g. birth distribution of fusion-products; energy, mass, and charge of the test particles, number of particle histories, and plasma parameters).

3) EROICA Code⁽⁴³⁾:

The EROICA is a modified version of SUPERLAYER-II⁽⁴⁴⁾ and was developed here at Illinois to calculate the perturbed energy for a given trial function to study the field-reversed mirror stability. The calculation of the orbit integrals required for stability studies as an additional numerical integration was added to the SUPERLAYER-II. Frequencies and growth rates are given as typical input data while the spectral energy and the autocorrelation energy are computed as typical outputs. This code is on file on magnetic tape storage (FILEM) on the NMFEC.

VI.3 Cyclotron Radiation Code⁽⁴⁵⁾

FOKN^(46,47) was modified to include cyclotron radiation by "externally" shifting the electron distribution function according to the radiation loss. The resulting electron distribution function is evolved by the Fokker-Planck operator at each time step. The iterative procedure solves for, from a given electron distribution function at time t and the time step Δt determined at previous cycle, the correct distribution function at time $t + \Delta t$, which satisfies both the Fokker-Planck equation and the cyclotron radiation loss. This iteration is repeated on every time step until the run is stopped in a way to conserve particles and energy. The code is available on the A-machine at the NMFEC.

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VIII. Staff Contributors and Acknowledgements

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IX. Reports, Theses, and Publications

- C00-2218-141 "Progress Report - 1 October 1978 - 30 September 1979 (185 pages)
- C00-2218-142 "User's Guide for FRMOD, a Zero Dimensional FRM Burn Code"
(D. Driemeyer and G. H. Miley, Internal Report, October 15, 1979).
- C00-2218-143 "User's Guide for MCFRM, a Monte Carlo Particle Code for FRM Studies"
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- C00-2218-145 "Transport and Stability Studies of the Reversed-Field Pinch"
(R. W. Moses, R. A. Nebel (LAS), George H. Miley, *APS*, 24, 1042 (1979))
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(George Robert Fenske, Ph.D. Thesis, Nuclear Engineering, Program, University of Illinois, 1979)
- C00-2218-149 "Overview of Nonelectrical Applications of Fusion"
(George H. Miley, *Proceedings, 2nd Miami International Conference on Alternative Energy Sources*, 10-13 December, 1979, Miami Beach, Florida)
- C00-2218-150 "A Study of Fusion Product Effects in Field-Reversed Mirrors"
(Daniel Edward Driemeyer, Ph.D. Thesis, Nuclear Engineering Program, University of Illinois, 1980)
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- C00-2218-154 "One-Dimensional Thermonuclear Burn Computations for the Reversed-Field Pinch Reactor (RFPR)"
(R. A. Nebel, G. H. Miley, and R. W. Moses, submitted for 1980 CNFD Student Award for Fusion Science and Engineering)
- C00-2218-155 "Fast Ion Thermalization Non-Circular Tokamaks"
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- C00-2218-156 "Fusion-Product Transport in Axisymmetric Tokamaks: Losses and Thermalization"
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- C00-2218-157 "User's Guide for ICSAVG/HUBIN, a Code Package for Calculation of Transport of Charge Exchange Neutrals Due to H₀ Injection Into Mirrors"
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X. Distribution ListINTERNAL

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