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ELECTRIC POWER RESEARCH INSTITUTE

PROCEEDINGS OF THE REVIEW MEETING ON ADVANCED-FUEL FUSION

EPRI ER-536-SR

Special Report

June 27-28, 1977
Chicago, Illinois

Prepared by

The Commonwealth Edison Company,
Electrical Institute of Chicago, and
The Fusion Studies Laboratory of the
University of Illinois at Urbana-Champaign   3 / / 2000

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PREFACE

This proceedings is comprised of papers of both oral and poster sessions presented at the *Review Meeting on Advanced-Fuel Fusion* held at the Commonwealth Edison Building, Chicago, June 27-28, 1977. Some other contributed articles which could not be presented at the Meeting are also included. As the first extensive report devoted to advanced-fuel fusion, this proceedings will hopefully provide an important background for continued planning of the research sponsored by the Electric Power Research Institute (EPRI).

Advanced-fuel fusion can potentially provide a major contribution to our long-term energy resources and simultaneously offer the most favorable fusion conditions possible in terms of minimizing radioactivity and maximizing efficiency. The main difficulty with the advanced-fuels is that, compared to D-T fuel, even higher fusing temperatures and better plasma confinement are required. However, the advantages are so important that this challenge cannot help but excite the scientific community. In fact the attitude towards advanced-fuel fusion has changed remarkably in the past few years as a larger segment of the fusion community recognizes the importance of advanced-fuel fusion as an ultimate goal. Individuals still differ on the question of how to achieve this goal. Consequently, the meeting had three important objectives:

(1) To review the present status of advanced-fuel fusion,
(2) To discuss the role of advanced-fuel fusion in the overall development of fusion power, and
(3) With conclusions from (1) and (2) in mind, consider how and at what pace advanced-fuel fusion should be developed.

In order to achieve these objectives, the meeting was arranged with a mixture of presentations, panels, and workshops; participation in the panel and workshop discussions was designed to provide all attendees with an opportunity to input views and ideas. Over 60 participants with wide range of backgrounds attended from universities, national laboratories, private firms, and the utility companies. The reader of this proceedings can, I feel, sense the intensity and excitement that resulted.

George H. Miley, Conference Coordinator
University of Illinois
at Urbana-Champaign

ACKNOWLEDGMENTS

The continued interest and support from the EPRI were the essential ingredients in organizing a meeting of this kind. Commonwealth Edison Company and the Electrical Institute of Chicago kindly provided us their excellent facilities without which this meeting could have never been possible. All those who participated in the meeting and worked through late hours in the workshop discussions are truly the ones to be credited for the success of this endeavor. After all, "fusion" could occur only from a collection of plasma "particles" not from a single "particle."

Ms. Chris Stalker of the University of Illinois did most of the editorial typing that was necessary to incorporate the various contributions into a consistent format.

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EPRI ADVANCED FUSION REVIEW

R. L. Bolger

June 27, 1977

Good morning and welcome to Chicago. We at Commonwealth Edison are pleased to act as hosts for your meeting and hope you will find it an interesting and profitable experience.

I would like to take about a minute of your time to discuss my experiences with the EPRI fusion program. Most of you are probably aware that EPRI has established an industry advisory structure to provide a medium for interchange of information between the Institute's staff and the utilities. I believe that this approach has been effective in directing EPRI's efforts toward areas which are of interest to utilities and making utilities aware of the goals and progress of the EPRI program.

From 1973 until December, 1976, I served on EPRI's Advanced Systems Task Force whose scope included the EPRI fusion program. During this period, a great deal of effort was devoted to establishing a proper role for the EPRI fusion program. It was recognized from the beginning that funding limitations would restrict the EPRI program to a small part of the national effort. This consideration lead to guidelines suggesting that EPRI's funds be directed to areas where a relatively small EPRI involvement would significantly direct an activity and assure that it was responsive to utilities' concerns and to areas which the national program appeared to overlook. Advanced fuel fusion appears to fall into at least one and possibly both of these areas. Since December, 1976, I have been on the Fossil Fuel and Advanced Systems Division Committee. Although the broader scope of this assignment necessarily limits our involvement in fusion, it remains a major concern of the Committee.

Without overlapping the presentations by Clint Ashworth and Howard Drew, I would like to tell you how at least one utility views the present energy situation. I feel strongly that the expression, "energy crisis," is a misnomer whose use may have an adverse impact on our coming to grips with our energy problems. It implicates that the problem is sudden and short-term and that all we need is to mount a major effort and we will find a solution. The problem did not arise suddenly and neither it nor the solution are short-term. The same can be said of the suggestion that our attempts to resolve energy problems are a form of "war." A war requires a total national effort but only for a limited period of time. Our energy problems require an extensive effort for an indefinite period but cannot be allowed to completely overshadow other social and economic concerns. The energy problem arises from two factors, the first of which is a misallocation of resources. Oil and gas comprise only 20% of our energy reserves but provide 75% of our energy. The misallocation was compounded by artificially low prices for these fuels which provided no incentive to use them wisely. The second factor is that we have been unable or unwilling to face up to the necessary trade-off between energy and the environment. As a result, the problem has been worsened and the solution has been delayed.

Although fusion is clearly a long-term option, let me first address the short-term solutions. Increased use of nuclear power to meet our energy deficit requires acceptance of some environmental trade-offs. This is even more true of an increase in the use of coal. The reluctance to face up to these trade-offs has led to an overemphasis on conservation and the so-called exotic methods of energy production such as solar, wind, and geothermal sources. The potential benefits from conservation are real and must be attained but they are limited. Conservation can buy us time but cannot eliminate the need to make use of other energy resources. Geothermal appears to have a real, though limited, potential in the near term and the others are clearly long-term options. There is no real choice for the near term but to conserve energy to the greatest practicable extent and to utilize the existing coal and light water reactor technologies.

While present energy policy appears to recognize these necessities, the real situation is far different. The call for increased utilization of coal is accompanied by tighter air quality standards while the acknowledgement of the role of the light water reactor brings with it a demand that we forego fuel reprocessing and plutonium and uranium recycle and accept a stricter regulation. Energy suppliers faced with these contradictions, uncertainties in growth forecasts, and financial problems have cancelled and deferred new facilities. Last winter, serious shortages of natural gas were experienced in some areas. There is a growing belief that by the mid-1980's we will be seeing significant shortages of electricity in many areas. In response, we may have to construct a large number of combustion turbines and further increase our dependence on petroleum products.

In the longer term, we can begin to look at other technologies, and we will have to look at the limitations of coal and LWR's. Although our estimated coal reserves are large, the problems of mining and transportation and the environmental impact suggest that the role of coal may be more limited than its advocates now believe. Aside from particulates and oxides of sulfur and nitrogen, many are concerned about the impact of carbon dioxide which is the unavoidable result of the combusion of fossil fuels. With respect to light water reactors, the concern becomes one of the availability of fuel if we delay or forego the breeder. I do not propose to rehash these arguments here, but it is certainly reasonable to conclude that we need to continue our search for new energy technology including fusion.

This discussion may suggest that we regard the breeder reactor and fusion as rivals and feel compelled to line up on one side or the other. Although it is no secret that Commonwealth Edison is a strong advocate of an intensive breeder development program, we do not advocate one to the exclusion of the other. The need for and the uncertainties of new energy technology are so great that we need to develop all of our options. Even if fusion becomes commercially feasible on the earliest schedule envisioned by its advocates, the need to provide an economical and dependable fuel supply for the light water reactors, which we will have built, justifies an active breeder development program. Conversely, the problems and uncertainties of the fission reactor program necessitate a strong effort in the fusion field.

It seems pointless to talk to this group about the potential advantages of fusion, but I would like to dwell a bit on one benefit which is especially applicable to advanced fuel fusion. The environmental and siting advantages of a method of generating electricity based on direct conversion are very worthwhile. Thermal efficiency is one obvious benefit from direct conversion but there is an even more basic one. Just as carbon dioxide emission may eventually limit the combusion of fossil fuels, water consumption has the potential to put a ceiling on total energy consumption. Proper design can mitigate the effects of thermal discharges but generally does so at a price in increased water consumption. Dry cooling has some potential but the price in capital cost and thermal efficiency may not be acceptable. If achievable and economically feasible, direct conversion is a most attractive approach.

The challenges are many but the rewards are great. I wish you a most successful meeting.

INTRODUCTION TO THE ADVANCED FUSION FUELS MEETING HELD AT THE COMMONWEALTH EDISON COMPANY CHICAGO, ILLINOIS

by

William C. Gough
Program Manager for Fusion Power
Electric Power Research Institute

Although advanced fusion fuels offer the utility industry the possibility for many environmental and operational advantages in the use of fusion power plants, very little effort has been devoted to date in exploring this option. This apparent lack of interest has been due to the more difficult physics requirements that the advanced fuels place upon the fusion plasma before a practical power plant can become a reality. However, as we enter this period of transition with fusion rapidly approaching conditions necessary for a burning core of D-T plasma, it is appropriate that we study more fully these difficulties and advantages of the advanced fuels. This combination meeting and workshop sponsored by the Electric Power Research Institute, has therefore been specifically devoted to the topic of exploring the advanced fusion fuel option. It represents the first comprehensive evaluation of the subject. The attendees at the meeting are from various backgrounds: Utilities, national laboratories, industry, universities, and the federal government. Their range of expertise is extensive including those doing detailed basic physics measurements, fusion experimentalists, engineers in energy conversion and nuclear systems, and managers and operators from major electric utilities.

Because of the potential advantages to the ultimate users of fusion power, the utility industry has had a long interest in understanding the trade-offs and options potentially available from advanced fusion fuels. In 1973 George Miley of the University of Illinois carried out a study for the Edison Electric Institute entitled, "The Development of High Efficiency-Advanced Fuel Fusion Reactors" that outlined a plan for obtaining this knowledge. At the Electric Power Research Institute we have been sponsoring since 1974 a number of programs and workshops that address various aspects of advanced fuels including the basic physics of new plasma confinement concepts, total power plant systems studies including integration with new and various forms of high efficiency energy converters, and detailed cross-section measurements and calculations to provide the basic data for furthering work in this area.

A brief description of each of the EPRI programs and workshops in advanced fuels appears at the end of this paper. Over the years the utility and other private programs have represented a large percentage of the national work in the area of advanced fuels. The largest contributions have been from privately funded work by EPRI, the Fusion Energy Corporation, and TRW.

The primary reason for the present increased utility interest in the area of advanced fuels can be described by the following four figures: First, the recent physics progress in fusion is moving us rapidly into a transition period where engineering and operational implications of fusion R & D should be carefully considered. Figure 1 shows that fusion progress has been moving at the rate of an order of magnitude per year in terms of the theoretical percentage of D-T fuel burned. This is a direct measure of the effectiveness of our ability to burn fusion fuel. Fusion machines now under construction should be able to burn sufficient fuel to move us into the regime of break-even conditions for D-T reactors. Thus, large expenditures could soon be starting on the engineering aspects of D-T systems. The technical hurdles that must be overcome before a D-T fusion system can become a successful commercial electric power plant appear formidable in terms of both environmental and plant operational considerations.

Figure 2 illustrates that an important role of the user, i.e., the utility industry, is to establish oeprational requirements so that they may be compared with the capability of fusion power systems being developed by the federal government. In some cases, such as in advanced fusion fuels, the comprehensive data base necessary for this comparison has been lacking.

Figure 3 gives an example of utility requirements for fusion power and some of the potential solutions that we believe will alleviate the problems now apparent from the initial conceptual designs of D-T fusion reactors. Advanced fuels with their lower number of neutrons have a definite advantage in terms of maintainability and reliability. The reduced tritium and neutrons also provide advantages in terms of environment and safety.

There is a long road in front of us before fusion power plants can be deployed for the utility industry. We therefore take the view, expressed in Figure 4, that there should be a trade-off made between physics risks and commercialization risks. Although advanced fusion power plants may require higher physics risks, they could lead to an easier commercialization than a fusion direction with a lower physics risk. Therefore, the purpose of this meeting is to initiate this type of an evaluation in terms of what are the potentials for advanced fuel fusion systems.

For fusion power, which is a capital intensive power source with almost nonexistent fuel costs, the cost per kilowatt hour of energy produced is essentially inversely proportional to the percentage of the time that the system is putting out power, except to the extent that components wear out with energy throughput. In D-T systems the first wall would probably represent components that could seriously affect plant availability. Advanced fuels should greatly aid in assuring high availability of the plant by reducing radiation damage and remote maintenance requirements. Because of the greater physics difficulties in achieving practical advanced fuel fusion systems, the total power plant must be considered as a single unit and optimized as a total system. The systems approach is far more important for advanced fuels than it is for D-T where much more leeway in the plasma physics exists. This means that emphasis must be placed on higher efficiency energy converters possibly involving new technology developments; plasma confinement concepts that lower synchrotron radiation such as surface confinement and other high beta systems; the use of fusion reactors that are power amplifiers and employ non-Maxwellian plasmas; and possibly multi-purpose plants which incorporate synthetic fuel production. All of these topics will be covered in the sessions that follow.

A meeting of this nature involves a great deal of work by many people. I'd like to take this opportunity to extend my thanks to the University of Illinois for organizing this meeting, in particular George Miley and Chan K. Choi; to the Commonwealth Edison Company who graciously agreed to host the meeting, particularly Ed Steeve, Robert Bolger, William Worden, and Martell Tuntland who so capably handled the detailed arrangements; to the Electrical Institute of Chicago for providing this fine meeting room, in particular Robert W. Turek, Managing Director of the Institute; and to F. Robert Scott of EPRI who is the Project Manager of most of our advanced fuel work and was instrumental in initiating and setting up this meeting.

FUSION RESEARCH PROGRESS

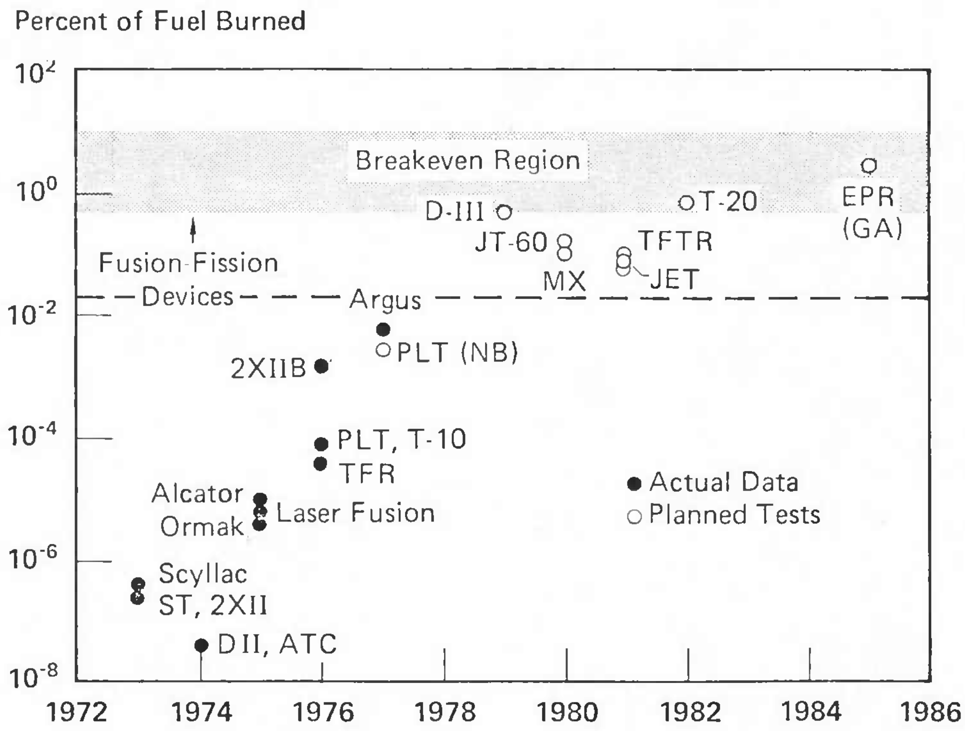

Figure 1

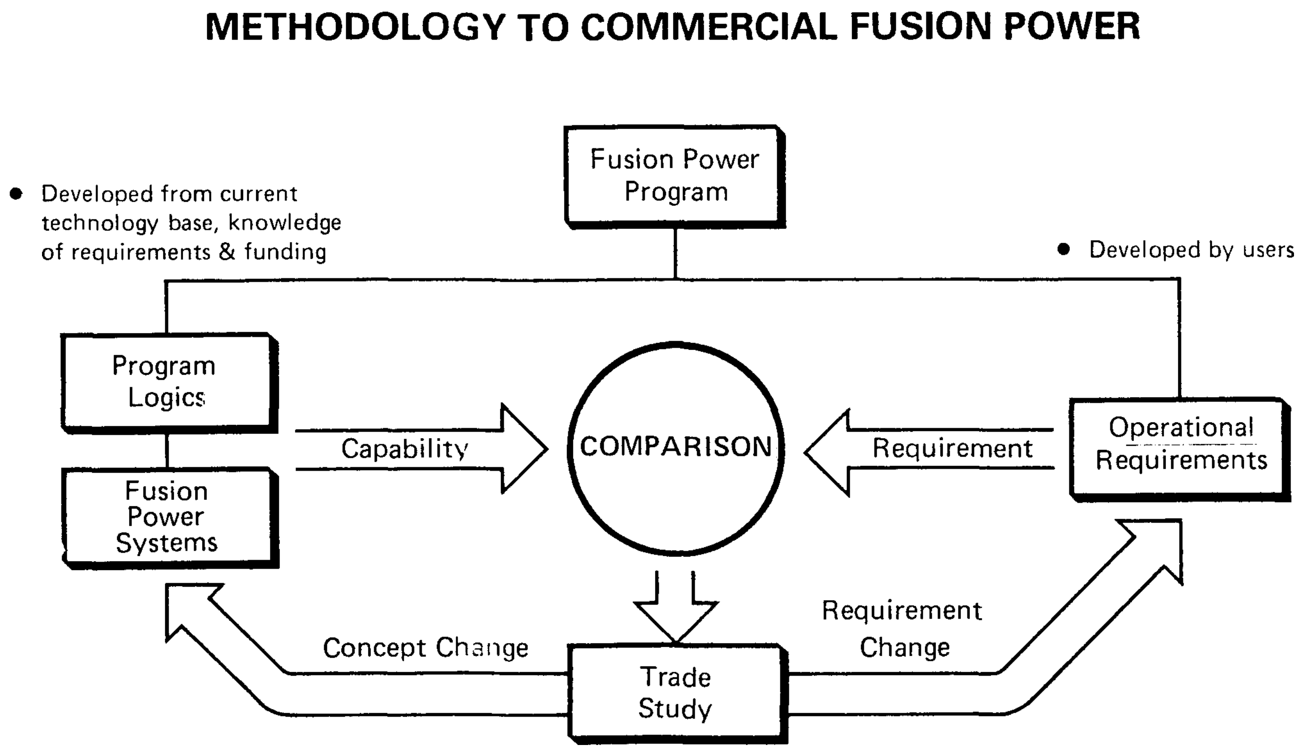
METHODOLOGY TO COMMERCIAL FUSION POWER
Fusion Power Program
• Developed from current technology base, knowledge of requirements & funding
• Developed by users
Program Logics
Fusion Power Systems
Capability
COMPARISON
Requirement
Operational Requirements
Concept Change
Trade Study
Requirement Change

Figure 2

FUSION POWER REACTORS

UTILITY REQUIREMENTS	*POTENTIAL SOLUTIONS*
Cost & Size	• Higher Power Density (alternative concepts) • Modular Design (alternative concepts) • End Stoppering of Linear Systems
Maintainability & Reliability	• Long Life & Low Activation Materials • Modular Design • Advanced Fuels – Less Neutrons • Improved Components – Energy Storage, Switches, etc.
System Performance	• Adaptive Control Systems • Reactor/Balance of Plant Integration • Reduce Start-up Power & Recirculating Energy
Environment & Safety	• Advanced Fuels – Reduced Tritium & Neutrons • Direct Energy Conversion • Low Activation Materials • High Temperature Materials

Figure 3

TODAY's VIEW OF TRADEOFFS FOR FUSION ENERGY DEPLOYMENT

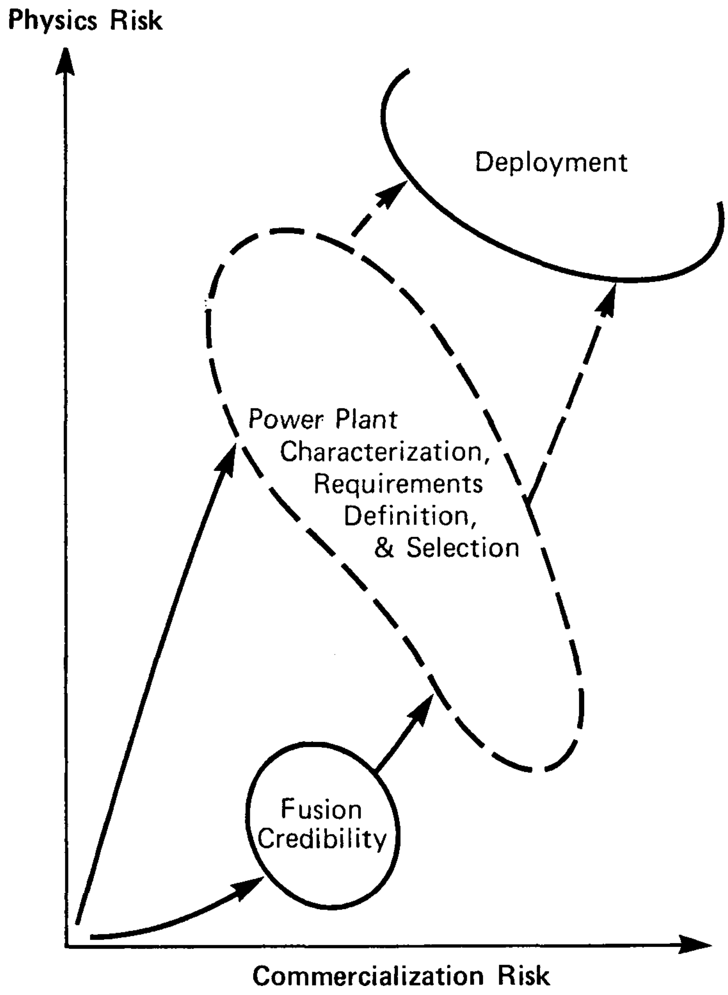

Figure 4

EPRI Programs Relating to Advanced Fusion Fuels

1. University of Texas, William E. Drummond

Describes a concept for directly extracting energy from a fusion plasma by magnetic coupling in a tokamak geometry. The $p\text{-}B^{11}$ and $D\text{-}He^3$ fuel cycles are discussed. Published in EPRI Report 96-2, September 1975.

2. Unviersity of Illinois, George H. Miley

"Conference Proceedings: Effects of Cyclotron Emission on the Power Balance in Fusion Systems," EPRI Special Report SR-16, September 1975.

3. Lawrence Berkeley Laboratory, Mort Levine

Experimental program on high beta tormac concept that holds promise for confinement of advanced fuels. Work in progress under RP-272.

4. Unviersity of Illinois, George H. Miley
   Brookhaven National Laboratory, Jim Powell
   Lawrence Livermore Laboratory, Ralph Moir

This project is evaluating the merits of the various advanced-fuel reactor systems relative to a conventional deuterium-tritium (D-T) system. The bulk of the effort has concentrated upon noncircular cross-section tokamaks using three basic fuels: catalyzed DD, DD, and DHe^3. Some exploratory studies of ultra-advanced fuels and reactor concepts were also considered. EPRI project RP-645, report in preparation.

5. Mathematical Sciences Northwest, Robert T. Tanssig

A special study of a "High Thermal Efficiency, Radiation-Based Advanced Fusion Reactor." The concept is highly beneficial for advanced fusion fuel cycles like $p\text{-}B^{11}$ and uses two new elements; the X-ray boiler concept for the first wall and the wave energy exchanger. An addition to EPRI project RP-645, report in publication.

6. University of Wisconsin, Robert Conn

Project will include a comparative analysis of alternative fuel cycles for laser-fusion power plants. EPRI project RP-237, work in progress.

7. University of California, Los Angeles, John Dawson

Computer simulation studies of the effects that background plasma has on the emission of synchrotron radiation. EPRI project RP-270, report to be published.

8. University of California, Los Angeles, John Dawson

Part C of EPRI assessment of alternate fusion reactor concepts addressing the possibilities of $p\text{-}B^{11}$ systems using multipole confinement, EPRI technica planning study, TPS-76-637, report in press.

9. EPRI Ad Hoc Advisory Panel

"The EPRI Asilomar Papers" with section "On the Possibility of Advanced Fuel Fusion Reactors," published as EPRI Special Report ER-378-SR, March 1977.

10. University of California, Los Angeles, Alfred Wong

An experimental program to provide sufficient data to permit the evaluation of the Surmac (Surface Magnetic Confinement) concept for fusion fuels, especially advanced fuels such as $p\text{-}B^{11}$. Project being started.

11. University of Iowa, Edwin Norbeck

Measurement of the $Li^6\text{-}Li^6$ reaction cross-section from 0.6 MeV to 5 MeV to permit an accurate evaluation of this fusion fuel's burn rate. Project just begining.

12. California Institute of Technology, Tom Tombrello

Measurement of the basic fusion fuel burn rate for $p\text{-}B^{11}$ over the range of interest to the fusion power program. EPRI Project TPS77-708, report in preparation.

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Comments on Advanced-Fuels and Workshop Objectives

by

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ABSTRACT

To provide additional background for the workshop groups, highlights from an 1973 report to the Edison Electric Institute on the development of advanced-fuel fusion reactors are reviewed.

I. INTRODUCTION

As indicated in the letter of invitation to this review meeting, participants were purposely selected to bring together a variety of points of view, including representatives from utilities, industry, national laboratories, universities, and funding agencies. The meeting format was designed to provice ample opportunities for exchanges of ideas through the inclusion of poster sessions, two panel discussion, and four workshop groups. My present objective is to provide some additional background for the workshop groups. Many of my thoughts on this matter stem from the report The Development of High-Efficiency-Advanced Fuel Fusion Reactors that I authored in November 1973 for the Fusion Task Force of the Edison Electric Institute (EEI). (Copies were sent to all registrants for this meeting.) At that time the EEI task force was interested in the same questions we are considering at this meeting: Are there advantages for advanced-fuel fusion? If so, how and when should an R & D program be implimented? Consequently a one-year study was commissioned which resulted in the report noted above. (As a part of the study, two workshop meetings, attended by a 30 workers from the fusion area, were held at the then AEC headquarters in Germantown along with intermediate reports to the task force during their meetings at Los Alamos and San Francisco.)

Indeed some parts of this report have subsequently been implimented through EPRI, but many of the questions and suggestions raised remain open for evaluation. Whether or not you agree with this document it remains as the only comprehensive discussion of the development of advanced-fuel fusion presently available. Consequently it is instructive to briefly review some highlights of the report.

II. HIGHLIGHTS OF THE EEI REPORT

The objective proposed was broadly stated as the development of an integrated advanced-fuel, direct-conversion reactor (referred to in subsequent sections simply as an advanced-fuel reactor). Such a reactor would hopefully have the following characteristics:

- high overall conversion efficiency with an ultimate goal of ultra-high efficiencies (> 70%).
- minimum tritium handling and breeding problems.
- minimum neutron fluxes, i.e. reduced

- neutron damage
- neutron activation of structure and other blanket materials
- shielding

The timing of such a project would be to establish the feasibility of an advanced-fuel concept prior to the point in fusion reactor development that extensive prototype and demonstration reactor programs are initiated. In this way an alternative to D-T might be provided in time to have impact.

Such a project would involve the integration of three elements:

- an advanced-fuel cycle with a charged-particle fraction approaching 100%,
- a fusion reactor capable of the temperature and conditions required to burn the fuel, and
- an high-efficiency (order of 90%) direct-conversion technique.

We will briefly consider each of these components in turn.

1. Advanced-Fuel Cycles

Prominent fusion fuels with charged-particle energy fractions over 90% are:

Table I: Prominent Advanced Fuels

	Energy fraction to Chg'D Particles*	T_i, keV for max $<\sigma v>/T^2$ **
D-D, product He^3 burned, not product tritium	0.907	$\gtrsim 30$
D-He^3	1.0	~ 70
p-B^{11}	1.0	~ 100
p-Li^6; D-Li^6	1.0	~ 500

*Neglects neutron-producting side-reactions such as D-D in a D-He^3 plasma
**The temperature for maximum power density in a magnetically-confined reactor

The selection of an optimum fuel cycle involves a number of considerations including: availability of the fuel; side-reactions that may produce neutrons, hard gammas, and tritium; temperature and confinement requirements. Different fuels may be selected for certain purposes, e.g. D^3-He may be best for near-term satellite reactors (obtain ^{3}He from D-D generators) but p-^{11}B could emerge as the "ideal" in the long term if methods to confine and burn it are found.

2. Reactors for Advanced Fuels

The desired reactor must have three key characteristics:

- a high Q-value to minimize inefficiencies inherent with large recirculation rates
- relatively low electron temperature to minimize radiation losses; and
- a relatively high β to reduce cyclotron radiation losses and to achieve reasonable power densities.

Consider magnetic confinement systems first. High-Q suggests a toroidal system, however existing approaches (Tokamak or Toroidal θ-pinch) would not simultaneously meet the other requirements. Thus it appears that new concepts must be considered. An example is the field-reversed mirror concept whereby a closed field region is created internally to an open mirror.

Inertial confinement systems could also be quite attractive for burning fuels like D-^{3}He or p-^{11}B. One obvious advantage is the elimination of cyclotron emission. However, more detailed studies are required to understand pellet performance. Also, the potential for developing lasers (or ion or electron beams) with both the required power and efficiency must be included in any evaluation.

A further problem is that some of the most promising reactor concepts pose special problems relative to coupling with efficient conversion devices.

3. High-Efficiency Direct Conversion Techniques

The following techniques have been considered for direct conversion:

- Direct Collection
- Magnetic Compression - Expansion
- Electric-Field Coupling
- MHD
- Other EM Coupling
- Non-electrical extraction

Of these, only direct collection is reasonably assured of approaching the required 80-90% efficiency. (Note that this is the efficiency of the converter alone and should not be confused with the overall plant efficiency.) Magnetic compression-expansion and electric field coupling (including bootstrap coupling) have not been sufficiently explored in a complete cycle sense to establish their capabilities. Rough estimates seem to place these approaches in the region of 60-70% efficiency,* and this is sufficiently encouraging to warrant further study. While the remaining methods cannot be ruled out, no studies are available to prove that any one of these methods has the promise of high efficiency.

In addition to direct-conversion, it must be remembered that a sizeable fraction of the output power from advanced fuel fusion may be in the form of radiation. Consequently, an equally strong emphasis should be placed on the development of techniques to efficiently handle this energy. This introduces the concept of high temperature blankets, perhaps using the working fluid in efficient turbine or MHD cycles.

In summary, of the three components required for an advanced high-efficiency reactor, the greatest uncertainty revolves around our ability to develop an integrated fuel-cycle--reactor concept. If this can be done, and if direct collection and a efficient blanket for handling output radiation can be coupled to the system, this should provide a viable approach to the goal.

A possible time schedule that I suggested in 1973 for this project is shown in Table I below:

TABLE I: OVERALL SCHEDULE

		Work	Years
Phase I		Basic & exploratory studies/ development of integrated fuel cycle-reactor-converter concept	1974-1977
	Decision*:	Is there a viable concept?	1977
Phase II		Major Experimental Program	1977-1985
	Decision:	Is the concept technically feasible? What level of continuing R&D should it receive compared to D-T systems?	1985
Phase III		Major Prototype and Engineering Studies	1985-2000
	Decision:	Is the concept suitable for commercial sales?	

*If negative answers are reached at any decision level, the program would, of course, be discontinued or redirected.

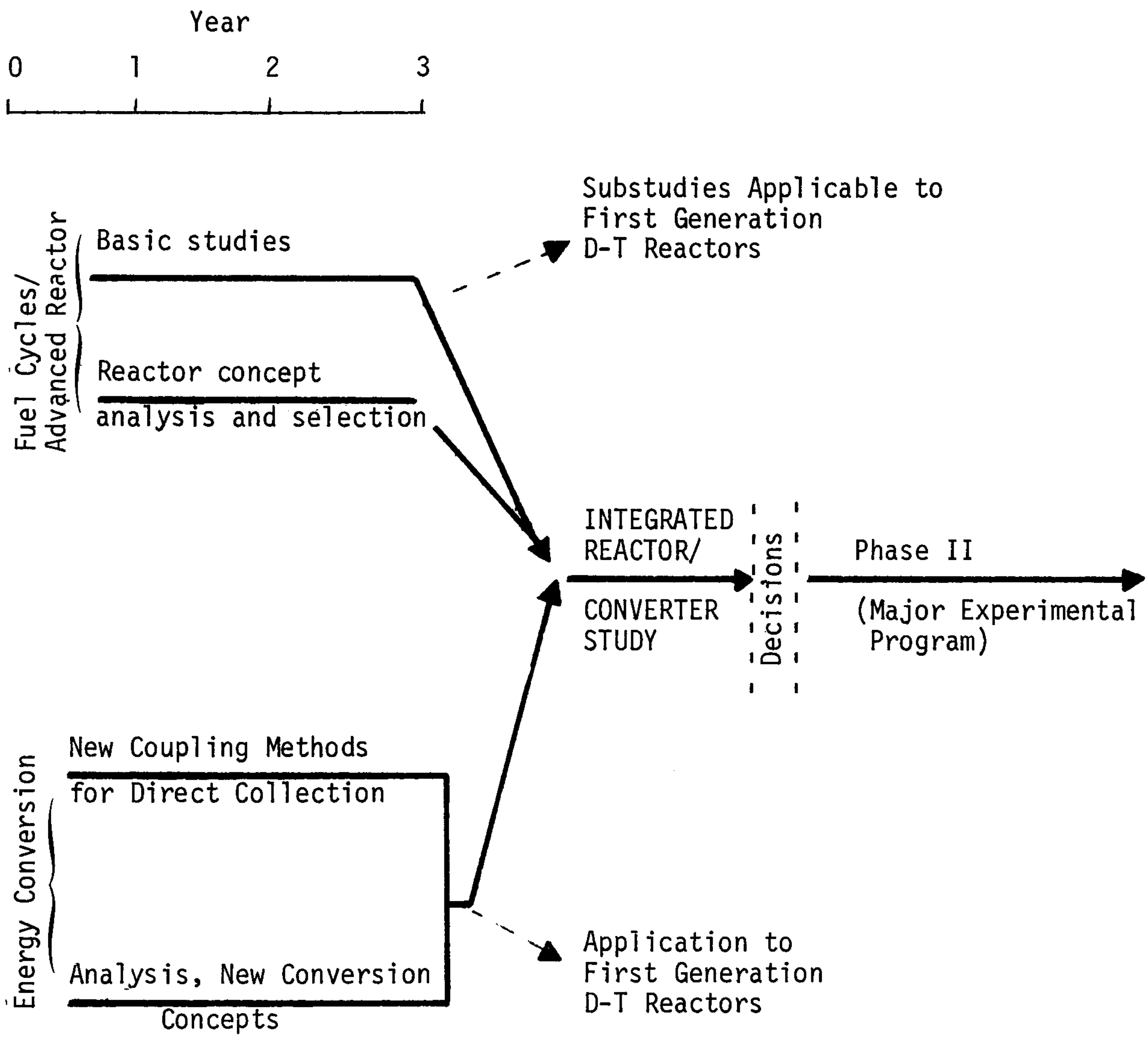

Figure 1. Time-Decision Graph for Phase I. As indicated, while the R&D indicated is directly aimed at the development of an advanced reactor, a part of the work completed by the end of the second year would also be applicable to first generation D-T reactor development.

According to this plan, we should know if there is a viable system concept in 1977! Perhaps that can be a topic for discussion in the workshops! With this overall plan in mind, we now turn to a detailed discussion of Phase I, illustrated in Figure 1. This schedule is geared to produce an integrated conceptual approach at the end of three years. This requires a tentative identification of the fuel-reactor approach by the end of the second year. It then becomes possible to choose a direct conversion technique to receive detailed study along with the reactor concept during the last year.

The EEE report contains a detailed estimate for the budget required to carry out Phase I. It suffices here to indicate that an overall cost for three years of $\sim$ \$5 million was proposed.

The program detailed above was designed as a "nominal" effort. An expanded program was also suggested that would permit the initiation of additional experimental work. Experimental studies that appear to be essential but yet sufficiently general so that they could be initiated without regard to the eventual type of reactor selected would include:

- Experimental study of cyclotron radiation transport and reflection.
- Basic cross-section measurements to improve the accuracy of current data and fill in gaps.
- Experimental study of ion slowing in high-Z plasmas.
- Radiation damage and voltage holding in direct-collection-type converters.
- Space-charge effects in direct-collection units.

These, plus other experiments which might be identified as critical as the analysis proceeded, could represent a significant increase in the budget, probably requiring an additional \$1 million/year.

In conclusion, although EPRI has initiated some advanced fuel studies, neither they nor ERDA have undertaken as agressive a program in advanced fuel R&D as suggested in the IEE report. The reason appears to be four-fold:

- lack of funds to do much more than D-T R&D
- a feeling that such work is premature
- indecision within the fusion community itself about the role of advanced fuels
- a feeling that advanced fuel systems will naturally evolve from D-T systems

Hopefully participants in the present workshops will consider the pros and cons of these points of view.

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A Utility View of Fusion

By

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ABSTRACT

We need from fusion a relatively problem-free way of generating electric power. The author feels that a large complex D-T fueled plant would prove to be a big disappointment in what it offers compared to a fission plant. Advanced fuels are seen as being where nearly all the commercially significant advantages offered by fusion lie.

The paper first takes a non-concept-specific look at 50-50 D-T versus advanced fuel blanket/shield dimensions and concludes that given an effective confinement (beta $\simeq 1$) practical advanced fuel reactors could be enough smaller and less difficult to engineer than D-T to make considerable difference in their timing and use.

The author urges more emphasis on tailoring confinement to fit the desired energy plant. An example is given.

A Utility View of Fusion

June 27, 1977
Clinton P. Ashworth

Without fusion there are already enough power generation technologies available or that surely can be made available to supply all the electric energy at reasonable cost this country, or the whole world for that matter, will ever need, at least for as far as any of us can forsee. But as I think about the project-stopping roadblocks that are increasingly inflicting the non-fusion technologies, I worry greatly that an era of shortages will befall us if we don't get moving on something substantially different and better.

I see no need for a fusion reactor that is not substantially different and better than, say, a fission reactor. A mere competitor with a few advantages of marginal real worth would not be good enough. Fusion reactors that are large, complex, and with radioactivity in the hundreds of megacuries would face the same problems that create energy supply doubts in the first place.

We need from fusion a relatively problem-free way of generating electric power. The advantages to be sought from fusion should be as dramatic as its billion-degree technology is different from other technologies.

This first slide, taken from my February 1977 AAAS paper, is a list of what I consider to be mission objectives for fusion power. Some of these have to do with keeping the time and financial commitment required for a project small enough that things can be caused to happen expeditiously without dragging on onerously. Others have to do with attractiveness, not only in the eys of the user but deserving of favorable recognition from neighbors and public servants. Most of the rest have to do with keeping the nuclear aspects to a level where the general public can perceive them to be no problem.

A favorable public perception depends on the "common man" being able to judge for himself the acceptability of the risks and benefits and not inviting that judgment to be made by a few self-appointed spokesmen. Huge, complex, issue-prone technologies defy common sense judgment by ordinary people. Thus, a recipe for successful fusion is as shown on the next slide.

Most fusion research is aimed toward trying to get the easiest to ignite fuel Z50-50 D-T) to burn. I believe that the plasma research job is not complete enough for application until (1) confining fields can be applied effectively to the plasma (beta $\underset{\sim}{\sim}$1), (2) extraneous energy input and loss mechanisms are plugged up, (3) sustained operation is assured, and (4) confined plasmas of reasonable proportions can be used in simple, easy device configurations. Until we can see our way through these basic accomplishments, any device we try to develop beforehand is likely to be premature and fail.

When plasma research reaches an adequate stage of readiness, 50-50 D-T could prove to be the wrong fuel. The Fusion Program Committee has repeatedly affirmed its desire for EPRI to pursue research on fuel cycles which produce more charged particle energy and less neutrons and tritium and on reactor concepts suited to such fuel cycles. Contrary to the plasma research oriented view, I believe so-called advanced fuels may offer the quickest route to commercial fusion energy. By advanced fuel, I mean anything substantially different from 50-50 D-T such as deuterium-rich D-T, all D, D-He^3, or the more exotic fuels such as p-B^{11}.

Let us consider the differences between 50-50 D-T fueled reactors and, say, symbiotic D-He^3 power reactors with all-D producer reactors fueling them.

To get some basics out of the way, consider the dominant reactions in any hydrogen and helium mix advanced fuel burn as shown in this next slide. The multipliers i, j, and k are partially controllable by setting the makeup fuel mix proportions, temperatures, burn time, and other factors. j and k cannot differ by more than 1.0 without an outside makeup source of either protons or helium 3. The tritium-breeding reaction is required if the sum of i and j is greater than one. The importance of the p-T reaction is its large cross section above 1 MeV and the large velocity of protons born at very high energies in the D-D and D-He^3 reactions.

This next table lists some of the key reaction parameters for ranges of i, j, and k. i represents the D-T reactions relative to pairs of D-D reactions, j/i is the ratio of p-T to D-T reactions, and "k_{min}" and "k_{max}" represent minimum and maximum burnup of He^3. "T/n" is the ratio of makeup tritium required per plasma neutron. Zero means no breeding blanket is required. Numbers in parentheses after some of the zeros indicate atoms of surplus tritium produced.

"kev" is the total reaction energy, "%*" is the percent charged, "PD*" is a rough estimate of the power density based on a field reversed mirror with an 8 tesla applied field, and "MR" is the roughly estimated minimum radiation. Units for PD* and MR are Mw/cubic meter. "%ch" is the stimated maximum percent of reaction energy available as input to a plasma direct converter.

At the bottom of the table are a few special cases. 50-50 D-T has a very large power density but only 15% of its energy can be delivered to a direct converter as charged particles. "Catalyzed D" corresponds to i=1, j=0, and k=1. All D-D and "ideal He^3 producer" are shown as having very low power densities. This may be a problem not only in getting cost per unit output to be reasonable but it also puts these cases on the ragged edge of being able to produce more power than is radiated away. p-B^{11} is also listed. It probably cannot produce more fusion power than radiation except at explosion densities.

Confinement systems which can hold 5 or 10 cubic meters of plasma at a beta of about 1.0 whould have no difficulty producing adequate net power almost anywhere in the table, particularly where the power density is indicated as being about 50 or greater, provided reasonable care is taken to minimize some of the energy wastes which can be tolerated in 50-50 D-T designs. Under these conditions, the D-He^3 reactor poses the least difficult engineering problems of anything shown in the table and possible 75% of its total energy can be delivered to a high efficiency direct converter.

It is not quite fair to compare a D-He^3 reactor with a 50-50 D-T reactor because most of the technical difficulty of the D-He^3 system is in the He^3 producer reactor which would presumably be sited and licensed somewhat like a fission power plant. On the other hand, it would not be fiar to compare 50-50 D-T reactors with the symbiotic He^3 producer-burner system without taking into account the transmission lines that would be replaced by He^3 transport and the existing oil-fired sites, cooling systems, and power lines thatcould continue to be used with the comparatively nuclear-issue-free D-He^3 reactors.

Because a D-He^3 reactor may have similarity to other advanced fuel reactors, let's just look at some differences in blanket/coil shield for a "Brand X" confinement which produces the power densities given in the reaction table. This next slide compares 150 Mwe D-He^3, He^3 producer, and 50-50 D-T reactor blanket/shields assuming cylindrical first walls with length to diamter ratios equal to ten. For the D-He^3 case, 1/10th of the vacuum space is assumed to be filled with plasma. For the He^3 producer and the 50-50 D-T reactors, size of the vacuum space is set to give reasonable wall loadings. Peak wall loadings are assumed to be 1.25 times the total power to the wall spread over the cylinder wall area.

The 150 Mwe D-He^3 reactor is shown as requiring roughly one-tenth the blanket structure and materials by volume as the same electric rated D-T reactor. Even the He^3 producer reactor is shown as requiring only 2/5ths the structure and materials volume of the D-T reactor. Incidentally, the producer is based on i=.5, j=.25, k=.15 and no breeding in the blanket. These assumptions would result in the immediate production of 1.1 atoms of He^3 per 18.66 Mev of reaction energy or enough for one producer reactor to fuel roughly 2 equal electric rated D-He^3 reactors. Thus, roughly 2 of the D-He^3 reactors plus one producer reactor could produce the same amount of electric power as 3 of the D-T reactors. No lithium or blanket-produced tritium is involved in the symbiotic system.

In addition to the basic size difference, the D-T blanket/shield has substantially more severe design requirements. It must make use of both neutrons and heat whereas the D-He^3 and producer reactor shields need to use just the heat, and that not very efficiently. In the case of the D-He^3 reactor shield, only 1/6 as much heat needs to be handled. The D-T blanket must contain lithium and possibly neutron multiplying materials and materials suitable for a high efficiency thermal cycle. The choice of materials for the other reactors is not so severely restrained. The wall loadings are also different.

The blanket/shield for the D-T reactor is thick compared to the diameter of the vacuum space. These geometric proportions for D-T in this size are so poor that large gains are to be made by scaling up to larger sizes of the plasma physics permits doing so. To get proportions comparable to the otherreactors would require increasing the size of the D-T reactor to at least 500 Mwe. The blanket/shield volume for a 500 Mwe D-T reactor would be perhaps 1000 cubic meters.

So, in the advanced fuel systems, we're dealing with roughly one-tenth as massive blanket/shield structures with substantially eased engineering and materials choice requirements as for D-T reactors. I believe these differences will have a dramatic effect on optimum size, total project cost, on cost per unit of energy output, and the time it takes to get things built and operating. In my opinion these differences are far from trivial.

Thus, unless the best plasma physics attainable dictates otherwise, it seems to me that 50-50 D-T will be large--possibly too large to develop expeditiously and to satisfy the need outlined at the beginning of my remarks. From an engineering point of view, advanced fuel reactors, particularly the He^3 system, offer more hope of being made practical in substantially smaller sizes and of being less difficult and quicker to develop.

But the bad news is that engineers cannot even begin down the advanced fuel road until the plasma physics quits leading down the primrose path of a half-done job--being satisfied with low beta approaches, extraneous energy flows, and inordinate complexity.

Proving out a confinement to go with the shields described is not going to be easy. The one shown in the next slide is, I think, interesting. It is basically a Lawrence Livermore Laboratory, Rand McNally, Hans Fleischmann idea and was described in my February 1977 AAAS paper. It is the rough equivaldnt of a billion degree gas turbine and appears suitable for the He^3 advanced fuel system, if the basic field reversed mirror concept works and provided ring sizes can be increased to tenths of cubic meters each.

The technical features that I like about this idea are (1) no extraneous plasma energy inputs are required to maintain the confinement conditions such as neutral beam injection at temperatures higher than burn temperature, (2) there is no problem of impurity or burn time control, and (3) energy transfer to and from the plasma is largely magnetic. These features offer the hope of a higher net energy return from advanced fuels than anything else we've looked at plus burn and impurity control comparable to a pulsed device but in continuous flow. Whether these hopes can be realized depends on demonstrating them experimentally.

There are surely many other promising advanced fuel concepts as or perhaps more attractive than this one, and that is what we are here to consider. But I think this concept illustrates a point. It is a complete energy producing idea and not just a plasma bottle. It hangs together well when viewed from angles besides just from the plasma physics alone. Let us not get so caught up with bottle ideas that we lose sight of which bottles lead to good energy plants and which do not.

Carrying my illustrative concept one step further, the next slide shows its advanced fuel system in commercial use. The probability that D-He^3 plants could be located close to where the power is used makes the beneficial impact of having the energy supply more obvious to the local public and makes the need for the plant less contestable. Getting energy users to be more aware of the direct impact of energy supply plants on whether or not they have energy is essential if we are to turn the tide against the political trend toward insufficient energy supply.

In effect we will have made a partial changeover in the way we do business--to more local generation using a "helium economy" for part of the energy, backed up by the interconnected electric system. The result--a truly optimal mix of local and grid-supplied electric energy with transport of clean "50 million mile per gallon" nuclear fuel taking the place of some transmission lines. This may be the kind of dramatic change that is needed to pump new vitality into the energy supply business.

The slide shows three local D-He^3 plants fueled by one producer. This ratio may be possible with the reactors described because the local plants with transmission serving mainly as backup would be more sensitive to local loads and would possibly operate with less load diversity (lower capacity factor) than the producers. A higher ratio of plants served per producer would require breeding tritium at the producer and awaiting its decay to He^3. This is a less desirable option which should be avoided if it can be.

How does the advanced fuel system measure up to the mission objectives described earlier? This next slide is my opinion of what is possible using fission power plants as a basis for comparison. A solid square indicates a potential solid advantage. An open square represents some advantage. A small circle indicates no substantial advantage and perhaps a serious problem.

As the slide shows, I feel that a large complex D-T fueled plant will prove to be a big disappointment in what it offers compared to a fission plant. I show a few more advantages for the small simple D-T plant but none of these deal with social issues. And as I indicated earlier, I doubt that a D-T fueled plant can be made small, simple and practical.

Advanced fuels are where I think nearly all the potential advantages offered by fusion lie. The slide shows the He^3 advanced fuel system. I think the D-He^3 reactor would be the least difficult fusion reactor to develop, if the physics research job were completed as described earlier. Perhaps "advanced" is the wrong word to use in describing these fuels. "Less demanding given a way to confine plasmas effectively" or "more satisfying of need" might be a better way to describe them.

I have not included the more advanced fuels, such as p-B^{11}, in this discussion because I have not yet been able to formulate any of the more advanced fuel concepts into complete and practical energy plant concepts as I have for the He^3 system. The essential ingredient of completeness in a practical way needs to be supplied before we utility evaluator types can make auy favorable judgments or recommendations. This should be one of our objectives here.

If my perception of what we need from fusion is right and of D-T fusion offering marginal advantages over what we already have, then advanced fuel fusion should be getting mainline treatment and not be relegated to a post D-T effort.

If there is a way to confine plasmas effectively, as required for advanced fuels, then I can't help but believe thatmore of the fusion research effort should be directed toward trying to tailor confinement to fit into the kind of plant we need and less toward tailoring the plant to fit each researcher's favorite confinement. Let's not let the cylinders totally dominate the design of our vehicles.

1. Projects easy to commit and carry out.
2. Plants economic in small size.
3. Project schedule time short.
4. Siting requirements met widely.
5. Plant cooling not difficult.
6. No extraordinary licensing, Q. A. or backfitting anticipated.
7. Release of radioactivity no significant public concern.
8. Radioactive wastes no significant public concern.
9. Misuse or proliferation of nuclear material no significant public concern.
10. Raw fuel is widely available and fuel supply can be of no public concern.
11. Is commercially available soon.

- Keep it simple
- Keep it small
- Keep it "clean"

Dominant reactions in a deutrium-rich burn of hydrogen and helium isotopes

2(D + D) → n(2.45) + p(3.02)* + T(1.01)* + He3(0.82)*

i(D + T → n(14.07) + He4(3.52)*)

j(p(0.76)* + T → n + He3)

k(D + He3 → p(14.68)* + He4(3.67)*)

Blanket reaction required if i + j > 1:

(i + j - 1)(n + Li6 → T + He4 + (4.8))

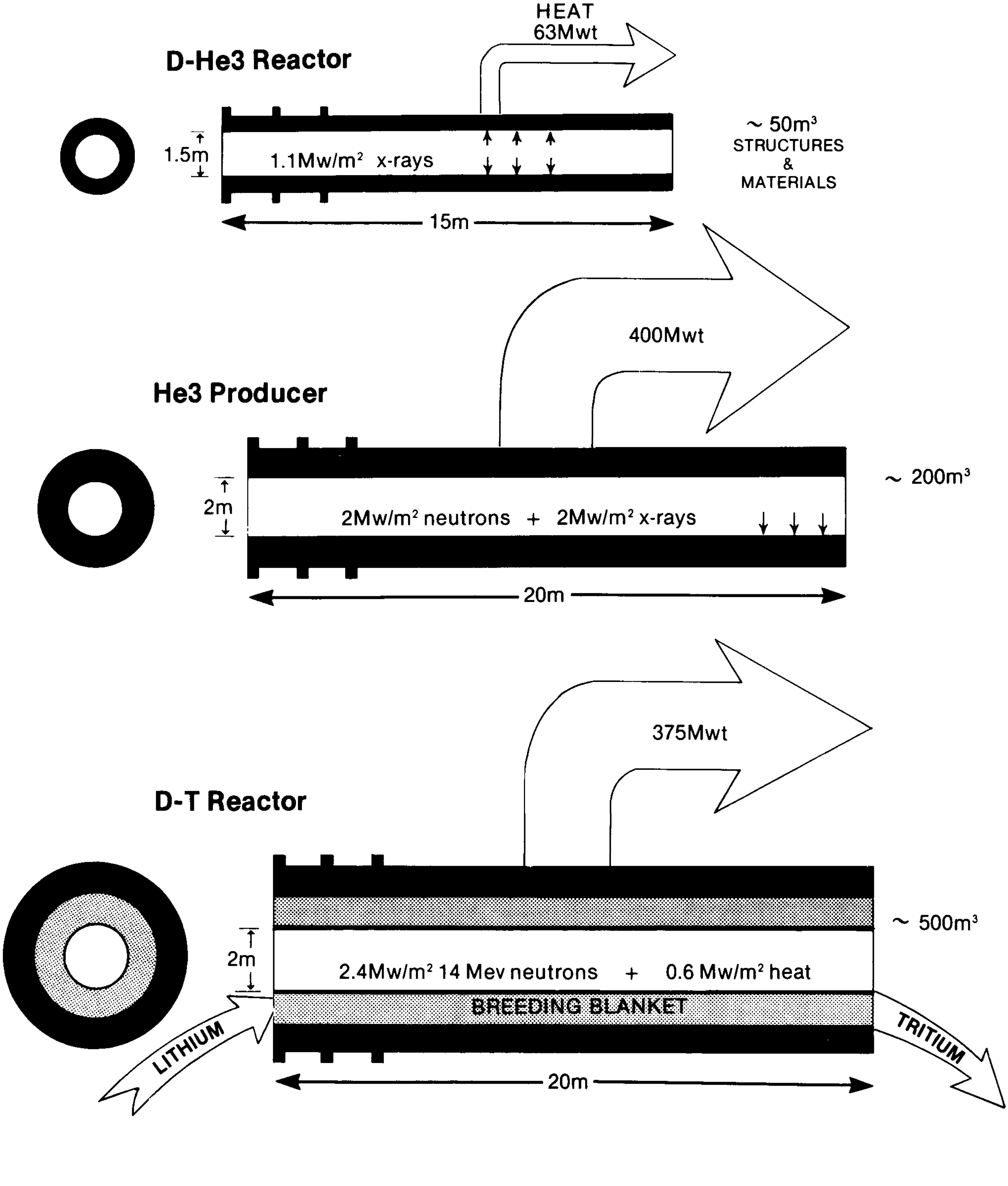
REACTION TABLE - DEUTERIUM-RICH BURNS

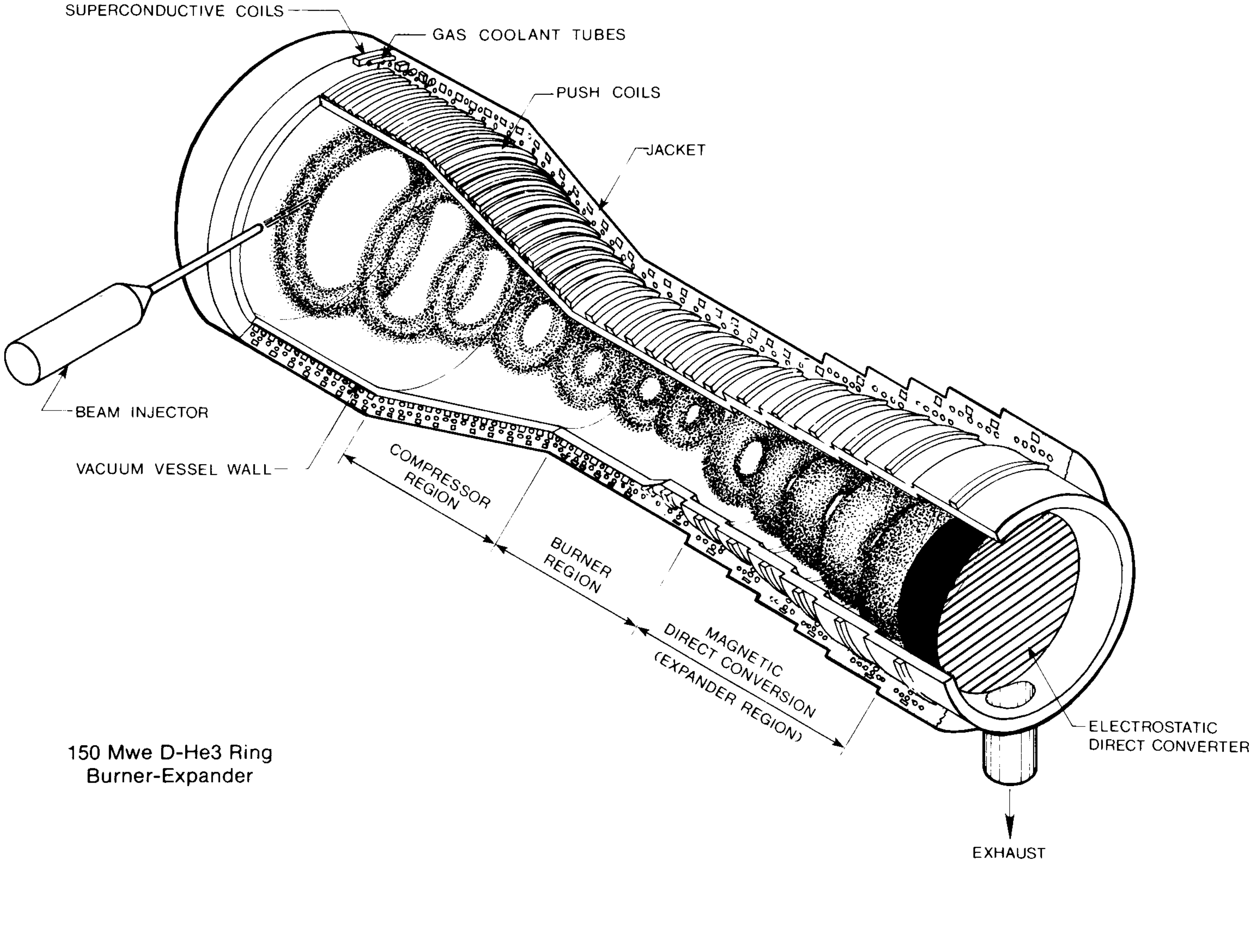
of H and He isotopes with minimum breeding in blanket

j/i	j	T/n	k_{min} (=j-1 or 0)					k_{max} = j+1				
			kev	%*	PD*	MR	%ch	kev	%*	PD*	MR	%ch
For i+j=.25		0(.75T)										
0	0		11.70	49.0	17	15	6	30.04	80.2	52	16	56
.125	.028		11.18	49.3	16	15	3	30.04	81.1	52	16	56
.25	.05		10.78	49.8	16	15	3	30.04	82.0	53	16	57
.5	.083		10.17	50.3	15	15	0	30.04	83.2	54	16	59
1	.125		9.40	51.3	14	15	-4	30.04	84.8	55	16	60
For i+j=.5		0(.5T)										
0	0		16.10	41.1	20	15	10	34.45	72.5	52	16	50
.125	.056		15.07	41.2	19	15	9	34.45	74.3	52	16	51
.25	.1		14.26	41.2	18	15	7	34.45	75.6	53	16	53
.5	.167		13.03	41.4	17	15	5	34.45	77.8	55	16	55
1	.25		11.51	41.6	15	15	0	34.45	80.4	56	16	57
For i+j=1		0										
0	0		24.89	33.6	25	15	13	43.24	61.8	52	17	42
.125	.111		22.85	33.1	22	15	11	43.24	64.6	55	17	45
.25	.2		21.22	32.6	20	15	8	43.24	66.9	56	17	47
.5	.333		18.78	31.7	18	15	5	43.24	70.3	59	17	50
1	.5		15.71	30.1	14	15	-2	43.24	74.6	63	17	54
For i+j=2		.33										
0	0		47.28	25.2	34	15	14	65.63	46.1	54	18	31
.125	.222		43.21	23.7	29	15	11	65.63	50.4	59	18	35
.25	.4		39.94	22.4	25	15	9	65.63	52.8	62	18	37
.5	.667		35.04	20.1	20	15	5	65.63	57.3	67	18	42
1	1		28.93	15.9	13	15	-2	65.63	62.9	74	18	48
For i+j=4		.60										
0	0		92.06	20.6	42	19	11	110.41	33.8	54	19	22
.125	.444		83.91	18.7	35	19	9	110.41	38.2	61	19	26
.25	.8		77.38	16.9	29	19	6	110.41	41.8	67	19	30
.5	1.33		73.71	20.8	34	19	9	110.41	47.1	75	19	35
1	2		73.71	30.8	50	19	19	110.41	53.8	86	19	42
For i+j=8		.78										
0	0		181.62	18.2	48	23	9	199.97	25.7	54	24	14
.125	.89		165.31	16.1	38	23	6	199.97	30.6	65	24	19
.25	1.6		163.27	19.8	47	23	10	199.97	34.5	73	24	23
.5	2.67		163.27	27.1	64	23	17	199.97	40.5	86	24	29
1	4		163.27	36.1	85	23	26	199.97	47.8	101	24	36

	kev	%*	PD*	MR	%ch
For i = very large (50-50 D-T): T/n=1	22.39	15.7	1200	45	15
For i = 0, j = 0, k = 0 (all D-D): T/n=0	7.30	66.4	20	15	17
For i=1, j=0, k=1, (Catalyzed D): T/n=0	43.24	61.8	58	17	44
For k = very large (50-50 D-He3)	18.35	100.0	100	25	75
For i=0, j=1, k=0(Ideal He3 producer): T/n=0	6.54	62.5	18	15	10
For p-B_{11}	8.7	100.0	13	40	-210

150 Mwe Reactors
Blanket/Shield Comparison





150 Mwe D-He3 Ring Burner-Expander

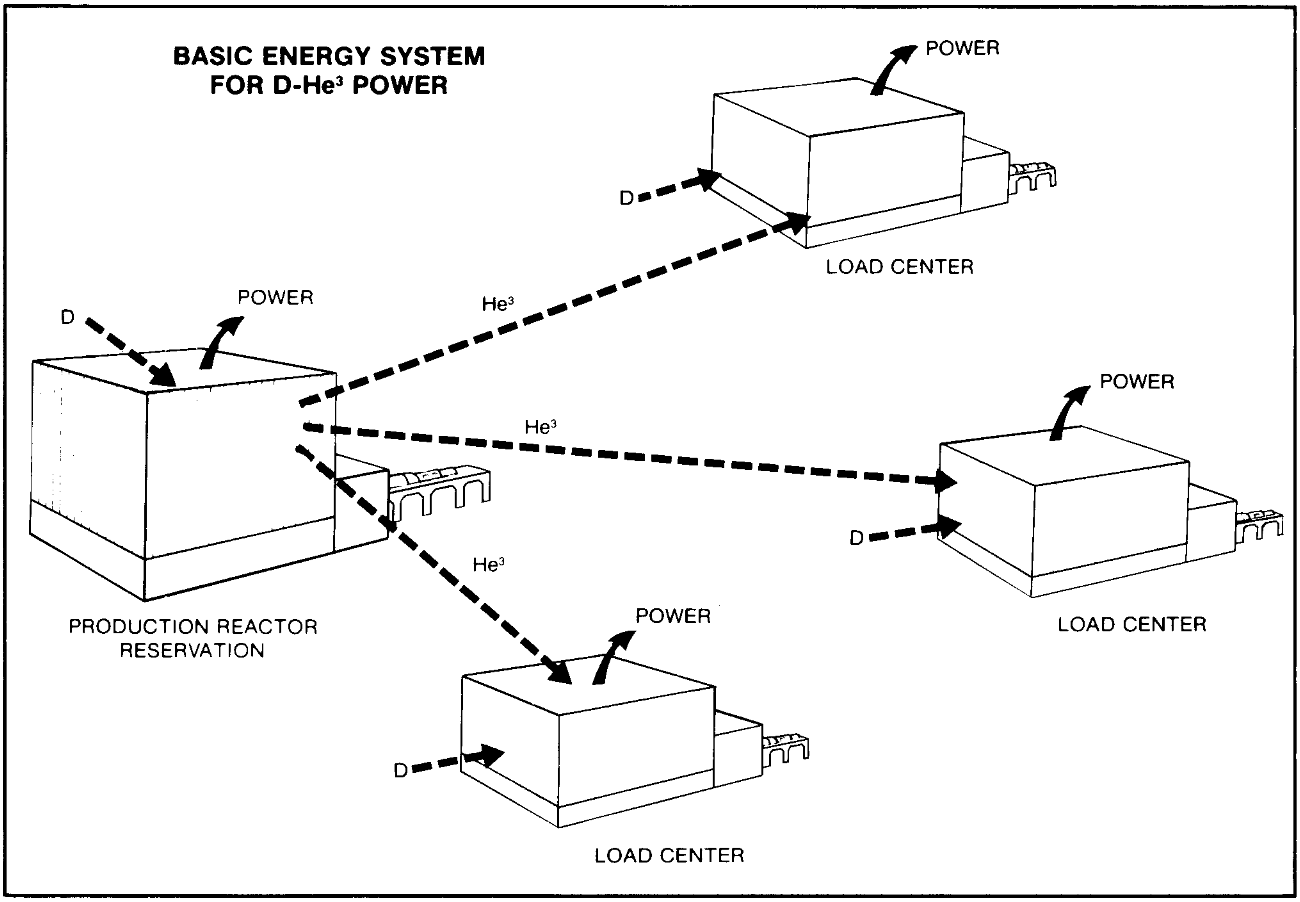


TABLE I

USER ADVANTAGES *(compared to fission plants)*

	1. DT	2. DT	3. SYMBIOTIC	
	Large Complex	Small Simple	DHe³ Small Simple	All-D Remote He³ producer
1. Projects easy to justify, etc.	○	□	■	□
2. Economic small	○	□	■	○
3. Short project schedule	○	□	■	□
4. Site widely	○	○	■	○
5. Plant cooling requirements	○	□	■	□
6. Licensing, etc., requirements	○	○	■	□
7. Radioactivity release concerns	○	○	■	□
8. Waste product concerns	□	□	■	□
9. Misuse, proliferation concerns	○	○	■	□
10. Fuel concerns	■	■	■	
11. Commercial soon	○	?	?	○

■ Definite advantage ○ No advantage
□ Some advantage ? Maybe

ADVANCED-FUEL FUSION SYSTEMS

The D-^{3}He Satellite Approach

(The ILB Reactors*)

by

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ABSTRACT

A system of ^{3}He-generator--D-^{3}He satellite fusion power plants is proposed. Either high-breeding-ratio D-T reactors or D-D reactors would serve as generators while several different concepts (D-^{3}He fueled tokamaks, bumpy tori, and field-reversed mirrors) are considered for satellites in order to offer satellite power levels ranging from ∿3GW(e) to 10MW(e). The satellites, due to their relative cleanliness and flexibility in size, could be located near the user while generators would be in more remote "parks." This requires shipment of ^{3}He, but this appears to be safe and straightforward. The freedom to design the satellite blanket without concern for tritium breeding makes simplifications possible which are essential for small plants where the "economy of size" no longer applies.

*Work sponsored by the EPRI. The results described represent a cooperative effort by the University of Illinois (plasma engineering), Lawrence Livermore Laboratory (injectors and direct convertors), and Brookhaven National Laboratory (blankets, magnets, and overall system).

I. Introduction

It is commonly agreed that an ultimate goal of fusion R&D is the achievement of D-D and/or p-^{11}B fusion in order to capitalize on an inexhaustible fuel supply (D, p, and ^{11}B), non-breeding blankets, and relative "cleanliness." The key question, then, is the optimum route to this goal. The approach presently followed by ERDA in its planning is to first develop a D-T reactor economy; D-D reactors would be introduced (if competitive) as later generation reactors followed, perhaps, by D-^{3}He and p-^{11}B reactors. This approach places almost total emphasis on D-T reactors while deferring advanced fuel studies until later. In contrast to this philosophy, the objectives of the present investigation were two fold: 1) to consider the possibility of an alternate approach; namely, demonstrate "scientific breakeven" with D-T, but if technology permits, then go directly to D-D or other advanced-fuel reactors, and 2) to perform a preliminary evaluation of the relative advantages of D-D and D-^{3}He reactors and identify key developmental problems.

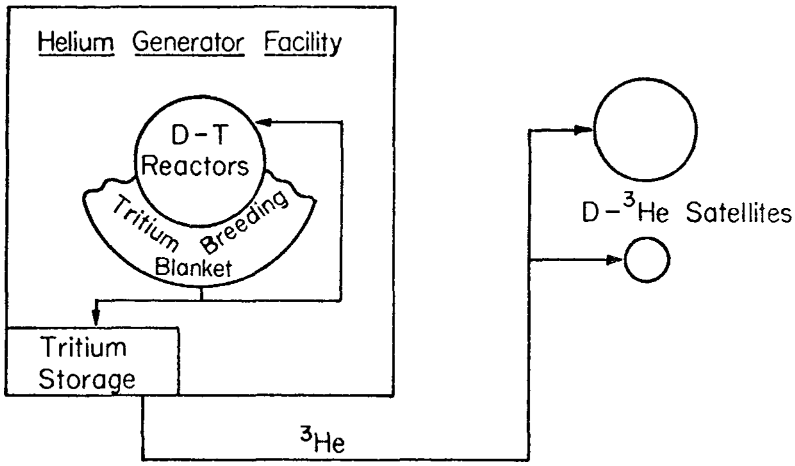
II. The D-^{3}He Satellite Approach

Since a bulk of present ERDA funding is directed at the mainline D-T tokamak approach, we have assumed that successful tokamak operation is demonstrated. Then, the three *Routes* indicated in Fig. 1 can be considered.

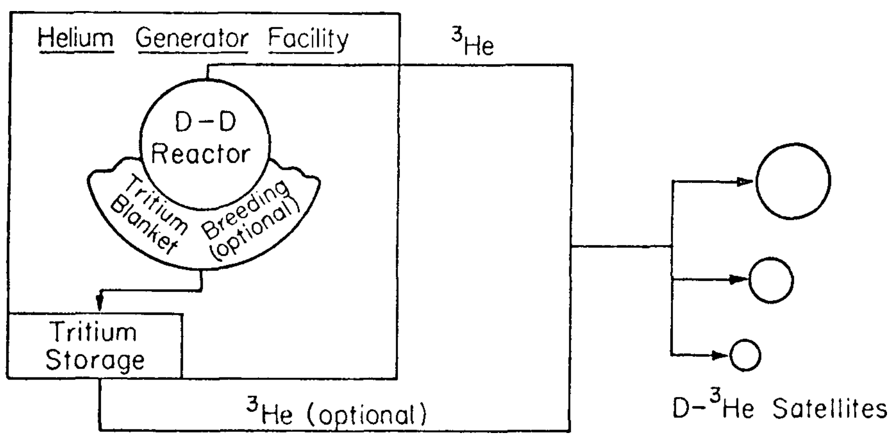
Assumption - mainline tokamaks achieved. Then -	
Route I.	D-D reactor system.[1] Reactors feasible, *but* large (>10 GWth) units required *unless* β raised above present 10% limit to $\sim$ 30%. With latter, 3-5 GW(th) plants possible.
Route II.	Conventional D-T reactor as generator for D-^{3}He satellites. *Advantage:* retain base D-T technology, design simpler D-^{3}He satellites first; phase D-D in later.
Route III.	Use D-D generator--D-^{3}He satellite system. Large generator stationed in remote reactor park, smaller, cleaner satellites near consumer. Variety of satellites feasible. *Advantage:* extensive system flexibility, relative overall "cleanliness".

Figure 1. *Routes* Considered

Indeed, the present study suggests that a D-^{3}He *satellite* system could be quite attractive. As suggested in Fig. 2, to achieve this, D-^{3}He satellite



Route II: D-^{3}He Satellites Initially Use D-T Generator (With optimum breeding, up to 1 MW satellite power per MW generator is possible.)



Route III: Satellites with D-D generator (1 MW satellite/MW generator possible without using tritium blanket; up to 4 MW/MW generator with a blanket)

Figure 2. Satellite Approaches

concepts could be developed simultaneously with D-T reactors. The satellites could be first run on ^{3}He from federal supplies stored at reservations like Savannah River. Later they would be fueled by ^{3}He from excess tritium bred by the D-T reactors. D-D generating power plants could be gradually "phased-in" to replace D-T generators.

This scenario has the advantages of 1) gaining the siting flexibility and relative "cleanliness" of D-^{3}He satellite plants early in the development of fusion power when D-T plants may face considerable environmental opposition, and 2) following a "natural evolution" in that D-^{3}He reactors seem to represent the smallest departure from D-T technology whereas D-D fusion appears to be yet more difficult.

A prime objective of the development of D-^{3}He satellites is to offer flexibility in size and energy split such that these reactors can fill a variety of tasks and siting assignments. While it may ultimately be possible to do this with a single reactor concept, the present approach has been to consider, as indicated in Fig. 3, four different concepts.

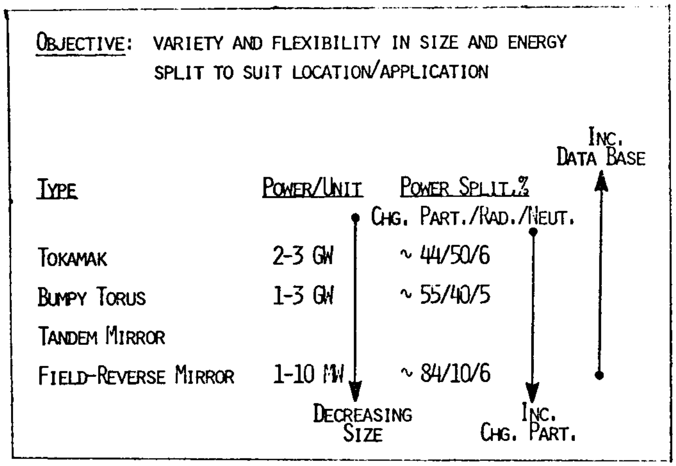


Figure 3. D-^{3}He SATELLITE CONCEPTS

The contrast in sizes is illustrated in Fig. 4 where "top view" sketches of the various reactor are compared with a deuterium-fueled generator (in turn comparable to D-T plant design such as the UWMAK series).

In the following sections we will briefly consider Tokamak, Bumpy Tori, Tandem Mirrors, and Field-Reversed Mirrors as D-^{3}He satellites. As illustrated in Fig. 3. This is in order of decreasing size and increasing risk in the sense of less experimental data base.

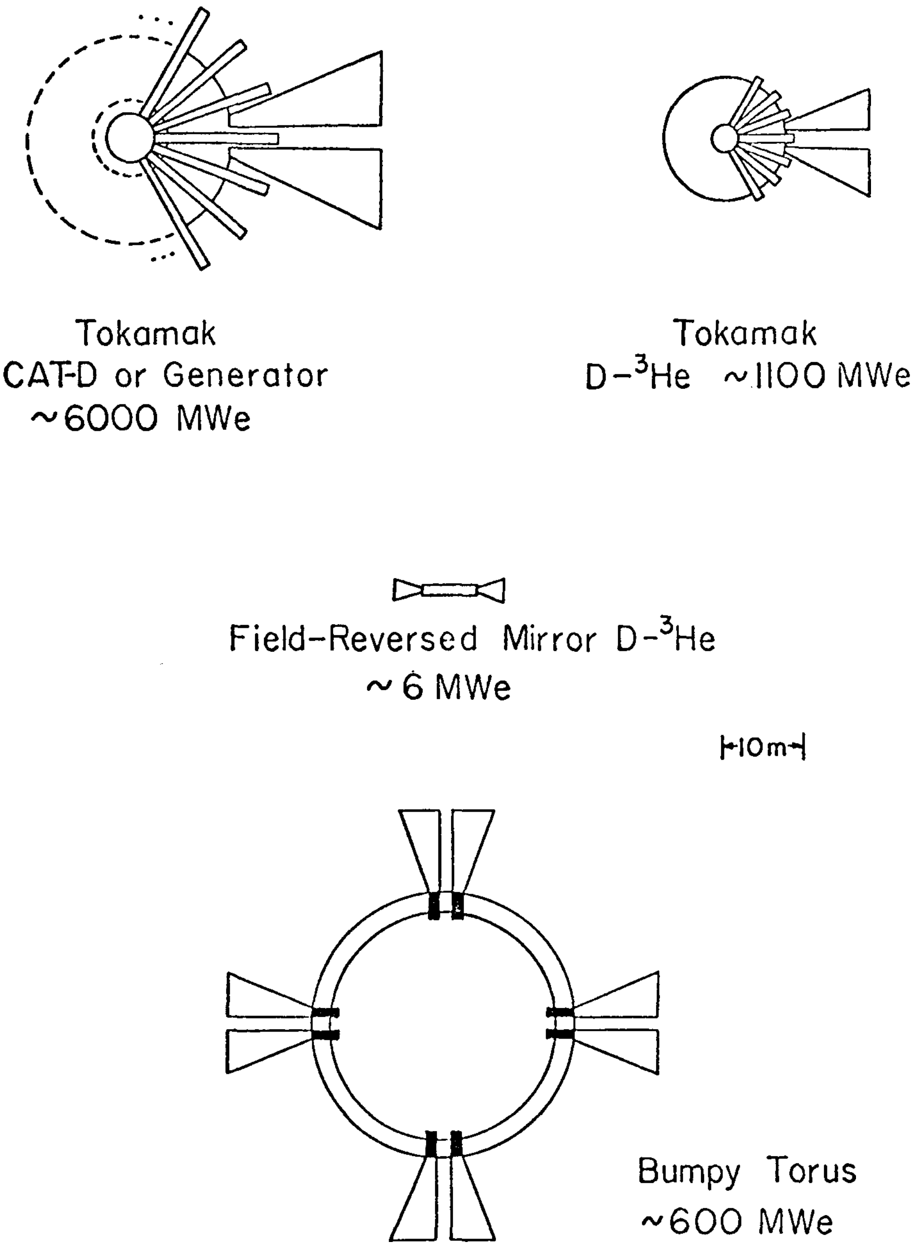


Figure 4. Comparison of "top views" of various reactor

II-(1) D-^{3}He Tokamaks*

Typical parameters for D-^{3}He fueled tokamak concepts are given in Table 1. This alternative offers a modest size device in the few GW range. A key advantage is that it represents a fairly straightforward extension of mainline tokamak technology. The low-beta design (β~10%) of Table 1 follows currently accepted β and $n\tau$ scaling, although admittedly many uncertainties in the scaling still exist.

* Further details concerning this and following satellite designs are presented in companion papers in this proceedings--see References 9-15.

TABLE 1

D-^{3}He Tokamak Power Reactors

	Low Beta	High Beta	High Beta-Ultra Clean
Major Radius, m	8.2	7.6	7.8
Minor Radius, m	2.7	2.5	2.6
Toroidal Magnetic Field, T	7.0	5.5	5.5
Electric Power, GW	1.1	1.3	1.1
Est. Plant Efficiency, %	55.0	65.0	55.0

It is encouraging that β is not so crucial to this design as for catalyzed-D reactors.(2) Thus, if 30% beta can be achieved (High-beta case), e.g. via flux-conserving methods, a lower magnetic field and somewhat smaller device are possible, but the low-β case remains acceptable.

The power split for this reactor is indicated in Fig. 5.

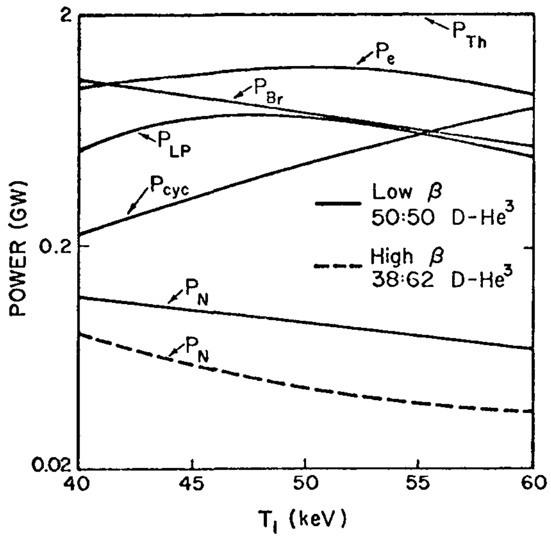


Figure 5. Power splits for D-^{3}He tokamak reactors vs. ion temperature: N, neutrons; cyc, cyclotron; LP, leaking plasma; Br, bremsstrahlung; e, electric; and th, thermal power

With a 50/50 D-^{3}He mixture, the neutron power fraction is generally less than 10% while radiation represents ~50% of the output power. Leaking plasma comprises the remainder or ~40% of the output power.

In effect, compared to D-T reactors, the neutron power has been reduced by a factor of 12 whereas radiation is increased by about 4 times. As illustrated in Fig. 5, neutron power could be reduced even further by use of ^{3}He-rich mixture (38/62: D/^{3}He - approximately the richest possible with ignited systems). This variation is probably not so important, however, as it is to consider the implications of the base case. First, as described in later sections, this design is compatible with a terphenyl-cooled, aluminum-structure blanket(3) that recovers the radiation power through a 37% efficient steam cycle, potentially offers > 30 year blanket lifetime, and provides excellent maintainance capability, including limited "hands-on" access. Further, efficient recovery of the leakage-plasma power appears possible using the bundle-divertor--direct-collector concept(3-5) developed earlier for catalyzed-D tokamaks. These various features would indeed provide a very attractive satellite.

II-(2) D-^{3}He Bumpy Torus

The Electron-Ring-Stablized Bumpy Torus (BT) potentially offers an attractive D-T reactor due to its high-β (30-50%), steady-state operation (no transformer driven currents), and ease of maintainance associated with an inherently large aspect ratio. These same advantages carry over to a D-^{3}He satellite version(6), and as indicated in Fig. 6, may even carry more significance.

- High-β scaling more firm than flux conserving tokamak.
- Neutron wall loading in D-T BT forces lower-β, hence reduced power density. D-^{3}He only β limited, not wall loading.
- Thin blanket in high aspect ratio device maximizes use of magnetic energy, reducing costs.
- Large aspect ratio of bumpy torus allows modular construction, and easier maintainance, which can be capitalized on with D-^{3}He.
- Toroidal divertor easily incorporated, improving direct convertor coupling.

Figure 6. *Why the Bumpy Torus (BT) is Well Suited to D-^{3}He*

A preliminary conceptual design suggests a D-^{3}He BT in the 1-3 GW range could be achieved using present scaling assumptions. For perspect[illegible] a cross section of this satellite is compared with the ORNL/DT bumpy torus reactor design and the UWMAK-I design in Fig. 7.

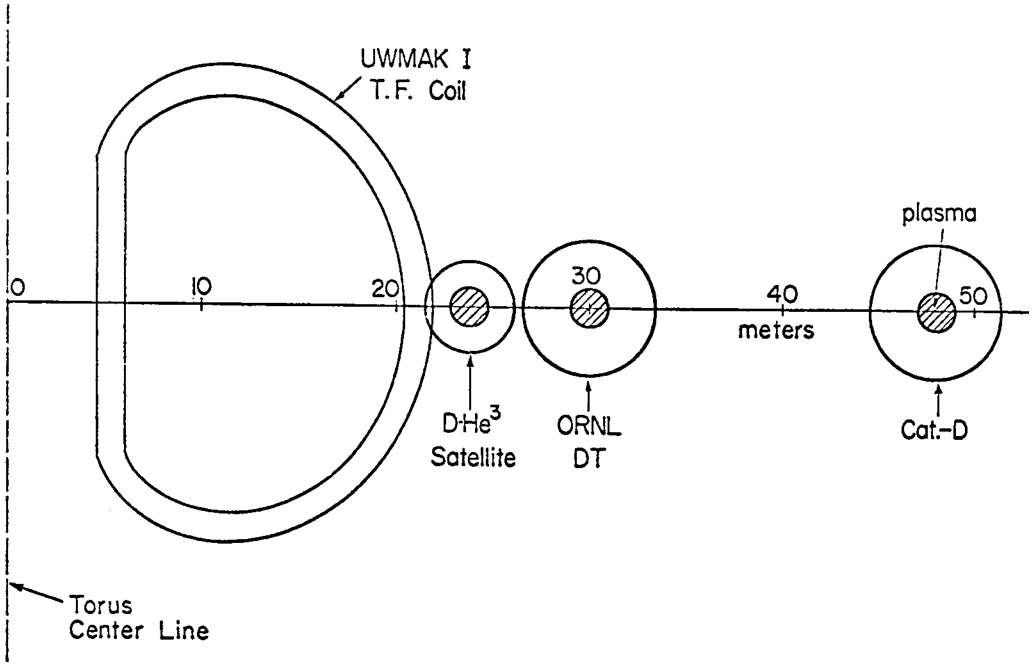


Figure 7. Comparative cross sections for several reactor concepts

The two BT's have comparable characteristics although the D-^{3}He version is slightly smaller. Both BT's display "bicycle-tire" type profiles as compared to the fat donut profile of the tokamak, and this in turn eases access and maintainance problems. This openness combined with the mirror-like field configuration also eases some of the problems of coupling a bundle-divertor--direct collector to a tokamak where it is difficult to exit through the system of shaping, vertical, and toroidal field coils. (see Fig. 8)

In one sense, the BT can be viewed as a substitute for a high-β tokamak. The ability of a BT to achieve 30-40% β appears to be on a much firmer ground than for a tokamak where ballowing modes may still limit β to order of 10%, even using flux-conserving methods. There are other speculative features of a BT, however, that makes its projected risk quite high. Planned experimental work on Bumpy Tori may clarify these features in the near future, however.

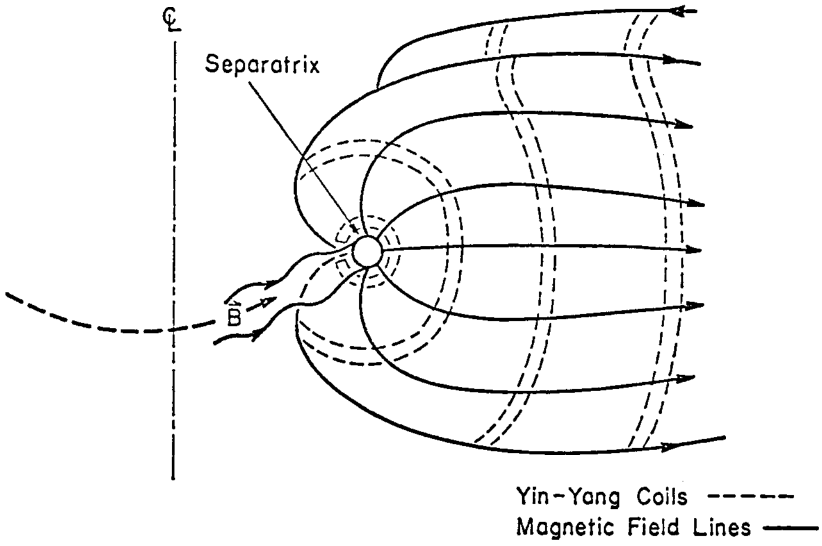


Figure 8. Schematic illustration of coupling a modified bundle divertor to the bumpy torus.

II-(3) Tandem Mirrors

The tandem mirror (TM), due to its electrostatic confinement properties, appears to be attractive for advanced fuels. Indeed, G. Logan at LLL has considered deuterium-fueled TMs (see paper, these proceedings). A rough extrapolation to D-^{3}He satellite suggests that this could offer an important option that falls between the BT and Field-Reversed Mirror in size. More detailed studies of this possibility are in progress.

II-(4) D-^{3}He Field-Reversed Mirror

The D-^{3}He Field-Reversed Mirror (FRM) potentially offers a unique satellite option because of its small size (few MW range), high-β ($\beta \sim 1$), and ease of coupling with a direct collection unit. These and other points, including some problems, are outlined in Fig. 9. The unique field configuration of the FRM, illustrated in Fig. 10, combines a closed toroidal region with a natural divertor action along the open lines. The toroidal region is created by a combination of injected and diamagnetic plasma currents, eliminating the need for actual toroidal magnets. It is thought that the entire plasma can be contained using simple circular magnets for the external field shaping. If so, this greatly simplifies magnet construction and costs compared to minimum-B type mirror coils, e.g. Yin Yang designs.

- Small size and relative cleanliness provide unique possibility for *both* early demonstration unit and localized satellite employment with minimial capital investment
- Stacking of units possible for larger output power
- Scale-up required beyond 2X-II and MFTR experiments modest, simplifying development
- $\beta \sim 1$ controls radiation losses despite relatively high temperatures

Problems

- feasibility yet to be verified experimentally (2X-II close!)
- stability requirements could restrict size to units where cost/kW (e) is high
- scaling, plasma modeling most uncertain of satellites considered

Conclusion: Advantages so significant that serious consideration merited despite uncertainties.

Figure 9. *Advantages of D-^{3}He Field-Reversed Mirror*

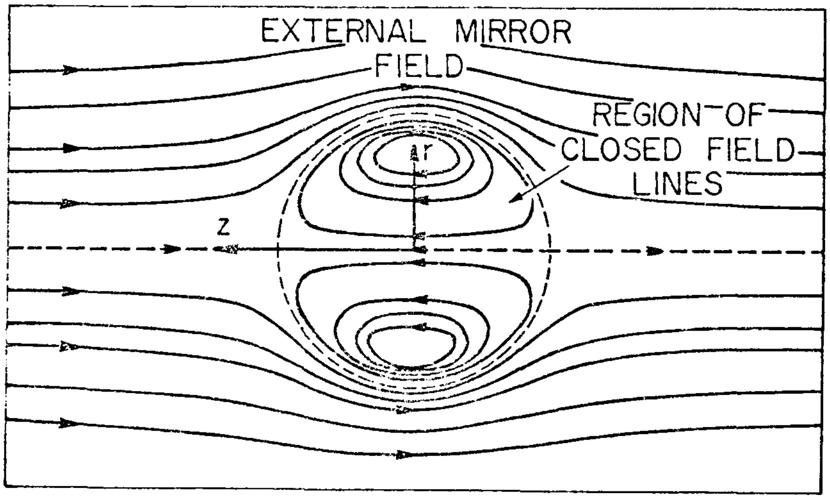


Figure 10. FRM Field Geometry

Typical parameters for a D-^{3}He FRM are given in Table 2, and some indication of the size and layout of such a reactor can be gained from LLL's design in Fig. 11.

TABLE 2

Typical D-^{3}He Field-Reversed Mirror

Vacuum Field	6 Tesla
Injection Energy	600 KeV
Plasma Volume	120 litre
Fusion Power	1.7 MW
Overall Efficiency*	$\sim$ 30%

*Inj, direct conv, and thermal effs = 80, 60, 40% respectively

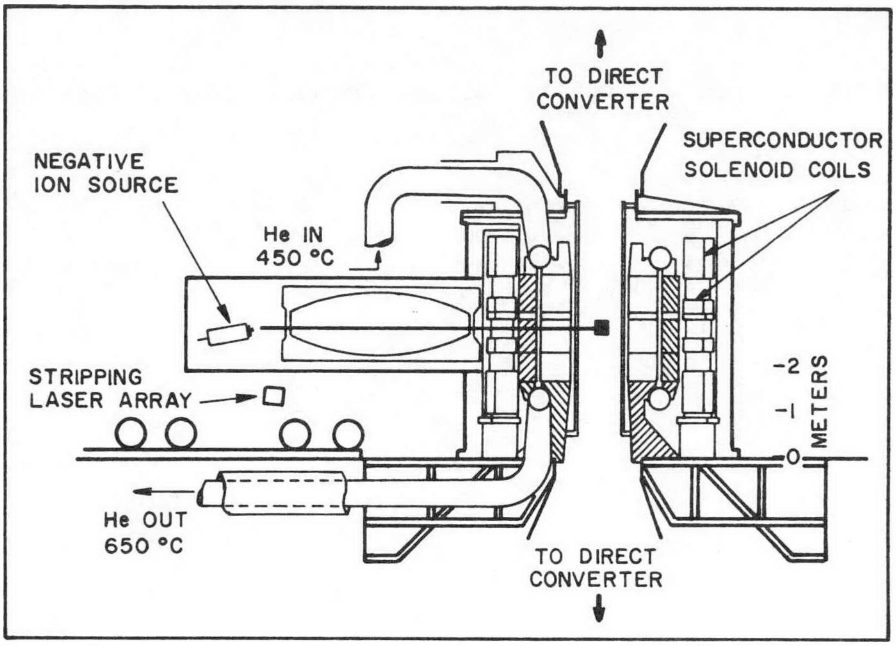


Figure 11. LLL concept of single-cell field-reversed mirror reactor using D-T. The D-^{3}He reactor would be similar but with a simplified blanket. Units could be stacked to obtain larger power levels.

The exceptionally small size of the D-^{3}He FRM suggests that this satellite could play a very important role in the development of fusion in general and advanced-fuel fusion in particular. The relatively small capital cost makes it an attractive candidate for early demonstration reactors. Initially, ^{3}He fuel would be purchased from government production facilities such as Savannah River. Later it could come for excess tritium bred in D-T reactors, and finally from D-D breeder reactors. Simultaneously, other sizes and types of satellites could be phased in.

III. ^{3}He Generator Facilities

It appears desirable to have reasonably large generator facilities in order to supply a larger number of satellites. With the generator being located in a special "plant park," the size should not present siting problems while advantage could be taken of the economy of scale, i.e., reduced \$ /MW(e) and \$ /gm of ^{3}He. It is perhaps a coincidence that deuterium-fueled reactors tend to be naturally largely due to the strenuous $n\tau$ requirements and the relatively low power density afforded. Considerable work along these lines was done earlier in this study in connection with catalyzed-D Tokamaks(1,2,9,10)(Catalyzed-D refers to burning the product tritium and ^3He in place to gain added fusion energy release in the plasma.) Some results from these studies are shown in Table 3 for Tokamak and Bumpy Torus reactors. (The actual generator would differ slightly from those shown since as much ^3He would be extracted as possible.) It is not clear that these designs are optimum but it is significant that reasonably attractive designs can be projected using currently accepted scaling.

TABLE 3. Generator Facilities

Basis: "partly" catalyzed-D
reasonable size (no. satellites > breeders)

Candidates: Tokamak (30% β if possible)
Bumpy Torus (BT)
Tandem Mirror (TM)

Some parameters*:

	10% β Tokamak	30% β Tokamak	40% β (BT)
Major Radius, m	13.2	10.6	48.0
Minor Radius, m	4.4	3.5	1.0
Ion Temp., keV	45	45	50
Toroidal Magnetic Field, T	7.7	4.7	7.4
Electric Power, GW (e)	6.5	1.8	4.1

*Cat-D designs - optimized generators may differ somewhat

In cases shown in Table 3, the power plants operate in the ignited mode, allowing attractive energy production concurrent with breeding.

IV. Blanket and Energy Conversion Considerations

As outlined in Table 4, both the D-^3He satellites and deuterium generators offer a number of advantages in such systems beyond the plasma. As indicated in Fig.12, the elimination of tritium breeding requirements combined with reduced neutron loadings make improved blanket designs possible. This could be an important feature often overlooked in considering advanced-fuel reactors. While plasma conditions are more difficult to attain than for D-T, the blanket design is simplified. Depending on the severity of radiation damage and maintainance problems in D-T reactors, this could reduce the development time required for this end of the plant (vs. more R&D for plasma heating and confinement).

TABLE 4. *Blanket, Magnet, and Energy Conversion Considerations*

	Cat-D (Generators)	D-^{3}He (Satellites)
Blanket	Graphite - internal radiation trap High-press He coolant, SiC tubes 1.1 to 1.4 MW(th)/m^2 > 30 yr. life (est.)	Aluminum - internal passages Low-press terphenyl coolant $\sim$ 0.7 MW(th)/m^2 > 30 yr. life (est.)
Thermal Cycle	800-1000^{o}C He/steam cycle: $\sim$38-40% eff.	400^{o}C organic/steam cycle: $\sim$37% eff.
Direct Conversion	Bundle Divertor/Venetian Blind Collector: $\sim$60% eff.	Depends on type satellite, e.g. multistage, > 60% eff.
Operation and Maintenance	Good	Excellent, including limited "hands-on" access.
Magnet Systems	"Deals" concept reduces penality for higher fields in Tokamaks	Depends on type satellite

- Minimum radiation damage - either via low neutron loads (for D-^{3}He) or ability to use non-breeding graphite structure (for D-D) capable of annealing at high temperature.
- Extended lifetime eliminates replacement (except possible unexpected failure), reduces down time, and provides cost savings.
- Ability to effectively use low activity materials (e.g. Al, SiC, and C) minimizes induced radioactivity, aiding maintainance. Cost reduction also possible, scarce materials (Be, Cr, etc) eliminated.
- Reduced thickness possible gives relative reduction in magnet size, i.e. cost.
- Elimination of tritium processing circuits and leak paths possible. No tritium storage needed.

Figure 12. *Key Advantages Confirmed for Blankets*

IV-(1). RADIATION POWER RECOVERY

A unique feature of both deuterium-generating power plants and the satellites is the relatively large radiation power produced. Assuming that the first-wall surface-temperature can be held in bounds, this is, in fact, a convenient form for energy recovery. Compared to neutrons, the radiation can be absorbed in a much thinner blanket. This, in principle, allows a compact, relatively high power-density and temperature blanket. In addition to reduced blanket costs, this allows use of smaller magnet structures.

The uniqueness of such blankets is illustrated in Figs. 13 and 14. The first shows a modular graphite construction designed at the BNL for use with the deuterium generating facilities.[3] A reasonably uniform temperature is maintained using the void regions (traps) through which heat is radiatively transported and absorbed in the SAP tubes containing helium coolant. This arrangement, using a 800-1000^{o}C He/steam cycle allows thermal efficiencies in the 38 to 40% efficiency range. The use of low induced-radioactivity materials combined with modern construction greatly simplifies maintainance.

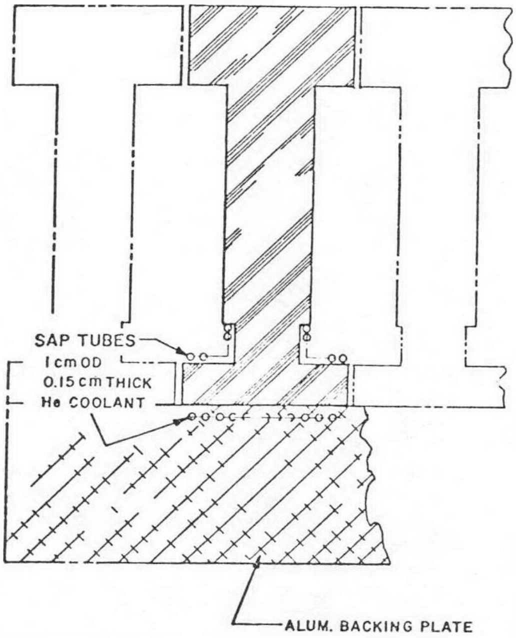


Figure 13. Radiation trap design enables efficient extraction of radiation energy hitting first wall of graphite-blanket module.

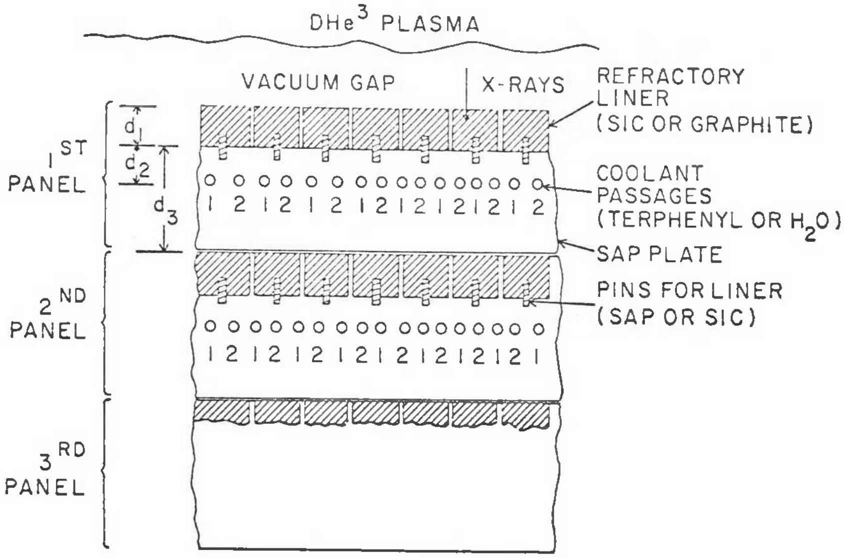


Figure 14. Aluminum Blankets for DHe^3 Reactors (Cross-Section Sketch).

A unique terphenyl cooled, aluminum blanket for the D-^{3}He satellites, also designed at the BNL(3), is shown in Fig. 14. This design offers the advantages of simplicity, low pressure operation, reasonable thermal efficiency (~37%), and excellent maintenance capability, including limited "hands-on" access. For normal operation, the blanket lifetime is expected to exceed 30 years in both toroidal and mirror-type advanced-fuel reactors.

V. SUMMARY AND CONCLUSIONS

The standard arguments against considering D-D or D-^{3}He reactors at this time are that the plasma temperature requirements are much higher than for D-T and the power density much lower, forcing uneconomically large systems. However, as outlined in Table 5, the present study indicates that these difficulties may not be as formidable as first thought.

Thus, to achieve the added heating required without excessive external power requirements, a matchhead (D-T core) ignition scheme illustrated in Fig. 15 appears to offer a practical approach(8). (Further details of startup studies are discussed in Ref. 11.) While $n\tau$ scaling laws must ultimately rest on experimental data, the concepts described here achieve adequate $n\tau$ for ignited operation (except for the TM and FRM which don't require full ignition) based on current knowledge. Careful selection of high-β operation helps

"Observable"	Actual Problem	Potential Solution
High temperature	Large energy/power for start up	Matchhead (D-T core) ignition
	Increased $n\tau$	Adequate for concepts selected
	Increased radiation losses/wall loading	High-β concepts "control losses--high-temperature wall design advantageous for thermal conversion
Low power density	Neutron wall loading, not power density, limits D-T	Reduced neutron loading allows maximum power density, competitive wall loadings

Table 5. Why are D-D/D-^{3}He thought more difficult to use than D-T?

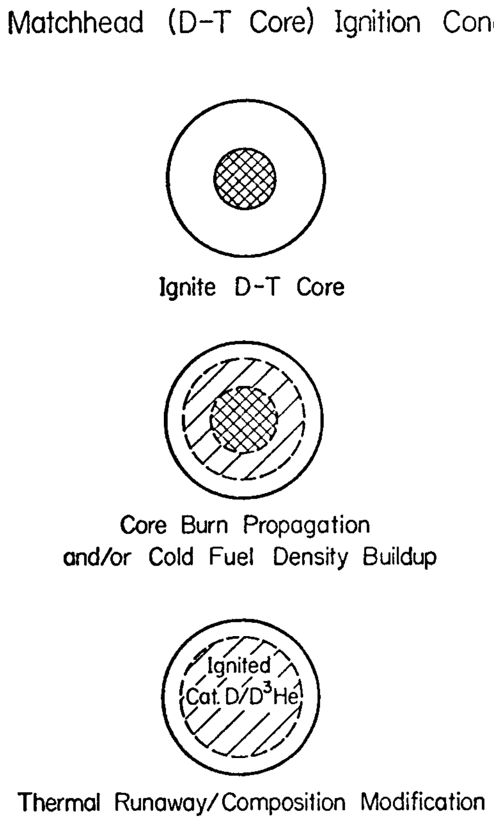


Figure 15. Matchhead (D-T Core) Ignition Concept

control radiation losses, and with the reduced neutron wall-loading, reasonably competitive power densities, hence sizes, seem feasible.

These "difficulties" must be weighed against the simplification in blanket engineering afforded with the elimination of tritium breeding. Further, the improved maintenance capability, especially with D-^{3}He devices, represents a very important gain. The reduction waste heat via improved energy conversion, including direct conversion, represents an important advantage in siting.

Such considerations suggest that advanced-fuel fusion reactors may play a vital role, even in the relatively "near-term" development of fusion. The present study has considered various aspects of this role, and the conclusions reached thus far are summarized in Fig. 16.

- The mose attractive scenario appears to be the D-D generator--D-^{3}He satellite system. This approach:
  - mitigates potential large-size of D-D reactors
  - capitalizes on relative "cleanliness" and flexibility of satellite D-^{3}He power plants
  - allows early, step-wise development of satellites using ^{3}He from "excess" tritium generators in D-T reactors
  - places emphasis on D-^{3}He satellites which represent smallest step past D-T in plasma technology and even simplier blanket/engineering technology
- Confirm advantages in non-breeding blankets including extended lifetimes, easier maintainance, efficient thermal cycles.
- Confirm direct conversion can provide an important reduction in rejected heat load for satellites and, to a lesser extent, for generators.
- Capital and operating cost differences are complicated and require more study.

Figure 16. Conclusions to date

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7. G. H. Miley and D. Driemeyer, "Approaches to Small-Size Field-Reversed Mirror Reactors," *Trans. Am. Nuc. Soc. 26*, 53 (1977). Also see *Bult. Am. Phys. Soc., Ser. II, 21*, 1161 (1976).

8. R. Stark, G. Gerdin and G. Miley, "Start-Up of a Catalyzed-D Tokamak," *Trans. Am. Nuc. Soc. 24*, 51 (1976).

In addition, more detailed discussions of the satellite concepts are presented in the following papers in these proceedings:

9. F. Southworth, "D-^{3}He Fueled Tokamak Reactor Designs," this proceedings.

10. F. Southworth, "Catalyzed-D Fueled Tokamak Reactor Designs," this proceedings.

11. G. Gerdin and R. Stark, "Start-Up of Advanced Fuel Tokamak," this proceedings.

12. G. Gerdin and F. Southworth, "Advanced-Fuel Bumpy Tori," this proceedings.

13. G. H. Miley and D. Driemeyer, "The D-^{3}He Field-Reversed Mirror as Minimum Size Satellites," this proceedings.

14. J. R. Powell, "Reactor Technology-Power Conversion Systems and Reactor Operation and Maintenance," this proceedings.

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Fusion Blankets for Catalyzed D-D and D-He^3 Reactors

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ABSTRACT

Blanket designs are presented for catalyzed D-D (Cat-D) and D-He^3 fusion reactors. Because of relatively low neutron wall loads and the flexibility due to non-tritium breeding, blankets potentially should operate for reactor lifetimes of ∿30 years. Unscheduled replacement of failed blanket modules should be relatively rapid, due to very low residual activity, by operators working either through access ports in the shield (option 1) or directly in the plasma chamber (option 2).

Cat-D blanket designs are presented for high (∿30%) and low (∿12%) β non-circular Tokamak reactors. The blankets are thick graphite screens, operating at high temperature to anneal radiation damage; the deposited neutron and gamma energy is thermally radiated along internal cavities and conducted to a bank of internal SiC coolant tubes (∿4 cm. ID) containing high pressure helium.

In the D-He^3 Tokamak reactor design, the blanket consists of multiple layers (e.g., three) of thin (∿10 cm.) high strength aluminum (e.g., SAP), modular plates, cooled by organic terphynyl coolant.

Work performed under the auspices of the Electric Power Research Institute

I. INTRODUCTION

Blanket designs have been developed for catalyzed D-D [Cat-D] and D-He^3 fusion reactors. Because of low neutron wall loads [for DHe^3 reactors] and the elimination of tritium breeding blankets have the potential to operate for reactor lifetimes of ∿30 years and without having to be replaced on a routine basis. Further, unscheduled replacement of failed blanket modules appears unlikely because of redundancy and good neutron damage resistance. If replacement of some modules is required, this can be done relatively rapidly, due to very low residual activity, by operators working either through access ports in the shield [option 1 (Fig. 1)] or directly in the plasma chamber [option 2].

The use of advanced fuels has a number of important potential benefits for fusion reactor blankets:

. Minimum radiation damage - either because of very low neutron wall loads [i.e., in the DHe^3 reactor] or ability to use non-breeding graphite blankets, which should anneal at high temperatures [i.e., in the Cat-D reactors].

. Elimination of the need to replace blanket modules during the life of the reactor [at least on a scheduled basis - non-scheduled replacement of modules due to unexpected failure may be necessary].

. Ability to effectively use low activity materials such as aluminum, SiC, and graphite which minimize activity and personnel dose rates in the reactor, and virtually eliminate radwaste handling and storage problems.

. Rapid replacement of failed blanket modules with essentially no remote handling.

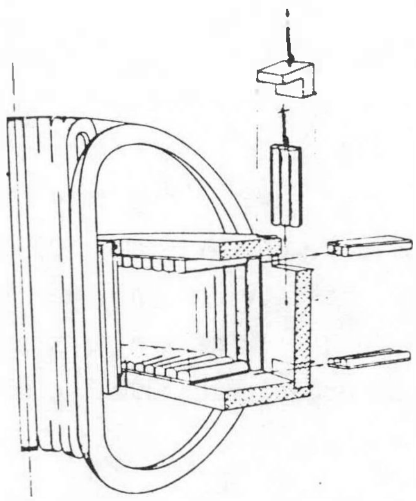

Figure 1. Schematic Showing Module Removal

. High plant availability associated with minimum need for shutdown for blanket maintenance and replacement.

. Low blanket cost.

. No use of scarce resources [Be, Cr, etc.] in blanket.

. Elimination of the tritium processing circuits in the blanket system.

. Elimination of tritium leak paths to the environment from the blanket and power conversion systems.

. Elimination of the need for tritium storage.

II. CATALYZED D-D BLANKET DESIGNS

Cat-D blankets have been designed for high [$\sim$30%] and low [$\sim$12%] β non-circular Tokamak reactors. The blankets are thick graphite screens, [Fig. 2] operating at high temperature to anneal radiation damage; the deposited neutron and gamma energy is thermally radiated along internal cavities and conducted to a bank of internal SiC coolant tubes [$\sim$4 cm. ID] containing high pressure helium. In option 1 the graphite blocks are mounted on heavy Al backing plates [cooled by He], which are supported from the fixed shield. The shield also provides the primary vacuum seal. In option 2 the graphite blocks radiate to a set of coolant tubes fixed to a separate backing plate which is also supported from the fixed shield. Non-neutron energy, i.e., bremsstrahlung, is transmitted almost completely through a thin, low thermal conductivity, low density graphite layer [either graphite felt or cloth] into the high temperature

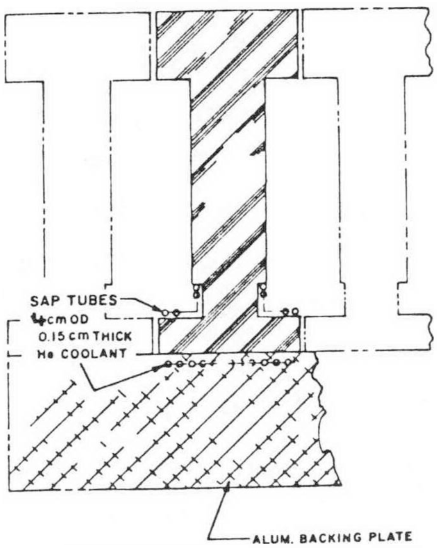

Figure 2. Graphite Blanket Module

bulk graphite blocks. Designs based on bulk graphite surfaces have also been examined.

Typical blanket sizes and weights for the Cat-D designs [option 1] are as follows: high β - The vertical inner and outer walls are ∿13 m long; each outer module is 1.5 m wide x 13 m x 1 m thick and holds 5 graphite blocks. The blocks are 30 cm wide x 13 m long x 84 cm thick [with a 16 cm thick Al plate] and have a total module weight of ∿35 Mt. Inboard modules are essentially the same. The maximum lengths of the top and bottom modules are ∿8 m. Because of their tapered width, the weights of the top and bottom modules will be less than either the inner or outer modules. There are approximately 18,000 coolant tubes in the reactor, with 20 tubes for each graphite block.

Low β - The vertical inner and outer walls are ∿26 m long. To facilitate handling and fabrication, two blanket modules are vertically stacked to form the side wall. The outer modules are similar to those for the high β design, but are somewhat larger, i.e., there are 7 blocks/module. Each block is 30 cm wide x 13 m x 84 cm thick, and the module is 2.1 m wide. The maximum lengths of the top and bottom modules are 10 m. There are approximately 35,000 coolant tubes in the reactor with 20 tubes for each graphite block. There are ∿200 modules in the reactor for both designs.

The Cat-D, high β design has a gross wall loading of 1.11 $MW(th)/m^2$ [bremsstrahlung and neutron blanket wall loading is 0.76 $MW(th)/m^2$ and the balance appears in the divertor]. Steady-state blanket temperatures are calculated by the computer code, CONRAD(1), using neutron and gamma heating distributions calculated by a 1-D 100 group P_3S_8 ANISN model. The effect of helium coolant outlet temperature from the SiC tubes is investigated for values of 800°C, 900°C, and 1000°C, with the inlet temperature fixed at 400°C. The effect of the number of coolant tubes in the graphite block is investigated so as to determine the effects on heat pickup and maximum blanket temperature. The range examined is from 16 to 24 tubes. The effect of He outlet temperature in the Al backing plate is also examined for the range of 300 to 400°C.

The maximum graphite first wall surface temperature is found to be ≤ 1800°C for all cases investigated. This temperature is well below the maximum allowable temperature of ∿2000°C established by graphite evaporation. Depending on coolant tube design, maximum helium coolant velocities range from 16 m/sec

32 m/sec, well within HTGR technology. The helium pumping power ratio [pumping power/heat pickup] is low, ranging from 0.5 to 1.0%.

For the Cat-D, low β design, the gross wall loading is 2.03 MW(th)/m^2 [bremsstrahlung and neutron blanket wall loading = 1.37 MW(th)/m^2, the balance appears in the divertor]. The same coolant temperatures range as in the high β design is assumed.

For the low β designs, the first wall surface temperatures exceed the design limit [∿2000°C] unless a low temperature radiation sink is used. A sink equivalent in area to 5% of the total first wall is found to provide a sufficient cooling area. This implies that 5% of thermal energy is not available for high temperature heat extraction; however, this energy can be used in a power cycle operating at a source temperature of 400°C. The thermal-hydraulic characteristics for the low and high β designs are summarized in Tables 1 and 2 and it is seen that they are essentially the same.

TABLE 1

Typical Thermal and Hydraulic Characteristics of the High β Graphite Blanket
[1.11 MW(th)/m^2 Gross Wall Loading]

	Maximum Coolant Temperature		
	800°C	900°C	1000°C
Number of Tubes/Block	20	20	20
Coolant	He	He	He
Inlet Temperature [°C]	400°C	400°C	400°C
Outlet Temperature [°C]	800°C	900°C	1000°C
Operating Pressure [atm]	60	60	60
Flow Rate [g/s]	135	108	90
Channel Velocity [m/s]	25.6	21.6	19.0
Heat Transfer Coefficient [W/cm^2°C]	0.139	0.122	0.103
Blanket Pressure Drop [psia]	2.2	1.56	1.12
Pumping Power/Thermal Power	.007	.004	.003

TABLE 2

Typical Thermal and Hydraulic Characteristics of the Low β Graphite Blanket
[2.03 MW(th)m^2 Gross Wall Loading]

	Maximum Coolant Temperature		
	800°C	900°C	1000°C
Number of Tubes/Block	20	20	20
Coolant	He	He	He
Inlet Temperature [°C]	400°C	400°C	400°C
Outlet Temperature [°C]	800°C	900°C	1000°C
Operating Pressure [atm]	60	60	60
Flow Rate [g/s]	136	109	91
Channel Velocity [m/s]	25.9	21.9	19.2
Heat Transfer Coefficient [W/cm^2 C]	0.141	0.118	0.104
Blanket Pressure Drip [psia]	1.24	o.883	0.672
Pumping Power/Thermal Power	.004	.002	.001

III. D-He^3 BLANKET DESIGNS

In the D-He^3 Tokamak high β reactor design, the major fraction of the blanket thermal power results from plasma x-rays [E_{AV}∿30KeV]. The x-rays essentially stop on the surface of the first wall, and the heat energy is conducted through the wall to a coolant system. The relatively high heat fluxes [70 w/cm^2] require that the first wall have good thermal conductivity to avoid large temperature differentials and excessive thermal stresses in the structure. Low conductivity materials such as stainless steel or titanium do not appear to be suitable. Aluminum appears to be the best choice for DHe^3 reactor blankets: it has very high thermal conductivity, low induced activity, and is plentiful and inexpensive. Since the integrated neutron wall load for the reactor lifetime [30 years] is low, no radiation damage problems are anticipated.

The blanket is formed from flat, relatively thin, aluminum sheets with internal coolant passages. Organic terphenyl coolant is preferable to helium or water, has a low operating pressure [∿100 psi], low pumping power, essential zero corrosion rate, and is stable at relatively high temperatures [$\sim$400°C] which results in good thermal cycle efficiencies [$\sim$36%]. Water would have a high operating pressure [$\sim$100 psi] but might cause corrosion problems. Thermal cycle efficiencies would be limited to $\sim$30%. Helium would have a high operating pressure [$\sim$1000 psi], high pumping power, and relatively low thermal efficiency [<30%]. The principal problem with terphenyl coolant, i.e., radiolytic decomposition by high energy neutrons and gamma rays is not of concern with DHe^3 reactors, because of the very low neutron wall loads.

The aluminum sheet blanket modules are covered with hexagonal SiC plates which are on the order of 2 cm thick. These protect the aluminum against erosion due to long-term sputtering and damage due to unexpected plasma dumps. If erosion of the SiC is excessive during the 30-year life of the blanket, the SiC plates can be regenerated in-situ by chemical vapor deposition.

The reference blanket design uses multiple layers [e.g., three] of thin [$\sim$10 cm] high strength aluminum [e.g., SAP] modular plates which operate at $\sim$400°C and are cooled by organic terphenyl coolant [HB-40 (Fig. 3]. The hexagonal SiC liner elements form the first wall. Each plate, which is typically 4 m long x 1.5 m wide, has internal coolant channels [0.5 cm x 1.0 cm] arranged in two independent redundant circuits. The multiple layer, independent circuit design gives very high blanket reliability, since leaking circuits can be valved off while the reactor is operating. If the reactor operator detects impaired plasma performance caused by a leak, he can successively shut off and

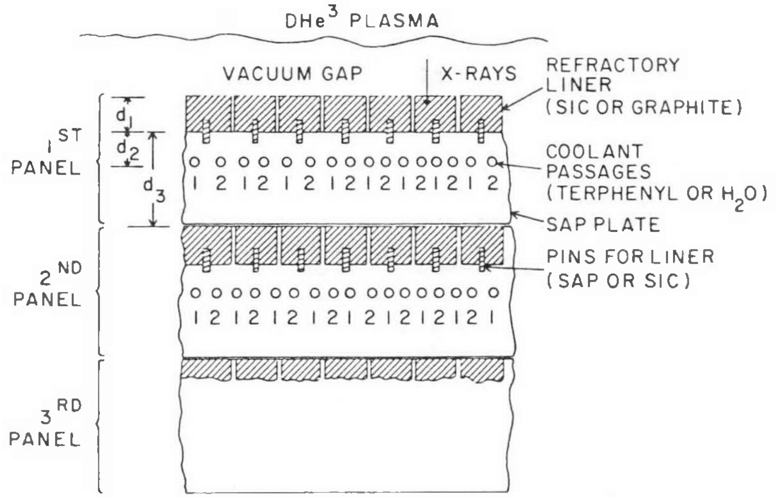

Figure 3. Aluminum Blankets for DHe^3 Reactors Cross Section

switch redundant circuits until the leaking circuit is detected, without having to shut down the reactor. Calculations of temperature distributions using the two-dimensional version of the Heating-3 conduction/convection code indicate that if one coolant circuit is shut off in a given plate, the remaining circuit can take over and the rise in plate temperature will be relatively small.

The integrated neutron wall load on the aluminum blanket is only 1.0 MW(th) yr/m^2 for a 30-year plant life [80% plant factor]. The aluminum should operate for a 30-year period, in view of the low integrated neutron wall load. If both coolant circuits on a plate should fail, which seems a very unlikely event, the reactor can be shut down and failed plate can be removed from inside the plasma chamber. The procedure is discussed in a later section. Plate replacement would not be necessary since the next inner plates can then take over the load. The inner plates will operate at a lower temperature [e.g., 50 - 100°C lower] than the plate facing the plasma. Since the thermal load of the inner plates is very small compared to the outer plate, the thermal power conversion efficiency is not significantly affected. The lower temperature should result in lower failure and pyrolytic decomposition rates.

A study has been made of the effect of various blanket parameters on blanket performance. The parameters studies include coolant velocity, coolant passage shape, nature of surface [finned vs. smooth], coolant passage size, module length, and module width. The optimum dimensions for a blanket module depend on its position in the reactor [Fig 4]. Table 3 gives the dimensions and weights for the first layer of blanket modules. Region 1 and 2 are the top and bottom halves of the outboard blanket, regions 3A, 3B, and 3C comprise the top part of the blanket, regions 6A, 6B, and 6C comprise the bottom part of the blanket, and regions 4 and 5 are the top and bottom halves of the inboard blanket. The maximum module weight is less than three metric tons, and replacement of a failed module should not be difficult.

Thermal-hydraulic parameters for a typical module are: coolant velocity, 8 m/sec; ΔP, 3 atm; inlet and outlet temperatures, 350 and 375°C; film temperature drop, 20°C; and pumping power/thermal power extraction ratio, 0.4%.

The thermal-hydraulic parameters are calculated for smooth rectangular coolant passages [0.5 x 1.0 cm] in the aluminum plate. Of the four cases

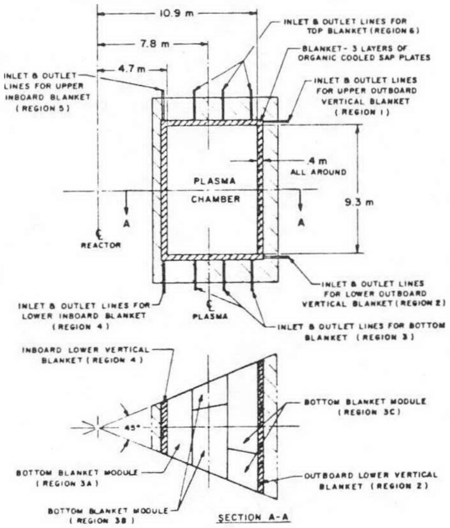

Figure 4. Organic Cooled Module Arrangement in DHe^3 Reactor

TABLE 3

Organic Cooled Blanket Module and Characteristics for DHe^3 Reactor

[First Layer of Modules]

Region	Module Length, m	Module Width, m	# of Modules in Reactor	Module Weight, kg
1	4.62	1.60	36	2595
2	4.62	1.60	36	2595
3A	3.70	2.04	8	2650
3B	4.06	2.04	12	2907
3C	3.86	2.04	16	2764
4	4.62	1.21	24	1962
5	4.62	1.21	24	1962
6A	3.70	2.04	8	2650
6B	4.06	2.04	12	2907
6C	3.86	2.04	16	2764
Total:			192	

Note: -Modules have the same construction in the following regions: 1 and 2; 3A and 6A; 3B and 6B; 3C and 6C; 4 and 5. Thus five different kinds of modules must be fabricated.

-There is a 2 cm gap between modules for ease in removal.

-Average density of 2.7 for module; thickness of 13 centimeter.

TABLE 4

Thermal-Hydraulic Parameters for DHe^3 Reactors

[HB-40 Terphenyl Coolant]

Design Parameter	Smooth Circular Holes	Finned Circular Holes	Smooth Rectangular Holes	Finned Rectangular Holes
Coolant Passage Size, cm	0.75	0.75	0.5 x 1.0	0.5 x 1.0
Coolant Velocity, m/sec	12.0	11.0	8.0	8.0
ΔP for Module, atm [4 meter length]	5.2	12	3	6.4
Film Temperature Drop, °C	30	14	20	8
ΔT for Module, °C [4 meter length]	32	36	25	28
Pumping Power/Thermal Power	0.0067	0.014	0.004	0.011

studied [smooth circular holes, smooth rectangular holes, finned circular holes, and finned rectangular holes] the smooth rectangular holes were preferable. Table 4 compares thermal hydraulic parameters for the four types of coolant passages at the optimum coolant velocity for each. Although the typical parameters given above are attractive and practical, a more optimized design with some change in coolant passage dimensions is probably achievable, and this would likely involve a reduction in both film and coolant transport [inlet to outlet] temperature differences. ΔP and $(R)^{-1}$ [pumping power to thermal extraction power ratio] would probably increase somewhat. Values for organic properties have been taken from the extensive work on organic cooled CANDU reactors.(2)

Because of the low neutron wall loading [0.04 $MW(th)/m^2$], the radiolytic organic decompisition rate is quite small. For a 2 GW(th) D-He reactor [1GW(e)] the total organic decomposition rate is ∿240 kg/hour based on HB-40 decomposition rate data(2).

Approximately one-half of the decomposition results from radiolytic effects [neutrons and gammas], with the remainder due to pyrolysis. Neutron and gamma energy deposition in the HB-40 coolant is calculated using a 1-D P_3S_8 ANISN model [100 group] with ENDF/B-IV cross-sections. At $2.00/kg for HB-40, organic makeup costs are low, ∿0.5 mills/KWH.

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Magnet System for Catalyzed D-D and D-He^3 Reactors

by

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ABSTRACT

A new concept for toroidal field magnets, called the DEALS (Demountable Externally anchored Low Stress) magnet systems, has been conceived and developed for advanced fuel reactors. This concept should result in substantial cost reductions for toroidal magnets. In addition, the conductor stress is relatively low and compressive in nature, which is very important if brittle Nb_3Sn conductors are to be successfully used, since advanced fuel Tokamak reactors require Nb_3Sn conductor if reasonable power densities are to be achieved. A final very important benefit of the DEALS magnet system is its demountability. Failed magnet segments can probably be replaced fairly rapidly (i.e., in approximately a month) without requiring movement and/or dissassembly of the blanket/shield regions and poloidal coils.

Due to the high degree of non-circularity present in the plasmas of the low-β Tokamak reactor designs which are detailed in this report, a system of poloidal field coils is necessitated to provide both shape and positional stability. In addition, the uncertainty regarding the accessibility of the high-β regimes discussed may also necessitate extensive poloidal field systems. Volt second requirements are such that they must be alleviated also by this poloidal field system.

Work performed under the auspices of the Electric Power Research Institute

I. INTRODUCTION

1. Toroidal Magnet Systems

The toroidal field magnets represent an important segment of the plant cost for DT Tokamak reactors, and will be even more important for Cat-D and DHe^3 advanced fuel reactors. For comparable power densities and plasma β's, Tokamaks operating on advanced fuels will require toroidal magentic field strengths 2 to 3 times greater than those operating on DT fuel.

Before summarizing the DEALS TF magnet designs for the advanced fuel reactors, it is helpful to briefly describe the concept. The TF coil segments are straight and the coil, when assembled, has a rectangular shape. To keep stress in the conductors and coil case low, the magnetic forces on the coil segment are transferred through a set of low thermal conductivity support pads (made from laminated expoxy fiberglass and stainless steel) to an external room temperature reinforcement structure. This structure, similar to the PCRV (Prestressed Concrete Reactor Vessel) used in some fission reactors, is a composite structure of concrete and reinforcing steel. Besides serving as the structural support for the magnets, it also serves as primary containment and biological shielding. Since the latter two functions require such a structure anyway, its use for magnet support offers the potential of large cost savings, since the expensive cold steel reinforcement required for conventional coils is not needed with DEALS. Adjustable hydraulic or mechanical wedges accommodate differential thermal movements between the magnet and its support structure. Joints between coil segments can either be soldered or made through multicontact pressure plates. The thermal load at $4^{o}K$ associated with the use of non-superconducting joints is quite small and is much less than 1 kW(th) for the entire magnet system.

2. Conductor and Magnet Designs

A. Conductor Fabrication

The conductor shown in Figure 1 is used for all four magnet coil designs. Basically, the finished conductor is a long thin copper plate. Three different types of conductor have been investigated. Figure 1a shows the cross section of a conductor with transposed braid super-

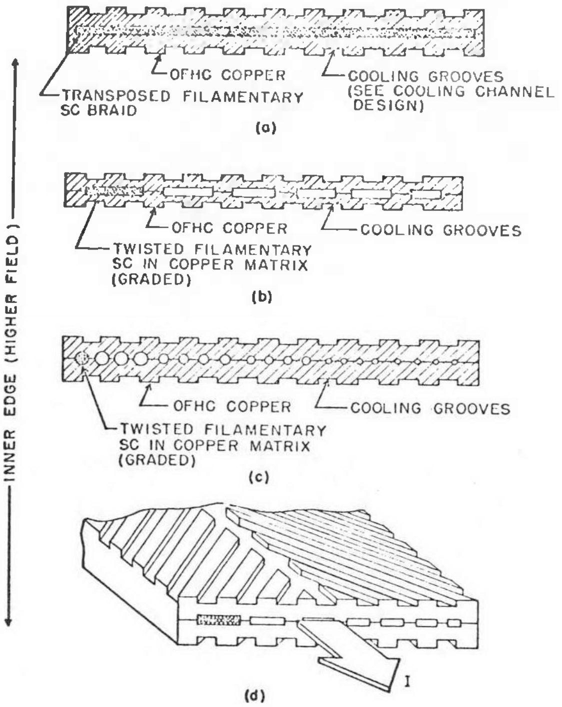

Figure 1. Conceptual Conductor Design

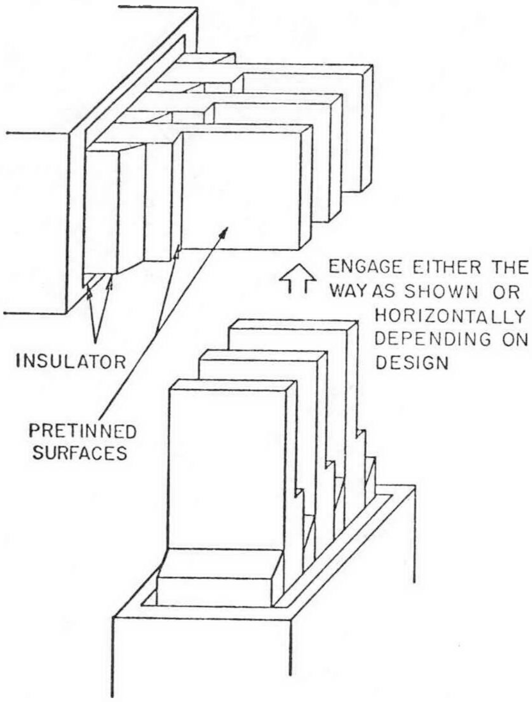

Figure 2. Coil Segment Joining Process

conductor sandwiched between copper plates; 1b and c show graded rectangular and round wires used in place of the transposed braid. Figure 1d shows a three dimensional view of a conductor section with cooling grooves on both surfaces of the copper plate.

B. Coil Segment Fabrication and Assembly

An important feature of this magnet design is that unlike other superconducting coils, the coils are not wound. The straight coil segments are fabricated in a factory. The coil segments are then assembled at the reactor site and the joints made. Transportation of the coil segments from factory to reactor is feasible with present railroad or truck systems.

The coil segment fabrication is the same for all four magnet designs. The process is simple and mass production techniques can be applied. The operation consists of stacking the prefabricated conductors and

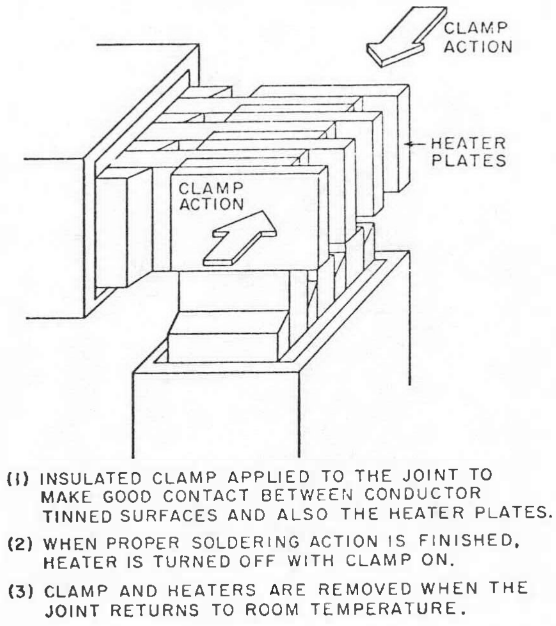

Figure 3. Coil Segment Joining Process

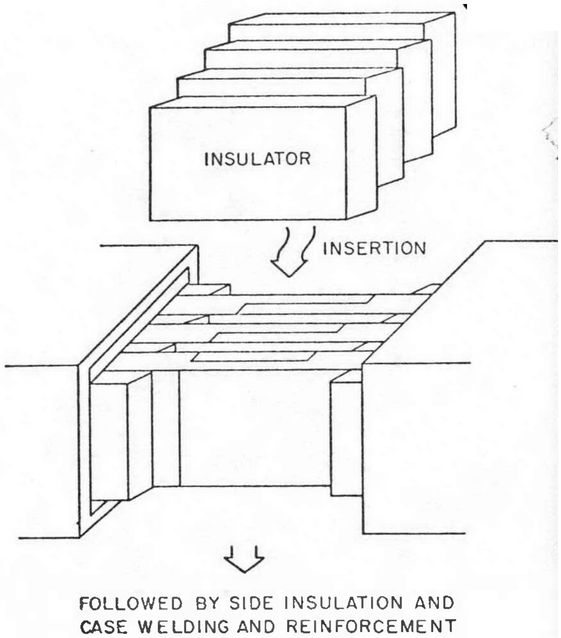

Figure 4. Coil Segment Joining Process

insulation plates and then welding the coil case. Figures 2, 3, and 4 show the conductor joining operation if soldered joints are used.

For joining - first engage the end together as shown in Figure 2, then insert the heater plate and clamp the joint as shown in Figure 3. The heater is powered until the pretinned surfaces are soldered. The heater is then turned off with clamp on. The heater and clamp are removed when the joint is cooled off and ceramic insulator plates are inserted. (Figure 4)

For disjoining - first remove the insulation, insert the heater plates, then clamp the joint and power the heater. Remove the clamp at proper temperature and slowly disengage the coil joints while keeping heater on. Both vertical and horizontal joints are shown to emphasize that conductor joining in the DEALS magnet concept can be made for a wide range of angles and locations, so that the specific requirements of many different types of magnet coil systems can be accommodated.

Another joining technique is the multi-contact pressure joint developed by LASL for high current density superconducting switch applications. A metal sheet containing a large number of tilted metal teeth, i.e,, louvers is pressed between the two conductor plates. Compression is applied, which results in point or edge contact between the perforated metal sheet and the two conductor plates. The current transfers from one plate through the louvers to the other plate. The advantage of this joining method is that limited relative movement of conductors is permitted. With solder joints, movement of the conductors will break the solder joint, drastically increasing the electrical resistance of the joint. The joint resistance of a multi-contact pressure joint will be a factor of 10 or more greater than the resistance of a solder joint but this is acceptable.

Either pool cooling and force cooling methods can be used in the DEALS magnet coils. The magnets are cryogenically stable if pool cooling is used. The cooling grooves are designed so that helium vapor bubbles quickly vent into large channels outside the conductors. If force cooling us used, the cooling passage length is relatively short ($\leq$ 50 meters). Heat from local hot spots can be removed in a few seconds. The magnet system parameters for the high-β D-He^3 reactor are presented in Table 1.

C. Heat Inputs and Refrigeration Requirements (Table 2)

Heat leak calculations were made for gas cooled power leads, magnet coil support pads, and the conductor joint. These heat leaks account for approximately 85% of the total refrigeration requirements. Other heat leaks such as nuclear heating, vacuum dewar losses and penetration losses (instrumentation), etc., account for the remainder. The eddy current loss in the copper stabilizer and the hysteresis loss in the superconductor are surprisingly small (just a few watts) due to the long burn time of approximately 3000 seconds. Therefore, 10 MW(e), 15 MW(e), and 23 MW(e) of room temperature refrigeration power should be adequate for the D-T, D-D high-β, and D-D low-β reactors respectively.

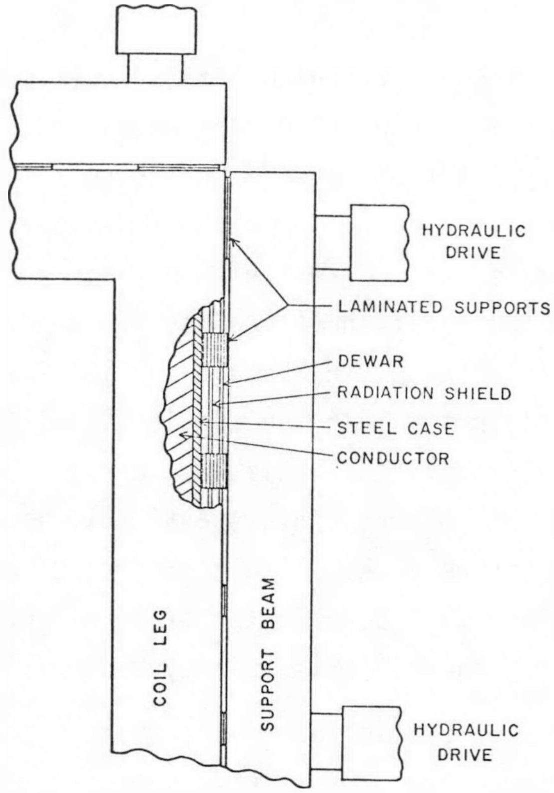

Figure 5. Coil Support System

3. Coil Support System

A. Description of Support System

Obviously, a rectangular coil geometry by itself is not an optimum configuration in terms of structural stresses. If the coil were supported only against the center column or at its four corners, unworkably high stresses would result. However, a series of intermediate supports along each coil span, will reduce stresses to very manageable levels in most cases, as the results given here indicate.

Figure 5 schematically shows the coil support system. The assembly of conductor plates and insulators in a given coil leg are enclosed in a steel coil case, several inches thick, which serves as the helium dewar. Laminated, low conductivity support pads bear directly on this steel case, penetrating through the vacuum dewar and radiation shield.

The loads on the coil are transmitted through these pads to an external warm reinforcement structure. The best arrangement seems to have the individual laminated pads mounted on a room temperature beam, which connects to the surrounding support structure by hydraulically or mechanically adjustable links. The adjustability is necessary so that the supports can be withdrawn for coil segment replacement, and it also compensates for thermal shrinkage of the coil during cool-down. The low conductivity pads can be made of many alternating layers of fiber-glass and stainless steel.

A Prestressed Concrete Reactor Vessel (PCRV) appears to be the best warm reinforcement structure for reactors of the size considered in this study. PCRV's have been used successfully in the U.S. and Europe as containments for certain types of fission reactors. Although the amount of prestressing and reinforcement steel in a PCRV is comparable to that needed for an all steel structure, the steel in a PCRV is cheaper. In addition, with smaller individual pieces, i.e., tendons and bands larger containment structures can be built more easily and cheaper with PCRV type construction than with steel only. Finally, since a PCRV can be made gas tight with steel liners, it can be a biological containment shield, as well as a reinforcement structure.

3. Stresses in Coils for Various Advanced Fuel Design Requirements

A plane, linear finite element code was used to investigate stress in the coils. Lorentz forces were calculated and applied as nodal loads. Support points were then varied until reasonable stresses were achieved. For all four magnet designs, a five-inch thick steel coil case was used. The finite elements modeling the copper and steel regions in the coil used the appropriate respective material properties.

Supports along the vertical leg are equally spaced; those along the horizontal are more closely spaced toward the axis of the torus since Lorentz forces are highest there. The inside leg experiences the largest forces and must be supported along a good fraction of its length. With a cold bucking column this leg could be continuously supported along its entire length. While stresses for the D-T reactor are lowest, those for the D-He^3 and D-D high-β reactors compare favorably and are quite acceptable. The stresses are much lower than in a self-supporting, constant tension design. Moreover, the highest stresses in all cases are found at the inside leg which is assumed to be support along 50% of its length. For a continuously supported leg, stress values would be even lower. The refrigeration requirements for the support pads are relatively low, as discussed previously.

The only design with unacceptable stress levels was the D-D low-β reactor where conductor stresses reached 41 ksi. The very high field and long coil leg spans made this design structurally unattractive. A thicker coil case and more support points would reduce stress levels somewhat, but the larger number of supports would increase heat leaks through the supports.

Only in-plane loadings have been considered here. Some additional support pads will be needed in all designs to compensate for lateral loads due to the poloidal coils. However, such loads will be much less than the in-plane loads, and the number of out-of-plane supports will be considerably less than the in-plane supports.

Since the corner joints are not designed to transmit loads, there is no net centering force and the loads on the bucking column are those transmitted by the inside leg. Even if the joints are soldered and not allowed to slip, correct adjustment of supports will prevent tensile loads from being carried through the joints. Other support and disassembly procedures as well as central buckling column configurations are discussed elsewhere.[1]

C. Thermal Stresses

Since the coil and case will operate at 4K, thermal stresses could occur due to the different thermal contractions of the materials in the coil. In cooling from room temperature to 4K copper contracts about .0034 in/in and 304 stainless steel about .0030 in/in while the ceramic insulators contract on the order of 10^{-6} in/in. They are therefore rigid compared to the copper and steel. During coil assembly, the ceramic insulators must be somewhat shorter than the copper conductor plates, leaving slight gaps which will be closed due to the copper shrinkage. Differences in shrinkage between copper and steel through the coil cross section can be compensated by appropriate thicknesses of the ceramic. The rigidity of the ceramic is not a problem since the insulator can be put between the conductors in long parallelogram-shaped sections whose ends could slide over each other to compensate for the contraction in the steel and copper. Over the length of a coil leg, however, the copper conductors will try to contract more than the stainless steel case. There are several solutions to this problem: The conductors could be anchored to the coil case at several points along the leg so that the ends of the legs experience only a small part of the total contractions. Also, movement in the joints may be prevented by a non-metallic bolt arrangement to give mechanical strength to the soldered joints. Designing bellow sections into the steel case is another solution. A very desirable solution would be the use of a steel whose contraction more closely matches that of the conductor.

Numerical detail not specifically stated in this paper are presented in a poster session at this meeting.

II. POLIODAL MAGNET SYSTEMS

The basic design calculations concerning poloidal field system have been performed according to the needs for shape and positional stability. Essentially these designs are based on two-dimensional MHD calculations.[2] The anticipated shape and current profile of the plasma are input parameters to the numerical problem and poloidal coil locations and currents represent the output of the design calculations. Figure 6 depicts the low-β "catalyzed" D-D reactor cross section. The dashed line present across the outboard shield of the figure shows the location of the ports for both the toroidal bundle divertor and the beam injector. Figure 7 is a similar representation of a high-β D-^{3}He Tokamak cross section; again the dashed line indicates the location of the divertor-injector opening. The specific coil currents and coordinates of coil centers are presented as part of a poster paper at this meeting. The purpose of this portion of the advanced reactor design study has been to estimate materials, power supply and cooling requirements for purposes of technological and, eventually, economic comparison with D-T Tokamak reactor designs.

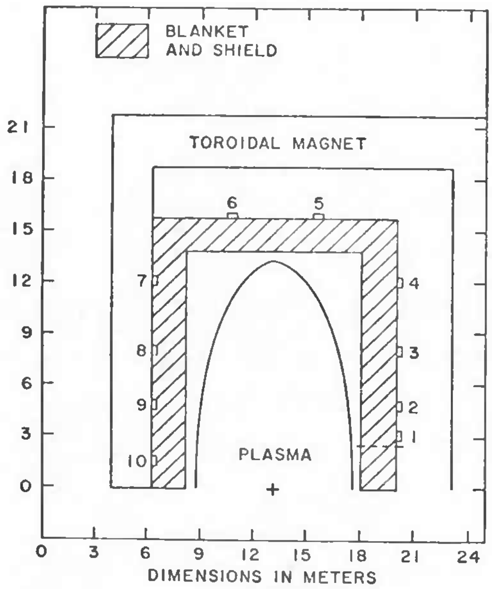

Figure 6. Catalyzed D-D Tokamak Reactor

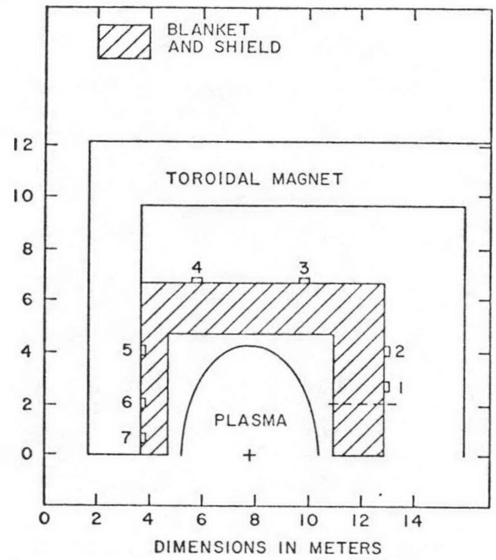

Figure 7. High Beta D-He^3 Tokamak Reactor

Neutronic calculations have shown that the inboard poloidal coils are sufficiently shielded in all the cases examined including the high-β D-^{3}He design where inboard blanket and shield thickness totals only 1.0 m. These shielding calculations were performed assuming superconducting magnets indicating that conventional magnets would also be quite well shielded. The poloidal magnet system is expected to be composed of both normal and superconducting magnets. The poloidal magnet system will be utilized in start-up to alleviate the volt-second requirements imposed on the ohmic-heating (OH) coils. The fast rise-time requirements during certain portions of start-up necessitate the use of quick response conventional coils (copper) while the slower rise-time portions could be handled by superconducting coils thus saving on power requirements.

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Table 1

D-He^3 High β Magnet Parameters

No. of Coils	18
Coil Size	14.5 m x 20.5 m
Coil Cross Section (including 4 side insulation, each 2 cm thick)	0.935 m x 1.04 m
No. of Turns Per Coil	60
Conductor Sizes (SC and Copper)	(1) 0.01 m x 1.0 m x 14.5 m (2) 0.01 m x 1.0 m x 20.5 m
SC Sizes * (Nb_3Sn)	(1) 10^{-4} m x 1.0 m x 14.5 m (2) 10^{-4} m x 1.0 m x 20.5 m
Insulator Sizes **	(1) 0.005 m x 1.0 m x 14.5 m (2) 0.005 m x 1.0 m x 20.5 m
Peak Field at Conductor (@ 3.7 m)	11.68 Tesla
Field at Plasma Center (@ 7.8 m)	5.54 Tesla
Operating Current	2×10^5 A
Amp-Turns Per Coil	1.2×10^7
Total Amp-Turns	2.16×10^8
Average Current Density Over Conductor	2000 A/cm^2
Average Current Density Over Coil	1300 A/cm^2
Required Heat Transfer Rate When All Current Flows in Copper Stabilizer	$\sim$0.26 W/cm^2
Inductance Per Coil	0.3 henry
Total Inductance	5.5 henry
Stored Energy Per Coil	6.1×10^9 joules
Total Stored Energy	1.1×10^{11} joules

* SC cross section was calculated for highest field of 11.68 Tesla

** Approximate Dimensions

Table 2

Losses and Refrigeration Power Requirement

	D-T		D-D High β		D-D Low β	
	4.5K (kw)	300K (Mw)†	4.5K (kw)	300K (Mw)†	4.5K (kw)	300K (Mw)†
Conductor Joint*	0.9	0.3	2.4	0.8	8.6	2.8
Coil Case Support Pads	1.4	0.5	1.8	0.6	2.3	0.8
Power Leads		7.6		11.0		16.0
Total		8.4		12.4		19.6

* Losses from multi-contact pressure joint are used. These losses are a factor of 10 higher than solder joints.

† Conversion factor 326w/w is used.

Direct Energy Conversion and Neutral Beam Injection for Catalyzed D and D - ^{3}He Tokamak Reactors*

by

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ABSTRACT

The calculated performance of single stage and Venetian blind direct energy converters for Catalyzed D and D - ^{3}He Tokamak reactors are discussed. Preliminary results on He pumping are outlined. The efficiency of D and T neutral beam injection is reviewed.

*Performed for the Electric Power Research Institute under Contract No. RP645-2 and under the auspices of the U.S. Energy Research and Development Administration under Contract No. W-7405-Eng-48.

I. INTRODUCTION

D-D and D-^{3}He fusion reactions yield a large fraction of their energy as energetic charged particles. In order to assess the potential benefit, we have studied the performance and technology requirements of direct energy converters when attached to such advanced fueled reactors. Direct converters can recover a portion of the charged-particle energy directly and send most of the remainder (in the form of heat) to a thermal bottoming cycle. Depending, of course, upon the ion energy distribution and the recovery technology, Venetian blind direct converter efficiencies as high as 65% have been calculated for more conventional, D-T fueled reactors. Our purpose was to extend these studies to Cat-D and D-^{3}He machines(1); we will present the calculated results.

The problems of fueling and heating plasmas by the neutral beam injection of deuterium and tritium has been extensively studied; the method appears to be practical in the case of Tokamaks. We will discuss the efficiencies that can be expected.

II. DIRECT ENERGY CONVERSION

We begin with a brief review of the types of direct energy converters that were studied. A Tokamak reactor is assumed.

- A portion of the Tokamak plasma diffuses across magnetic field lines until it encounters a line that leaves the reactor interior through a bundle divertor.
- The plasma follows the field line out of the reactor and enters a conical "expander" tank several 10's of meters long, such as is shown in Fig. 1.
- Because of the magnetic mirror-like action of the diverter coil, the reactor plasma on the field lines linked by the divertor coil will be at a positive ambipolar potential with respect to the reactor wall. As plasma travels toward the grounded grid

(1)In particular, we considered direct energy conversion on five different reactor designs, developed by the University of Illinois and BNL.

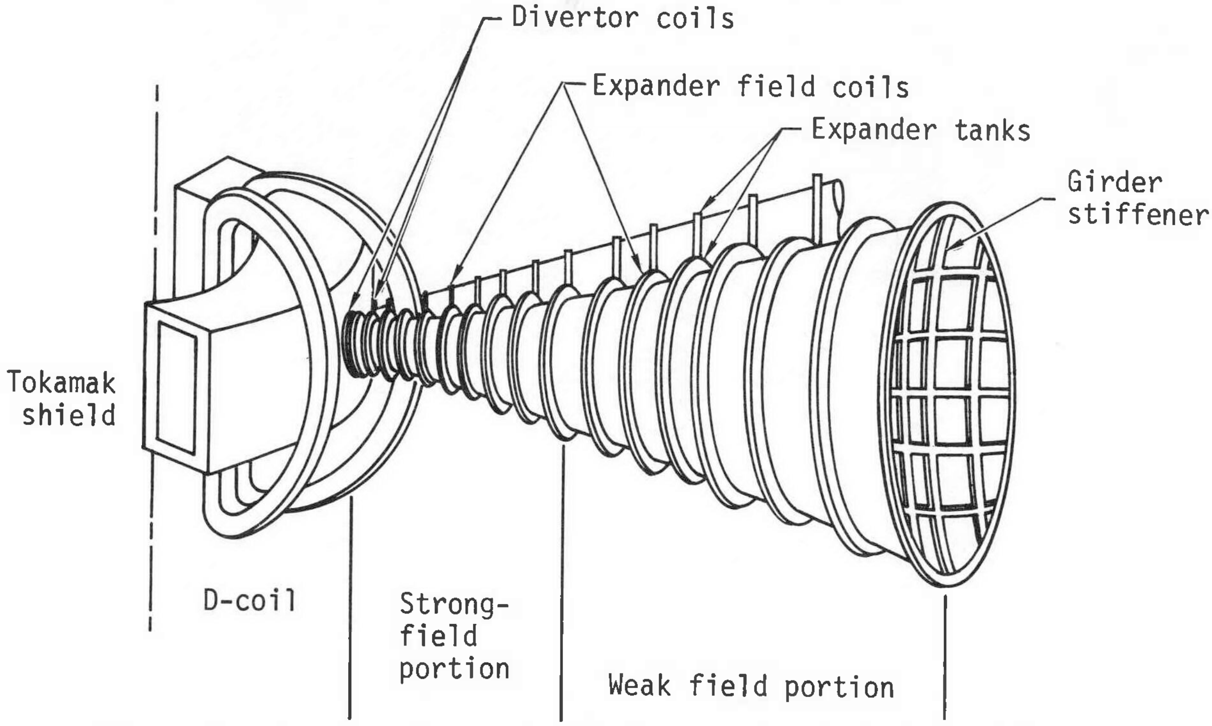

Figure 1. A pair of expander tanks are sketched along with a portion of the Tokamak reactor.

at the far end of the tank, the ions will be accelerated by this potential; the electrons will be slowed down.

- The plasma stream expands along field lines as it travels down the tank. This implies a decreasing magnetic field, which in turn converts the ion velocity component, $V_\perp$, into $V_\parallel$. The charged particle power density and the charged particle density are both reduced.

- Near the wider, far end of the tank, the direct converter electrodes are encountered. These are shown in Fig. 2. Some of the electrons (a number equal to the ion current) strike the grounded grid and are removed. The remaining electrons are reflected back toward the reactor by the negative grid.

- The ions are accelerated between grids, but slow down in velocity after they pass the negative grid and approach the electrically positive collector.

- The slowed down ions strike the positive collector. Their excess energy is given up as heat.

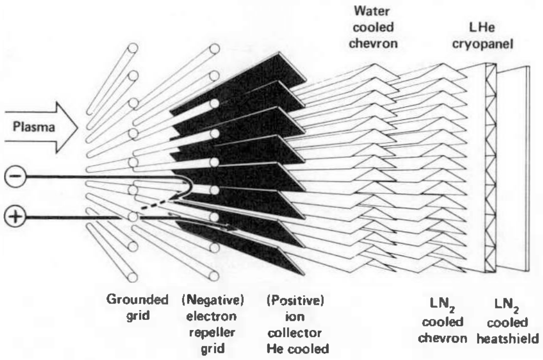

Figure 2. The single-stage direct converter electrode structure

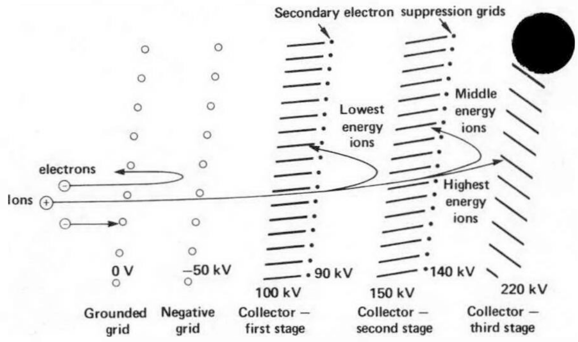

Figure 3. The three-stage Venetian blind direct converter electrode structure

The multistage collector shown in Fig. 3 can be used to increase the electrical output. The higher energy ions are collected on high voltage collectors. Their excess energy, upon collection, is smaller. More electrical power and less heat is produced.

III. MODELING OF THE PLASMA

The analyses of the reactor plasma were zero dimensional, and yielded only the magnitudes of the charged particle power, ion currents, and the electron current. This necessitated some assumptions about the statistics of the ions and electrons escaping through the divertor. The populations were assumed to be Maxwellian with temperatures T_e and T_i. The ratio T_e/T_i was left as a parameter which was studied over the range $1.0 \geq T_e/T_i \geq 0.25$. The Maxwellian plasmas were assumed to be accelerated through an ambipolar potential of $4.5\ kT_e/q$. By selecting a value of T_e/T_i and combining the preceding assumptions with values for the escaping ion current, electron current, and charged particle power, we determined a distribution for the charged particle energies. The results of supplying these charged particle distributions to the direct converter are discussed in the next section.

IV. CAT-D AND D-^{3}He DIRECT CONVERSION - AS CONTRASTED WITH PRIOR D-T MIRROR RESULTS

The processes by which energy is lost from direct conversion are noted in Table 1.

Since prior direct energy conversion studies for D-T mirror reactors constitute the bulk of the presently available results, it may be useful to point out the major differences that resulted from the switch to Cat-D and D-^{3}He fuels.

- The mean ion energies are lower and charge-exchange neutralization is no longer negligible. A direct-conversion efficiency decrease of as much as 10% is attributed to charge-exchange in many of the cases.
- The electron energy is higher, over the assumed range of T_e/T_i, and since electron energies are not recovered, the direct conversion efficiency will be decreased.
- The grounded grid size has to be increased to accommodate the higher electron energy that is deposited there. The larger grounded grids intercept more ions before they can reach the collector. The grids contribute more to the efficiency decreases seen in these studies than was attributed to the grounded grid in prior Mirror studies.
- Large D_2 and ^{3}He gas flows must be pumped out of the expander tank. Since conventional cryogenic systems pump D_2 well, but pump He (and by assumption ^{3}He) very poorly, some portion of the pumping surface had to be occupied by diffusion pumps. Because these diffusion pumps pump D_2 gas with less speed than the cryogenic pumps, the D_2 is pumped at a reduced speed, and the background D_2 concentration and the associated charge-exchange neutralization are increased.

Table 1 - The various energy loss mechanisms and the resulting heat depositions are listed.

1) Charge-Exchange Neutralization: Ions from the reactor are changed into energetic neutral atoms before reaching the collector.	Energetic neutrals give up energy by colliding with electrodes, principally the collector. Resulting low energy ions strike the grounded and negative grids.
2,3) Electron and ion interception on the grounded grid.	The grounded grid is heated.
4) Ion interception on the negative grid.	The negative grid is heated.
5) Ion energies in excess of the ambipolar potential ("ideal collection loss").	The collector is heated.
6) Collector voltage lowering to accommodate: a) finite magnetic field expansion b) transverse deflection of ion as it passes near a grid.	The transverse velocity component of the ion delivers its energy to the collector as heat.
7) Thermionic emission from the negative grid: 7G) Current flow to the collector results 7C) Current flow to the collector results	The emitted electrons travel to the grounded grid and the collector where they deposit their energy as heat.
8) Secondary emission from the negative grid.	Secondary electrons give up their energy to the grounded grid as heat.
9) Coolant pump power to the grounded grid.	The resulting heat is extracted along with the grid wire coolant.
10) Structural elements intercept plasma.	The structures which support the direct converter electrodes are heated.

Fig. 4 and Fig. 5 plot, in a cumulative fashion, the contributions of the various loss processes: Fig. 4 for the Cat-D, low-β design[1] and Fig. 5 for the D-^{3}He [1], 1:1, high-β design. These are representative examples. Two different collection losses are listed, that of a single stage collector and that of a 4-stage collector. The two lowest curves represent the direct conversion efficiencies (directly converted electrical power/charged-particle power) for these two designs.

It is interesting to note that for large T_e/T_i, the ion energies (per ionic charge) are closely grouped near the ambipolar potential, and the four-stage collector offers only a few percent improvement over the single stage design. For smaller T_e/T_i, the single stage collector is less efficient, and a four-stage design offers a 10% to 20% improvement in direct conversion efficiency.

The loss processes listed in Table 1 result in the production of heat. Most of this heat can be collected and sent to a thermal bottoming cycle. By assuming that the efficiency of this cycle is 40%, one can calculate a plant efficiency (total power out) for the charged particle power such as is shown in Fig. 6, for a Cat-D design. Fig. 8 shows an outline (top view) of this reactor with attached direct converter tanks. The total charged particle power and the assumed power density determine the expander tank size. Fig. 7 and Fig. 9 show the same results for a D-^{3}He design.

The increase in total electrical power out, which occurs in some cases between 100 W/cm^2 and 150 W/cm^2, is a result of the grid selection process. Radiatively cooled grids whose thermal output can be recovered only in part are used at lower power densities. In the case of the convectively cooled grids used at power densities above 100 W/cm^2, all the thermal power can be recovered and sent to the bottoming cycle.

V. EXTRACTING MAGNETIC FIELD LINES

The toroidal magnetic field lines can be bent and thereby directed through the reactor blanket and shield by bundle diverter coils. The continued radial travel of these lines, in the manner shown in Fig. 10, depends upon two additional coil systems.

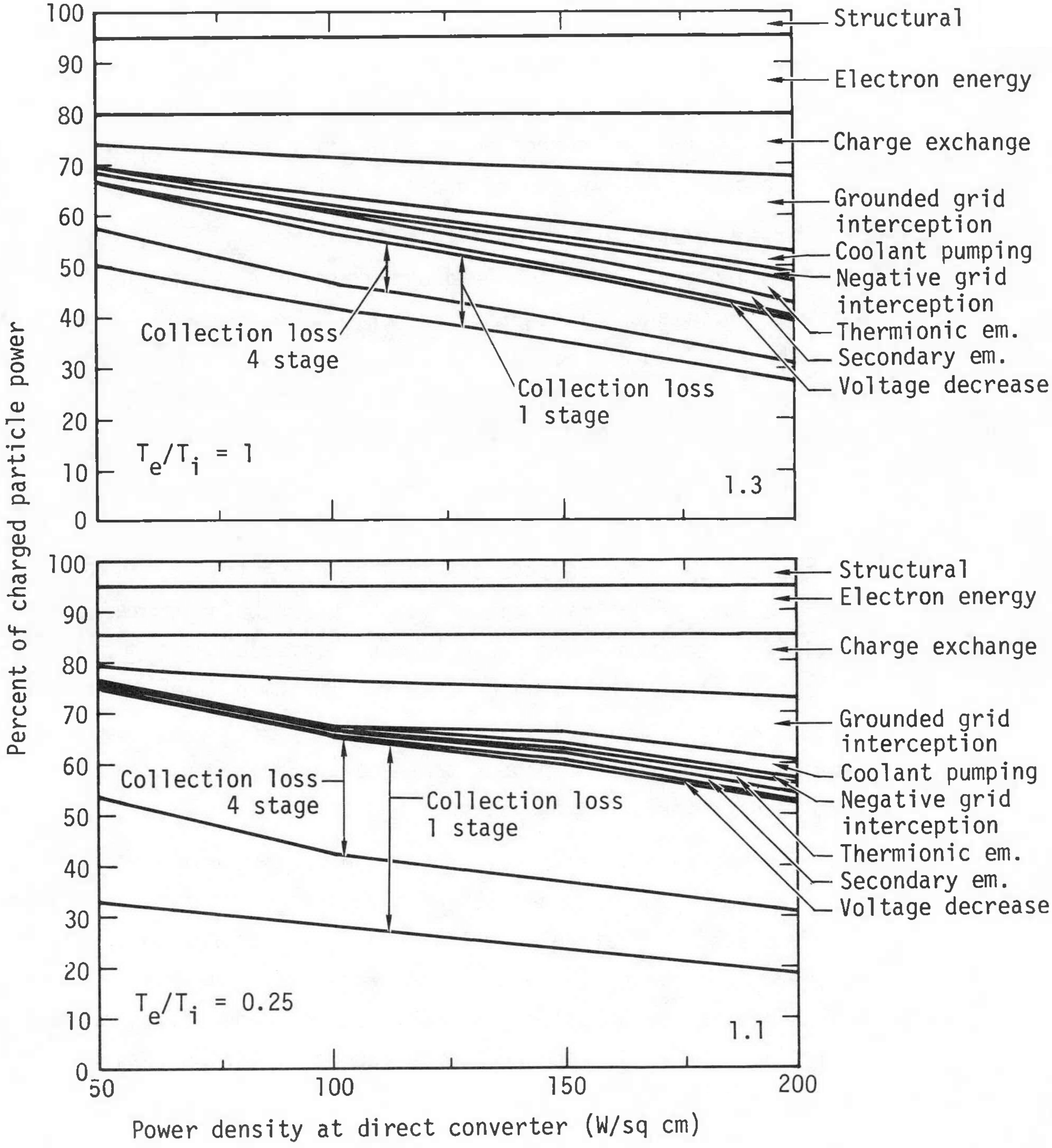

Figure 4. The losses for the low-β, Cat-D design are cumulatively plotted. The upper figure is for T_e/T_i = 1; the lower figure is for T_e/T_i = 0.25.

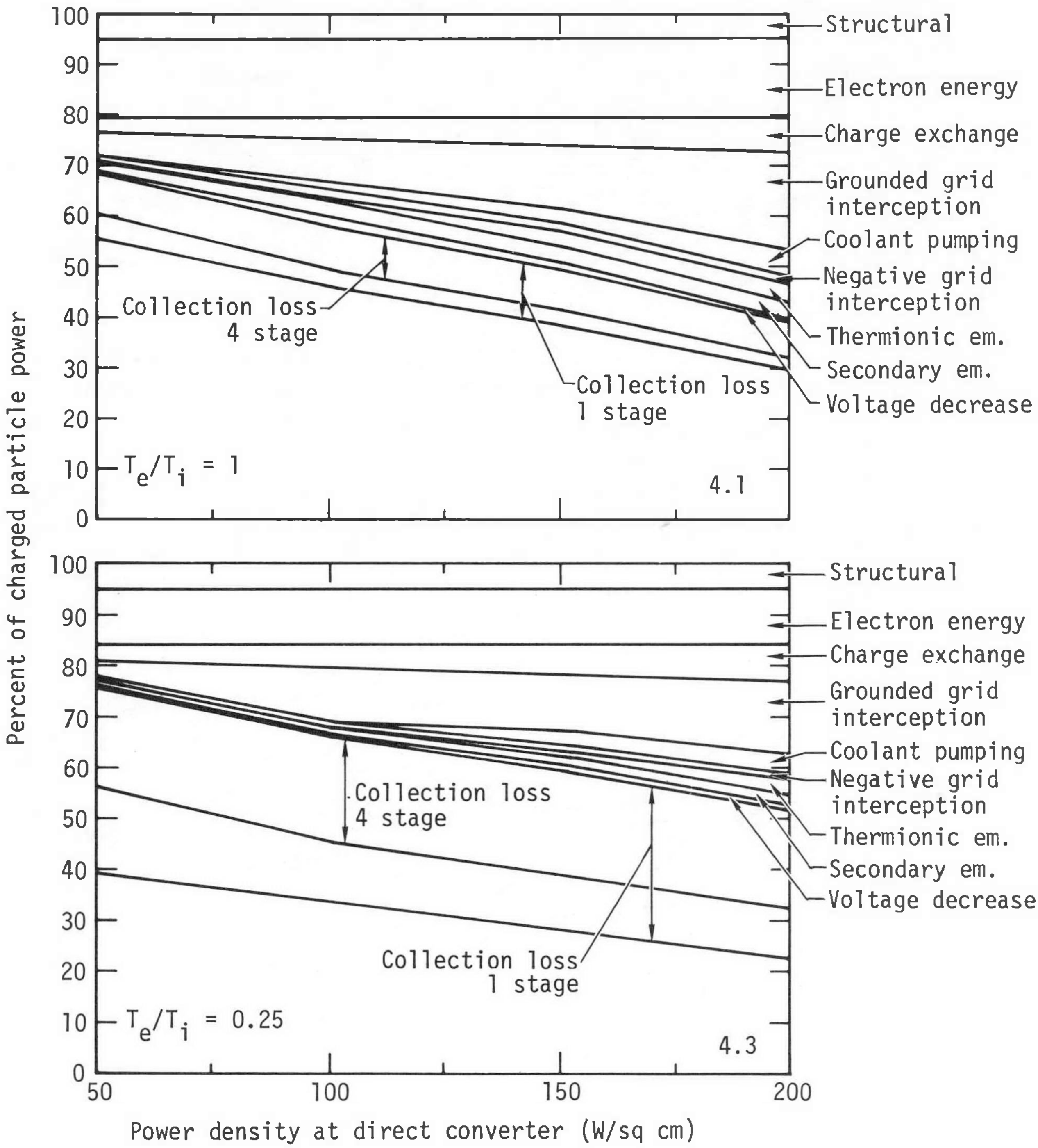

Figure 5. The losses for the D-^{3}He, 1:1 high-β design are cumulatively plotted. The upper figure is for T_e/T_i = 1; the lower figure is for T_e/T_i = 0.25.

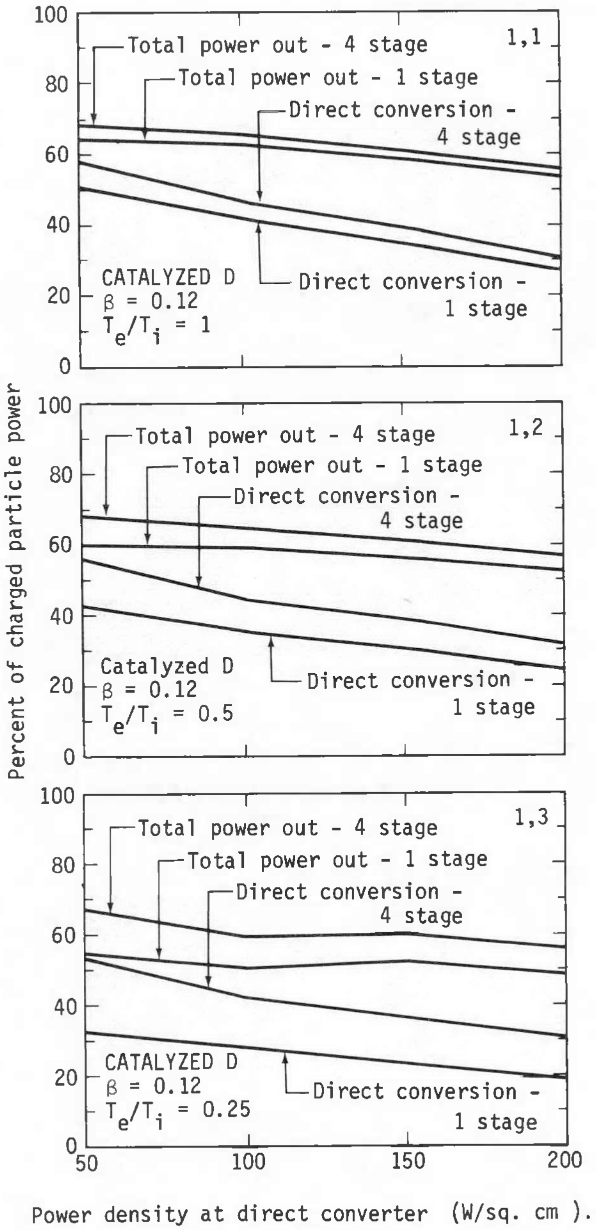

Figure 6.

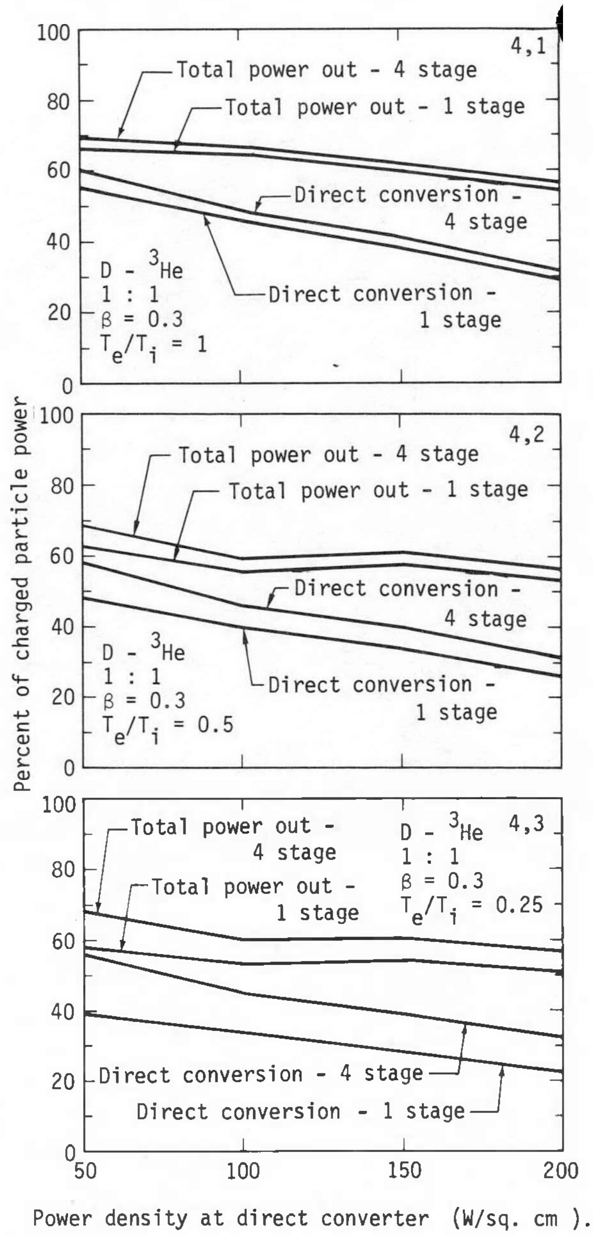

Figure 7.

The direct conversion efficiency and the total electrical power (extracted from the charged particles), assuming a 40% efficient thermal bottoming cycle.

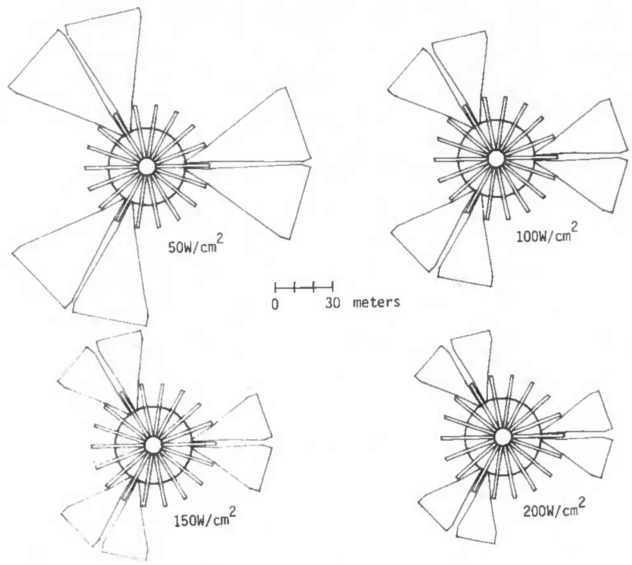

Figure 8. A top-view outline of the Cat-D, low-β reactor and direct converter tanks for differing power densities at the direct converters.

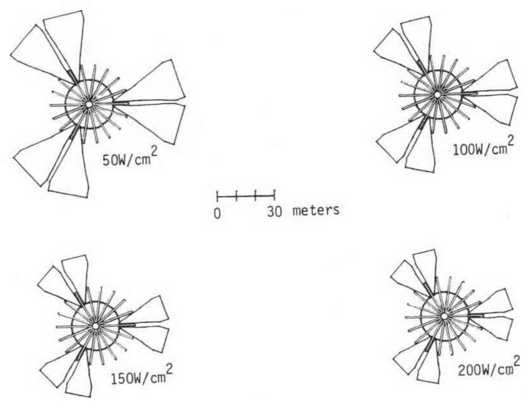

Figure 9. A top-view outline of the D-^{3}He, 1:1, high-β reactor and direct converter tanks for differing power densities at the direct converters.

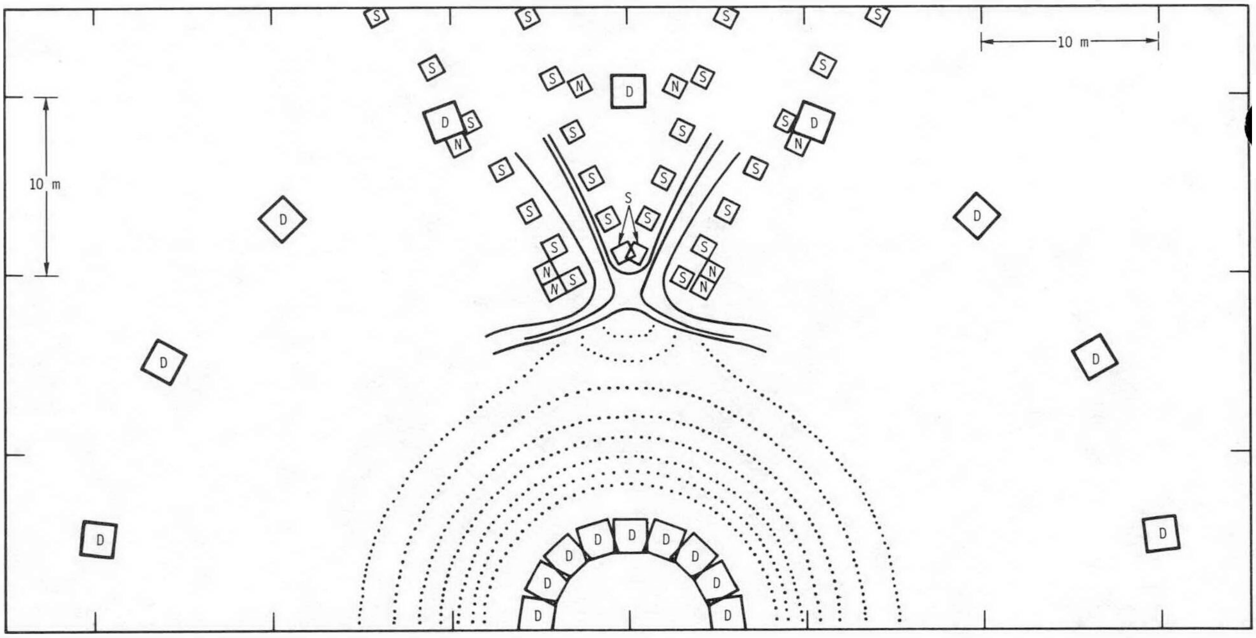

Figure 10. The magnetic field lines (solid curves) are bent into a radial direction by the ring coils, denoted by an s, and the toroidal field nulling coils, denoted by an n.

First, in addition to the bundle diverter coils, a series of ring-like coils, as shown in Fig. 11b, must be spaced along the expander to provide an expanding cone of lines inside. The squares marked s, in Fig. 10, denote sections through these coils.

Second, the toroidal field which cuts through the expander must be canceled. The coils shown in Fig. 11a perform this task. This arrangement surrounds the portion of the expander within the toroidal field. The squares marked N in Fig. 10 denote sections through these coils.

VI. HELIUM PUMPING

For the purpose of this study, it has been assumed that diffusion pumps will be used to remove the large flows of ^{3}He gas. On the other hand, cryogenic pumps would allow structural simplifications, higher pump speeds, and freedom from back streaming. For this reason we have investigated their application to our He pumping requirements. Only very preliminary results are available, but it now appears that when combined with an N_2 or Ar gas spray, ^{4}He can be pumped at speeds

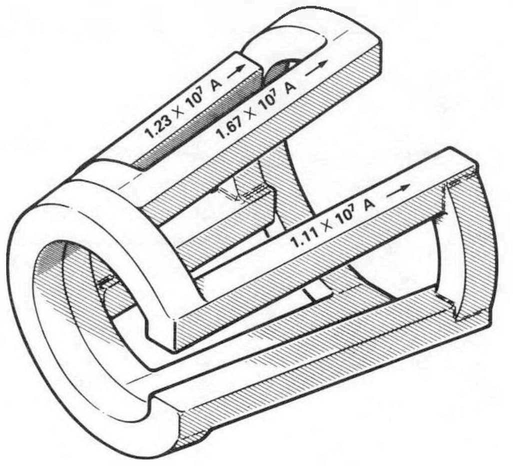

Figure 11a. The toroidal field nulling coil surrounds the portion of the expander within the toroidal field.

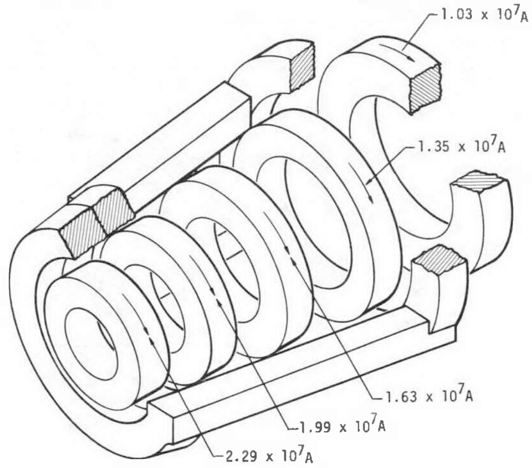

Figure 11b. The ring coils are enclosed by the TFNC.

approaching those with which cryogenic pumps remove D_2(2). No back streaming of the N_2 or Ar gases were observed. Experimental work on ^{3}He is now beginning.

VII. NEUTRAL BEAM HEATING

Neutral beam heating is required only during startup. By the nature of the startup procedures, the power requirements are modest: 100 MW for 62 sec. (this worst case corresponds to the high-β Tokamak startup). In order to penetrate the plasma, ion energies of 200-350 keV are required. D and T beams at these energies can be produced at high efficiency by accelerating negative ions, D^- or T^-, and then neutralizing them via photodetachment or stripping in a cesium vapor cell. For these two methods of neutralizing negative ions, the percent ratio of -- neutral beam power to line power into the power supply -- is shown in Fig. 12(3) and Fig. 13(3).

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(2)This work is being performed by Dr. Thomas Batzer, Lawrence Livermore Laboratory.

(3)J. H. Fink, W. L. Barr, and G. W. Hamilton, UCRL-52173, Nov. 1976.

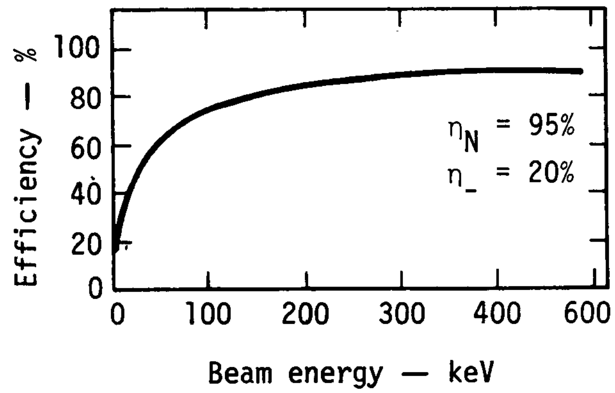

Figure 12. Deuterium neutral beam injection efficiency using negative ion acceleration and photodetachment.

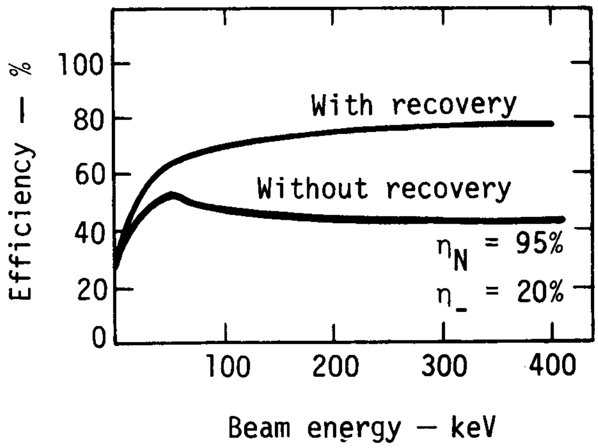

Figure 13. Deuterium neutral beam injection efficiency using negative ion acceleration and cesium cell stripping. Energy recovery includes both direct and thermal energy recovery.

An Electric Utility View of Fusion

by

H. R. Drew
Executive Vice President and Secretary
Texas Atomic Energy Research Foundation

ABSTRACT

The history of the development of electric power plant technology has, until recently, been one of steadily increasing efficiency and declining plant cost per kilowatt of generating capacity. Fusion reactions which produce energetic charged particles offer a dramatic opportunity to continue these trends. In spite of the difficulties, the potential is so attractive that a substantial research effort is warranted.

Having been an early advocate, on behalf of the utilities, of stepped-up research in advanced fuels for fusion, I am particularly gratified that this meeting is taking place. As a non-scientist but long-time fusion supporter, I can perhaps best augment your program by reviewing the reasons we are so interested in this work.

I doubt that any industry has faced as many challenges in recent years as has the electric utility industry. Rapid technological developments and ever-expanding social expectations make long-range planning more essential than ever. Yet, shifting and often conflicting government policies, international events and the transitory nature of public concerns make long-range planning more difficult than ever.

When the Texas Atomic Energy Research Foundation* began its support of fusion research in 1957, fusion power seemed much less difficult of an achievement than it turned out to be. The early optimism was premature but today we seem to be on the verge of achieving the feasibility goal that has eluded us for so long. Now that we can see the light at the end of the tunnel, some in the fusion community wonder why the utilities aren't eagerly pushing the development of this marvelous new energy resource.

A utility executive's perception of fusion is influenced by many factors. He has urgent problems in supplying the energy needs of his customers as cheaply and reliably as possible. He must be sure that his facilities will comply with a mass of regulations on all manner of environmental and social matters. He finds it increasingly difficult to site new power plants. He has to raise vast sums of money to build the plants and to pay for the fuel they use but regulatory commissions frequently refuse to give his company the rates needed to do the job. It is hardly surprising that the utilities' research and development funds are largely devoted to the solution of these immediate and pressing problems.

For the longer range, substantial utility R&D funds have been committed to the breeder reactor - a commitment which the federal government urgently solicited. The breeder is a technology which has been demonstrated. It is a

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*The Texas Atomic Energy Research Foundation is composed of the following electric power companies of Texas: Central Power & Light Company; Community Public Service Company; Dallas Power & Light Company; El Paso Electric Company; Gulf States Utilities Company; Houston Lighting & Power Company; Southwestern Publ[ic] Service Company; Texas Electric Service Company; Texas Power & Light Company; and West Texas Utilities Company.

commercial process in other countries and there is no doubt that it can extend the usefulness of nuclear power for many decades into the future. The feasibility of fusion remains to be demonstrated scientifically and the technological problems which must be solved before it can be considered a practicality are tremendous.

The coin has two sides of course. If the breeder were to be severely curtailed worldwide, the need for fusion would become urgent. That prospect does not seem likely to me, in spite of the current arguments against the breeder by the present administration and some segments of the public. I believe we have the time and resources to pursue a reasoned fusion research program. To do so we need to be sure that our research goals are consistent with the ultimate national needs for fusion.

There is a need to evaluate all of the fusion alternatives, including the mainline efforts, from two points of view: (1) the scientific and technical difficulties of achieving their utilization and (2) their relative attractiveness for electric power generation. The criteria for the latter evaluation should include capital and operating cost estimates, size, environmental and safety aspects, serviceability and maintainability, operating characteristics (flexibility, ability to follow load, ability to stand loss of load and downtime required for maintenance), resource use and problems of disposal of radioactive materials.

As the national program approaches some major decisions regarding very large and expensive experiments, the need for such a comprehensive evaluation will be obvious. Without it, we cannot set the R&D priorities that will be needed to set the most logical and least expensive course for the overall effort. The Tokamak concept, although it may offer a promising route towards proof of ability to control fusion reactions, is not necessarily the best concept for fusion power reactors. To a utility engineer, it presents some problems. Conceptual Tokamak power reactors are exceedingly complex, combining several new and unproven technologies into a machine which will be both costly to build and hard to maintain. We foresee great difficulty in operating and maintaining a reactor which incorporates circulating molten lithium, huge cryogenic coils of complex shape, sophisticated fuel injection and waste removal systems and inner walls which must periodically be replaced. The very large size of the Tokamaks will add to siting difficulties and will reduce system operating flexibility.

Their complexity is certain to make the Tokamak power plants expensive to build. The Clinch River breeder plant will cost about $5000/kw (380 MW, $1.95 billion) as compared to $800/kw for current generation light water reactor plants. Conceptual Tokamak designs seem to be at least as complex as Clinch River. I cannot believe that they would be any less expensive.

There is a strong incentive to strive for power conversion methods which are more efficient than those we must use today. Problems in dissipating the waste heat from thermal plants are severe and it is becoming increasingly difficult to site large power facilities on that account. From a thermodynamic point of view, it seems almost sinful to generate energy at the high temperatures of fusion reactions and to then degrade that heat down to temperatures that can be utilized to generate steam. That approach would be acceptable if we knew that no alternatives existed, but they do exist, at least in principle. "Neutronless" fusion reactions which produce only energetic charged particles offer an opportunity to achieve a much higher efficiency in electric power generation than is posssible in thermal plants, through the direct conversion of particle energy into electricity. Boilers, turbine generators, condensers and the associated pumps and other auxiliary equipment would be eliminated. Problems with radioactivity, vacuum wall deterioration and the thermal blanket might be eliminated or greatly reduced. Conceptually, a direct conversion fusion plant should be less complex than a D-T plant and therefore less expensive.

It is easy to point out the advantages and minimize the difficulties. This meeting will explore the many scientific and engineering problems that must be overcome if one of these advanced fuel systems is to be realized. Because these concepts are still undeveloped, it would be a mistake to place too much reliance on alternative cycles at this stage of development. We must continue the mainline fusion program, at least to the feasibility demonstration stage, but I believe the nation has the resources to pursue multiple paths in fusion. A modest program which seeks to solve the physics problems and advanced fuel cycles need not penalize the D-T feasibility effort.

In a recent article in SCIENCE magazine[1], 24 representative government-sponsored demonstrations of new technologies were analyzed to determine why some demonstrations were successful in producing a commercial product while

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(1) SCIENCE, 27 May 1977, pp. 950-957

others were not. The study showed that one characteristic associated with success in speeding commercialization of a new technology was participation in the demonstration phase by those who would take responsibility for further diffusion of that technology.

Industrial participation in the national fusion program is minimal. If fusion is to become a commercial reality, the manufacturers who will eventually be expected to produce fusion reactors and the utilities who will be expected to build and operate the power plants should be given more opportunity to participate in the national fusion program. That participation must include planning of the kind that led to today's meeting.

Multipole Fusion Reactors Using Advanced Fuels

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Presented by

D. Arnush

ABSTRACT

A progress report on a study of the use of advanced fuels in a multipole plasma confinement reactor is presented. As an example, an octopole employing $P + {}^{11}B$ is investigated with regard to power flow balance. The results are encouraging, assuming that recent octopole diffusion scaling measurements may be extrapolated to the fusion regime. A conceptual reactor design is used to identify outstanding engineering and physics uncertainties and the conceptual design of an experimental levitated octopole to resolve most of the key issues is presented.

I. INTRODUCTION

Fusion reactors using advanced fuels with multipole confinement geometries appear to offer an exceptionally attractive alternative to the current mainline programs (d-t tokamaks and mirrors). This conclusion is based on recent experimental, theoretical and engineering feasibility studies conducted at the University of Wisconsin, General Atomic, TRW Systems and UCLA.

TRW has initiated a detailed assessment of this concept. To this end, we formulated a study plan with the objectives evaluating and selecting the most promising advanced fuels for potential reactor applications, to establish an initial reactor reference design as an example of an advanced fuel concept, and to prepare a preliminary design of an experiment to determine the scaling of multipole confinement properties from the present low temperature and small $n\tau$ regime to quasi reactor conditions (density of order $3 \times 10^{13} cm^{-3}$ to $10^{14} cm^{-3}$, temperatures of a few keV and machine experimental time constants of seconds).

Advanced fuel reactions (producing no neutrons, or in any case much fewer than d-t), have been of interest[1-7] because of characteristics which reduce or eliminate many engineering problems associated with commercial reactor design and operation. An ideal reactor fuel should have a large reaction cross section, small nuclear charge, non-radioactive electrically charged end products, high natural availability, low cost and ease of handling. No individual fuel has all of these desirable properties; however, several candidate fuels exhibit many advantages as compared to d-t. For examples: If the reaction products are non-radioactive and electrically charged, then plant life is increased since there is less neutron damage to the structure; the waste management problem is alleviated since neutron activation is reduced; the environmental impact is reduced since there is no gaseous (tritium) radioactivity release; and the plant is inherently safer and more reliable because the working fluid is not radioactive and an intermediate coolant loop is not needed. If the fuel has high natural availability, the capital investment in fuel is reduced; the plant is simplified since a fuel breeding blanket is not required; and no fuel separation facility is needed. The use of advanced fuels has been impeded by the high temperature required (~300 keV for p = ^{11}B) and the consequent power losses due to electron synchrotron radiation and bremsstrahlung.[6,7]

Another well known fusion concept is the use of multipole configurations for confinement. There had been a significant program to develop this approach for some fifteen years,[8,9] involving both theory and experiments. The latter have been very successful, albeit at low density and temperature, showing confinement for as much as several hundred Bohm times. In recent experiments performed at the University of Wisconsin,[10,11] direct measurements of the diffusion coefficient yielded data consistent with a loss mechanism[12] (convective diffusion) which would scale very favorably to the reactor regime. Preliminary experiments have also been carried out by Alfred Wong[13] at UCLA to test the properties of high order multipole or surface magnetic field configurations (surmac), whose loss mechanisms promise to scale even more favorably. The principal difficulty of multipole confinement is the loss of plasma due to the presence of mechanical supports or electrical leads[14] for the current carrying hoops embedded in the plasma. To avoid mechanical supports or electrical leads, one could use magnetically levitated, superconducting coils but it is difficult to shield these from the intense 14 MeV neutron flux produced on the d-t reaction.

John Dawson[6] perceived a synergistic relation between advanced fuels and multipole geometries. Relatively high order multipoles (octopole or higher) result in a significantly weaker magnetic field over most of the plasma volume which results in a large reduction of synchrotron radiation. This makes the energy losses due to the high electron temperatures associated with advanced fuel reactions tolerable while the lack of neutrons allows the hoops to be superconductors thus solving one of the major problems associated with this confinement configuration. (In studies carried out recently,[15] it has been suggested that multipoles may be suitable even for d-t reactors. However, engineering feasibility of the concept remains to be demonstrated).

An advanced fuel, multipole reactor appears to satisfy the basic requirement of any alternative concept since it has significant advantages over the mainline approaches providing the presently identified problems of physics and technology can be solved satisfactorily.

II. CANDIDATE NEUTRONLESS REACTIONS AND PLASMA ENERGY

Useful fuels in a CTR application should exhibit some or all of the following desirable features, listed in order of approximate decreasing importance:

1. Nonradiactive End Products - Of major importance is the avoidance of reactions leading to the production of radioactive isotopes as an end product.
2. Large Cross Section - The fusion cross section should be large at low kinetic energies, and the reaction should release much energy (high $Q_{nuclear}$).
3. Small Nuclear Charge - In order to reduce the bremsstrahlung losses, the nuclear charges should be small. Boron with $Z = 5$, must be close to the tolerable limit.
4. Charged Products - The reaction should preferably release most of its energy in the form of charged particles, which can give their energy to the plasma to maintain its temperature. Charged products also have the potential for direct energy recovery, which greatly increases the plant efficiency.
5. Natural Abundance - Ideally, the fuel should be readily available, cheap and easily handled. These factors will greatly influence the final economics of the reactor concept. Tradeoffs should include the breeding of nonradioactive fuels, such as ^3He.

A number of advanced fuel cycles have been proposed. For the purpose of illustration, we consider two sets of reactions; however, as the study of the physics issues progresses and possible tradeoffs are evaluated a different fuel chain may be considered:

$$p + {}^{11}B \rightarrow 3\,{}^4He \qquad Q_{nuclear} = 8.7\ \text{MeV}$$

$$p + {}^6Li \rightarrow {}^4He + {}^3He \qquad 4.0$$

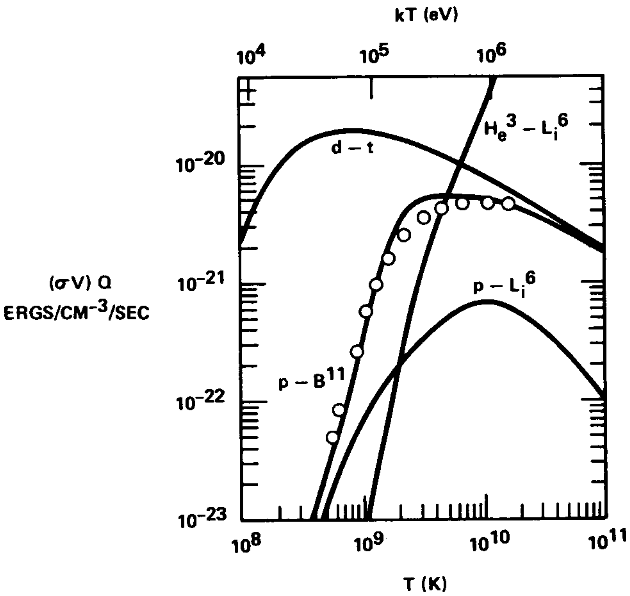
$$^3He + {}^6Li \rightarrow 2\,{}^4He + p \qquad 16.8$$

$$^6Li + {}^6Li \rightarrow 3\,{}^4He \qquad 20.9$$

The thermonuclear power production rates of these fuels peak at high temperatures (Figure 1). At T_i = 300 KeV the p - ^{11}B reaction has a yield $<\sigma v> Q_{nuclear}$ larger than the d-t yield at 10 KeV. The peak for the p - ^{6}Li is an order of magnitude lower than for p - ^{11}B and it also occurs at a high temperature (1 MeV), this reaction can not yet be ruled out as a candidate because its ^{3}He product reacts very effectively with ^{6}Li, a detailed knowledge of the cross section of ^{3}He + ^{6}Li, above 1 MeV is needed to assess the utility of this candidate fuel cycle.

FIGURE 1

ENERGY PRODUCTION RATE AS A FUNCTION OF ION TEMPERATURE

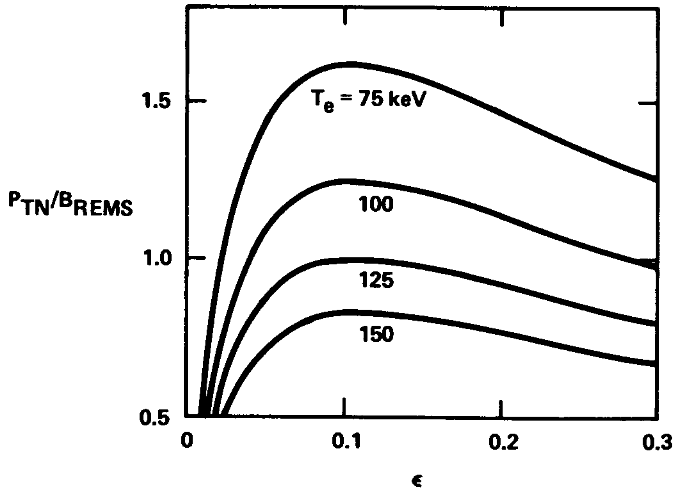


Our attention is then focused on the p - ^{11}B reaction for which energy balance calculations have been carried out.[6,7] Thermonuclear power production exceeds bremsstrahlung losses only if the electron temperature is held below 125 KeV, the optimum of P_{TN}/P_{Brems} as a function of $\epsilon = n_{boron}/n_{proton}$ is **broad (Figure 2) and leaves enough flexi**bility for optimization against other parameters. The charged particle products (α - particles) deliver more than 90% of their energy to the ions of the fuel by Coulomb collisions. This power input to the ions can not balance the collisional heat loss to the electrons if electron temperatures below 150 KeV are contemplated. Ignition is not possible but the 30% power deficit at Te = 125 KeV can be made up by direct ion heating which can be

FIGURE 2

OPTIMIZATION OF P_{TN}/P_{BREMS}

$\epsilon = n_B/n_p$, Ti = 300 keV



accomplished by neutral beam or RF heating. This input power is then amplified by the thermonuclear reactions and the bremsstrahlung output is recovered with a resulting plasma amplification factor $Q \sim 3$. Recovery of the bremsstrahlung radiation can be done efficiently: the hard X-ray radiation passes through a low Z wall and heats a high Z gas in which it is absorbed, the high temperatures reached by this gas allow a very efficient thermal recovery of the X-ray power output.

Synchrotron losses have been evaluated by Dawson;[6] although the estimate is an upper bound which is not specific to the multipole geometry it points out that this loss mechanism does not rule out p - ^{11}B as a thermonuclear fuel. At Te = 150 KeV the upper bound losses are intolerable but at T_e = 100 KeV they are smaller than one quarter of the bremsstrahlung losses. A detailed calculation for the multipole geometry is needed but our expectation is that octopoles or higher order multipoles have such large field free volumes that the synchrotron losses will be much smaller than Dawson's estimate.[6]

Plasma confinement requirements are easy to establish now that we have established that the dominant heat loss is bremsstrahlung radiation. The cooling time $\tau_{brems} = \frac{3}{2}\,\frac{neTe}{P_{brems}}$ leads to an $n\tau_{brems}^{*}$ figure of merit in the neighborhood of 10^{15} cm^{-3}. s, the plasma confinement $n\tau$ should then be much larger. It is quite difficult to extrapolate the transport scaling laws observed in past multipole experiments ($n \lesssim 10^{12}$ cm^{-3}, $T \lesssim 10$ eV) to the reactor regime ($n \gtrsim 10^{14}$ cm^{-3}, T = 300 KeV).

Experiments in the General Atomic dc Octopole have established collisionless diffusion at a rate $D = D_{Bohn}/1000$, extrapolated to the multipole reactor regime this yields $n\tau = 2\times10^{14}$ $cm^{-3}S$ which is inadequate. Experiments at the University of Wisconsin have shown that at low density and with strong magnetic fields(10,11) $D = 2\times10^{8}\left(\frac{T}{m_i n \ell_{11}}\right)^{1/2}$ a diffusion scaling which was predicted by Okuda and Dawson (12) as due to thermally excited convective cells. This result scales favorably to the reactor regime with an $n\tau$ of $6\times10^{15} cm^{-3}S$ which compares favorably with the $n\tau_{brems}$ of 10^{15}.

III. MULTIPOLE REACTOR PRELIMINARY DESIGN USING AN ADVANCED FUEL CYCLE

Several advanced fuel reactions were considered in Section II. Since the primary reaction produces no neutrons, the p + ^{11}B reaction is an attractive candidate for an advanced fuel fusion reactor. In this section we present a preliminary design and configuration data for a multipole reactor utilizing the p + ^{11}B reaction. The methodology presented, while specifically designed for a p + ^{11}B reactor, is generally applicable to any advanced fuel concept.

1. Design Considerations

A 1000 MW(th) unit is used as a reasonable gross power rating to determine typical physical dimensions of an octopole reactor based on the $^{11}B(p, \alpha)2\alpha$ reaction cycle. The plasma parameters and the size for such a device are listed in Table 1.

Thermal Power (MW)	1000
Plasma Volume (m^3)	1000
Major Radius (m)	7
Effective Plasma Radius (m)	2.6
R (inner hoops, m)	5
R (outer hoops, m)	9

TABLE 1

PLASMA PARAMETERS FOR A p-^{11}B OCTOPOLE FUSION POWER REACTOR

B_{bridge} (kGauss)	100
Te (keV)	150
Ti (keV)	300
$n\tau_E$ ($cm^{-3}s$)	1.2×10^{15}
$\varepsilon = n_B/n_p$	0.2
$n_p(cm^{-3})$	1×10^{14}
$<\sigma v>$ Q ($MeV\text{-}cm^3/s$)	3.4×10^{-15}
Power Density (W/cm^3)	1

The reactor is toroidal with four current-carrying hoops embedded in the plasma. The main field is the poloidal field and only a weak toroidal field (~5 kG) is required. In addition, the octopole hoops are superconducting and a transformer core is not required. The coils outside the vacuum chamber levitate the four internal octopole rings. Thus, unlike Tokamak systems, one has significant access space near the centerline of the device and the toroidal field coils do not dominate the design, either in physical dimension or in cost.

2. Octopole Coil Design

The design of the four internal current hoops is one of the most difficult design problems of octopole reactors. These coils must be superconducting to eliminate power losses and levitated to minimize or eliminate supports. Consider first the design of the conductor itself and the cooling of the coil. The maximum field on the surface of the inner hoop is 10T. We assume that it is not necessary for the coils to withstand the magnetic loading but that external coils built into the structure will be used to levitate the coils and to cancel the magnetic loading. This implies that special precautions must be taken to ensure that the current in any of the coils is not activated without the corresponding current in the force-canceling coil being activated simultaneously.

For a cryogenically stable coil operating in a 10T field, an overall current density of 4000 A/cm^2 is reasonable assuming a 20 percent void fraction for liquid helium and a 3 cm zone outside the liquid helium barrier. A material with a large heat capacity, such as Pb, may be provided around the outside of the coil to give thermal capacity at cryogenic temperatures. An analysis of the hoops based on the above assumptions yields the coil parameters listed in Table 2.

TABLE 2

PARAMETERS OF OCTOPOLE REACTOR COILS

	INNER COILS	OUTER COILS
Surface Field (kG)*	100	80
Coil Current (MA)	23	15
Major Radius (m)	5	9
Minor Diameter (m)	0.96	0.76
Superconductor**	NbTi	NbTi
Stabilizer**	Cu	Cu
Current Density (A/cm^2)**	4000	4000
Superconductor/Cu Ratio	1:33	1:40

* Fields corrected to be uniform around coil.

* S/C Could be Nb_3Sn. Stabilizer could be Al.
Current density could be raised to 5000 A/cm^2.

The most promising conductor is that designed by Cornish et al at Lawrence Livermore Laboratory. The Cornish conductor employs workhardened copper as both stabilizer and structure. To provide maximum cooling surface, each conductor is manufactured in four pieces and then soldered together.

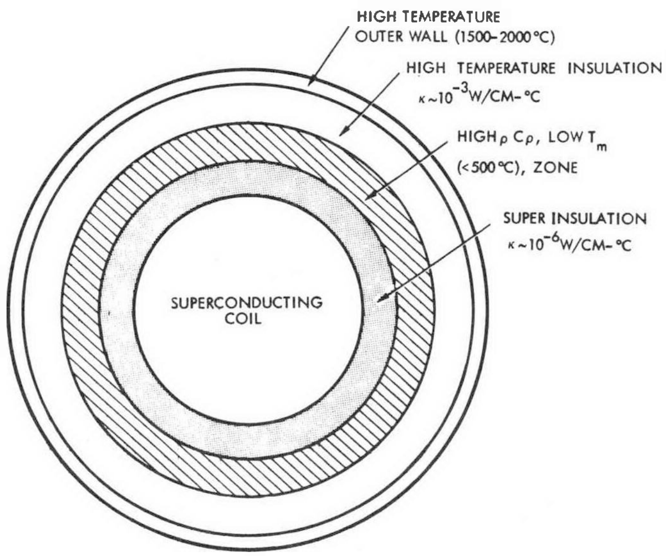
3. Coil Cooling

The octopole coils are imbedded in the plasma and therefore are subjected to an intense surface heat load. Two possibilities for coil cooling exist. The first possibility involves levitating the coils but allows small leads to carry an external coolant. It is found that incident heat loads of approximately 100 W/cm^2 can be removed by using 4 to 8 coolant leads per hoop. A simple calculation shows that, if unprotected, such leads would be subjected to enormous heat loads plus the effect such plasma losses would have on confinement. Thus, magnetic guarding will be essential if leads are used. Guarded leads have been used experimentally on the LASL quadrupole[16] but much further work is required. Such experiments can be done with present and future machines.

The second possibility is to design levitated coils with a large thermal capacity so that days of continous operation are possible without external cooling. The generic design for such a coil is shown in Figure 3

FIGURE 3

GENERIC DESIGN FOR A LEVITATED OCTOPOLE COIL WITHOUT EXTERNAL COOLING



An estimate of the surface heat load to the coil exterior is approximately 100 W/cm^2. The outer wall must therefore be designed for high reflectivity and operation at about 2000°K. To minimize the heat leak, three additional zones are required; one, a superinsulating zone around the cryogenic coil; two, a zone of high thermal capacity that operates up to several hundred °C, and three, a zone of high temperature insulation. Actually, the detailed design of each general zone may consist of several zones and different materials. The middle zone of high (ρC_p) and relatively low melting temperature, (T_m), serves the dual purpose of separating the low and high temperature insulating zones while providing a heat sink within the coil. Lithium is a logical choice for shielding the middle zone from neutrons produced by side reactions. Start with Li at 77°K and allow it to heat to its melting point of 459°K until melted. In this way, the enthalpy consists of both the temperature rise and the heat of melting.

A simple one-dimensional heat transfer calculation has shown that, with the lithium zone and high temperature insulating zones at 22 cm each, fifty hours is required to melt all of the lithium. During this period, the heat leak to the superconducting bath is only 15 W. By using Pb in the superconducting zone as a heat sink to raise the temperature to 12-15°K and using Nb_3Sn, the time required to melt the Li is the shortest time constant. Thus, with a reasonable zone thickness achieved, two or more days of continous operation are possible. Therefore, this design approach seems feasible and should be examined in greater detail.

4. Power Cycle Considerations and Plasma Q Values

Since it is likely that the p + ^{11}B cycle will require supplementary heating and operation as a high-Q amplifier, it is important to have high thermal cycle efficiency. All of the fusion reaction energy is deposited in the plasma by alpha heating and subsequently radiated to the first wall from which the heat energy can be directly used in the power cycle. Therefore, a p + ^{11}B octopole reactor is the fusion equivalent of a coal-fired boiler. For the power conversion the potassium topping cycle system is attractive. This approach has been considered for the 1000 MWt octopole reactor and it is assumed that the operating temperature of the first wall is 1000°C. The gross electrical output is 114 MWe from the K turbine, 67 MWe from the gas turbine and 364 MWe from the steam turbines which yields a gross cycle efficiency of 54.5 percent.

Other advanced conversion concepts may be possible with the low neutron fuel cycle. For example, the bremsstrahlung radiation can be transmitted through a low Z first wall and stopped in a clean gas to be used as the working fluid in an MHD cycle. Advantages gained would be the lack of mineral impurities associated with coal-derived gases and potential cycle efficiencies of 65-70 percent.

An alternate approach to the question of thermal cycle efficiency, is to ask what gross efficiency, η_g, and what plasma amplification factor Q are required to produce a plant of acceptable net thermal efficiency. A simple reactor power balance permits calculation of approximate requirements by omitting auxilliary factors such as the energy required to heat replacement fuel. Assuming net plant efficiency of 30 percent as an indicator of minimum economic interest, one requires plasma amplification factors between 4 and 6 when η_g = 54.5 percent and the recycle power efficiency (η_{rec}) is between 70 and 80 percent. An η_{rec} of 70 to 80 percent is the efficiency range required of 1 MeV proton beam injectors. In perhaps a most optimistic case where η_g = 65 percent and η_{rec} = 80 percent, the Q value for η_{net} = 0.3 is about 2.8. From plasma physics considerations, Q values approaching 3-4 appear feasible.

IV. SUMMARY OF POTENTIAL ADVANTAGES OF MULTIPOLES AND ADVANCED CYCLE FUSION REACTORS.

There are many potential advantages to multipoles and advanced fuel cycle fusion reactors:

- Multipole system allows p + ^{11}B fuel with magnetic confinement.
- The reactor is the fusion analog of a coal-fired boiler and could have the balance of plant costs similar to coal systems.
- The fuel costs will be less than those for a fission plant.
- An intermediate coolant loop is not required.
- Other safety problems are similar to coal plants rather than nuclear.
- The minimal neutron output will not cause damage to structural materials and there will be low neutron induced activity.
- Heating and fueling functions can be separated.

- The fuels are abundant.
- The system has a low environmental impact considering air pollution, radiation release (no gaseous radioactivity), mining and long term waste.

V. FUTURE DIRECTIONS IN PHYSICS AND TECHNOLOGY

The preceeding preliminary reactor analysis has suggested important physics information required to make an improved assessment of advanced fuel cycle multipole fusion reactors. Many important areas can be studied by near term experiments. Clearly, the scaling of the diffusion coefficient, D, and the plasma thermal conductivity, K, with machine size, plasma collisionality, and B are critical. Impurity effects and alpha particle diffusion are very important for reactors and will probably determine the longest feasible burn time. Such effects can be studied in a near-term machine by injecting helium into a p + ^{11}B plasma without concern for the radioactive contamination associated with d-t operation. Also, using p, B, and α ensures that subtle mass effects are not missed. The behavior of the multipole plasma in the high β regime needs to be analyzed. Cooling tubes or leads to hoops can change the concept of an octopole reactor so the effects of hoop leads on diffusion and methods for guarding leads are equally important. An experiment should be designed with fully levitated rings to allow both modes of operation. Also, as found earlier, the self-field of each current hoop is very large in reactor size machines. Some of the coils are in tension while others are in compression, even when the hoops are fully levitated. Therefore, it is necessary to use external coils to reduce these self forces. Any resulting effect on plasma confinement is an important physics question.

Two nuclear physics areas are unsettled and the answers are very important. The first is an accurate measurement of the $^{11}B(p, \alpha)2\alpha$ and other candidate reactions cross section from 50 keV to about 5 MeV, which permits a more accurate determination of the power balance and break-even conditions. The other needed measurements are of the neutron-producing side reactions since only with these measurements can an accurate assessment be made of radioactivity and damage to superconducting hoops.

Reactor studies generate basic questions affecting near-term experimental studies and they can guide policy with respect to the ultimate practical applicability of a given concept. These studies also suggest technological areas for further analysis and experimental work. This preliminary study indicates the pursuit of a more complete reactor study including costs, comparison of power generation concepts and an analysis of multipoles will be most worthwhile. A study of octopoles and higher order multipoles using several advanced cycles is of special importance. The assessment of potential system economics and a comparison with coal, fission, and other approaches to fusion will require a very detailed study of multipole reactors. Finally, as part of any larger reactor analysis, several specific problems and detailed design questions must be addressed. These include: (1) separation of fueling and heating functions, (2) ash removal, (3) analysis of heating with 1 MeV proton beams including source design, (4) design of coils with an without cooling leads, (5) design of the levitation system, trimming coils and control system, (6) analysis of the power cycle, and (7) assessment of neutron damage and activation effects.

VI. MULTIPOLE SCALING EXPERIMENT

Multipole reactor configurations using advanced fuels cycles have potentially great advantages over mainline reactor configurations provided the previous scaling laws and technology forecasts hold true.

1. Experiment Objectives

The primary objective of the multipole scaling experiment is to determine if the favorable results obtained in current laboratory multipoles can be extrapolated to the plasma conditions expected in a realistic fusion device. TRW has formulated the scaling experiment study plan with the primary objective divided into the following issues:

a) Plasma Physics Issues

- Determine and analyze the plasma transport scaling laws over a wide range of plasma parameters 10^{10} cm^{-3} < n$\sim 10^{14}$ cm^{-3}, 1 eV < T $\sim$ 1 keV.
- Demonstrate plasma production and heating and determine their effects on transport properties.
- Provide at high plasma densities and temperatures the experimental support for diffusion and transport theory that multipoles have provided historically in the low density and temperature range.

b) Reactor Physics Issue

- Examine such issues as high β effects, synchroton radiation, impurity control, fueling and ash removal, and the feasibility of magnetic guarding of supports.

c) Technological Issues

- Examine, in the experiment, the technological issues which are raised by the design and development of levitated rings, and which have a direct bearing on the feasibility of an internal ring fusion reactor.

2. <u>Experiment Design Requirements</u>

In the previous section, the objectives of the multipole scaling experiment were defined. To initiate the design of a multipole device, the designer first needs the desired plasma parameters (temperature, density, volume, imbedded magnetic fields, etc.); the power requirement to maintain quasi steady state conditions; the allowable variation, both in time and space, of the specified parameters; and the time sequence of the contemplated experiments. A short discussion follows:

3. <u>Choice of Plasma Parameters</u>

The regime of present multipole experiments ($n \simeq 10^{12}$ cm^{-3}, $T_e \simeq 10$ eV) lies so far from the reactor regime ($n \simeq 10^{14}$ cm^{-3}, $T_e \simeq 150$ keV, $T_i \simeq 300$ keV), that a significant extrapolation of the present scaling laws is required before their applicability to reactors becomes credible.

a) Present multipoles: $n = 10^{12}$ cm^{-3}, $T_e = 10$ eV, $T_i = 1$ eV, $B = 3$ kG, $R = 1.5$ m

b) Proposed TRW multipole: $n = 10^{14}$ cm^{-3}, $T_e = T_i = 1$ keV, $B = 20$ kG, $R = 1.5$ m

c) Small Reactor: $n = 10^{14}$, $T_e = 150$ keV, $T_i = 300$ keV, $B = 100$ kG, $R = 7$ m

The demonstration that the TRW octopole design conditions which are similar to those in a Tokamak ($n \sim 10^{14}$ cm^{-3}, $T \lesssim 1$ keV) can be achieved in a multipole device, would be in itself a success. It would also provide a sufficient data base from which the next generation of multipole experiments could be designed.

4. Power Requirement

Plasma transport dictates the heating power required to maintain a steady state. Confinement times and power requirements can be estimated from the previously presented diffusion coefficients.

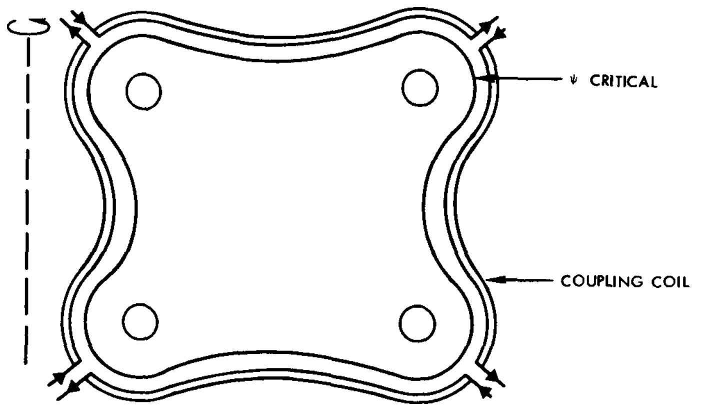
$$\tau = a^2/D , \quad P = nVT/\tau$$

Ignoring the differences between energy and particle confinement times since heat transport measurements have not been completed in past multipole experiments and considering a volume of ten cubic meters, an average field line length of ℓ = one meter, a typical gradient length a = 5 cm and a magnetic field strength B = 20 kG, we obtain the power requirements of 3 MW ($\tau \sim 50$ msec Poloidal Bohm scaling) or 300 kW ($\tau \sim 0.55$ sec convective cell scaling).

5. Plasma Heating

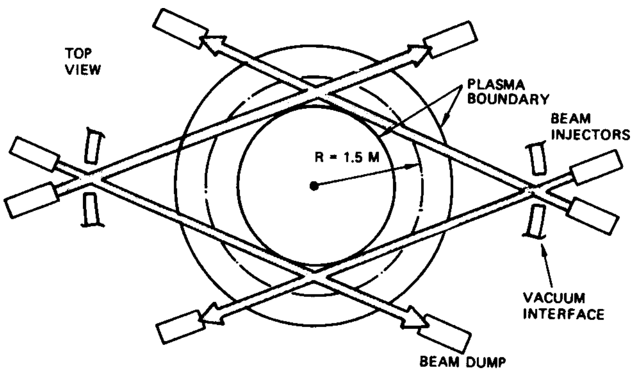
Plasma heating in a multipole geometry to 1 keV temperature and maintaining a density of 10^{14} cm^{-3} is a difficult experiment. The current plan is to use poloidal ohmic heating to form a fully ionized ($\sim$100 eV) target plasma and to raise it to the final temperature (several keV) using neutral beam injection. Figure 4 shows schematically the ohmic heating coil and the injection geometry into the octopole. While neutral beam heating has been extensively investigated in Tokamaks, poloidal ohmic heating requires further study and development.

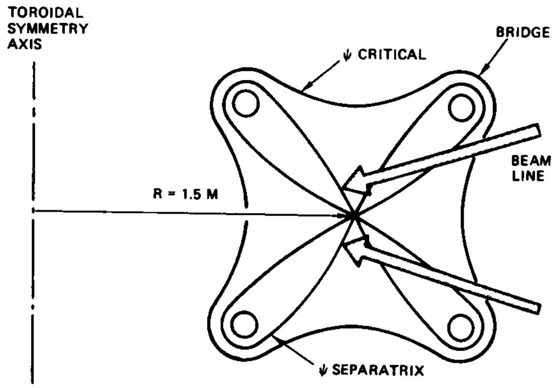
FIGURE 4 A



Schematic of the Poloidal Ohmic Heating Primary Coil.

FIGURE 4 B



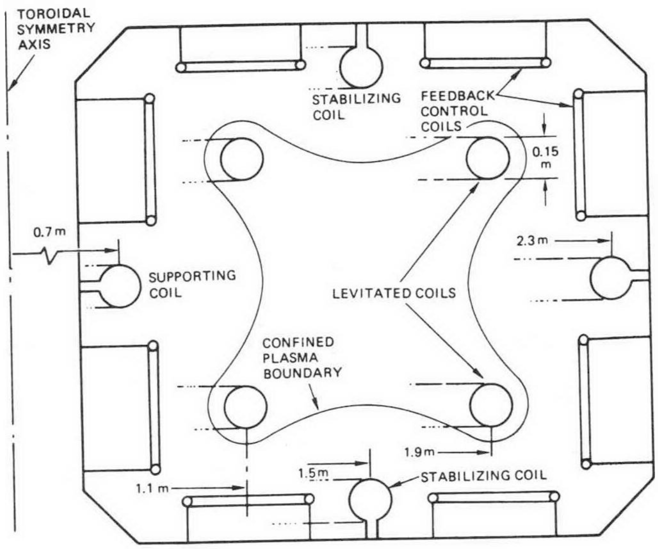


Neutral Beam Injection into the Octopole.

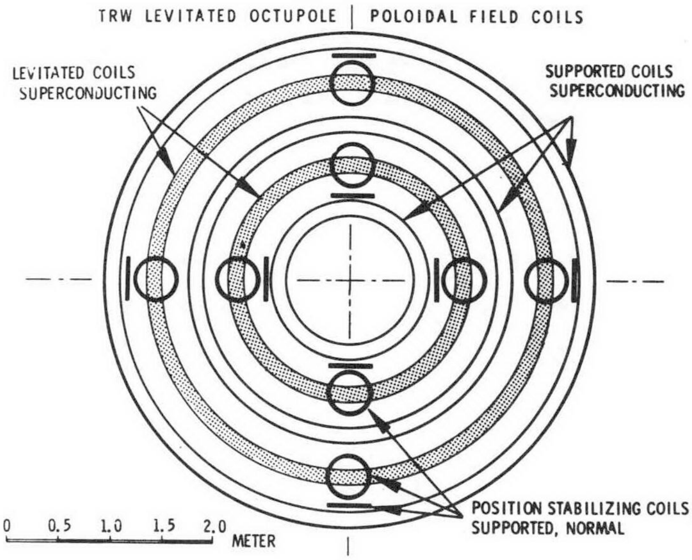
6. Conceptual Levitation and Stabilization Configuration

Figure 5 shows the levitation and stabilization configuration considered most attractive at present and selected for further tradeoffs and performance evaluations during the study. The concepts and technology involved in this approach have been successfully applied in other fusion research projects (Princeton, Lawrence Livermore, Culham Levitrons). This experience provides valuable data and helps to reduce costs and development risk. Steady-state levitation and separation forces are produced by a set of four superconducting coils fixed to the walls. The currents in these coils flow in opposite directions to those of the levitated rings and are selected to provide stable vertical equilibrium at the desired positions. On the assumption of unstable horizontal and rotational motions, the conceptual baseline includes a set of 32 normal coils which provide feedback control stabilizing forces. These feedback loops operate with error signals provided by a configuration of optical sensors which measures the actual positions of the levitated coils.

FIGURE 5



Schematic Cross Section of the Levitated Octopole.



Levitation and Stabilization Conceptual Drawing.

7. System Conceptual Design

The TRW multipole device concept is based on both the particular physics requirements of the experiment and on existing designs used in existing internal ring machines. Some preliminary analyses have been made to determine the configuration of the key elements of the device. The systems diagram of the device is shown in Figure 6 .

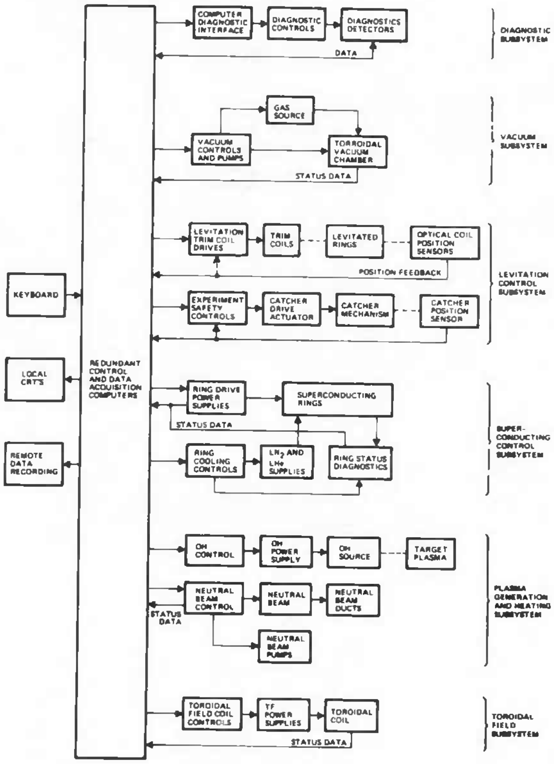


FIGURE 6 Multipole Device Systems Diagram

VII. SUMMARY

We have presented a brief report about the work currently in progress at TRW on a study of advanced fuel multipole reactors. A preliminary analysis of advanced fuel cycle candidates suggests that the use of P + ^{11}B is feasible in a device with low synchrotron radiation and hence, low magnetic field. An octopole was selected as the basis for a conceptual reactor design. Preliminary designs of key systems based on University of Wisconsin octopole diffusion scaling results uncovered no serious engineering difficulties. Several issues of basic physics scaling remain unresolved. A preliminary design of an experimental device to verify the assumed scaling was begun.

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Implosion of Advanced Fuels Using High Energy Heavy Ions*

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ABSTRACT

The use of high energy heavy ions for igniting the fusion reaction in DT pellets appears most promising. It is relatively simple to extend this concept to the implosion of pellets of advanced fuels. An accelerator configuration designed for DT fusion would clearly test advanced fuel pellets in a meaningful way. To obtain useful output power from the catalyzed D reaction, and likely from $D\text{-}He^3$, appears to require a high accelerator efficiency. This requirement limits the choices of accelerator configurations to that of a full energy linac filling several storage rings. The feasibility of meeting the requirements appears quite high. The trend raises the question, yet unanswered, concerning the potential for obtaining useful output powers from more exotic fuels such as $P\text{-}B^{11}$.

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I. INTRODUCTION

The concept of using high energy heavy ions for the implosion and ignition of pellets of thermonuclear materials is closely related to similar concepts using lasers and e beams as the ignition source. The latter have been under development for several years. The very rapid evolution of the promise of heavy ions for the pellet fusion application is due in part to the work done in these earlier programs on advanced pellet designs and calculations on energy deposition requirements.

Another factor, however, is the remarkable impedance match of existing high energy accelerator technology to the energy deposition and power requirements for the pellet fusion application. In this respect, one might note the fact that the ranges of heavy ions with GeV energies are sufficiently short to deposit their energy in reasonable thicknesses of the pellet shell. The use of GeV ions rather than the MeV energies of electrons or protons implies the need for kiloamperes of heavy ions rather than the megamperes of the lighter particles for the same power, and production of kiloamperes of high energy heavy ions appears conceptually straightforward with existing technology. Also important is the fact that accelerator technology for high energy physics is highly developed, and megajoules of beam energy (with protons) exists in many operating machines (the Fermilab accelerator, the 400 GeV SPS at CERN, and each ring of the CERN intersecting storage rings). Plans for construction of 200 GeV colliding beam storage rings, ISABELLE, are far advanced with projected circulated beams of protons of 20 MJ each. The transport of ion beams over long distances and subsequent focusing of these beams onto small targets is common practice in the high energy field. In addition, the normal high repetition rate of conventional accelerators is well matched to the needs of an ignition source for pellet fusion.

For these reasons, the proponents of ion beam fusion view the concept of using ion beams as the ignition source for pellet fusion as one of the most promising of all current ideas. These views were strengthened by the conclusions of the 1976 ERDA Summer Study [1] involving accelerator and pellet experts.

There are, of course, some reservations of the feasibility of the techniques required in addition to some physics questions. Existing experience on high energy accelerator technology for intense ion beams applies almost exclusively to the acceleration of protons. Heavy ions, even with the same currents of singly charged ions, present some unique problems. These involve maintenance of very high vacuum in order to avoid beam loss due to charge changing collisions with residual gas and a fundamental limitation of accumulation time due to charge changing collisions between the ions themselves. The adequacy of heavy ion source currents with a beam quality adequate for acceleration in conventional accelerators has not been demonstrated. In addition, the revolution times of GeV ions in circular machines with reasonable magnetic fields is of the order of a few microseconds; therefore, longitudinal compression to shorten the beam duration to the nanosecond times required for pellet fusion is necessary. Accelerator physicists have some experience with longitudinal compression (bunchers preceding linacs, debunchers between the linac and ring to narrow the energy spread, and bunching for phase stability in rf acceleration in synchrotron), but the experience does not extend to such large compression factors as required for ion beam fusion.

Two solutions for the longitudinal compression have been proposed. Argonne suggested (2) filling the storage ring with many separate bunches each of a time duration equal to the desired beam time on the target. All bunches would be extracted simultaneously and transported to the target in as many independent beam lines. A. Maschke suggested (3) that bunching techniques with very high rf fields could accomplish the required longitudinal compression. If this were done within a ring, the nominal space charge limit of that ring would be exceeded by a very substantial amount; however, Maschke pointed out that if this were carried out fast enough, the ions would cross resonances sufficiently rapidly that the normal beam blowup would not occur. A combination of these two concepts, that is, some rf longitudinal compression and a few simultaneous beams transported to the target, seems to be more straightforward than either alone.

To date, the development of the ideas for heavy ion fusion has dealt almost exclusively with the ignition of DT pellets since fusion of this material represents the minimum requirements on the ignition source. However, the target group of the 1976 ERDA Summer Study did present a calculation (1) of the requirements for igniting DD reactions with seeding by T and He^3. The amounts of the latter used are produced in the D burning; hence, no independent breeding of these materials would be required. This result is shown as case 3 of Table I, which is partially reproduced from the published report.(1) The other cases of Table I apply to the requirements of DT ignition, If one accepts this calculation as representing the requirements for igniting the catalyzed D and D-He^3 reactions, then one can draw some conclusions regarding the promise of very high energy ions for implosion of advanced fuel pellets.

Table I. Target Requirements

Case No.	E (MJ)	ΔT (nsec)	P (TW)	ε (J/g)	φ (%)	Rep Rate (sec^{-1})	Special Features	Confidence Level
1	10	10	600	3×10^7	10	2	Nominal	High
2	1	6	100	2×10^7	10	20	Advanced	Moderate
3	10	10	600	3×10^7	50	20	D Burner	Moderate
4	10	10	600	3×10^7	50	10	Noncryogenic	Moderate

II. DISCUSSION

From the data of Table I, one can see that the requirements of implosion for advanced fuel pellets is roughly an order of magnitude greater than for DT pellets. For cases of moderate confidence, cases 2 and 3, ignition of catalyzed D reactions requires ten times the energy and six times the power level as that for DT. Comparing case 3 with the high confidence level DT case 1, an order of magnitude higher repetition rate and higher efficiency are required for the catalyzed D reaction. In particular, the requirement for high accelerator efficiency has most important implications for pellet fusion with advanced fuels.

Operation of a linear accelerator with 50% of the line power transferred to the beam seems relatively straightforward. On the other hand, it is unlikely that greater than 10% efficiency can be achieved with high energy synchrotrons; therefore, if this requirement for high accelerator efficiency for fusion with advanced fuels is valid, then acceleration to full energy in a linear accelerator may be necessary to obtain useful power output.

Attainment of 50% overall efficiency in a linear accelerator requires that the accelerated beam current represent an appropriate load impedance to the rf system in comparison to the shunt impedance of the linac structure. The latter, of course, depends on the structure. For the purpose of the illustration presented in this paper, I will assume that a current of 300 mA can be accelerated in an Alvarez linac at 50 MHz with an efficiency equal to or exceeding 50%. Since the anticipated heavy ion source current (taking into account rf capture losses) is no more than 50 mA, some mechanism to match the generated beam current to the linac for high efficiency is required. There have been three methods of current amplification in the linac suggested, and all seem relatively straightforward. We have suggested[4] that one or more circular accumulator rings operating between two parts of the linac could be employed for this purpose. A. Maschke has suggested[5] that combining beams from several linacs might be useful or that switching[6] a linac beam into several transport lines with different delay times for subsequent recombination could increase the linac current. The energy at which this linac current amplification should take place is a detailed question which has not been addressed. It will depend on many factors such as the appropriate stripping energy if high charge states are to be utilized. Nevertheless, the energy at which the current is to be matched to the shunt impedance of the structure will be very low (perhaps tens of MeV) compared to the full energy of the linac (possibly tens of GeV). The lack of good efficiency in the source, preaccelerator, and early part of the linac is, therefore, inconsequential.

To claim that ion beam fusion can be accomplished with existing accelerator technology requires setting realistic constraints on various parameters. These are discussed briefly in an earlier report[7] which concentrated on the requirements for igniting the DT reaction with linear accelerator systems. The same constraints apply to implosion of advanced fuel pellets. However, the requirement for high accelerator efficiency and, therefore, an assumed linac current of 300 mA from the source (after rf capture loss in the linac), the multiplication factor must then be six. Any of the techniques of current multiplication will enlarge the transverse emittance of the resultant beam and decrease the number of injected turns which can be accommodated in a given storage ring. We, therefore, assume that a realistic maximum of the number of injected turns into a storage ring be reduced by the same factor.

This change is reflected in the curve of Fig. 1, which displays the required current multiplication factor K of the 300 mA linac beam to the fusion target as a function of linac voltage for the 600 TW target requirements of the catalyzed D reaction. The current multiplication factor K is made up of the product of the number of injected turns into each storage ring S, the longitudinal beam compression L, and the number of beams simultaneously focused onto the target N_B. If we assume that the maximum value of the components of K to be

$$S \leq 400/6$$

$$L \leq 100$$

$$N_B \leq 100$$

then K cannot exceed 6.7 x 10^5, which is shown by the horizontal dashed line of Fig. 1. The intersection of this line with the solid curve indicates a minimum linac voltage of GV. One notes that the product of this target current (300 mA x 6.7 x 10^5 = 200 kA) with a voltage of 5 GV would result in a peak power of 1000 TW. However, pulse shaping is required, [(1)] and only 60% of the total energy is utilized in the last 10 nsec of the pulse, giving the peak incident power of 600 TW. Higher linac voltage implies a smaller required

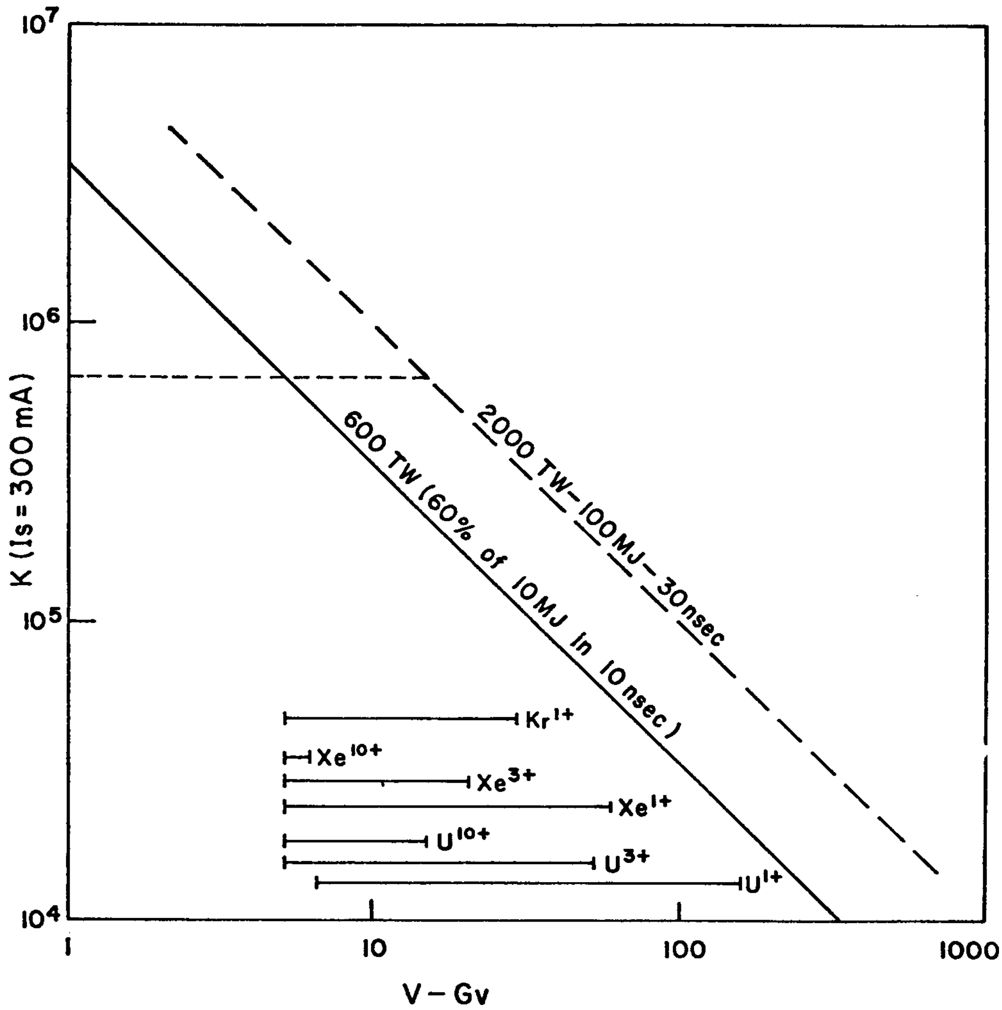

Fig. 1 Current Multiplication Factor Vs. Linac Voltage for High Confidence Target Case

current and current multiplication factor K. Technical feasibility is therefore increased, although the cost of the accelerator system likewise increases.

To assess the implication of the curve of Fig. 1 on any ion or charge state, we make use of the target constraints designated[1] by the target group of the 1976 Summer Study; namely,

$$1 \text{ mm} \leq r \leq 1 \text{ cm}$$

$$\text{and} \qquad Q/M = 30 \text{ MJ/g}$$

These criteria, combined with the range curves, set upper and lower bounds on the possible energies of any given ion. For instance, the upper limit above which the target radius would become too small is 155 GeV for uranium, 58 GeV for xenon, and 28 GeV for krypton. The lower limits are all below the minimum linac voltage of 5 GV with the exception of uranium, for which the lower limit is 6.3 GeV.

The possible range of charge states for any ion is also given by these limiting energies, where the energy of the ion is given by qV when q is the ionic charge. Here it is assumed that the source will always be operated to produce singly charged ions in order to obtain the highest particle current, that stripping to higher charge states would occur at an energy that is low compared to the final linac energy, and that the efficiency of stripping is such that the electrical current remains nearly constant. The assumption of 50 mA of source current (after rf capture losses) independent of ion mass may not properly take into account realistic source performance ($\sim 1/\sqrt{A}$) but will serve for purposes of this illustration.

The very large range of possible ions and charge states which can satisfy our criteria is indicated in the lower left of Fig. 1. Although we have not investigated this entire range in detail to ensure that no other criteria are violated, it seems that one can always design an accelerator system for a given value of K and linac voltage such that none of the individual components of K (S, L, or N_B) exceed the limits stated. In addition, the minimum number of beam lines N_B is determined by the power transmission limit (due to space charge defocusing) of a quadrupole transport line with a given pole tip field. N_B must likewise equal or exceed the number of storage rings required to accumulate the required number of ions without exceeding the space charge limit of any one storage ring.

In all of the above, one has assumed that the target requirements specified are valid. One might explore the possibilities under the assumption that an order of magnitude higher energy (100 MJ) were required (as could be the case for $P\text{-}B^{11}$). The dashed curve of Fig. 1 gives K vs. V when the peak power (again 60% of the total energy) is 2000 TW. One easily sees that there still is a reasonable range of ion species which could satisfy this requirement.

III. CONCLUSION

This exercise indicates some of the very great potential of ion beam fusion. The same accelerator system designed for fusion of DT pellets would be applicable for implosion of advanced fuels. The research and development required for either is therefore identical, unlike other approaches to fusion. If such ion beams can be produced, and the probability appears very high, then there seems little doubt that one can initiate the fusion reaction in pellets of advanced fuels as well as DT. Rather than feasibility, the main question appears to be one of economics, which is not yet clear. Judging by the higher efficiency requirement of the catalyzed D reaction compared to DT and extrapolating this trend to the $P\text{-}B^{11}$ reaction, one might question whether useful output energy were a possibility even though one could ignite the reaction. The need for more calculations on pellet requirements is quite clear.

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MIGMACELL - A LOW-GAIN "DRIVEN" FUSION POWER AMPLIFIER AS AN INTERIM ENERGY SOURCE

(PRIOR TO INFINITE-GAIN "IGNITED FUSION REACTORS)

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ABSTRACT

A fusion program less ambitious in its objectives and technology than the official programs, but such that can be reduced to practice before the "ignited" fusion power reactors with "infinite gain" become a reality, is described. "Migmacell" is designed as a "driven" power amplifier with a gain of 1.5 to 3, energized by fusion. By recirculating a significant fraction of the output power, migmacell would become a self-sustained power source of 100 KW_e to 5 MW_e. Although small (1 m. in dia.), it can operate at much higher "temperatures" than plasmas, by using direct nuclear collisions, instead of heating; and utilize the environmentally acceptable "advanced fuels". Larger plants are envisioned as a plurality of standardized migmacells. Three stages of the Program are completed. Equipment for Stage 4 (1 year) has been assembled and tested. Stage 5 is a 5-year plan to build a demonstration power unit. Economic projections indicate a competitive power source. State-of-the art technology is assumed throughout.

1. The "Philosophy" of the Migmacell System Design

The Migma Program uses the concepts and technologies of the 1970's, not those of the 1950's. Migmacell is a combination of 17 ideas, concepts and inventions, many of them borrowed from operational devices.[1-12]

The "philosophy" of the migmacell design differs from that of the governmental fusion power reactors with regard to (1) size, (2) power gain, (3) power output per unit, (4) method of initiating and maintaining the fusion process, (5) time structure of operation, and (6) extraction of fusion products.

2.1 Size. For the ignited reactors it is known that the larger the device is, the more likely it is to work. In contrast, our premise is that the smaller the fusion device if, the more likely it will work. The stabilizing effect of Larmor radii large relative to the size of the device is theoretically and experimentally known.[12] Furthermore, the smaller the device, the nearer the fuel to the superconductor, the more advantage it will derive from the magnetic field since the fusion power is proportional to the fourth power of the field strength. The fuel ions will be compressed to a smaller volume readily accessible to external "manipulation": instabilities may be controlled by a number of "corrective actions" used in electronic circuits and tubes. Migmacell is about 1 meter in diameter. The fusion reaction volume is 0.01 to 0.1 m^3, depending on the strength of the magnetic field on the superconductor that can be achieved.

Experiments with such small devices are relatively easy to commit. The hardware of an experimental unit costs less than $1 million each. Theories can be tested fast and rejected fast; designs and directions can be easily changed.

2.2 Power Gain. The power gain of the "ignited" reactors, which are the aim of the governmental programs, is infinity[13]. In contrast, our physics and engineering studies have convinced us, that the lower the fusion power gain aimed at, the more likely it is that the fusion power device can be made operational and useful. This is, of course, true only as long as the system is self-propelled, i.e. $Q > 1$. Our detailed calculations (see Sec. 3) based on the state-of-the-art technology give the gain:

$$Q = \frac{\text{power input}}{\text{power output}} = 1.5 \text{ to } 3.5. \qquad (1)$$

The advantages of the low gain are consequences of the physical and engineering advantages of high-energy fusion (in MeV range), as opposed to thermonuclear fusion (in KeV range). (See references 1, 4, 9, 11, 14).

The adverse effect of the low gain is the high "circulating power" needed. A large fraction of the fusion power generated must be fed back into the system. This results in:

- an increased fuel consumption
- the requirement that the conversion of the fusion energy into electricity be done with the efficiency higher than that of thermal conversion.

These difficulties may be offset by the facts that

- the cost of fusion fuel, such as Deuterium or Boron may be negligible compared to the capital cost
- the use of advanced fuels whose fusion products are charged particles, makes high efficiencies less difficult because the direct conversion into electricity becomes feasible. As shown in Sec. 6, the efficiencies in the 80-90% range that are desirable, appear to be possible in direct conversion. Yet, these high efficiencies are not essential. If the "burn" percentage can be increased, efficiencies as low as 30% will make migmacell operational (See Table 2), although only marginally so.

2.3 Power Output per Unit. While the typical design figure for the ignited reactors is 5,000 MW_e per unit, we are aiming at a power output per migmacell in the range of 100 kW_e to 5 MW_e. A large power station can be built by multitude of power cells. Advantages of having the power stations consisting of 10's to 1,000's of identical low to medium power units are:

- economy of mass production. This would introduce standardization in the power plant construction, a feature long desired by the utility industry.
- Power stations can be made large or small. As a result, the pattern of the power distribution in the nation will change. Local stations (e.g. for peak load) would become possible. The costly power transmission can be reduced. A study of the favorable effect of an eventual migmacell based power production on distribution of electric power has been presented by the Schenectady "think tank"[23]
- Planning and construction time would be reduced, thus reducing the

already high and increasing indirect costs of inflation and interest during construction.

- Power plant size can be increased with load eliminating the need for large overcapacity to allow for future growth.

2.4 Method of Initiating and Maintaining the Fusion Process. Ordered motions of the fuel ions are the key to the operation of migmacell. It is our basic premise that random motions of fuel ions have been the main obstacle to achieving the controlled fusion conditions in plasmas. Random motions can be confined by gravitational and inertial methods (stars and the fusion bomb), but not necessarily by the magnetic confinement. The nature of the magnetic forces acting on the randomly moving particles of both signs of charge is intrinsically different from the gravitational and inertial forces. The latter are independent of the sign of charge and the direction of motion; the former are not.

Migmacell is based on ordered motions of ions, thus the magnetic field acts like a guiding field rather than the "pressure"field. The ambipolar potential of migma is always maintained positive, by electronic feedback control of the number of electrons. In such a system in which both the electric and magnetic fields are present, the electron motions become quasi-ordered too. The electrons in migmacell are expected to exhibit motions similar to those of the "solid body rotation" of the electron space charge in a magnetron[16], as well as oscillations[7] (as in a triode oscillator.)

2.5 Time Structure of Operation. The mode of injections into migmacell is either DC or a continuous train of "buckets", or "bunches", like in linacs. This is in contrast to the colossal bursts every few seconds of the proposed ignited machines. We reject the idea that one should build a fusion engine the same way one builds a fusion bomb. The engineering considerations give practical preferences to steady operational-energy generators. The utilities have no experience with electric generators operating in big pulses.

2.6 Extraction of Fusion Products. The small size of the device and the relatively low average fuel ion density renders it possible to extract the charged fusion products at the energies near to their original energy as released from the reaction, that is in the MeV range. This contrasts with other approaches in which it is assumed that the fusion products would thermalize before extraction. This also facilitates removal of high charge nuclei which "poison" fusion reactors.

By having the nuclei produced in fusion in the energy range 1 to 10 MeV, one can apply magnetic beam-shaping methods such as magnetic cusps, to transform the curly radial migma motions of these nuclei into a beam. (This process is the reverse of the "electron ring acceleration" in which a beam is transformed into a curly, radial movement by a magnetic cusp.)

Once the nuclei produced in fusion are made into a beam, their kinetic energy can be converted into electric energy by directing this beam into a decelerator i.e. an accelerator at a repulsive potential. As the beam is being slowed down, it will give its kinetic energy to the rings of the decelerator via the vacuum displacement current. This, in turn, will result in megavolt potentials suitable for power transmission. Furthermore, the exiting beam can be modulated to obtain the AC output.

3. Fusion Power Generated in Migmacell.

Viability of migma as power amplifier depends on the power balance. Our economic projections[19] have shown that, in order for a migmacell to be viable, it must have fusion power in excess of 100 kW. In a detailed self-consistent calculation[18], the migma theorists have shown that migmacell can operate in the regime in which the ratio of the kinetic and magnetic energy density is unity, $\beta \simeq 1$.

We present here the results of calculations[20] of the power balance which shows that tens of Megawatts per m^3 of fusion power density can be produced in migmacell. This program requires much computer time and is expensive. It has been used to develop analytic models which can show trends in important parameters. However, only the first stage of the calculations has been completed, which assumes the mirror ratio produced by the external magnetic field only (case A, "semi-open" mirror). For the mirror ratio enhanced by the diamagnetic effect, this author presents his estimates (case B, "semi-closed" mirror). No numbers are given for "plugged mirror". Therefore, the presented power balance results are conservative.

3.1 Three Fuel Cycles. Three fuel cycles have been considered:

1- pure ^3He

2- pure D

3- mixed D + ^3He.

For each of the fuels, the results for two cases, A = "semi-open" mirror, B = "semi-closed" (diamagnetic mirror ratio) in Tables 1, 2 and 3.

3.2 Results for Case A: "Semi-Open" Mirror.

Pure ^3He (1A) and pure D fuels (2A).

Results show a fusion power density of 10 Megawatts/m^3 for pure D fuel, for a 7 tesla external field, and the scaling proportional to the fourth power of the field strength. This is shown in Figure 1.

Mixed D-^3He fuel (3A).

Using the results for the pure fuels, we estimate a fusion power density of 30 Megawatts/m^3 for D-^3He fuel. These fusion power densities, 2 to 30 Megawatts/m^3, are achievable at the average ion densities in the range 5 x 10^{13} - 10^{14} ions/cm^3 which have already been reached in the major plasma devices.

3.3 Results for Case B: "Semi-Closed" Mirror. Diamagnetism increases the effective mirror ratio which, in turn, increases the burn.

Pure ^3He (1B) and pure D (2B).

Results show a fusion power density of 6 Megawatts/m^3 for pure ^3He and 60 Megawatts/m^3 for pure D.

Mixed ^3He-D (3B).

Using the results for pure fuels, we estimate a fusion power density of 100 Megawatts/m^3 for the mixed ^3He-D fuel.

3.4 Power Flow Diagram. The results have been displayed in terms of the Power Flow Diagram shown in Figure 2. The accelerator (ACCEL) injects an ion current I (amp) at the accelerating voltage V (MV), that is, the beam power P_I (MW) into the MIGMACELL. The accelerator efficiency is $\eta_I = P_I/P_{circ.}$, where $P_{circ.}$ = the circulating power. The beam trapping efficiency to make migma is not a free parameter, but is compiled as a function of migma density and other ractors. The fusion power released is determined by the computed quantity: BURN percentage = (equiv. fusion rate): ion loss rate due to multiple coulomb and nuclear elastic scattering.

The fusion rate plus power output from the cell is partitioned between charged particles, CP; neutrons, n; electro-magnetic radiation, r; and leaking electrons, e. Charged particles are split into two components:

charged fusion products and non-fused fuel ions. The power carried out of the cell by the sum of these two components is labeled P_{CP}, and that taken away by the neutrons, P_n. The power carried by radiation comes only from the stored fuel ions, because the fusion products storage time in migma-cell is short. The sum of the radiation and electron carried power is labeled $P_{r,e}$.

The efficiencies for direct and thermal conversion are η_{DC} and η_{TC} respectively, so that $\eta_{DC} = P_{DC}/P_{CP}$ and $\eta_{TC} = P_{TC}/(P_n + P_{r,e})$. The waste heat $HEAT_{DC} = P_{CP} - P_{DC}$ (and similarly for TC). The net electric power output is $P_{net} = P_{DC} + P_{TC} - P_{circ}$. The waste heat from ACCEL is $HEAT_{acc} = P_{circ} - P_I$.

3.5 Discussion of Efficiencies. The efficiencies are displayed in Table 1.

Accelerator Efficiency, η_I. So far, there has been no demand to the accelerator manufacturers to increase the efficiencies of ion accelerators. Efficiencies of the commercially produced 2 MeV electron accelerators with the isolated core transformer, ICT, are: power line - to - tube = 0.9; and tube - to - beam = 0.9. In a mature design, P_{circ} from direct converter (decelerator) to the accelerator will flow directly, from each ring of decelerator to the corresponding accelerator ring, i.e. deceleration and acceleration will be in the same tank. Thus, we take: $\eta_I = 0.9$. We think the eventual injector will be Linac whose η is known to increase with the beam power and can probably exceed 0.9 under a development program.

Thermal conversion efficiency $\eta_{TH} = 0.4$ was used in all cases.

Direct conversion efficiency η_{DC}. Studies of direct conversion collectors in keV range at LLL have indicated [21] $\eta_{DC} = 0.88 - .97$. The migma program envisages a more efficient system using the decelerator technique.

Minimal efficiency for direct conversion η_{min} is obtained by seeking the value of η_{DC} when $P_{net} = 0$ which implies "engineering breakeven". η_{min} are given in Column 3 of Table 1. We see that the migma system will be self-sustained for η_{DC} as low as 0.3 for mixed fuels. The situation is most difficult for pure ^3He, which requires η_{min} - 0.7 with the state-of-the-art technology. We note, however, that in arriving at η_{min}, we assume the simplest conversion into heat, $\eta_{TH} = 0.4$, without the advanced thermal conversion methods such as electrochemical with $\eta_{TH} \geq 0.65$.

3.6 Technical Problem I: Superconductive magnet. The state-of-the-art superconducting magnet that can be designed and made today would not allow for the space between the migma chamber and the magnet structure. This excludes the neutron moderator and absorber. Pure D fuel will be releasing 1.1 MW, and ^{3}He - D, 0.1 MW of neutron power. This power cannot be taken by the superconducting magnet cooling system, as it will result in very high rate of liquid helium boiloff. Only pure ^{3}He is neutron free. Therefore, we are forced to consider only ^{3}He fuel as the possibility for our first migma fusion power amplifier.

3.7 Technical Problem II: Ion current. Column 1 of Table 2 reveals the problem No. II: ion currents of the order of 1 to 2.4 amps are needed at Megavolts accelerating voltages. Technology of ampere ion accelerators at Megavolts is under development for the super high energy accelerator at CERN in Geneva. Also, R. Martin is developing an 0.5 amp 4 MV machine for inertial fusion at Argonne. From an accelerator that may be built commercially, we hope to get 0.03 amps of ^{3}He. This limits the power output to about 1% of the values listed in Columns 6-8, i.e. from 6 MW to 60 KW (case A), and from 180 KW to 2 KW (case B).

4. Conclusions

Referring to Tables 1, 2 and 3, a number of conclusions can be drawn. We selected some of them:

Conclusion I. Using mixed fuel D - ^{3}He, driven power amplifiers with gain Q = 3 would produce net useful power of 2 MWe, if direct conversion efficiency of 0.9 can be achieved. However, the superconductive magnet, giving 7 tesla in the middle of the cell, of sufficiently large size needed to avoid the neutron-induced liquid helium boiloff is not within the state-of-the-art.

Conclusion II. Mixed fuel D - ^{3}He migmacell would operate at the engineering breakeven, with as low a direct conversion efficiency as 0.3, if the neutron problem can be handled.

Conclusion III. Pure D fueled migmacell could generate 0.6 MWe with direct conversion efficiency of as low as 0.7, when 2 amps of D_2^+ ions current at 2 MeV becomes available, and if the neutron problem can be handled.

Conclusion IV. Pure ^3He fuel is the only one for which a large enough superconductive magnet can be built with certainty, but it requires an accelerator delivering 1 amp of $^3He^+$ ions at 4 MV, in which case it will have a fusion gain of 2 and a useful power output of ~1 MWe, provided a direct conversion efficiency of 0.9 is possible.

Conclusion V. With the state-of-the-art accelerator which could probably deliver 30 ma of $^3He^+$ at 4 MV, a pure ^3He migmacell can be built with a fusion gain of ~1.05 and delivering 10 KW of unconverted net fusion power.

5. How Have Our Experiments Progressed?

We have built four operational laboratory models of migmacell named Migma 1, Migma 2, Migma 3 and Migma 4. We did experiments with the first three models. We have achieved:

5.1 - a reliable production of deuterium (and ^3He) migma (Fig. 3);

5.2 - an energy confinement time of 2 seconds, which is about 100 times longer than that of any presently operating devices (Fig. 4);

5.3 - a collisional energy of about 1 MeV. To achieve the same collisional speeds in a plasma, plasma would have to be heated to 10 billion degrees centigrade, this is 10^2 to 10^3 higher than the equivalent temperature of plasma fusion devices (Fig. 5);

5.4 - the total number of deuterium ions stored in Migma 3 of 1.4×10^{10}. This number is limited by the injection rate and vacuum, which were 0.070 ma and 6×10^{-8} torr, respectively. The average ion density was $n = 3.5 \times 10^8$ d^+/cm^3. Migma has exhibited no instabilities at this density. In contrast, a plasma machine with the similar embodiment, DCX-1 of the 1950's, had shown clear "negative mass" instability at 100 times lower density. Our theoretical calculations show that the migma orbit configuration ("rosette") is responsible for this stability;

5.5 - the product of density and confinement time achieved is: $n\tau = 10^9$ sec cm^{-3}. This was a 10^4 -fold improvement over

Migma 2 (Fig. 5);

5.6 - the overall performance indicator[23,24], the product of temperature, density and time reached is $Tn\tau = 3 \times 10^9$ MeV sec cm^{-3}. A glance at Figure 8 shows that this value $Tn\tau$ has brought migma in the same ball park as the leading fusion devices of the governmental fusion programs;

5.7 - Migma 4 is expected to leap 100 fold to 1,000 fold in the overall performance from Migma 3. This is expected by the 100 fold increase in density and 10 fold increase in the confinement time. To accomplish this a 100 times better vacuum, and 7 times better ion injection is needed. In the best runs in December 1976 both have been achieved. With the pumping speed of 100,000 liters/sec, a vacuum of better than 10^{-9} torr has been maintained while 0.5 ma of ion beam was being continuously injected into the chamber.

The experiments with this system could not be done because of lack of funds. The tests with Migma 4 are considered "critical" because the migma fuel will enter the regime in which the ω_p/ω_c exceeds 1.

6. Economic Projections of Migma Fusion

An elaborate economic study[19], using the generally established procedures in estimating and projecting capital, operating and R & D costs for power plants shows that migma fusion can generate electricity for 1¢/kWh to 6¢/kWh, depending on level of development which is competitive with present power sources. It also indicates that present power sources are subject to large increases in cost.

These projections have been made without invoking a crash program, i.e. on the conservative assumption of no extraordinary intensive development effort that may be dictated, for example, by a crisis or by defense applications.

Thanks are due to Robert A. Miller for his invaluable aid in preparation of this paper.

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d) Design Considerations for a Migma Advanced-Fuel Fusion Reactor by J.E. Golden, R.A. Miller, B.C. Maglich, S.R. Channon, J.R. Treglio; p. 1018;

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12. Rebuttal of the ERDA-appointed Panel ("Robson Panel") to evaluate the Migma Program of Controlled Fusion entitled "Analysis of the Final Report" by Fusion Energy Corp. Staff. We list titles of the main articles:

1 - A "self-consistent" analytic calculation of diamagnetism,
2 - Numerical relaxation calculation of self-consistent diamagnetism,
3 - Radiative loss,
4 - Fusion Power Generation,
5 - Migma 4 Parameters,
6 - The relevance of DCX data to migma,
7 - Numerical simulation of negative mass,
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24. See e.g. B. Maglich, "Unified Criterion of Controlled Fusion" in Ref. 2.

TABLE 1: Efficiencies (See Fig. 2)

Fuel	Case	η_1 accel.	η_{DC} dir. con.	η_{DC}^{min} engin. break-even	η_{TH} therm.
1 ^3He	A	0.9	0.9	0.8	0.4
	B	0.9	0.9	0.6	0.4
2 D	A	0.9	0.9	0.7	0.4
	B	0.9	0.9	0.5	0.4
3 ^3He D	A	0.9	0.9	0.5	0.4
	B	0.9	0.9	0.3	0.4

TABLE 3: Power Ratii (See Fig. 2)

Fuel	Case	P_{circ}/P_{net}	P_{heat}/P_{net}	Fusion Gain Q	Engineering Gain G
1 ^3He	A	7.6	2.2	1.5	1.1
	B	2.47	0.8	1.8	1.4
2 D	A	4.5	2.7	1.9	1.1
	B	2.0	1.4	2.3	1.5
3 ^3He D	A	1.7	0.9	2.2	1.6
	B	0.7	0.6	3.5	2.4

Case A: "Semi-Open" mirror (mirror ratio increased by migma focusing action, but without diamagnetic effect). Case B: "Semi-Closed" mirror (mirror ratio increased by both focusing and diamagnetic action).

TABLE 2

Power Flow (See Figure 2)

(Power in MW, Voltage in MV, Current in A) Efficiencies are listed in Table 1 for each case.

Fuel	Case	I ion current	V accel.	P_1 acc.	$Heat_{acc.}$ Waste	BURN	P_{CP} charg. part.	P_n neutr.	$P_{r,e}$ rad., elec.	P_{DC} dir. con.	P_{TC} therm. con.	$Heat_{DC}$ Waste	$Heat_{TC}$ Waste	$P_{circ.}$	$P^{electric}_{net}$
1 ^3He	A	1.05	4	4.1	0.4	0.3	5.4	0	0.7	4.9	0.3	0.5	0.4	4.5	0.6
	B	0.75	4	4.0	0.4	0.5	6.6	0	0.7	5.9	0.3	0.7	0.4	4.4	1.8
2 D	A	2.4	1	2.4	0.24	0.5	3	1.1	0.5	2.7	0.6	0.3	1.0	2.64	0.6
	B	2.0	1	2.0	0.2	0.7	2.9	1.3	0.4	2.6	0.7	0.3	1.0	2.2	1.1
3 ^3He D	A	1.7	1	1.7	0.17	.13	3	-	0.8	2.7	0.4	0.3	0.5	1.87	1.1
	B	1.5	1	1.5	0.15	.35	3.8	0.1	1.3	3.4	0.6	0.4	0.9	1.65	2.35

Case A: "Semi-Open" Mirror

Case B: "Semi-Closed" Mirror

$P_{circ} = P_1 + Heat_{acc}$

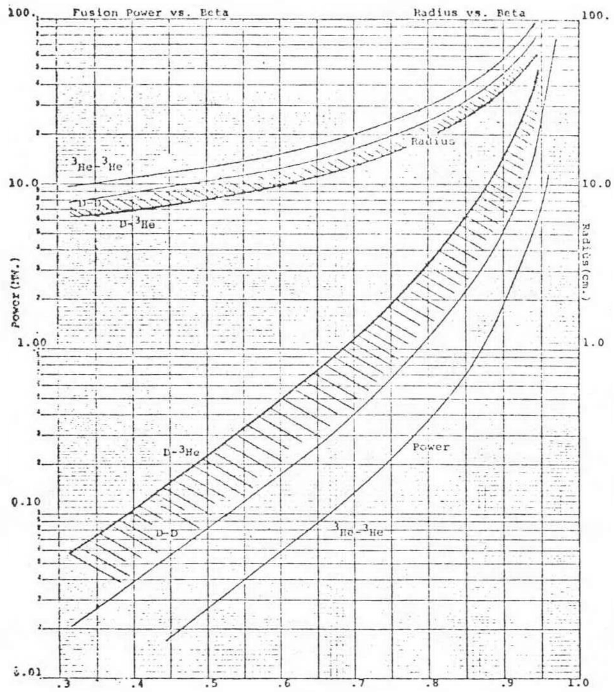

FIG. 1. FUSION POWER IN MW (LEFT) AND RADIUS OF MIGMA IN CM (RIGHT), vs RATIO OF KINETIC-TO-MAGNETIC ENERGY DENSITY, BETA, [β_m]. FOR 3 FUELS: PURE D, PURE ^{3}He AND 50:50 MIXED. B_o = 7 T, AXIAL MIGMA LENGTH, L = 50 CM: RATE ENHANCEMENT DUE TO INCREASED CENTRAL DENSITY λ = 3.

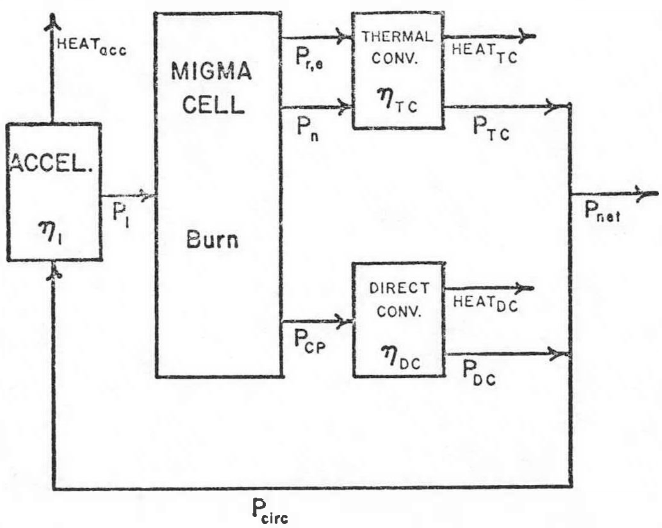

FIG. 2. POWER FLOW DIAGRAM

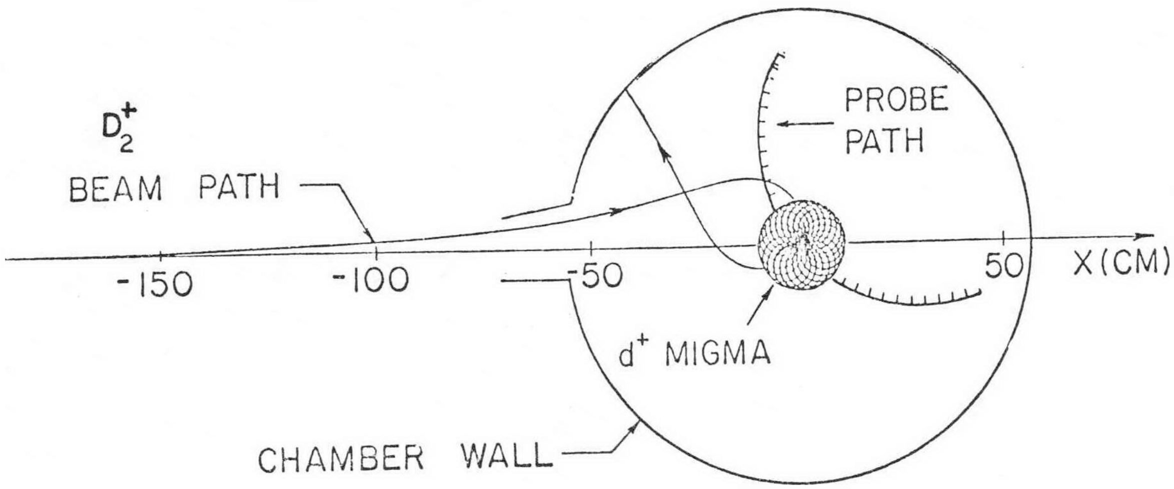

FIG. 3. METHOD OF MIGMA PRODUCTION IN MIGMACELL MODE L 3. D_2^+ BEAM OF 1.2 MEV IS INJECTED FROM THE LEFT AND DISSOCIATED WITHIN ± 0.5 CM FROM THE CENTER OF THE SUPERCONDUCTIVE MAGNET WITH CENTRAL FIELD STRENGTH OF 3 TESLA. THE 2-STEP DISSOCIATION PROCESS STARTS WITH THE LORENTZ AND GAS DISSOCIATION, CREATING "SEED" MIGMA, THEN THE COLLISIONAL DISSOCIATION TAKES OVER. THE STATISTICAL FLUCTUATIONS IN THE NUMBER OF IONS IN MIGMA ORBITS INDUCE RF CURRENTS IN A DISC (NOT SHOWN) PLACED ABOVE THE MIGMA. THIS IS PICKED UP AS AN INCOHERENT RF SIGNAL WHOSE POWER IS PROPORTIONAL TO THE NUMBER OF IONS IN MIGMA. AT 10^{-7} TORR, 10^{10} d^+ WERE STORED, LIMITED BY VACUUM.

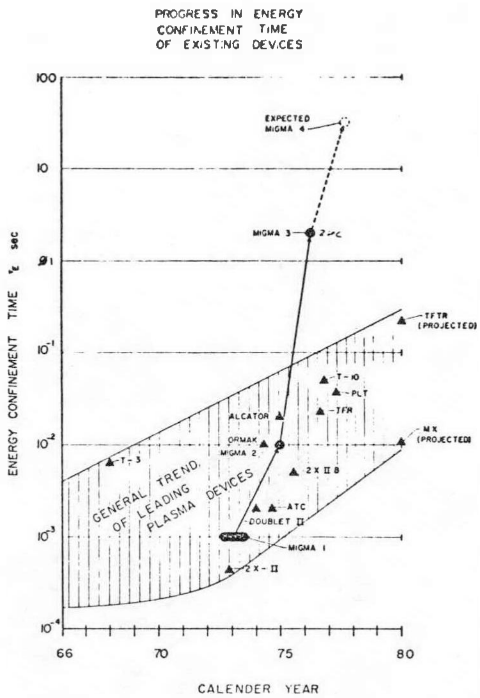

FIG. 4. 10-YEAR PROGRESS IN ENERGY CONFINEMENT TIME OF EXISTING AND PROJECTED PLASMA DEVICES IN THE USA AND USSR, COMPARED WITH THAT OF THE MIGMA PROGRA PROGRAM.

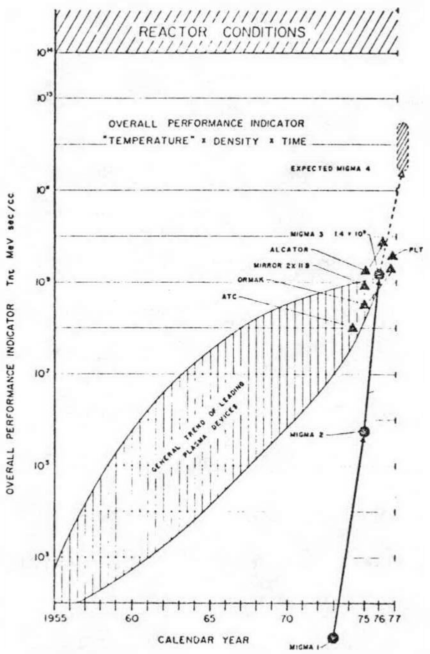

FIG. 6. TREND OF THE 20-YEAR PROGRESS IN THE OVERALL PERFORMANCE INDICATOR, $Tn\tau$, ("TEMPERATURE" x DENSITY x CONFINEMENT TIME) FOR THE US MAINLINE FUSION DEVICES AND MIGMA. THE LAST MIGMA EXPERIMENTS TOOK PLACE IN APRIL 1976.

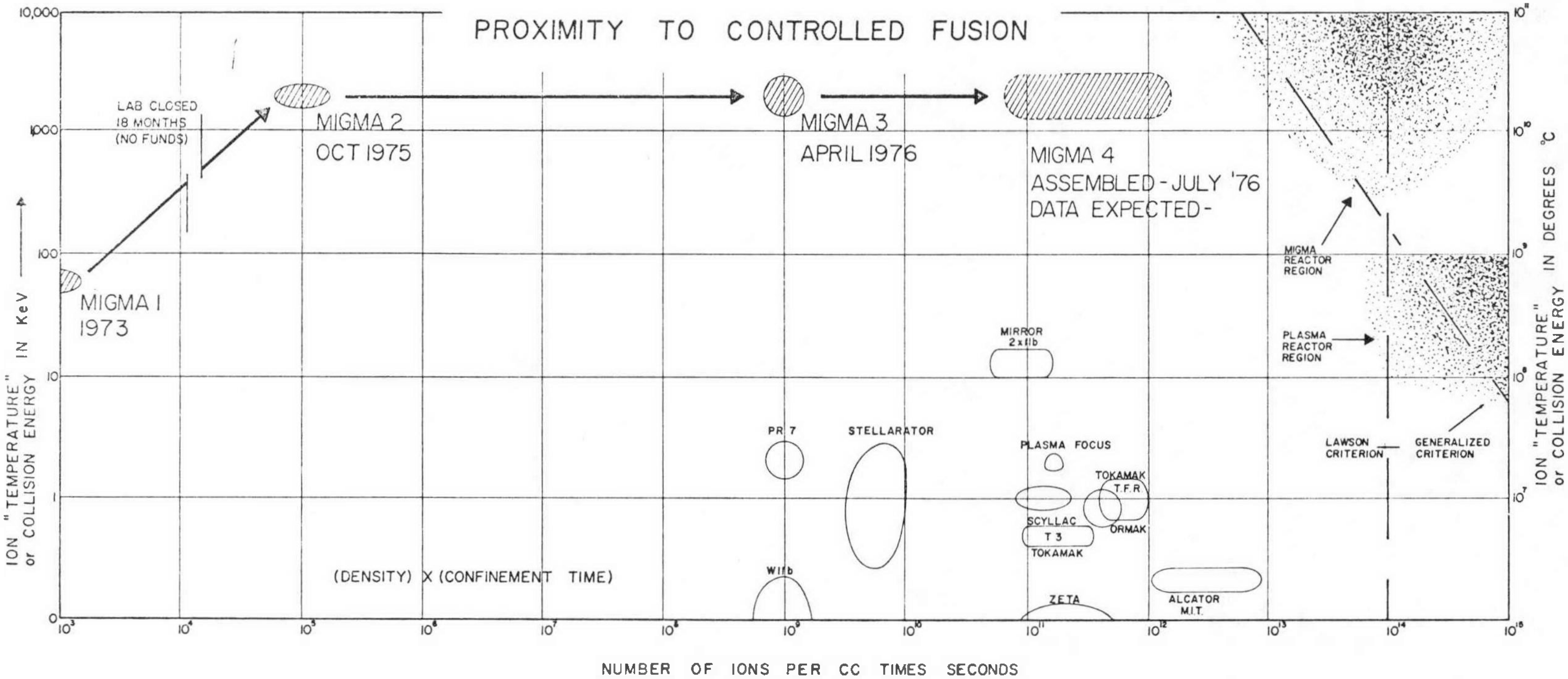

FIG. 5. PROGRESS OF THE MIGMA PROGRAM IN THE (EQUIVALENT) "TEMPERATURE" versus $n\tau$ PLANE, COMPARED WITH THAT OF THE EXISTING AND PROJECTED MAINLINE FUSION PLASMA DEVICES, (THE LATTER ARE TAKEN FROM PEASE, PHYSICA 82C, 1976). TWO HEAVILY SHADED AREAS ON THE RIGHT CORRESPOND TO THE REACTOR CONDITIONS FOR THE ADVANCED FUELS AND DT, RESPECTIVELY.

Laser-Fusion Employing Direct Nuclear Pumped Lasers and Advanced Fuel (D-D-T) Pellets*

by

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ABSTRACT

Direct Nuclear Pumped Lasers (DNPLs) that employ neutrons to pump gas lasers by creating MeV-charged particles through nuclear reactions in the laser cavity have been demonstrated experimentally. It is potentially attractive to use neutrons from a fusion-pellet microexplosion to pump the laser since this avoids the inefficiencies and cost of converting the neutron energy to electrical energy as required for conventional lasers. A deuterium-rich advanced fuel (D-D-T) pellet is especially attractive since the reduction in tritium breeding requirements provides more freedom in blanket design, making more neutrons available for pumping the DNPL. Indeed, preliminary calculations suggest that adequate neutrons can be obtained for a successful laser-feedback device.

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*This paper represents a combined oral presentation and poster-session display. The oral presentation was substituted in the program when a paper by L. Wood (LLL) was withdrawn due to classification difficulties.

I. INTRODUCTION

A DNPL utilizes MeV ions produced by neutron-driven nuclear reactions to pump the laser. Examples of nuclear reactions that have been used in experiments include:

$$n(^{3}He,p)T: \quad {}^{1}n_{0} + {}^{3}He_{2} \rightarrow {}^{1}H_{1} + {}^{3}H_{1} + 0.8 \text{ MeV}$$

$$n(^{235}U,f)ff: \quad {}^{1}n_{0} + {}^{235}U_{92} \rightarrow 2ff + \nu n + 165 \text{ MeV}$$

and

$$n(^{10}B,Li)\alpha: \quad {}^{1}n_{0} + {}^{10}B_{5} \rightarrow {}^{7}Li_{3} + {}^{4}He_{2} + 2.3 \text{ MeV}$$

where ff indicates energetic fission fragments, ν is the number of neutrons/fission and other symbols follow standard convention.

Two broad classes of gaseous lasers are possible: lasers using boron or uranium coated tube walls, or alternately, designs using mixtures containing gases such as ^{3}He, UF_6, or BF_3. The latter are best suited for high-pressure operation since MeV ions are produced throughout the volume of the laser medium. Neutrons to drive the reactions in experimental devices are presently obtained from high-flux pulsed fission reactors although other sources such as particle accelerators or fusion devices, such as a plasma focus, are possible.

MeV ions slow down in gases via both excitation and ionization collisions. High-energy secondary electrons produced in ionization events carry off a major portion of the ion's energy, and at the pressures of interest here the subsequent ionization-excitation produced by these electrons provide the prime energy flow channel.[1-5] The high-energy "tail" on the distribution is a distinguishing feature that can lead to non-equilibrium excitation. (One important exception is the CO laser where a significant portion of a fission fragment's energy can be transferred directly to vibrational states in molecular gases.[6-7]) In this sense, DNPLs are similar to electron-beam driven lasers.[8] The key difference, from a practical point of view, is the possibility of pumping large volumes using neutron penetration.

Since electric fields are absent, the electron temperature in the DNPL plasma is characteristically low, nearly in equilibrium with the gas temperature.[9] In this sense the DNPL plasma resembles the "afterglow" regime in gaseous discharges, and recombination provides another important mechanism for selective excitation.[10]

II. DNPL SURVEY

While the concept of a DNPL laser virtually dates back to the discovery of the laser itself,† experimental verification was not achieved until 1974 when MacArthur and Tollefsrud[6] obtained lasing in CO using a uranium coated tube in the SANDIA fast burst reactor. Closely thereafter, Helmick, et al.[12] achieved direct pumping of a He-Xe mixture and DeYoung, et al.[13,14] reported lasing with Ne-N_2. These and other results reported through Jan. 1977 are summarized in Table 1.

With DNPL research only in its infancy, many lasers beyond those listed in Table 1 can safely be anticipated. The output powers indicated in Table 1 are small, but as shown in following sections, high powers are predicted with scales up to large-volumes and higher pressures. Perhaps the most attractive laser listed from a potential power-efficiency point of view is the CO case where an efficiency (laser output/nuclear energy in) well over 1% is predicted.[6]

The He-CO[18] and Ne-N_2 lasers[13,14] offer the lowest threshold neutron requirement. They are, in fact, the only lasers to date to have been achieved using a TRIGA reactor, as opposed to a "fast-burst" reactor. The He-Hg laser[16,17] represents the first DNPL with visible output, although gain has been reported on the 8446-Å oxygen transition in a He-Ne-O_2 mixture. Among other uses, output in this range appears most attractive for laser-fusion coupling.

III. SCALED-UP DNPL EXPERIMENTS

If DNPLs are to be used for laser fusion, it is imperative that MJ laser devices be built in the near future to test concepts. In this section we consider the feasibility of doing this using existing fast-

†The first unclassified DNPL study known to the authors is by L. Herwig in 1964.[11]

Table 1. A summary of DNPL's achieved to date.

Laser	Ref.	Wavelength	Thermal Flux Threshold (n/cm^2-sec)	Peak Laser Power
He-Hg	16,17	6150 Å	$\sim 1 \times 10^{16}$	~1 mW
He-CO;He-CO_2	18	1.4543μ	$\sim 3 \times 10^{14}$	~2 mW
CO	6	5.1-5.6μ	$\sim 5 \times 10^{16}$	>2 W
He-Xe	12	3.5μ	$\sim 3 \times 10^{15}$	>10 mW
Ne-N_2	13,14	8629 Å and 9393 Å	$\sim 1 \times 10^{15}$	~2 mW
^{3}He-Ar	15*	1.79μ	$\sim 2 \times 10^{16}$	~50 mW

*These workers also indicate (unpublished reports) lasing in the 1-2.5μ range for He mixtures with Kr, Xe, and Ne.

burst fission reactors.

Experimental lasers to date have simply used small, single laser cells that only intercept a small fraction of the total neutrons available. It seems quite feasible, however, to design a large-volume laser system that would efficiently utilize neutrons from a fast-burst reactor. These reactors(19) typically consist of a uranium-alloy core in the form of a right cylinder of only 30-cm radius and height. They are capable of delivering 6 to 14 MJ in pulses lasting 100 to 200 μsec.

Indeed, Sandia researchers(20,21) have considered a possible DNPL system such as illustrated in Fig. 1. Neutrons from the reactor enter a surrounding subcritical uranium region which, in turn, produces fission fragments that escape into and excite the laser gas. The subcritical (laser "driver") region consists of laminated plates having a thin

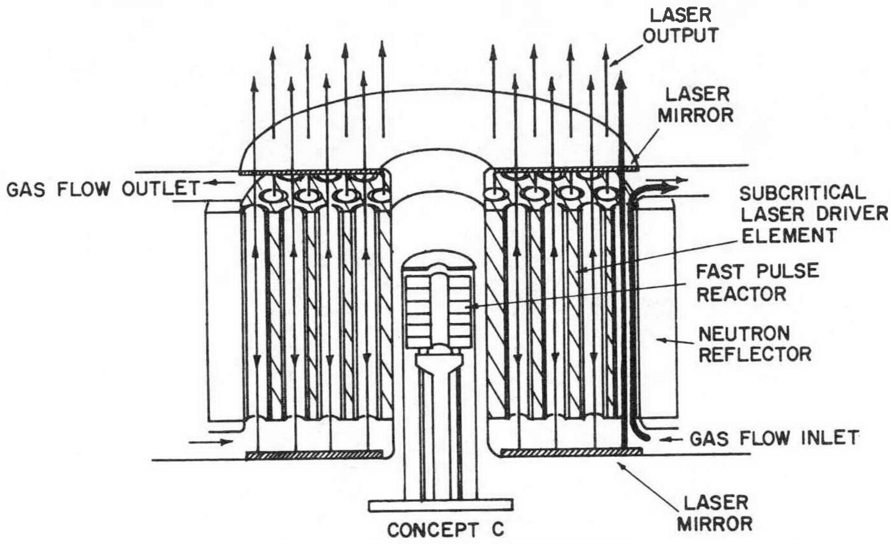

Figure 1. Conceptual designs for coupling a DNPL driver to Sandia's SPR-III reactor (after Schmidt and McArthur[21]). A design using a combined excitation-laser region is shown. Gas-flow rates can be slow, depending only on cooling and gas renewal requirements.

(~3 micron thickness) coating of uranium metal on neutron-moderator slabs of ~0.2-cm thickness. These slabs thermalize the neutrons to provide a better interaction with the uranium; they also serve as a heat sink and provide structural strength for the uranium. The thickness of the uranium coating is determined by the range of fission fragments (~10 microns) in uranium.

It is estimated[21] that up to 21 MJ can be deposited in the sub-critical region resulting in an ultimate 2.1 MJ laser. Due to the good efficiency for transfer of energy to vibrational states in CO by direct fission fragment interaction, it is anticipated that over 50% of the energy deposited in the gas can be extracted by gas lasing. Since ~20% of the energy released in the driver enters the gas, this gives an efficiency of ~10% for the driver-laser. The laser-driver system would only occupy ~6m^3 while the overall unit (including the reactor) is approximately double this size.

In conclusion, it appears that using technology at hand, it should be possible to design and build a pulsed DNPL capable of delivering $\gtrsim$ 1 MJ/pulse. Further, the system is self-contained and sufficiently compact for use in such applications as satellites or remote sensing stations.

IV. UF_6 DNPL CONCEPTS

Perhaps the "ultimate" DNPL system would be one in which the fissioning process takes place directly in the laser medium. With ~80% of the nuclear energy (~165 MeV/fission) carried by fission fragments, this would efficiently deposit large amounts of energy throughout the laser volume.

The potential for using UF_6 in a DNPL has been considered by NASA workers.[22,23] A key unanswered question at this time is whether UF_6 itself will lase under nuclear pumping, or if a mixture with another lasing gas will work. No definitive information is available on UF_6 lasing, but some encouraging data on mixtures has been reported.[8,23,24]

A crucial consideration is the absorption cross section of UF_6. Lasing at wavelengths >400 nm is attractive since the absorption is small in this region out to the infrared. Also, as Lorents, et al.[8] point out, the window at ~340 nm closely matches important I_2^* and XeF^* transitions.

To investigate the possibility of a XeF^*-UF_6 laser, Lorents, et al.[8] measured e-beam induced fluorescence of XeF^* from mixtures of Ar/Xe/F_2 with various amounts of UF_6 added. With 760 Torr Ar, 40 Torr Xe, and 4 Torr F_2, no change in XeF^* intensity occurred with 4 Torr UF_6 added and the intensity only fell by one-third with 50 Torr added. Measurements at NASA-Langley[24] with an electrical laser employing Xe-UF_6 confirm laser action is unaffected with UF_6 concentrations up to 5%. Earlier measurements[23] of emission intensities show strong quenching of the N_2 lines while several Ar-lines (750 and

772 nm) are only quenched at >10% UF_6. Based on these various data, the possibility of finding a UF_6 mixture that lases seems quite promising. It is less certain, however, that sufficient UF_6 concentration (probably >20%) can be achieved to attain a high energy density during neutron bombardment.

V. LASER-FUSION DNPL FEEDBACK SYSTEMS

A basic roadblock to the ultimate achievement of commercial laser fusion power involves the development of the "Brand-X" laser (see Ref. 25). In addition to requirements on wavelength, pulse shape, and peak power, the Brand-X laser must have a high efficiency. Unless an energy multiplication from the pellet burn exceeding 100 is achieved, laser efficiencies exceeding 10% are necessary to prevent excessive recirculation of power.[25,26]

The energy flow presently envisioned is illustrated in Fig. 2. Neutron and plasma energy from the pellet burn are first converted to heat then to electrical energy (probably using a steam cycle), then to high-voltage direct-current or other high-grade form of electrical energy necessary for the laser excitation. If, on the other hand, direct nuclear pumping is assumed, several energy flow schemes, illustrated in Fig. 3 are possible. Assuming that DT pellets are employed, 80% of the energy is released with 14-MeV neutrons which can be moderated and used to drive a DNPL in much the same fashion as envisioned for fission systems. Alternately, the fusion energy carried by the leaking plasma might be employed.[27]

The use of a DNPL in the feedback mode can play two important and distinctive roles in laser fusion. First, this provides a way to bootstrap the startup without requiring large and expensive energy storage facilities that would be necessary for a conventional laser. Thus, Wells[28] estimates that starting with 1 kJ conventional laser and imploding 300 DT pellets so as they energize a direct nuclear pumped laser having a 1% efficiency would make it possible to bootstrap up to an energy of 1 MJ. The DNPL could subsequently be employed for steady-

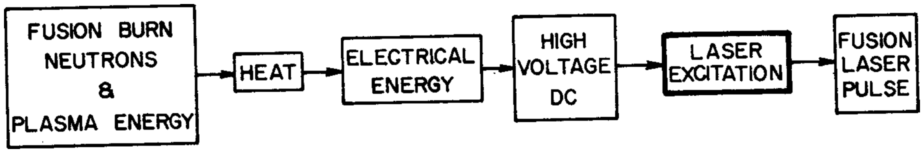

Figure 2. Energy flow in a conventional laser-fusion system.

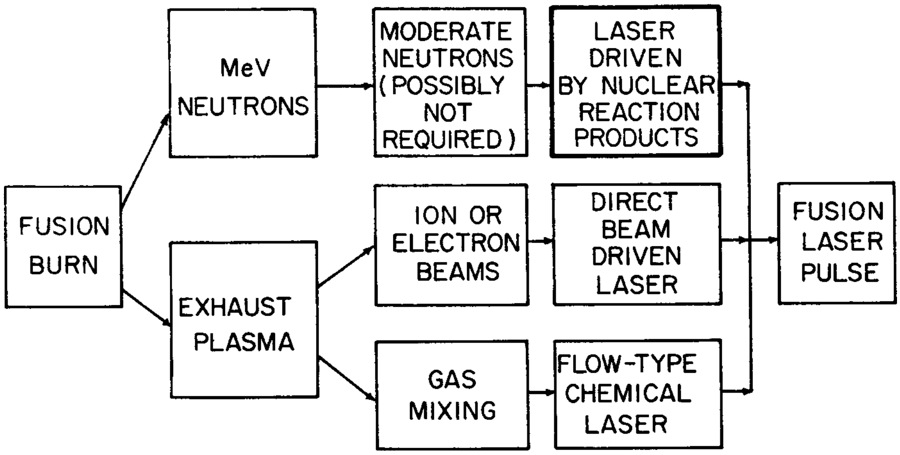

Figure 3. Potential energy paths for a feedback-type DNPL-fusion system.

state operation of the laser fusion device, and this would be its most crucial role.

For the remainder of this discussion we will assume that neutron coupling is employed. Aside from the laser itself, there are three main obstacles to a DNPL feedback system, namely: the neutron economy must satisfy tritium breeding requirements and still provide sufficient neutron flux for laser pumping; an internal energy-storage system must be incorporated to provide proper timing; and suitable pulse-shaping techniques must be developed and radiation resistant optics or replaceable optics employed.

Energy storage perhaps poses the most unique and crucial problem.[29] The neutron energy released in the pellet microexplosion must be stored for a time approaching the interval between laser pulses, which, for a typical laser fusion power plant is ~1 sec. In a conventional electrical laser system, storage is accomplished through a capacitor bank or equivalent. For a DNPL, however, some other energy storage technique must be developed. Approaches considered thus far include a transfer-type flowing laser[29] and a special blanket designed to lengthen neutron moderation-propagation times.[30] Alternate approaches can also be envisioned. For example, a sub-threshold electron voltage sustainer could be used with a recombination-type DNPL to maintain a relatively high electron temperature in the DNPL cavity for a short period after the neutron pulse.[31] Turning the voltage off would accelerate recombination and lasing.

VI. D-D-T PELLET AND BLANKET-DELAY CONCEPT

The authors and colleagues[30,32] have proposed the concept illustrated in Fig. 4 to provide DNPL feedback with a deuterium rich (D-D-T) pellet. This design is intended to provide improved neutron economy compared to D-T pellets and, by reducing tritium breeding requirements, makes it possible to use a special graphite-D_2O blanket that effectively achieves energy storage through a lengthened neutron propagation time.[33,34] While the D-D-T pellet requires a larger laser energy than a D-T pellet, this obstacle is mitigated by the favorable energy-cost scaling of the DNPL compared to a conventional laser.

As seen from Table 1, the lowest neutron threshold for a DNPL reported to date is $\sim 3\times10^{14}$ thermal neut./cm^2-sec. Such fluxes are difficult to achieve with D-T pellets due to the lithium-blanket required for tritium breeding. To avoid this, D-D-T pellets are proposed, i.e. a deuterium pellet containing a D-T "seed" for ignition propagation. (Alternately "excitation heating"[35] might be used to reduce ignition requirements.) Present estimates are that, compared to an equivalent D-T pellet, ~2 times the energy input is required for ignition. However, the added 2.54-MeV D-D neutron production provides an attractive coupling source and allows operation with a tritium breeding ratio <<1. Thus, the present design can utilize a thin lithium section followed by a helium-cooled graphite "moderator-propagator" region. A bulk of the neutron

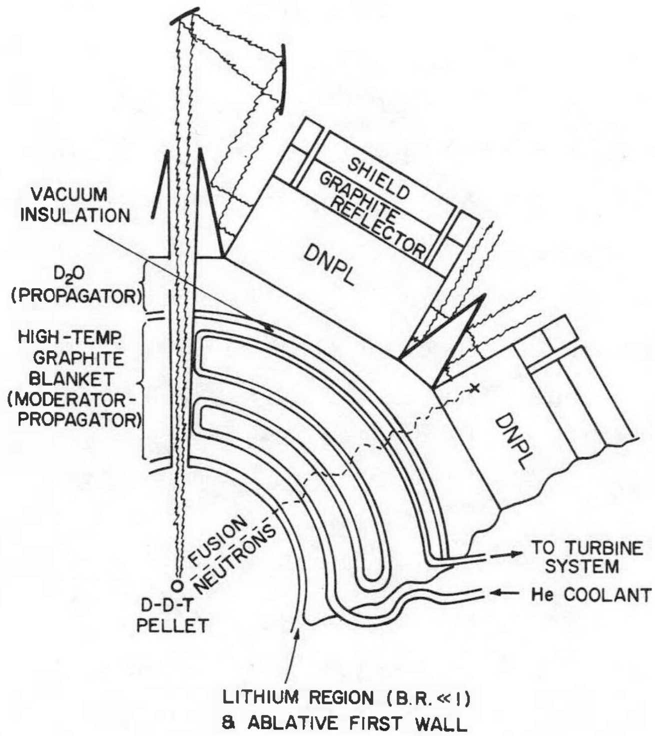

Figure 4. A D-D-T pellet, neutron propagation blanket concept for feedback coupling to a fusion reactor.

kinetic energy is recovered as heat processed through a helium-turbine cycle to produce electricity. The laser pump energy is mainly provided by neutron-induced reactions in the laser; consequently, once the DNPL neutron threshold is achieved, the net electrical efficiency is determined by the thermal cycle. If a further time delay is desired, a low temperature D_2O region can be used after the high-temperature graphite (D_2O has the advantage of even lower absorption and slower neutron propagation times than graphite). The present design uses a blanket-moderator (graphite-D_2O plus structure) design with a neutron thermalization plus thermal propagation time of ~50 msec.[33,34] A Q-spoiling technique is then employed for pulse shaping. A flow system provides cooling of the laser medium, but unlike the preceding concept, this scheme decouples the rate from energy storage requirements.

Neutronic calculations, based on a reference 100-MJ output per pellet, indicate a neutron production of ~3×10^{20}/pellet which, with the present blanket, delivers ~3×10^{19} thermal neutrons to the DNPL. This is adequate to pump, in the feedback mode, a 10% efficient BF_3 fueled laser, or alternately 0.1% or 0.01% efficient UF_6 or AmF_6 fueled systems, respectively. A disadvantage of the latter lasers is the introduction of fission products into the system. The radioactive inventory need not be large, however, and the laser medium would be well subcritical. Consequently, such a system presents fewer problems than conventional fission-fusion hybrid concepts, frequently proposed to overcome the energy recirculation problem.

In conclusion, the D-D-T neutron-coupled DNPL concept is shown to meet the key objectives of energy storage and neutron economy. In common with other laser-fusion concepts, however, a number of other technological problems must be overcome to attain a practical power plant.

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Fusion by Grand Catastrophe

by

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ABSTRACT

The plasma focus, an accidentally discovered natural phenomenon, provides a pulsed power source of 10^{12}-10^{13} watts in a variety of forms: electron bursts, ion bursts, and the stagnation and direct field heating of the snowplow. High density fusion work being conducted at Livermore with a plasma focus with maximum bank energy of 500 kJ at 40 kV is described. The primary purpose of the project is to employ the plasma focus as a pulsed power source to explore various fusion microexplosion concepts. The first exotic fuel experiments have been carried out at this facility by operating the focus on D_2+He^3 at 120 kJ and 27 kV.

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I. Introduction

The French mathematician Rene Thom[1] introduced a new branch of mathematics called catastrophe theory that identifies the possible classes of "catastrophic almost discontinuous jumps" between relatively well-behaved states of a physical system. From our point of view the importance of this new development is that it focuses attention on the violent almost discontinuous behavior of physical systems that are less completely studied than the more tractable, relatively quiescent states.

The normal engineering approach to a problem is directed toward producing structures and machines that are well behaved and subject to human control throughout their operating range. This approach, while self-evidently desirable for most applications, has the consequence that the study of uncontrollable catastrophic events tends to be ignored. There appears to be little work directed towards learning how to increase the violence of natural catastrophies.

The achievement of controlled fusion has proved to be an extremely difficult task, and the lesson of much of the research in this field is that plasmas "like to have" instabilities. The elimination of instabilities has proved difficult to such an extent, that the quest for a system not plagued by instabilities may be unnatural, and a different approach might prove to be more rewarding. That is, one can search for a particularly violent plasma instability, study it with the purpose of making it as bad as possible, and then see if the resulting energy concentration might be utilized to produce thermonuclear fusion rather than prevent it.

As a slightly humorous case in point, consider one of the more remarkable weather phenomena—a tornado. A warm mass of air in collision with a colder air mass will on occasion penetrate beneath the cold air creating a strong temperature inversion. This unstable state can result in the opening of a natural drain through which the cold air descends and the warm air rises, forming a tornado. This path of catastrophic transition between the inverted and uninverted states attains a high rotational velocity. One can speak of a typical tornado as catastrophic. Occasionally conditions are "just right" and a tornado of relatively enormous proportions occurs. Such a tornado we choose to call a "grand catastrophe".

When a tornado forms over a body of water a water spout is produced.[2] It is amusing to comtemplate the replacement of the body of water by an inflamable liquid to produce a high octane gasoline spout and then see what happens if the column is ignited. The serious point behind this perhaps bizarre thought experiment is that one has supplied an additional energy source to an already remarkable natural phenomenon and, this additional energy may act to intensify the phenomenon to produce a "truly grand catastrophe".

In the material that follows we wish to discuss a highly minaturized analog of this thought experiment that is being persued at Livermore. In this program we are trying to see if another remarkable natural phenomenon, the plasma focus (in many respects similar to a minature electrodynamic tornado or ultraminiature sun-spot) can be utilized to drive a fusion micro-explosion.

Many people have contributed to this effort, but two names require special mention, Edward Teller and John Luce. Dr. Teller had the courage to initiate this unconventional approach to controlled fusion. He stated at one point "I have the sneaking suspicion that no one really knows how the plasma focus works—it might be amusing to place a drop of DT in the focus and see what happens." Our plasma focus project is based on earlier work of John Luce and his research group carried out prior to 1968. Dr. Luce has been primarily responsible for achieving the first successful experimental results in the project at Livermore.

II. What is Plasma Focus

The earliest work on controlled fusion concentrated on the Z pinch. These efforts, in addition to being frustrated by instabilities in the final pinch phase that led to large neutron yield of a beam target nature, were also plagued by the fact that the current which initially struck over the inside surface of the insulating vacuum wall would cling to this surface for a period of time and evolved high Z contaminates into the fill gas prior to the onset of pinching this gas in snowplow fashion to the axis by the magnetic field pressure. Filippov,[3] in an effort to eliminate this problem, employed a metal vacuum wall that served as a cathode and placed the insulator separating the anode and cathode behind the anode plate. In this configuration the current still struck across the surface of the insulator separating the anode and the cathode, but resulted in the formation of an inverse pinch. The current lifted off the insulated surface snow-plowing plasma ahead of it, rounded the corner and collapsed rapidly to the axis to produce a very intense r-z pinch. The plasma focus was independently discovered by Mather[4] in the course of work on plasma guns and has led to the plasma focus geometry shown in Fig. 1, which is known as the co-axial or Mather geometry.

The r-z pinch can reach a density > 10^{19} ion/cm^3, a temperature of about a kilovolt, and produces record neutron yields. The pinch differs in a significant way from a Z-pinch. One illustration of this point is the fact that the Z-pinch is not affected by interchange of anode and cathode, while in the case of the plasma focus the neutron yield decreases by an order of magnitude when the outer metal wall is the anode. At the time of maximum pinch roughly 1/2 of the initial capacitor bank energy is inductively stored in the magnetic field concentrated around the pinch. After a period of the order of 10 ns, a complex series of processes involving both strong turbulence and the production of a resistive high-Z vapor at the anode center result in rapid conversion of about one half of the magnetic field energy into particle energy.

The process of rapid conversion of a major portion of the field energy into particle energy appears to be compulsory, particularly for systems with pinch currents I > 1 MA. In a plasma focus operated on D_2 gas, the sudden conversion of field energy into particle energy is accompanied by large neutron yields. These yields in most systems under most operating conditions are dominantly of a beam-target nature. However, there are certain operating conditions in specific systems where most of the neutron yield may be of a nearly thermonuclear nature.[5]

The process of sudden conversion of field energy to particle energy is known to be sensitive to a number of factors in addition to the electrical parameters of the system, including the geometry and materials of the anode center, the type of fill gas, the fill gas pressure, and the presence of a few percent of high-Z impurities in fill gas. In high pressure operation the field energy tends to go into thermal energy of the plasma, while in lower pressure operation it can appear as a concentrated relativistic electron burst and an oppositely-directed, accelerated ion burst whose production is accompanied by the emission of significant microwave bursts in the vicinity of 10^{10} Hertz and strong nonthermal infrared radiation in the 1-10 μ range. It is this conversion of 1/2 the magnetic field energy into particle energy in a time of 30-60 ns that is the most significant property of the focus and constitutes in large systems a pulsed power source of 10^{12}-10^{13} watts.

The most recent plasma focus research has resulted in DD neutron yields of 1-2 x 10^{12} achieved in 400 kJ 50 kV operation by the now defunct Mather Group at Los Alamos, and by Bernard at Limiel, and in Frascati by Maisonnier operating at partial capacity in the 1 MJ 40 kV Frascati focus. The neutron yield scales as power of the pinch current I^{α}, $3 \leq \alpha \leq 5$, over the entire operating range so far explored in higher energy systems. The result obtained by Bernard's group at Limiel, $\alpha = 3.3$, is based on many operating energies by the same research group with the same gun design.

III. The Focus As A Pulsed Power Source

If the focus is viewed as a pulsed power source we see that two means of energy storage and compression are present. During the collapse phase field energy is fed into kinetic energy of the snowplow over a time longer than the time in which this directed energy is converted into heat as the high velocity snowplow ($v \geq 3 \times 10^{7}$ cm/sec) stagnates on axis to form the r-z pinch. At the time of maximum pinch, about 50% of the bank energy is inductively stored in a concentrated form surrounding the pinch, and a major portion of this energy is rapidly converted into particle energy. Both the snowplow stagnation and the conversion of field into particle energy constitute pulsed power sources in the 10^{12}-10^{13} watt range. The field energy can go into thermal energy of the pinched plasma or into concentrated electron and ion bursts.

In the following discussion we will concentrate primarily on the electron burst produced in low pressure operation of the focus and simply note that the ion burst and the snowplow stagnation followed by its rapid heating by the stored magnetic energy also provide pulsed power sources of great interest for driving fusion microexplosions; in each case one must employ targets designs specifically adapted to the source.

IV. The Plasma Focus Produced Electron Burst

In 1968 the 250 kJ 20 kV plasma focus operated by J. Luce was employed in a low pressure mode with a small diameter tungsten rod protruding on axis from the center of the anode, in an effort to enhance the hard x-ray yield. Peak yields of more than 100 J of x-rays above 100 keV were obtained with pulse width $T < 40$ ns due to thick target bremsstrahlung from energetic electrons impinging a spot about 1 mm in diameter at the end of the tungsten rod. Most important of all, it was noted by John Luce that the largest x-ray yields occurred when a previous shot had damaged the tungsten rod to form a more or less conical point with a fine whisker inexplicably growing from its tip. This observation was a major motivating factor in establishing the Livermore plasma focus program because, in addition to demonstrating in a graphic way the sensitivity of the electron burst production to the geometry as well as the materials of the anode center, it suggested that a small fusionable target inserted into the focus could cause an avalanche of energy from the focus into the target.

From this data it was possible to estimate that at least 5% of the bank energy was present in the electron bursts with current density 10^7 A/cm^2 and pulse width $T \leq 10$ ns. This pulse width is shorter than the rise time of the most recent generation of advanced conventional relativistic electron beam machines. Subsequent developments at Livermore and other laboratories has confirmed the original deduction concerning the size and pulse width of the focus produced electron burst.[6] Record electron bursts with yields of 22 kJ of 300 kV electrons in 60-70 ns at a current of 1.5 MA and a current density in the range $10^7 - 8 \times 10^8$ A/cm^2 produced by a 94 kJ Filippov plasma focus have been reported in the USSR.[7] This result is most remarkable when one realizes that an undesigned accident of nature results in electron beams comparable with that of the most advanced generation of relativistic e-beam machines at a current density exceeding that achieved to date with these machines.

One mode of electron burst production in a plasma focus studied by Filippov[8] is particularly attractive partly because of its conceptual simplicity. As the cylindric pinch column of the focus proceeds toward the axis, sufficient energy can be carried to the anode surface to evolve a resistive anode vapor ("a high-Z vapor spout"). As a result of this increased resistance the current is forced to diffuse ahead of the snowplow into regions where the anode has higher conductivity; but this results in the production of more vapor forcing the current to proceed rapidly toward the axis where it has no place left to run away from the self-produced anode vapor, and as a consequence, the voltage across this anode vapor must increase until the current can be carried on axis even in the presence of the high-Z gas. A particularly attractive feature of this mode of electron burst production is that the current and field penetrate inside the snowplow only in a relative narrow region above the anode, and as a result, the added inductance is sufficiently small to permit the concentrated relativistic electron burst to have nearly the same current as the peak current of the focus.

V. Conditions For Fusion Microexplosion

It is now well know that to obtain energy production with a fusion microexplosion, convergent compression is essential (like all universal truths this statement may have exceptions). Conservative estimates set the power requirements for driving exothermic fusion microexplosion at least as high as 10^{14} watts. However, certain target concepts[9] may permit lower power operation at modest compression or even at densities a little less than that of uncompressed solid DT. We believe that it may ultimately prove possible to reduce the power requirements to a value as low as 10^{12} watts. The conversion of 10% of the bank energy of a single plasma focus with bank energy $\leq$ 1 MJ into electron burst of duration $T \leq 10$ ns provides a pulsed power source that permits significant fusion microexplosion research. A convergent array of $E \geq 120$ kJ, $V \geq 50$ kV plasma foci of the Filippov type e.g. mounted on the 12 faces of a dodecahedron (see Fig. 2) could provide energy $E \geq 1$ MJ at a power in excess of 10^{14} watts.

VI. The Livermore Focus and Experiments

In a series of experiments carried out by J. Luce[10] at 120 kJ and 27 kV with a portion of the 500 kJ 40 kV bank, three microballon were initially fired in the focus. The targets employed (courtesy of the Livermore Laser Fusion Program) were 100 μ diameter glass spheres with a wall thickness of 2 μ containing 8 ng of DT mounted on 30 μ diameter dielectric stems; they were inserted on the axis from a direction opposite to that of the anode and positioned about 1 cm below the anode center, and the focus was operated with H_2 fill gas. One of the targets produced a $(5 \pm 2) \times 10^6$ 14 MeV neutrons yield with a pulse of $T \leq 1$ ns. The other two targets produced null results. Three mechanisms were proposed as the possible explanation of the neutron yield. (1) Target implosion by current sheet driven snowplow. (2) Target implosion by a proton burst with ion energy $E > 1$ MeV with a range in gm/cm^2 slight in excess of that of the DT filled glass shell. (3) Some type of collective acceleration process may have taken place in the DT gas inside the glass shell. The data permitted setting an upper bound of 100 kV on the deutrons that might have been responsible for the yield if an ion acceleration process was involved.

Sophisticated 1D computer calculations based on (1) and (2)[11] showed that the observed neutron yield is compatible with either model. Present experiments with improved diagnostics provided by Wm. Pickles tend to favor hypothesis (2) or (3) over (1) and should ultimately permit a choice to be made between (2) and (3).

We conclude by noting that the first experiment with the exotic fuel, D_2+He^3, have been carried out in our plasma focus,[12] as a result of John Luce's long standing interest in this reaction. The direct use of the plamsa focus, or the focus combined with advanced targets, may prove to be an ideal means for the study of exotic fuels.

The problem of achieving net energy release even with DT is sufficiently difficult that we believe a new discovery may be required to permit the early application of the exotic fuels. One candidate for the missing "x" factor might be super-elastic collision with the He-like triplet state of multiply ionized high Z ions, an idea first introduced by Rand McNalley, et al.[13]

to explain the discovery that in energetic carbon arcs, ion energies can be substantially greater than the energy of the electrons. Like so many of the topics considered in this paper or by other authors at this conference, John Luce was present at the conception—the energetic carbon arc is also know as Luce arc.

VII. Summary

The plasma focus is a remarkable natural phenomenon that in an appropriate operating range can give rise to conversion 10-25% of the bank energy into a concentrated relativistic electron burst with power in the range 10^{12}-10^{13} watts, at a current in excess of 1 MA and current density of the order of 10^8 A/cm^2. In addition to the electron burst the plasma focus can yield an accelerated ion burst of comparable power, or provide a power source of 10^{12}-10^{13} watts in the form of direct field heating of the snowplow. Each of these pulsed power sources can be employed to drive fusion micro-explosion designed specifically for the source. At Livermore the first exotic fuel studies in a plasma focus have been carried out by employing D_2-He^3 fill gas. The present plasma focus (500 kJ-40 kV) is being employed to study various DD and DT fusion microexplosion ideas and may also provide an ideal means for the study of the exotic fuels of primary interest at this conference.

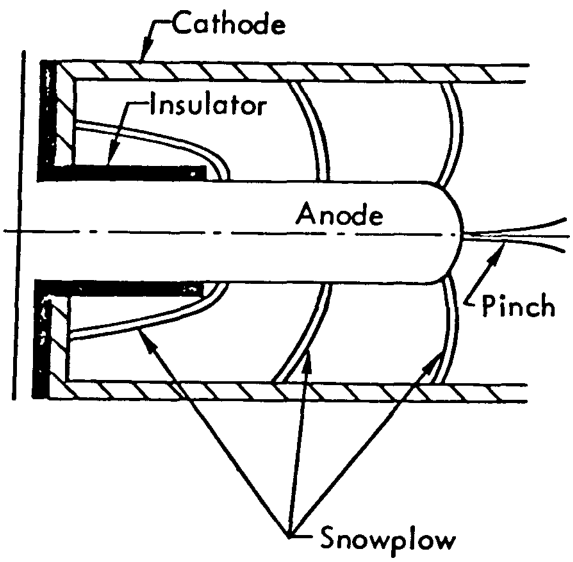

Fig. 1 A plasma focus in the Mather or Co-axial geometry.

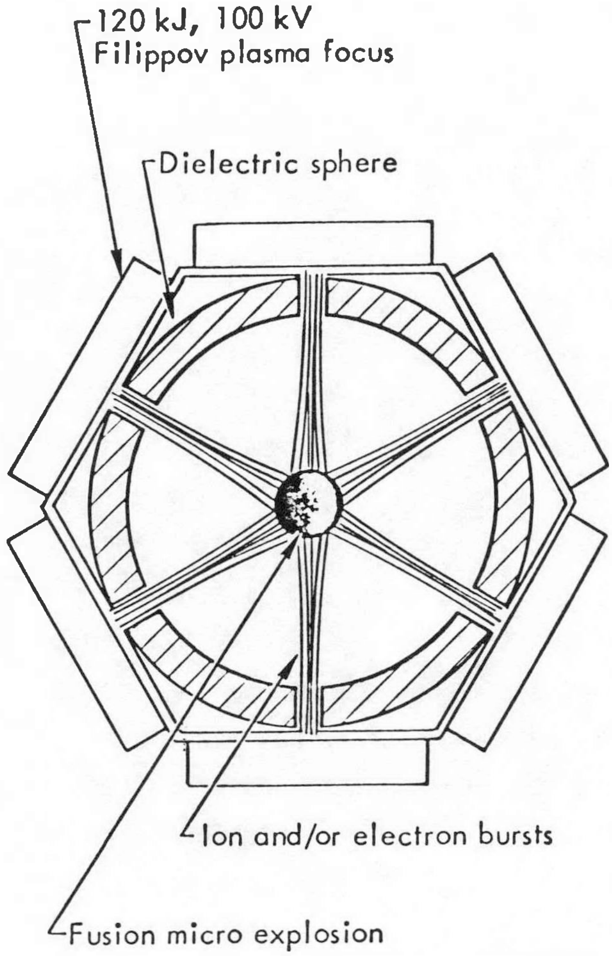

Fig. 2 Convergent Array of Filippov Plasma Foci.

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High Thermal Efficiency Advanced Fuel Fusion Reactors*

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ABSTRACT

A new radiation energy conversion scheme is presented which has the potential for operating at high conversion efficiencies on the order of 60 to 70 percent. Two new device concepts, an x-ray radiation boiler and an energy exchanger, permit radiation energy extraction through a low-Z reactor first wall to a high-Z working fluid which is heated volumetrically to temperatures of 2000 to 3000 °K. The energy exchanger transfers the high temperature working fluid energy to a lower temperature working fluid which drives a conventional turbine. This scheme is applied to radiation energy conversion from a $p^{11}B$ fusion reaction, using a single reheat topping and bottoming cycle combination. The high thermal efficiency of this cycle permits an otherwise marginal advanced fusion reactor to become a more attractive net power producer.

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I. INTRODUCTION

The benefits of advanced fuel, neutron-free reactors to the utilities include a significant reduction of nuclear issues (accidental releases, waste and diversion of radioactive materials) and a cheap, plentiful supply of fuel (e.g., proton-boron 11 fuel). Advanced fusion fuels are capable of producing extremely high grade energy output x-ray radiation which is characterized by the high fuel temperatures. Consequently, there is a potential for high efficiency power generation. Advanced fuel fusion reactors also have particularly low Q values, where Q is defined as the ratio of the fusion energy to the input energy. High efficiency energy conversion is therefore an absolute necessity for these reactors to be serious contenders for commercial fusion power.

This paper reports on a new radiation energy conversion scheme for extracting the high quality energy discussed above and shows that for this scheme high efficiency conversion appears to be feasible. This new reactor concept employs a novel x-ray radiation boiler and a new thermal conversion device called an energy exchanger. The first walls of the radiation boiler are semi-transparent to x-rays and are kept cool by incoming working fluid, which is subsequently heated in the interior of the boiler by volumetric x-ray absorption. The high-temperature working fluid from the boiler then compresses a low-temperature gas in the energy exchanger by transmitting expansion work. The low-temperature output gas of the energy exchanger drives a conventional turbine. The overall thermal efficiency of the cycle is characterized by the high temperature of the working fluid. These temperatures can exceed 3000 °K to give net reactor thermal efficiencies as high as 70 percent.

The advantages of the radiation boiler depend on its ability to produce a very hot working fluid and to couple this energy efficiently to conventional electricity-generating equipment. The radiation boiler can be a compact part of the reactor shell because x-rays are absorbed readily by high-Z materials. This feature should help reduce reactor size and cost. The high overall efficiency also reduces the waste heat for a given recirculated power fraction, allows smaller reactors to be built for a given reactor output, and enhances the internal energy balance for advanced fuel cycles which release most of their energy as x-ray radiation. A sketch of a possible design for an advanced fuel reactor is shown in Figure 1, including enlargements of the radiative boiler-first wall of the reactor and the energy exchanger.

II. BREAKEVEN CONDITIONS

The usual Lawson condition for breakeven requires that the total thermal power output from a fusion reactor is equal to the energy required to sustain the plasma divided by the thermal conversion efficiency. This criterion must be modified for advanced fuel reactors to include the effects of higher effective Z of the radiating plasma and near relativistic plasma temperatures. For a neutral beam sustained reactor, the expression for the energy confinement time is

$$\tau_E = \frac{E_{Plas}\left[1-\eta_P\eta_H(1+F)\right]}{\eta_P\eta_H P_{TN} - P_{Rad}\left[1-\eta_H(\eta_R+\eta_P F)\right]}$$

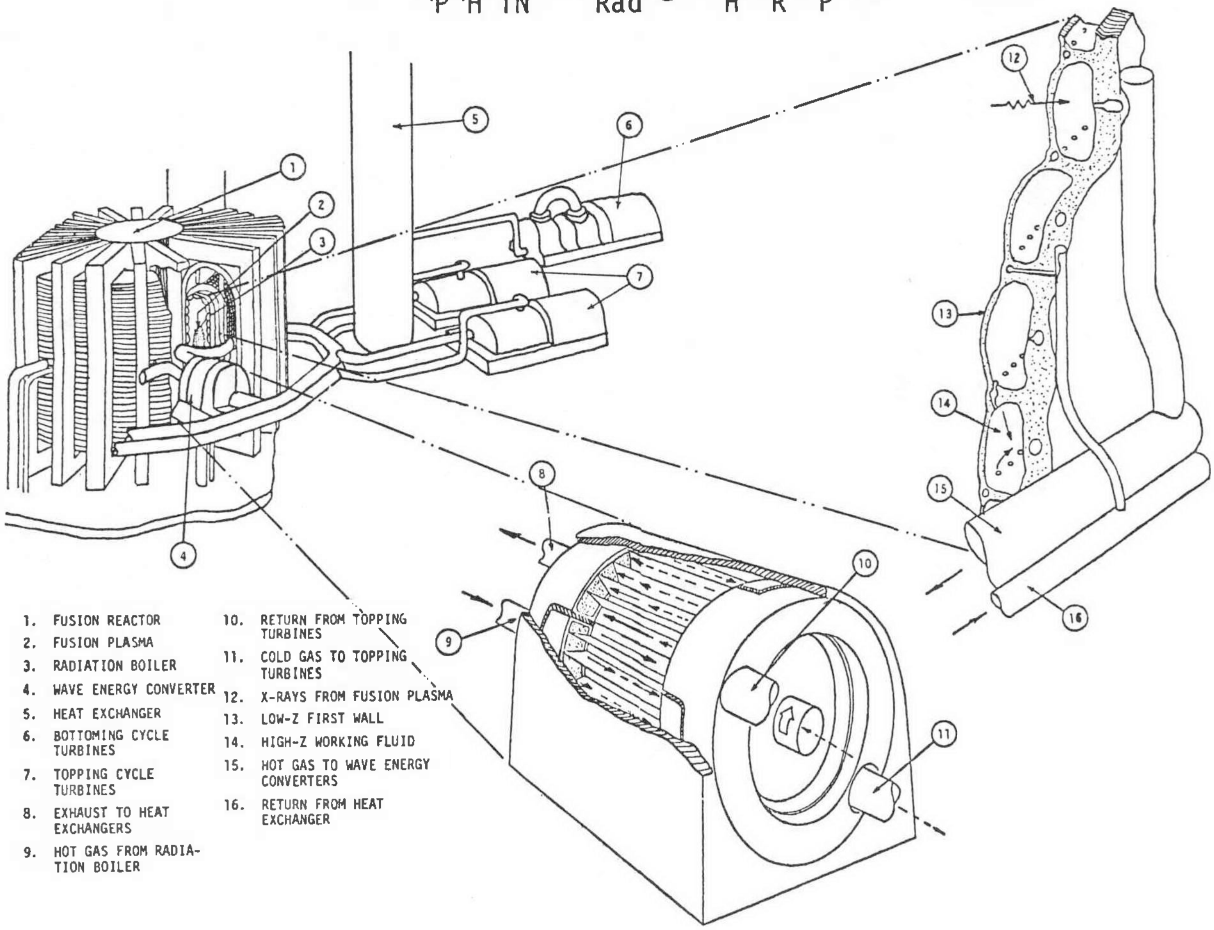

Figure 1. Schematic of a Radiation-Based Fusion Reactor with Energy Exchanger and Radiation Boiler

where the power to sustain the plasma is $P_{Rad} + (E_{Plas}/\tau_E)$, η_R and η_P are the efficiencies for conversion of x-ray radiation and particle energy to electrical energy, and η_H is the plasma heater (neutral beam) efficiency. The plasma energy density is E_{Plas}, while P_{TN} and P_{Rad} are the thermonuclear and x-ray radiation power densities, respectively. Injected beam particles exhibit enhanced reactivity compared to Maxwellian ions. Before slowing down, the beam particles result in the production of F times the beam power in fusion reactions. It may be seen that the required energy confinement time is strongly sensitive to the energy conversion efficiency for both plasma particles ($\eta_P\eta_H$) and radiation ($\eta_R\eta_H$). To illustrate this, we equate both quantities ($\eta_P\eta_H = \eta_R\eta_H = \eta$) and plot the density independent Lawson product $\eta\tau_E$ for D^3He and $p^{11}B$ in Figure 2. The advantage of high efficiency energy conversion is illustrated for $p^{11}B$ by comparing the two curves for $\eta = 0.56$ and $\eta = 0.70$. The minimum $\eta\tau$ required for ignition is nearly a factor of 6 smaller for $\eta = 0.70$. In systems such as D^3He and $p^{11}B$ where radiation dominates the

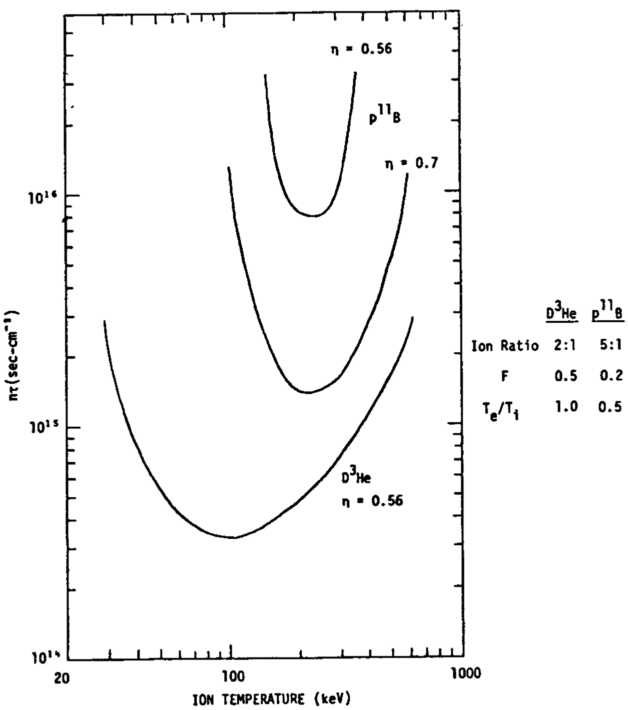

Figure 2. Modified Lawson Condition for D^3He and $p^{11}B$ for 100 Percent Recirculated Power (f=1)

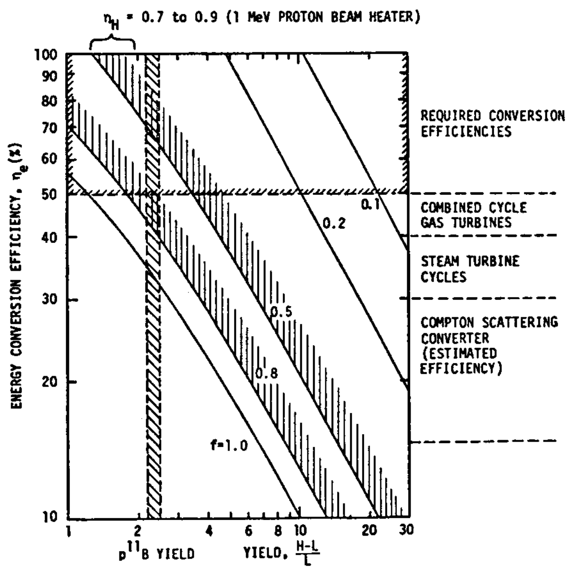

Figure 3. Energy Conversion Efficiency Versus Thermonuclear Yield

plasma energy release, efficient recovery of radiated energy makes breakeven possible at relatively low values of τ_E.

III. NET POWER PRODUCTION

The importance of high thermal plant efficiencies for low-yield fusion cycles which must produce a net power output (i.e., recirculated power fraction, f, less than 1) is illustrated dramatically in Figure 3. A very simplistic energy balance,

$$L = \eta_e \eta_H f H$$

has been assumed where H is the fusion energy out, L is the heater energy in, η_e is the energy conversion efficiency, and η_H is the heater efficiency with a range of values consistent with proton accelerators, neutral beams, and relativistic electron beams.

The trend is very clear; for low-yield systems such as $p^{11}B$, energy conversion efficiencies close to 70 percent are required in order to attain realistic recirculated power fractions. None of the schemes noted in Figure 3 have the capability, as presently conceived, of achieving the requisite efficiencies. For this reason we have chosen to study a novel concept which concentrates on converting the radiation from the plasma to electric power at efficiencies on the order of 70 percent. Together with 70 percent efficient conversion of escaping particle energy, e.g., with electrostatic converters, the total conversion efficiency will be 70 percent as required according to Figure 3.

IV. THE RADIATION REACTOR CONCEPT

We have considered two basic designs for the radiation boiler as shown in cross section in Figures 4a and 4b. Both designs suppose that the first wall is made of tubular sections under tension to support the high-pressure working fluid. The curvatures of the first wall would be reversed if the wall material was stronger under compression.

Candidate wall materials include low-Z, high strength elements such as carbon (e.g., graphite) and beryllium, the oxides, nitrides and carbides of beryllium, boron, and silicon, and alumina. Most of these materials were

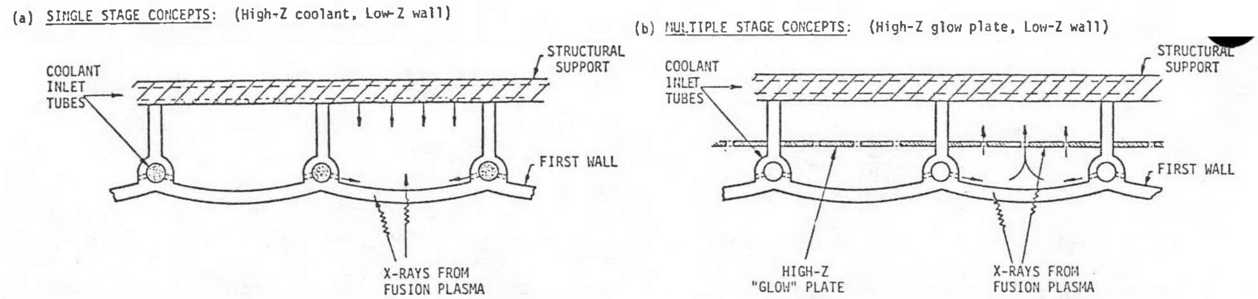

Figure 4. Radiation Boiler/Fusion Reactor First-Wall Concepts. Figure 4a shows x-ray energy being volumetrically absorbed in a high-Z working fluid. In Figure 4b, a lower-Z working fluid can be used in conjunction with a high-Z absorbing glow plate or other structural materials.

the subject of a recent General Atomic study; some of their data have been used in the selection of first-wall materials in this section. The opposing constraints on first-wall thickness are an upper limit imposed by the need for x-ray transparency and a lower limit imposed by the need to retain a high pressure.

The initial selection of first-wall materials made on the basis of these constraints includes B_4C, Be_2C, BN, and Be. The first three materials do not have the ductility of beryllium but do have advantages in terms of their high temperature resistance. Carbon doped beryllium has many of the best features of pure beryllium, including high thermal conductivity and an increased melting temperature characterisitc of the complete carbide. The relatively low-temperature limits (e.g., 500 °C) placed on beryllium could possibly be maintained by active cooling with a low-Z gas such as helium.

The energy exchanger is a device which transmits the work of expansion of a high-temperature gas through a gas piston-like interface to a colder, lower molecular weight gas, which in turn can be used to drive a conventional turbine. The gas interface within the energy exchanger effectively separates the high and low-temperature working fluids and yet allows all of the work to be transmitted from one side of the device to the other. The transfer of energy across a gas interface by a compression wave is facilitated by a condition called impedance matching which requires that no acoustic wave be reflected from the interface, i.e., the original wave be transmitted in full strength. In terms of the state of the two gases, impedance matching means that the specific heat ratio, γ, and the product of the density and sound speed, ρa, must be the same on each side of the interface.

Further, since we are dealing with compression processes, shock losses must be avoided. Consequently, an isentropic compression is ideal. The optimal energy exchanger process thus consists of gasdynamic compression of a low-temperature driven gas by a high-temperature driver gas, under impedance matched and completely reversible conditions; that is, isentropic compression followed by isentropic expansion. Under these ideal conditions, the enthalpy of the hot driver gas is diminished precisely by the amount that the enthalpy of the cold driver gas is increased. The piston-like nonsteady wave action in the energy exchanger is shown in Figure 5 for ideal gas compression in a single tube of such a device.

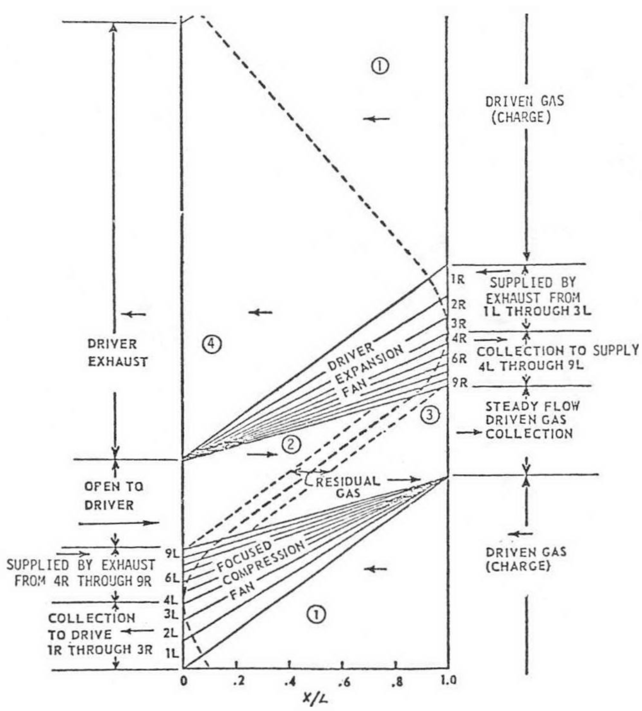

Figure 5. Ideal Energy Exchanger Cycle. Smooth compression and expansion with no wave reflection from the ends of the tube is accomplished by rerouting some of the driver and driven gases from one end of the tube to the other. The temperature ratio between states 1 and 3 is proportional to the molecular weights: $T_2/T_3 = m_2/m_3$.

In order to produce a steady flow process, multiple-energy exchanger tubes are mounted on the rim of a rotating drum as shown in Figure 1. As the drum rotates past a series of shaped nozzles and compression walls, the gasdynamic cycle is repeated consecutively in each tube, approaching a steady flow conversion of high-temperature driver gas energy to high-pressure, lower temperature driven gas energy. High-temperature material problems are eliminated or mitigated to a large extent in the energy exchanger because the tube walls are continually cooled by the cycling of the low-temperature driven gas through the system.

V. CYCLE CALCULATION

The basic energy conversion system consists of the radiation boiler, the energy exchanger, topping cycle gas turbines, and a bottoming steam cycle shown schematically in Figure 6.

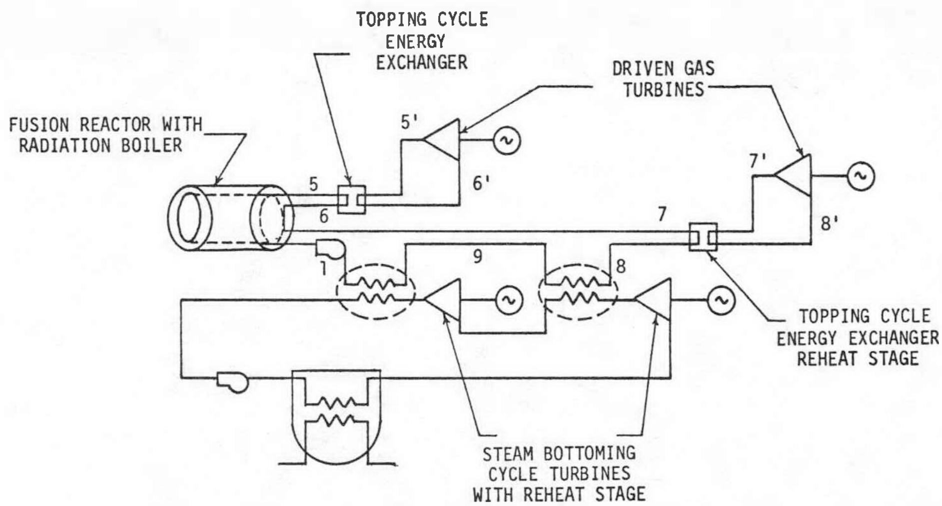

Figure 6. Systems Diagram for a Single Reheat Mercury-Steam Combined Cycle Utilizing a Radiation Boiler and Two Wave Energy Exchanger Converters. High-temperature mercury comes from the boiler and transfers gasdynamic expansion work (5 to 6) through the wave energy converter to the driven gas, which in turn delivers the work (5' to 6') to a low-temperature turbine generator. The driver gas returns to the boiler for reheating (6 to 7) and the process is repeated at lower pressures (7 to 8 and 7' to 8'). The remaining thermal energy of the driver gas is transferred to a steam bottoming cycle (8 to 9 to 1).

The use of reheat stages in both topping and bottoming cycles raises the combined cycle efficiency significantly. Figure 7 depicts the thermodynamic cycle where the peak temperature for the mercury topping cycle is 3000 °K at 100 atm and 14 atm, and the peak temperatures in the steam cycle are 922 °K (136 atm) and 866 °K (4.4 atm).

The topping cycle efficiency η_T = 40 percent and bottoming cycle efficiency is η_B = 42 percent for a combined cycle efficiency of

$$\eta_C = \eta_B(1-\eta_T) + \eta_T = 65 \text{ percent}$$

where an energy exchanger efficiency of 85 percent and turbine generator efficiencies of 93 percent have been assumed. The combined cycle efficiency

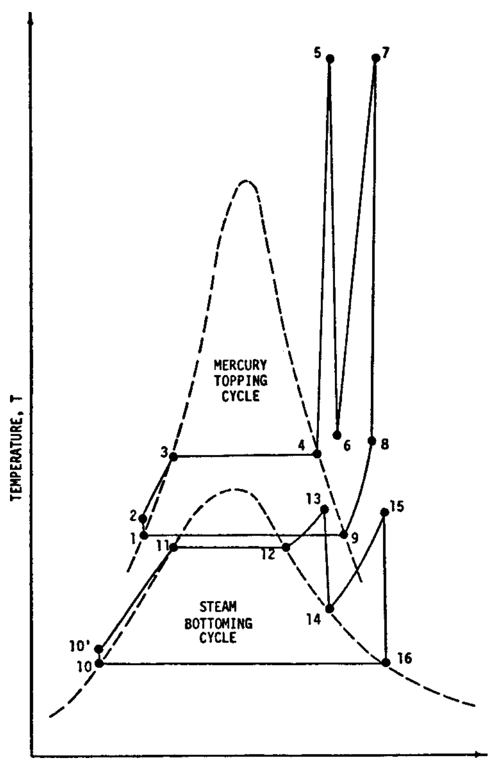

Figure 7. The Thermodynamic Cycle for a Mercury Vapor Topping Cycle and a Steam Bottoming Cycle Each with a Single Reheat Stage

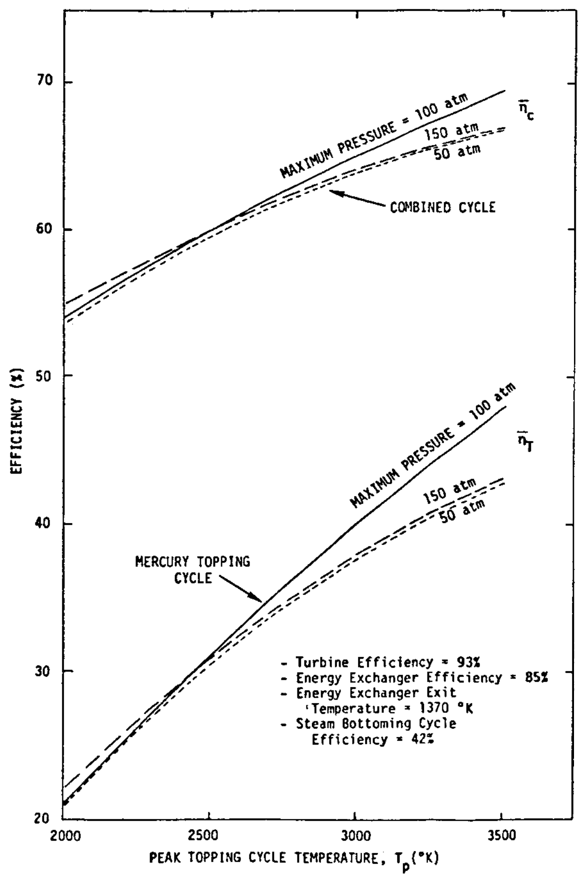

Figure 8. Topping Cycle and Combined Cycle Efficiencies as a Function of Peak Cycle Temperature for Various Topping Cycle Pressures

is well within the desired range of efficiencies appropriate for advanced cycles such as $p^{11}B$.

The effects of the topping cycle peak pressures and temperatures on topping cycle and combined cycle efficiencies are shown in Figure 8 for a mercury/steam combined cycle. For the parameters chosen, it would appear that 100 atm is nearly ideal since the efficiencies decrease to higher and lower pressures in the regime of interest. Combined cycle efficiencies on the order of 70 percent appear achievable at 3500 °K, and higher efficiencies can be reached as the peak temperature increases. Other working fluids besides mercury and steam were investigated and may yield equivalent combined cycle efficiencies. The above is intended only as a sample calculation and is in no sense an optimum choice.

VI. CONCLUSIONS

This paper has presented the results of an exploratory study in a new fusion energy conversion concept which utilizes advanced neutron-free fusion fuels such as $p^{11}B$. Most of the advanced fuels release significantly less fusion energy in the plasma per unit of input power, thus placing a premium on the conversion process efficiency. The reactor concept studied here utilizes the unique output of the $p^{11}B$ reaction (e.g., 50 percent or more in x-rays) to heat a working fluid in an x-ray boiler, and subsequently to transfer that energy through an efficient energy exchanger, another new cycle element, to a conventional turbine generator.

The low Q of the $p^{11}B$ reaction requires a high overall cycle efficiency to be at all practical ($\eta \cong$ 50 to 70 percent). A 60 to 70 percent cycle efficiency requires a binary cycle, probably with reheat and a peak working fluid temperature between 2000 °K and 3000 °K. The energy transfer processes have been analyzed and volumetric heating to temperatures on the order of 3000 °K by x-ray absorption appears feasible with a number of possible gases. An energy exchanger with good energy transfer efficiency and high temperature capability appears feasible but has not been built to date. However, small, lower pressure energy exchangers are commercially available as diesel superchargers, and related compression wave machines have already been built for re-entry physics experiments which reached temperatures as high as 4500 °K.

Radiation Emission and Production of

Synthetic Fuels

by

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Abstract

The radiation emitted by fusion reactors has unique characteristics which can be used to produce synthetic fuels such as hydrogen. Direct radiolytic chemical coversion of neutron or X-ray energy is possible. High temperature ($2700^{\circ}K$) blankets, suitable for efficient chemical processing can be designed using radiative heat transfer. The efficiencies of some of these new fuel production techniques are shown to be comparable to or greater than electrolytic hydrogen production using electricity produced by a thermal cycle. The advantages of dual purpose, electricity and fuel producing reactors are discussed. The areas of research and development needed to better define the process are outlined.

I. INTRODUCTION

Fusion reactors offer unique new ways of production of synthetic fuel that could be used for the portable fuel segment of the U.S. energy economy. Hydrogen appears to be the most likely synthetic fuel produced by the reactor itself. The variety and accessibility of the different forms of energy generated by the various fuel cycles possible make this new energy resource unique. A DT reactor would release 80% of its energy in the form of 14 Mev neutrons, while an advanced fuel like p-B" would release over 90% of its energy as X-rays and produce a negligible flux of neutrons. Neutrons and X-rays have two characteristics that make a central station power plant producing them unique. First, they can heat matter radiatively rather than by conduction or convection. Second, they can produce chemical reactions directly via radiolysis.

II. PRODUCTION METHODS

Most methods for production of synthetic fuel in fusion reactors involve the breakup of relatively simple molecules such as H_2O and CO_2. Water can be converted to hydrogen and oxygen. CO_2 would be converted to CO, which would then produce hydrogen via the water shift reaction ($CO + H_2O \longrightarrow CO_2 + H_2$). A recent meeting of EPRI and ERDA contractors was held by ERDA to assess the potential for synthetic fuel production in specially designed blankets. Six processes are considered to be viable candidates for fuel production. They are listed in Table I, along with some rough estimates of overall reactor efficiencies. While only one process was thought to be capable of exceeding the efficiency of electrolysis using electricity from a thermal cycle reactor, it should be noted that a number of processes are capable of producing fuel with significant efficiency.

(1) Electrolysis

Present efficiencies for conversion of electrical energy to hydrogen are in the 70 to 73 percent range. Projections of efficiencies of 80 to 85% have been make. Thus, if the power plant's electrical production efficiency is 40%, the overall efficiency for hydrogen production would be about 32%. This straightforward approach represents a baseline for judgement of other techniques.

(2) Thermal - Enhanced Electrolysis

Thermal-enhanced electrolysis would utilize the high flow stream temperatures available from fusion reactors to increase the effectiveness of the electrolytic process. In this case, high process heat temperatures of 1000^{o}C would be used to provide a significant fraction of the energy required for hydrogen production. This type of operation is actually a high temperature fuel cell, with solid-electrolyte membranes, working backwards. Steam is injected on one of the electrodes, the cathode, the water dissociates, and leaves H_2 on the side where the contact first occurs, $+O^{--}$. The O^{--} diffuses through the electrode, made of a conducting oxide (ZrO_2 for example). On the anode side, the O^{--} becomes O_2. The entry gas is steam mixed with hydrogen and the exit gas is mostly hydrogen mixed with steam.

(3) Radiolysis

Neutrons and X-rays are capable of directly converting water to hydrogen via radiolysis in specially designed reactor blankets. The best data available for radiolytic production of gases show conversion efficiencies in the 30% range. (α radiolysis of $CO_2 \longrightarrow CO + 1/2\ O_2$).

The difficulty in use of these reactions in a practical system comes in the design and incorporation of a radiolytic processing blanket in the reactor. Gas phase reactions require pressures on the order of 200 atmospheres to be able to absorb the neutron energy in a thickness the order of one meter. Blankets with thicknesses on the order of 1 meter or less are necessary for magnetic fusion reactors because of the high price of magnets. (This requirement could be much less restrictive in the case of laser fusion reactors).

The thick walls of the vessel needed to contain these pressures will absorb at least 1/3 of the primary neutron flux energy, according to calculations at Brookhaven National Laboratory. A self-sufficient Fusion reactor based on the D-T fuel cycle must also breed tritum in a portion of the blanket. It was estimated that this process would also take about 1/3 of the primary neutron flux energy.

Thus, while up to 30% of the neutron energy can reasonably be assumed to be convertible to H_2 on a fundamental basis, consideration of engineering factors would make only a third of the neutron energy released by the fusion reaction available for the process. The system efficiency would thus be on the order of 10% or less.

(4) Radiolytic Thermochemistry

Radiolytic-Thermochemical cycles would use a radiolytic process as one stage of a cycle that used thermal reactions at other stages.

One such process would be:

(a) $SO_2(g) + Br_2 + 2H_2O = H_2SO_4 + 2HBr$ (70-100^oC)

(b) $H_2SO_4 = H_2O + SO_2 + 1/2\ O_2$ (800^oC)

(c) $2HBr \xrightarrow{rad} H_2 + Br_2$

The efficiency for production of hydrogen via the above process could approach 32% in a fully engineered blanket.

(5) Thermal Direct

It is possible to design blanket process regions which will maintain temperatures of 2700-3200^oK. Such blankets would have carbide or oxide central regions contained by gas cooled metallic walls. Penetrating neutron or X-ray radiation would raise the temperature of the carbide or oxide matrix to between 2700 and 3000^oK. Direct decomposition of H_2O or CO_2 can take place at these temperatures.

$$2H_2O \longrightarrow 2H_2 + 2O_2 \qquad (3000^oC)$$
$$2CO_2 \longrightarrow 2CO_2 + O_2 \qquad (2000^oC)$$

The CO can then be used to produce hydrogen by the conventional shift reaction:

$$CO + H_2O \qquad CO_2 + H_2 \qquad (500^oC)$$

It is estimated that the above process could take place with a 35% conversion efficiency of fusion energy to stored chemical energy. (The efficiency includes blanket design considerations.) Sensible heat remaining from the process could convert an additional 20% of the fusion energy to electricity for an overall plant efficiency of 55%.

(6) Thermochemistry

Thermochemical processes can also make use of the radiative heating capability of fusion neutrons. The thermochemical concept is an entirely thermal cycle. An example of this is the FeO/Fe_3O_4 process in which water is hydrolized by reaction with FeO at relatively low temperature, e.g. 1000^oK to form H_2 and a higher valence oxide, e.g. FeO_4. The higher valence oxide is then heated to a temperature of 2000^oK, and the oxygen driven off. Such cycle could have an efficiency as high as 60%.

III. ADVANTAGES

(1) Synthetic fuel production offers its greatest advantages to small, less advanced fusion reactors, which require large amounts of circulating power. The size reactors needed for net power production could be reduced significantly thus allowing them to be built for relatively low capital cost, ($400-600 million). This would make it much more probable that a number of such reactors would be built at an earlier date than the large ($1-3 billion) units presently envisioned.

(2) Dual purpose fusion reactors which can vary the ratio of electrical to synthetic fuel production could provide an ideal solution to peak-to-base loading problems. When little electricity is in demand, the reactor would produce mostly synthetic fuel.

(3) The lifetime of existing fossil fuel plants could be extended by the use of synthetic fuel produced by fusion plants. Existing plants are already sited conveniant to urban centers. Thus, remote fusion plants could utilize existing pipelines to supply fuel to the closer-in installations.

(4) The capital investment in existing pipelines could be preserved.

(5) Hydrogen produced in fusion reactors would be ideally suited to fuel cells. This could make fuel cells the ideal way of adding increased electrical generation capacity in areas where addition of new transmission lines is difficult or impossible.

IV. RESEARCH AND DEVELOPMENT NEEDS

A two day workshop on the use of Fusion Reactors for Synthetic Fuel Production was held on February 9 and 10, 1976 under the sponsorship of the Electric Power Research Institute. It brought together a group including plasma physicists, fusion reactor designers, radiation chemists, and fuels scientists to discuss the potential uses fo fusion reactors in the production on synthetic fuels. The general conclusion of the meeting was that there were a number of reasons to vigorously pursue a program to define and explore synthetic fuels production by fusion reactors.

The workshop recommendations, in order of priority, were:

1. Modeling of Reactors -- Develop simplified reactor computer models which evaluate energy output as a function of reactor operating conditions.
2. Develop a priority list of suitable product fuels.
3. Use existing neutron sources to develop fundamental data on the interaction of 14 Mev neutrons with chemical systems.
4. Determine the extent of induced radioactivity in low z materials exposed to large fluxes of 14 Mev neutrons.
5. Design reactor blankets capable of process chemistry. Problems which need definition and study include tritium contamination, separation of the desired product, prevention of back reactions and estimates of overall thruput and economics.
6. Initiate studies of high temperature electrolysis (thermoelectro-chemistry), direct use of high temperature ($>1000^{\circ}C$) process heat, high frequency r-f induced chemical reactions.

It is clear this subject is in its infancy. However, the potential pay-off is enormous -- even crucial to the economic survival of our economy. Efforts should begin now to put this puzzle together. Total funding by appropriate organizations (ERDA and EPRI) has been less than $350,000.

TABLE I

ESTIMATED EFFICIENCIES OF HYDROGEN PRODUCTION IN FUSION REACTORS

METHOD	ESTIMATED EFFICIENCY *
Electrolysis	35%
Thermal Enhanced Electrolysis	$\geq$ 35%
Radiolysis	10%
Radiolytic-Thermochemistry m	$\leq$ 32%
Thermal-Direct	30%
Thermochemistry	35%

* Note that these are very rough estimates and are based on little hard data. The actual num bers could be more or less than these values

General Requirements and Approaches to Advanced Fuel Fusion*

by

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ABSTRACT

A new fusion ignition parameter ($Tn_e\tau_E$) is proposed which is proportional to β^2B^4 (or n^2T^2) and inversely proportional to the fusion power density of a reacting plasma. Ignition curves are given for many potential nuclear fusion fuels, and comparison is made with existing experiments. Prospects are presented for some advanced fusion fuels.

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*Research sponsored by Energy Research and Development Administration under contract with Union Carbide Corporation.

This is a condensation of (1) "The Ignition Parameter $Tn\tau$ and the Energy Multiplication Factor k for Fusioning Plasmas," ORNL/TM-5766 (May 1977), and (2) "Some Fusion Perspectives," in Proc. ERDA-ORAU Institute on Energy Sources for the Future, CONF-760744, 1976, both of which should be consulted for additional material and detailed references.

I. INTRODUCTION

At the first Atoms for Peace Conference held in Geneva in 1955, the Conference President, Homi Bhabha of India, stated, "The historical period we are just entering in which atomic energy released by the fission process will supply some of the power requirements of the world may well be regarded one day as the primitive period of the atomic age. It is well known that atomic energy can also be obtained by a fusion process as in the H-bomb, and there is no basic scientific knowledge in our possession today to show that it is impossible for us to obtain this energy from the fusion process in a controlled manner. The technical problems are formidable, but one should remember it is not yet 15 years since atomic energy was released in an atomic pile for the first time by Fermi. I venture to predict that a method will be found for liberating fusion energy in a controlled manner within the next two decades."

Now, 22 years later, this method has not been found — or if it has been found, it has not gained acceptance by the total fusion community. Yet, news reports still glow with the prospect of unlimited energy — clean, nonpolluting, safe — from nuclear fusion. Having been associated with the controlled thermonuclear research (CTR) program during these 22 years, I have seen new hopes arise and then dim, but there has been a steady progression in our knowledge of and ability to handle increasingly hotter, denser, and better confined plasmas. On the other hand, this knowledge has revealed that nuclear fusion has its own set of problems such as radioactivity, afterheat, neutron damage, and gamma (γ) rays, and therefore the glamour of benign fusion has somewhat dimmed.

The fact that different fusion fuels can be ignited has been demonstrated in the 1951 U.S. Greenhouse test of DT burning, the 1952 U.S. Ivy-Mike test of DD burning, the 1953 Soviet "Joe-4" test of LiD burning, and the U.S. Bravo and other tests of the Castle series in 1954 (LiD, LiD-U). Whether any of these fuels can be burned in a controlled manner for peaceful applications, such as in limited microexplosions after the fashion of the automobile combustion chamber or in steady-state nuclear burners using magnetic fields for confinement, remains to be determined. Fusion-fission hybrid reactors had their progenitor in the Castle series of explosive fusion-fission.

II. NUCLEAR FUELS AND REACTIONS

The classical fusion fuel for fusion reactors is DT (all others are called advanced fuels), which reacts much faster at low temperatures than any other charged particle combination. It involves a highly excited compound nucleus $^5He^*$ and reacts best at a deuteron bombarding energy of 0.107 MeV (or at a kinetic temperature of about 60 keV). Since tritium is almost nonexistent in nature, it must be regenerated by neutron reactions in a lithium blanket. Thus, with a tritium generation factor of about 1.2, the DT reactor is essentially a D(T)-Li reactor, and although deuterium is abundant in water ($\sim$1/6500 of ordinary hydrogen), the lithium must be obtained economically from the earth's crust (20 ppm), from salt brines ($\sim$50-300 ppm), or from the sea ($\sim$0.2 ppm).

The practical advanced fusion fuels include DD and D^6Li (both of which appear to be suitable fuels for steady-state magnetic containment reactors) and the more exotic and potentially much "cleaner" fuel, D^3He. The latter is dependent on an excess tritium inventory from DT reactors ($T \rightarrow {}^3He + \beta + \nu$) or on DD or D^6Li reactors which can breed ^3He and tritium. The fuels — $^3He^3He$, p^6Li, p^9Be, and $p^{11}B$ — do not appear to be practical at this time either because they probably won't ignite at all or because their ignition temperature is too high and energy return too low (e.g., $p^{11}B$) to permit steady-state operation. One might visualize a large, advanced fuel DD reactor operating in a reactor park and providing ^3He and tritium as source material for several urban-sited, clean D^3He reactors. The ultimate clean fuels ($^3He^3He$, p^6Li, $^3He^6Li$, and $p^{11}B$) merit further detailed studies.

In a driven, catalyzed DD burner, the unburned tritium and ^3He would be isotopically separated from the p and α ashes and returned with deuterium as fuel makeup to the reactors or transported to storage areas for tritium decay to ^3He in the reactor park. Subsequently, the ^3He would be transported to urban-sited, relatively clean D^3He reactors. Preliminary indications are that tritium in the plasma, 14-MeV neutrons, and the power output from neutrons can all be reduced by at least two orders of magnitude in a D^3He reactor compared with an equivalent power DT reactor. The DD reactor (catalyzed or noncatalyzed), on the other hand, is only marginally better (by a factor of two or more) than the DT reactor with reference to 14-MeV neutrons; however, the tritium abundance in the plasma is down by a factor of $\sim$20-30 and total tritium holdup may be only about 10^4 Ci. DD burners which have not been tritium catalyzed would require the cumulative storage of very large quantities of the excess tritium — but this would be stored in cold, static systems for ^{3}He production. It should be emphasized that any fusion reactor capable of producing copious free neutrons may serve as a breeder of either ^{3}He fusion fuel (via tritium production) or ^{233}U or ^{239}Pu fission fuel. This may pose a safeguards problem.

III. NUCLEAR BURN CHARACTERISTICS

Figure 1 schematically shows the burn sequence of a DT plasma as a function of ion temperature. Below ∿6 keV, the power expended in radiation losses (bremsstrahlung and synchrotron) exceeds the power deposited in the plasma by the α particles; thus, once the externally supplied energy source is removed the plasma will quench. At the ignition temperature the plasma is unstable, and a slight increase in temperature will lead to a thermal runaway to high temperatures (possibly 300 keV) independent of whether the

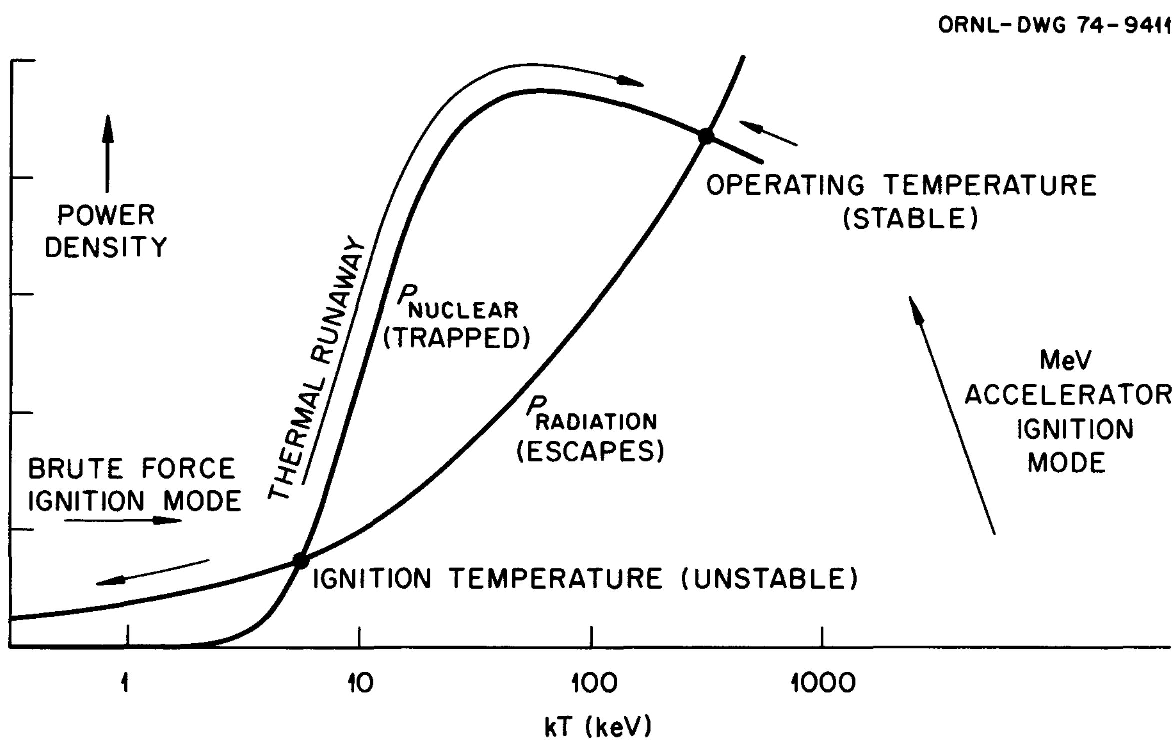

Fig. 1. Schema of burning characteristics of radiation dominated DT plasma.

external energy source is active. The steepness of the radiation power loss curve at 300 keV is due mostly to synchrotron radiation losses (which vary about as T_e^3).

At the operating or burning temperature, the plasma exhibits a stable behavior provided the fuel mix does not change (such as results from build-up of fuel or ashes). This operating point has a characteristic negative temperature coefficient like that of a fission reactor; if the temperature is perturbed the plasma tries to stabilize itself and will do so provided the perturbation is not too large. On the other hand, if the fuel density or mix changes, the plasma will seek a new operating point. Increase of fuel density leads to more reactivity, and since the synchrotron radiation is better absorbed in denser plasmas, the temperature increases. Thus, the plasma has an undesirable property — a positive density coefficient — which must be controlled by fuel feed or ash control or by manipulation of the magnetic field. A negative feedback control must be introduced to stabilize the nuclear burn. An increase of ash in the plasma will cool the plasma.

The presence of still other losses (particles or energy) in addition to radiation will lower the operating temperature point. Most of these losses (e.g., pseudoclassical diffusion across the magnetic field) are rather modest; however, if severe temperature-dependent losses such as the dissipative trapped ion (DTI) loss mode predicted for tokamak fusion reactors occur, the operating temperature may shift down to the T_i = 10-20 keV range. The DTI has not yet been observed experimentally, so it may not be operative in a tokamak or in other potential fusion devices. If it is not present in tokamaks, the possibility of a high beta ($\beta = 8\pi\Sigma nkT/B^2$) flux-conserved tokamak may be a real tokamak option, even permitting the prospect of burning advanced fuels. Instability of the plasma-magnetic field configuration can also lead to severe losses.

IV. THE $Tn\tau$ IGNITION CRITERION

The fusion power density is an important criterion for evaluating the economics of an ignited advanced fuel fusion reactor. The inverse of this quantity is proportional to $Tn_e\tau_E$ whose value defines the plasma ignition condition for the particular fuel at a particular temperature. The expression for $Tn_e\tau_E$ is

$$Tn_e\tau_E = \frac{3/2\ n_e[n_e + \alpha_{ij}(n_i + n_j) + n_x]}{\alpha_{ij}n_in_jQ_{ij}} \qquad \frac{T^2}{<\sigma v>_{ij}}$$

for $T = T_i = T_e$ and where $\alpha_{ij} = 1$ when $i \neq j$ and 1/2 when $i = j$, τ_E is the energy confinement time (and represents all losses), x is ash or impurity, and Q_{ij} is the fusion energy release in charged particles. Curves of $Tn_e\tau_E$ are given in Fig. 2 for DT, DD, DD (catalyzed), D^3He, $^3He^3He$, p^6Li (catalyzed)

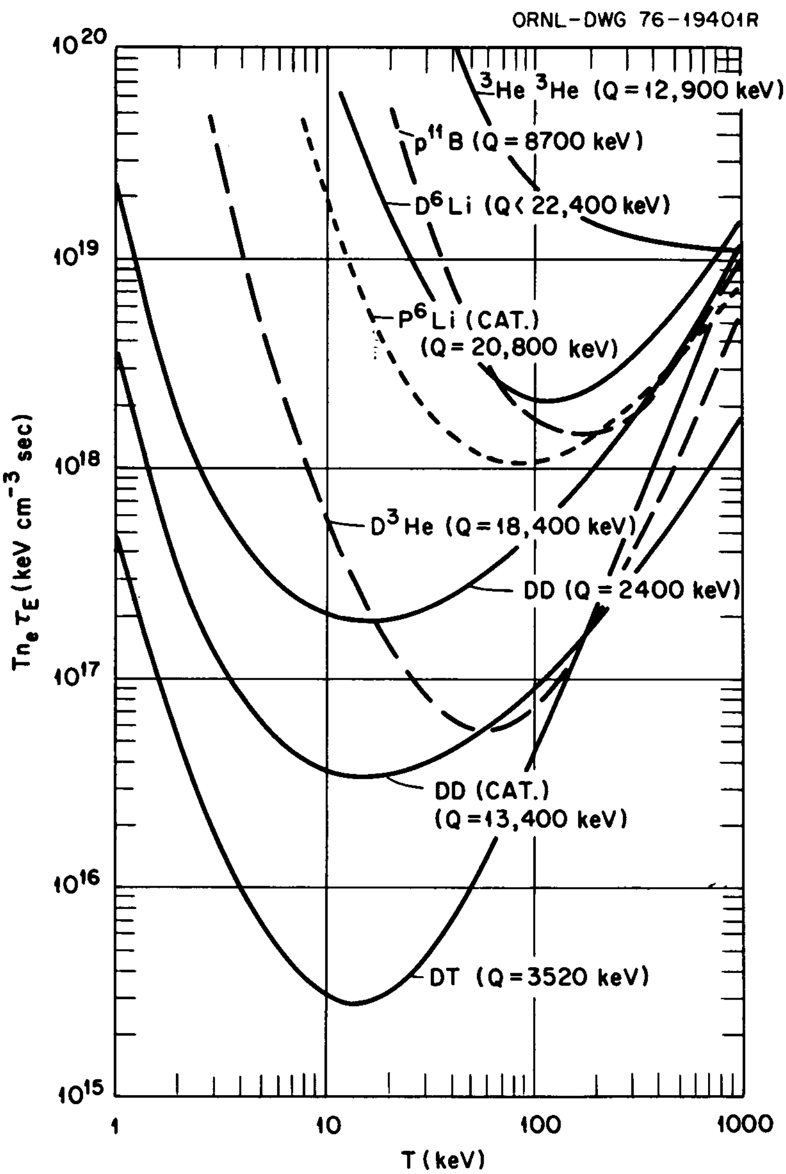

Fig. 2. Ignition criterion $Tn_e\tau_E$ vs T for various fusion fuels ($T_i = T_e$, x = 0).

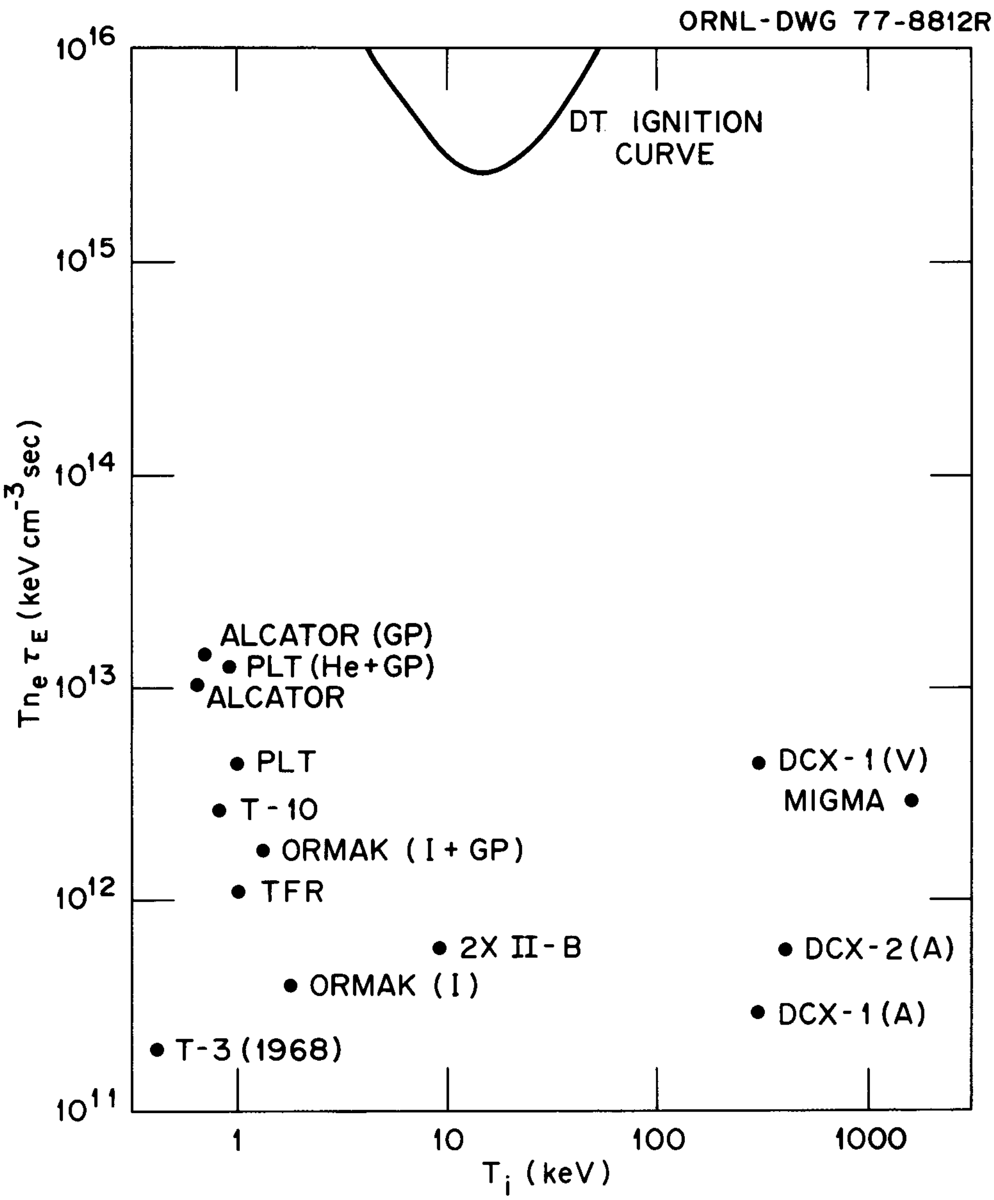

Fig. 3. Comparison of selected plasma experimental $Tn_e\tau_L$ vs T_i(keV) with the DT ignition curve (I = injection, GP = gas puff, A = arc, V = high vacuum, He = helium).

D^6Li, and $p^{11}B$. Minimum values of $Tn_e\tau_E$ range from 3×10^{15} keV cm^{-3} sec for DT at 15 keV to 10^{19} keV cm^{-3} sec for $^3He^3He$ at 1 MeV; the latter would necessitate very high β and magnetic field (B) values to provide interesting fusion power densities since $P \propto \beta^2 B^4 \langle\sigma v\rangle/T^2$. However, high B mitigates against low synchrotron radiation so that very high β plasmas or surface magnetic configurations (e.g., SURMAC) would be required. <u>Power density losses</u> can be expressed as $Tn_e\tau_E = 3/2n_e[n_e + \alpha_{ij}(n_i + n_j) + n_x]T^2/P_L$ and $1/\tau_{E,L} = \Sigma\, 1/\tau_L$; these loss curves define an upper bound for $Tn_e\tau_E$.

The energy utilization factor f in existing devices is defined as $f = P_{Fusion}/P_{Loss} = (Tn_e\tau_L)/(Tn_e\tau_E)_I$. In experimental plasmas, f has increased by about two orders of magnitude in the past decade and now exceeds 10^{-4} (a "nearest" f* exceeds 10^{-3}, see ORNL/TM-5766) as shown in Fig. 3. The f factor is analogous to its fission counterpart in the four-factor neutron multiplication factor, $k = f\eta\varepsilon p$, where f is the neutron thermal utilization factor. Thus, the simple ignition parameter $(Tn_e\tau_E)_I$ is only a first cut at evaluating the plasma energy criticality or ignition. The physical interpretations of both $(Tn_e\tau_E)_I$ and k_{Fusion} are discussed elsewhere. In Fig. 1, k = 1 at the ignition (unstable) temperature as well as at the burning or operating (stable) temperature.

The classical fusion fuel DT has the lowest $(Tn_e\tau_E)_I$ curve and hence would have the highest power density for given $\beta^2 B^4$. It may be too reactive in large systems in the range T < 80 keV; thus, it may be important to evaluate the advanced fuels with this aspect in mind as well as considerations of tritium inventory, first wall damage, and 14-MeV neutron radiation damage. To attain the same power density with the advanced fuel reactors as with DT fuel at T < 80 keV, the advanced fuel reactors would need larger values of $\beta^2 B^4$. Catalyzed DD fueled fusion reactors have an ignition parameter only about 10 times higher than pure DT at 10 keV, not 100 times as is commonly implied.

Should DT plasmas burn best at T = 100-300 keV, the catalyzed DD and cleaner D^3He-fueled reactors would have a real competitive economic advantage as well as comparable power densities. The more advanced fuels — p^6Li, $p^{11}B$, $^3He^3He$, or $^3He^6Li$ — would require a significantly larger $\beta^2 B^4$ value in this temperature range. Recent studies suggest that catalyzed DD, DD, D^6Li, D^3He can operate in principle as steady-state reactors, provided $\beta \gtrsim 20\%$, $n_e\tau_E \gtrsim 2 \times 10^{15}$ cm^{-3} sec, and T > 50-100 keV.

The five tokamaks and the 2XII-B mirror device shown in Fig. 3 emphasize large power, pulsed heating technologies (large ohmic heating currents and/or intense neutral beam injection). Such approaches pose potential deleterious gas loading and first wall power loading problems which must be overcome to permit the $\gtrsim$1000 fold extrapolations to the ignition requirement. However, significant progress has been made over the past nine years as shown by the 1968 T-3 result with present-day Alcator and PLT results.

There is increased optimism for smaller sized, high toroidal beta (β_T) tokamaks for D-burning based on the flux-conserving tokamak ideas of John Clarke (ORNL). Equilibrium configurations having β_T up to at least 0.28 and preliminary MHD stability limits of 0.05-0.10 have been deduced theoretically. Since power output varies as β_T^2, and conventional tokamaks were thought to be operable at $\beta_T \leqslant 0.03$, a significant reduction in size (minor radii of 1.25 by 2 m) would be necessary to achieve comparable power loading of the walls. It should be noted that the poloidal beta ($\beta_p \sim \beta_T B_T^2/B_p^2$) has exceeded unity on a transient basis in some tokamaks even though this violates the pinch condition (but satisfies flux conservation). There is also some theoretical evidence that density profiling of flux-conserved tokamak plasmas may not only permit high average β_T but also the higher temperatures needed for some advanced fusion fuels. The higher temperature operation would lead to slightly larger flux-conserved tokamak sizes due to the lower power densities (see Fig. 2 and recall $P_{Fusion} \propto 1/Tn_e\tau_E$).

The steady-state DCX-1 magnetic mirror experiment at high vacuum (DCX-1V) utilized modest beam power ($\sim$10 mA of 600 keV H_2^+). Attempts were made to achieve an exponential or vacuum buildup of the fast ion density (exponentiation occurs when the plasma trapping rate exceeds the charge exchange loss rate) which, if successful, would have given about a three-thousand-fold increase of density provided ECH (electron cyclotron heating) were used to reduce the stopping power of the cold electrons. The energetic protons were confined for τ_p up to 150 sec; however, the highly organized proton orbits in DCX-1 drove a severe negative mass instability which prevented attainment of the exponential condition and eventually the experiment was terminated. The 2XII-B mirror experiment did achieve exponential density buildup, but because of the extremely large charge exchange loss cross section at the low beam energies required about 300 A (eq.) of injected neutral deuterium atoms in the 5-liter volume.

The multi-MeV MIGMA mirror experiment has used 3-MeV D_2^+ injection and will attempt exponential buildup as well as Lorentz trapping, and may ascertain the severity of instabilities in a well-spread ion velocity distribution. Exponentiation appears much more favorable at very high ion energies (especially for protons) because of the million-fold reduction of charge exchange losses compared to the 2XII-B case. If instabilities are surmountable, it appears very likely that intense ECH will be required to permit long ranging exponentiation or Lorentz trapping buildup in devices using only modest injection currents ($\sim$1 mA).

The simple magnetic mirror devices may not allow economic fusion power unless the ends are stoppered in some way or one goes to the beam-driven fusion-fission option or the beam-driven, large circulating power pure fusion devices. Figure 4 illustrates a potential Ion Layer with E core as a possibility for end closure. The two reports cited on the title page

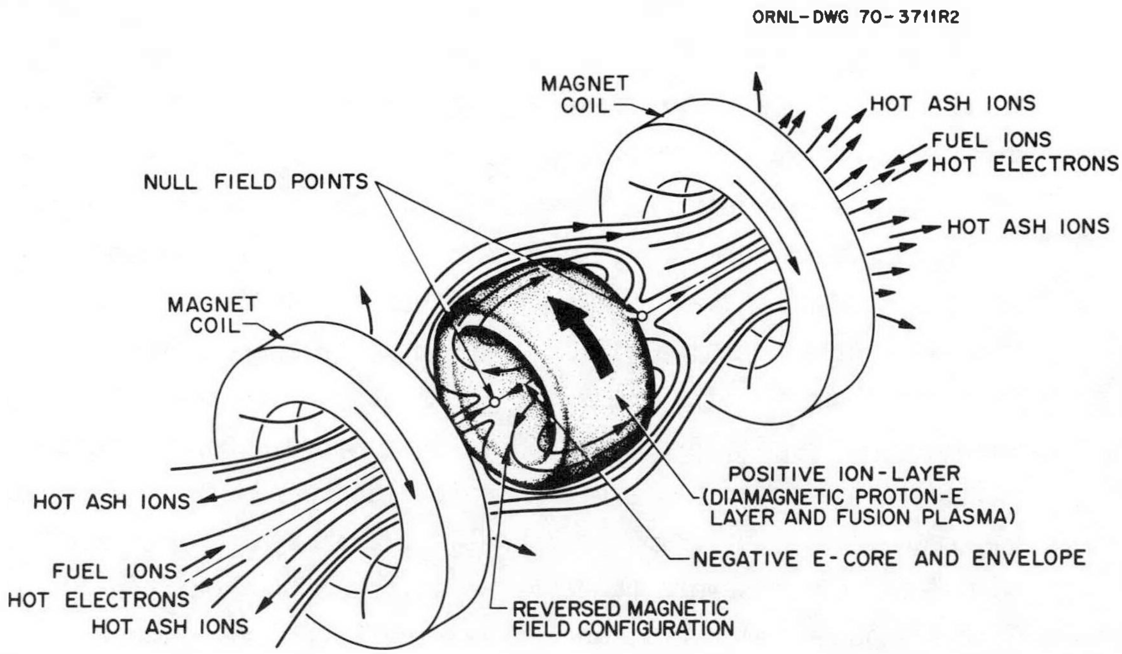

Fig. 4. The Ion-layer, E-core plasma electromagnetic configuration showing axial feed and heating of cold fuel and charge separation of departing ash ions and electrons.

give some details on Ion Layer production, including the use of GeV ions to produce large, reactor-size systems.

V. ADVANCED PELLET FUSION FUELS

The classical pellet fusion fuel is solid DT (ρ_o = 0.21 g/cm^3). Other DT-enriched chemical compounds have much higher initial densities (up to $\sim$1 g/cm^3) so that the compression required for ignition of the pellet is significantly relaxed, i.e., $C = (\rho R/\rho_o R_o)^{3/2}$. ρR is the analogous Lawson condition and $\rho R = n_e \tau v_s M_e$ expressed in g/cm^2 (v_s is the sound velocity $\simeq \sqrt{5nk(T_i + T_e)/3\rho}$ and M_e is the mass per electron). The value of ρR must be of order 1 g/cm^2 for ignition so that if R_o = 0.3 mm, for which m $\sim$ 22 microgram of DT, ρR must be about 150 times larger than $\rho_o R_o$ and the required compression is about 1800. By using the advanced pellet fuels at much higher initial densities, the compressional requirements are relaxed by $(5)^{3/2}$ and compression need be only 160. Already $(CD_2)_n$ pellet experiments have given compressions of thirty-fold. Thus, a slight increase in core radius (say to R_o = 1 mm) would reduce the required compression to the experimentally achieved value of about 30.

It should be noted that solid DT ignites at 3 keV provided the ρR condition is achieved. The advanced pellet fusion fuels — $LiD_{0.5}T_{0.5}$, Be DT, $BD_{1.5}T_{1.5}$, $(CDT)_n$, CD_2T_2, $ND_{1.5}T_{1.5}$, ODT — will have higher bremsstrahlung losses because of the high Z of the carrier element; consequently some increase in ignition temperature is to be expected. If we define Q_p to be

$$Q_p = \frac{P_+}{P_{Brem}} = \frac{n_D n_T \langle\sigma v\rangle Q_+}{\alpha n_e \Sigma n_i Z_i^2 \sqrt{T_e}} > 1 \qquad (1)$$

for ignition (α = numerical constant), then the density coefficient K is

$$K = \frac{n_D n_T}{n_e \Sigma n_i Z_i^2} , \qquad (2)$$

and is given below.

Pellet	State	ρ_0(est.)	K
DT	solid	0.21 g/cm^3	0.25
$^6LiD_{0.5}T_{1.5}$	solid	0.83	0.006
^9BeDT	solid	0.97	0.009
$^{11}BD_{1.5}T_{1.5}$	solid	0.77	0.010
^{12}CDT	solid	1.16	0.003
$^{12}CD_2T_2$	liquid	0.57	0.010
$^{14}ND_{1.5}T_{1.5}$	liquid	1.03	0.004
^{16}ODT	liquid	1.16	0.0015

The relative reactivity coefficient K is 25 times larger for DT than for DT-enriched, liquid methane (CD_2T_2); however, $\langle\sigma v\rangle/\sqrt{T_e}$ varies as T^3 in the range T = 3-10 keV. Thus, the ignition temperature for CD_2T_2 is only about 3 times higher than that for solid DT. In fact, the degree of difficulty, as measured by the product $T_{ign}C$, favors some of the advanced fuels compared to solid DT. In an actual pellet, the initial compression will lead to a faster reactivity response whereas subsequent expansion of the pellet will give reduced reactivity.

At an electron density as low as 10^{14} e/cm^3, the CD_2T_2 plasma will still burn at $T \gtrsim 50$ keV, so that if additional DT fuel can be added to the plasma and the carbon and ashes gradually flushed out, one can visualize igniting and perhaps sustaining a magnetically confined plasma. Eventual feed of DD, D^6Li, or especially D^3He would ensure a somewhat cleaner burn. The exploding pellet plasma would introduce a large-scale diamagnetism of the hot plasma if located in a magnetic mirror field and this may permit development of a natural Ion Layer configuration. Thus, pellet-fusion microexplosions could serve as the "match" to ignite steady-state fusion reactors.

Some years ago, a 6-m-long, magnetically confined carbon arc was found to have ions 100 times hotter than the electrons ($T_i \sim 500$ eV, $T_e \sim 5$ eV). An hypothesis called "excitation-heating" was proposed to explain this unusual phenomenon. It involved the transfer of electron kinetic energy to excitation energy of doubly ionized carbon ions to form metastable ions (C^{2+*}).

At 5-10 eV an inverted population of C^{2+*} ions is produced, and most ionic collisions occur between two highly excited C^{2+*} ions. These collisions would be superelastic and the stored energy converted to kinetic energy of the ions; multiple cyclic processes lead to runaway ion temperatures. The individual steps in the cycles are:

$$\text{Excitation:} \quad e + C^{2+}\ (^1S_0) \rightarrow e + C^{2+*}\ (^3P^0_{2,1,0}) - 6.5\ \text{eV}, \qquad (3)$$

$$\text{Heating:} \quad 2C^{2+*}(^3P^0_{2,1,0}) \rightarrow 2C^{2+}(^1S_0) + 13.0\ \text{eV}. \qquad (4)$$

A dense pellet plasma containing carbon may prove even more efficient as a catalyst for promoting runaway ion temperatures in the dense pellet for three reasons. 1) The slowing down of fast ions in dense, low temperature plasmas decreases markedly because of the density effect on the effective stopping power. 2) The electrons form a degenerate Fermi gas [$T_{F(keV)} = 2.4 \times 10^{-18}\ n_e^{2/3}$] which reduces the stopping power. 3) Excitation-heating might occur very efficiently provided T_e could be maintained at 10 eV and $n_e > 10^{25}$ e/cm^3.

Whether such catalytic ignition of DT-enriched polyethylene or liquid methane could be induced is a problem for the future. Suffice it to say that special care will have to be exercised in view of the hazardous nature of pellet microexplosions; a 1-mg DT burn is almost equivalent to 0.1 ton of TNT.

"CROSS SECTION REQUIREMENTS"

by

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ABSTRACT

Nuclear cross section requirements for potential advanced fuels will be discussed. The emphasis will be on three systems: P-^{11}B, P-^{6}Li, and D-^{3}He. A suggested order of priorities for these cases will also be included. Other fuel options, less promising than the three cited above, will be mentioned briefly.

I. Introduction

Within the Division of Magnetic Fusion Energy (DMFE) of ERDA, the Applied Plasma Physics (APP) Program and the Development and Technology (D&T) Program have nuclear physics responsibilities. However, the interests of each are quite distinct. D&T is concerned primarily with neutron cross sections in the blanket for breeding tritium and shielding. Presently, APP does support some dosimetry work but focuses upon cross sections for fusion fuels, i.e., D-T and the advanced fuels. The Experimental Branch of APP executes the cross-section tasks, while the Advanced Fusion Concepts Branch examines alternatives to tokamaks and mirrors; feasibility for advanced fuel operation and energy conversion schemes are among the criteria used for evaluation. Thus, the APP Program recognizes that the issue of advanced fuels consists of a package, i.e., cross sections, device, and energy conversion. This talk will deal with one aspect of that package, the cross sections. The organization of APP relevant for advanced fuels is shown in Figure 1.

APPLIED PLASMA PHYSICS

EXPERIMENTAL BRANCH
- A) Advanced Fuel Cross Sections

ADVANCED FUSION CONCEPTS BRANCH
- A) Advanced fuel confinement devices
- B) Energy conversion methods

APP ORGANIZATION FOR ADVANCED FUELS

FIGURE 1

DMFE believes that investigations of advanced fuels and advanced fusion concepts are timely and appropriate at this point. As noted above, alternate devices will be required so that advanced fuel operation will not be constrained by tokamak or mirror imposed limitations. DMFE is studying the situation to determine whether the development of advanced concepts along with reasonable extrapolations of existing technologies indicate that advanced fuels may possibly exist as alternatives to D-T on a shorter timescale than previously thought.

A questionnaire was sent recently to a group of prominent investigators whose names and affiliations are listed in Figure 2. The purpose of the questionnaire was to establish present needs for advanced fuel cross-sections. The format was to consider three classes of fuels: near term (i.e., D-D, D-^{3}He), longer term (i.e., Li, Be and B) and very high temperature (i.e., ^{3}He-^{3}He, ^{6}Li-^{6}Li). The respondents were asked to comment on the strengths, weaknesses, and experimental and theoretical cross section needs for each fuel. A portion of this talk will be a report on the responses. As of this date, two-thirds of the responses have been received. The greatest interest appeared to be in three fuels: D-^{3}He, P-^{6}Li, and P-^{11}B. Each of these will be discussed in turn.

II. D-^{3}He

There is interest in this fuel from two perspectives: first, as a fuel in its own right and second, as an important factor in the catalyzed D-D cycle. The reactions and energies involved are given in Figure 3.

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FIGURE 2

$D + {}^3He \rightarrow {}^4He + p$ ($Q_+ = 18.4$ MeV)

CATALYZED D-D: $D+D \rightarrow n + {}^3He$ ($Q_+ = .8$ MeV)

$\rightarrow p + T$ ($Q_+ = 4.0$ MeV)

$D+T \rightarrow n + {}^4He$ ($Q_+ = 3.5$ MeV)

$D+{}^3He \rightarrow {}^4He + p$ ($Q_+ = 18.4$ MeV)

SUMMARY: $2\,D + 4\,D \rightarrow 2n + 2p + 2\,{}^4He$ ($(Q_+)_{AVERAGE} = 13.4$ MeV)

Q_+ = ENERGY RELEASED IN CHARGED PARTICLES

FIGURE 3

D-^{3}He has several positive aspects such as:

1. a large Q_+ value (18.4 MeV)
2. a peak in the reaction rate $<\sigma v>$ below 100 keV ($\sim$ 50-70keV)
3. at 100 keV, the $<\sigma v>_{D^3\text{-}He}$ is comparable to $<\sigma v>_{D\text{-}T}$
4. parasitic D-D neutrons may be significantly suppressed by injecting D onto a cold ^{3}He target.
5. the D-^{3}He channel makes a substantial contribution to the Q_+ value (13.4 MeV) of the catalyzed D-D cycle.

The primary weakness of D-^{3}He is that ^{3}He does not occur naturally and must be obtained from tritium decay (as is the case today) or bred, either remotely or in a catalyzed situation. In either case much of the rationale for seeking an advanced fuel will have been compromised, i.e., tritium handling and 14 MeV neutrons would reappear as major concerns. However, it may be safe to assume that experience with D-T reactors will provide methods and means for dealing with, and possibly alleviating, these problems. Thus, D-^{3}He should still be considered.

There was some disagreement among the respondents regarding the quality of present D-^{3}He cross sections below 100 keV. A low-intermediate priority perhaps should be given to cross section measurements in the 10-100 keV region since these additional measurements would essentially refine the current data. Refined cross sections may be very helpful in reaching a final determination of the D/^{3}He fuel mix which in general should be lean in D and rich in ^{3}He.

Other pertinent issues are correcting the $\langle\sigma v\rangle$ as a function of temperature and σ as a function of energy from assumed maxwellian distributions to include beam-plasma and thermal effects. Additional factors arise in the catalyzed D-D approach. Two of the most important would be first, the impact on the reaction rate of suprathermal ions (since D-D produced ^{3}He has an energy of 800 keV) and second, a sound knowledge of elastic cross sections and the effect of elastic collisions on the distribution function in these energy regimes.

III. P-^{6}Li

With P-^{6}Li, a new region is entered in several respects:

1. high temperature operation, i.e., several hundred keV, is imperative
2. tokamak and possibly mirror systems are not applicable due to high β and radiation considerations among others,
3. unlike the D-^{3}He case, ignition for P-^{6}Li is questionable,

4. greatly increased number of side reactions,

5. radiation loss requires $T_e < T_i$, low z, low $|\underline{B}|$

Despite these factors the P-^{6}Li system is worth pursuing for the following reasons:

1. both fuel components not only occur naturally, but also are abundant

2. the system can, and in fact for feasibility, must be operated in a fully catalyzed mode which is described in Figure 4.

3. the catalyzed system has a very high Q_+ (20.8Mev)

4. the system is clean, i.e., neutron and tritium free

CATALYZED P-^{6}LI:

$$p + {}^6\mathrm{Li} \rightarrow {}^3\mathrm{He} + {}^4\mathrm{He} \quad (Q_+ = 4.0\ \mathrm{MeV})$$

$${}^3\mathrm{He} + {}^6\mathrm{Li} \rightarrow p + 2\,{}^4\mathrm{He} \quad (Q_+ = 16.8\ \mathrm{MeV})$$

$$\mathrm{He}^3 + \mathrm{He}^3 \rightarrow {}^4\mathrm{He} + {}^2p \quad (Q_+ = 12.9\ \mathrm{MeV})$$ (POSSIBLE SIGNIFICANT CONTRIBUTION TO REACTION RATE)

FIGURE 4

There are several cross section needs for the P-^{6}Li system. Further work on refining and confirming present data is appropriate. However, the most pressing need is for ^{3}He-^{6}Li cross-sections below 1 MeV where little, if any, data now exists. If catalyzed P-^{6}Li is to be seriously considered, then obviously the ^{3}He-^{6}Li cross sections must be known over a wide energy range. Suprathermal ^{3}He in the MeV range is generated by P-^{6}Li. In this energy regime ^{3}He-^{3}He reaction rates, computed with the usual assumptions about maxwellians and current cross sections, are relatively large. ^3He-^3He cross sections should be measured down to a few hundred keV. The branching ratios of possible side reactions should also be well established. Finally as discussed with the catalyzed D-D cycle, elastic cross sections for suprathermal products must also be known.

The theoretical needs are in general similar to those mentioned with D-^3He, i.e., corrections on reaction rates and cross sections and determinations of optimal operating temperatures for each species and the mix of the fuel ions as a function of confinement device.

IV. P-^{11}B

The points made about P-^6Li regarding new operating considerations are also applicable to P-^{11}B. In addition, the factors justifying continued interest are partially applicable. Specifically, the statements about catalyzed cycles do not apply since ^4He-^{11}B has neutron and ^{14}C producing branches and the Q_+ for P-^{11}B is a rather modest 8.7 MeV. The basic reaction along with undesired side and secondary reactions are shown in Figure 5.

$$p + {}^{11}B \rightarrow 3\ {}^4He \quad (Q_+ = 8.7\ MeV)$$

SIDE AND SECONDARY REACTIONS:

$$p + {}^{11}B \rightarrow n + {}^{11}C$$

$$^{11}B + {}^4He \rightarrow {}^{14}N + n$$

$$^{11}B + {}^4He \rightarrow {}^{14}C + p$$

FIGURE 5

Recently groups at Cal Tech and LLL completed measurements of P-^{11}B cross sections down to $\sim$ 100 keV with very good agreement. The primary cross section needs for the P-^{11}B system are for the side and secondary reactions shown in Figure 5 and also for ^{11}B (p,p), ^{11}B $(\alpha\alpha)$ ^{11}B. The ^{11}B(p,p)^{11}B cross section should be measured from 100 kev-1 MeV, the others from 100 kev - 4.5 MeV. The theoretical needs are very similar to those given for P-^{6}Li.

V. Other Possibilities

As stated earlier in this talk, three fuels would be emphasized. However, there are other possible advanced fuels and it would be appropriate at this point to mention why they may not be promising. If an advanced fuel is defined as anything past D-T, then advanced fuels would begin with D-D.

D-D: neutron and tritium problems, low Q_+

catalyzed D-D: discussed in section on D-^{3}He

D-^{6}Li: has neutron branches and a tritium branch, must also prevent D-D reactions

P-^{7}Li ^{7}Li is neutron rich

D-^{7}Li must also prevent D-D reactions

P-^{9}Be: ^{9}Be is toxic and relatively scarce

^{3}He-^{3}He: does not occur naturally, very high temperatures required

^{6}Li-^{6}Li: Operating temperatures are too high

VI. Summary

The needs for the three major systems will be given with a priority listed for the fuel:

1. D-^{3}He:

   Needs - refine existing data in 10-100 keV range
   consider beam-plasma and thermal effects in $\langle\sigma v\rangle$ and σ
   correct reaction rate for suprathermal ions
   determine elastic cross sections for suprathermal ions

   Priority - low - intermediate due primarily to ^{3}He production problems

2. P-^{6}Li:

   Needs - refine and confirm existing data on P-^{6}Li
   measure ^{3}He-^{6}Li cross sections below 1 MeV
   measure ^{3}He-^{3}He cross sections below 2 MeV
   determine elastic cross sections for suprathermal ions

   Priority- intermediate - high since the reaction has attractive possibilities and can be used as a back-up for P-^{11}B

3. P-^{11}B:

   Needs - measure cross sections of side and secondary reactions along with $^{11}B\,(p,p)^{11}B$ and $^{11}B\,(\alpha\alpha)^{11}B$

   Priority - high since this is the most promising advanced fuel and questions regarding ignition and feasibility may be answered in a relatively short time.

In closing, it should be noted that a general need expressed by several respondents was for a survey of existing data and a compilation of that data.

"Cross Section Measurements and Needs"

by

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ABSTRACT

There are only a limited number of nuclear reactions between charged particles that have a significant cross section at energies available in a fusion reactor. Cross sections should be determined for all of these reactions. A nuclide may be introduced into the reaction region as a fuel component, as a reaction product, or as an impurity. The ^{6}Li-^{6}Li and ^{3}He-^{6}Li reactions have not been adequately considered from a fusion point of view. These two reactions are discussed in detail both for their intrinsic importance and as examples of the kind of reactions that occur between the more complex nuclei.

I. INTRODUCTION

What kind of nuclear physics information is needed for the design and operation of fusion reactors? The answer is that no nuclear reaction that could take place inside of a reactor should be ignored. One misjudgement in the design of a large thermonuclear facility caused by deficient nuclear information could cost a thousand times as much as the total cost of acquiring all of the nuclear information.

The items of information that are needed are cross sections. All other nuclear parameters, such as $\langle\sigma v\rangle$ values, are derived from the cross sections.

It is interesting to compare the charged particle cross section requirements of the fusion program with the neutron cross section requirements of the fission program. With neutrons, an enormous number of similar measurements is required. Cross sections are needed for the interaction of neutrons with every type of atom that might be found in a reactor. With fusion, the only elements involved are H, He, Li, Be, and B. The neutron energy of interest varies over a factor of about 10^9 while for charged particles the range is usually less than a factor of 10^2. Neutron cross sections are characterized by resonances where the cross section bounces up and down rapidly as a function of energy (Figure 1). Fusion cross section curves are smooth and can usually be characterized by a dozen or so points. The number of fusion cross section data points is not quite as small as these comments imply. One beam-target pair can result in a large number of reaction products. Table 1 shows the products of the ^6Li-^6Li reaction. A different experimental procedure is required for each item in the list. In some cases the products are difficult or virtually impossible to separate from each other and from the much larger amount of scattered beam. With the aid of a suitable theoretical model, some of the inaccessible data points can be

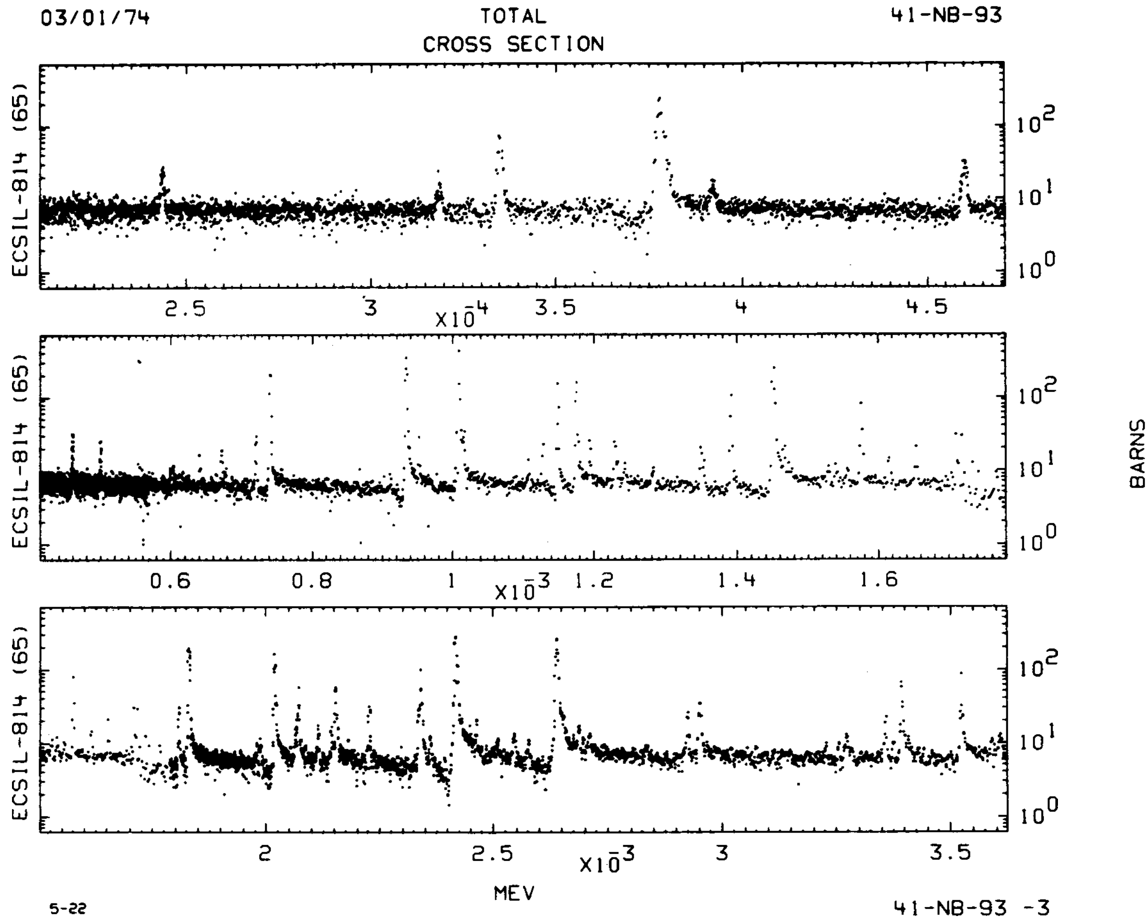

Figure 1. A small portion of the neutron cross section as a function of neutron energy for ^{93}Nb.

calculated. The conclusion is that single fusion cross section point is more expensive to obtain than a neutron point but much fewer are needed.

Neutron cross sections have been mass produced, mostly in national laboratories. Fusion cross section measurements require many different kinds of short experiments. The emphasis is on flexibility and ingenuity. This is the kind of work that university laboratories do well.

The total cost of determining all of the fusion cross sections will be small if it is done on a small scale over a number of years. If each cross section were to be studied on a crash basis, the total cost would be much higher.

Table 1

^{12}C	+	γ	+	28.2 MeV
^{11}C	+	n	+	9.4
^{11}B	+	p	+	12.2
^{10}B	+	d	+	3.0
^{10}B	+	p+n	+	0.8
^{9}B	+	t	+	0.8
^{10}Be	+	2p	+	1.0
^{9}Be	+	^{3}He	+	1.9
^{7}Be	+	α+n	+	1.9
^{7}Li	+	α+p	+	3.5
3α		+		20.9
2α	+	t+p	+	1.1
2α	+	^{3}He+n	+	0.3

Products from the $^{6}Li + {}^{6}Li$ Reactions
with Positive Q Values

For many years, the general feeling was that the only cross sections worth knowing were D-T and D-D. We now know that it is worthwhile considering other fusion fuels. There is great interest now in finding the best nuclear reaction that does not produce too many neutrons. The decision as to which is best will depend not just on the primary reaction but on the sum total of all the reactions that occur in the plasma. For example, a reacting D-^{6}Li plasma involves D-D, ^{6}Li-^{6}Li, and D-^{6}Li along with additional reactions involving the products of the original components.

Beryllium is mentioned occasionally as a candidate for the first wall. The metal has a high thermal conductivity. It is exceptionally transparent to photons and is resistant to sputtering. The oxide, which has similar virtues, has been suggested as a refractory lining to protect an underlying metallic wall. Since material from the first wall can easily find its way into the reacting plasma, cross sections are needed for reactions of ^{9}Be with the various fuel candidates.

The consequences of the accidental introduction of a reactive material into a fusion plasma should be given careful consideration. For example, a small leak in the first wall of a D-T reactor could allow lithium from the tritium breeding blanket to enter the plasma. Would this cool the plasma or would it cause a dangerous temperature excursion?

II. ^{6}Li-^{6}Li Reaction

There is much still to be known about this reaction, although some information is available. Figure 2 shows the cross section as a function of energy for some of the products. The top two curves were from measurements by McGrath[1] of the γ rays from the first excited states of ^{7}Li and ^{7}Be. The numbers are somewhat different than those given in the reference. The original numbers, as explained in McGrath's thesis, were calculated assuming an effective charge of +2.5e for the ^{6}Li ions as they left the target. The numbers in Figure 2 were calculated assuming an effective charge derived from

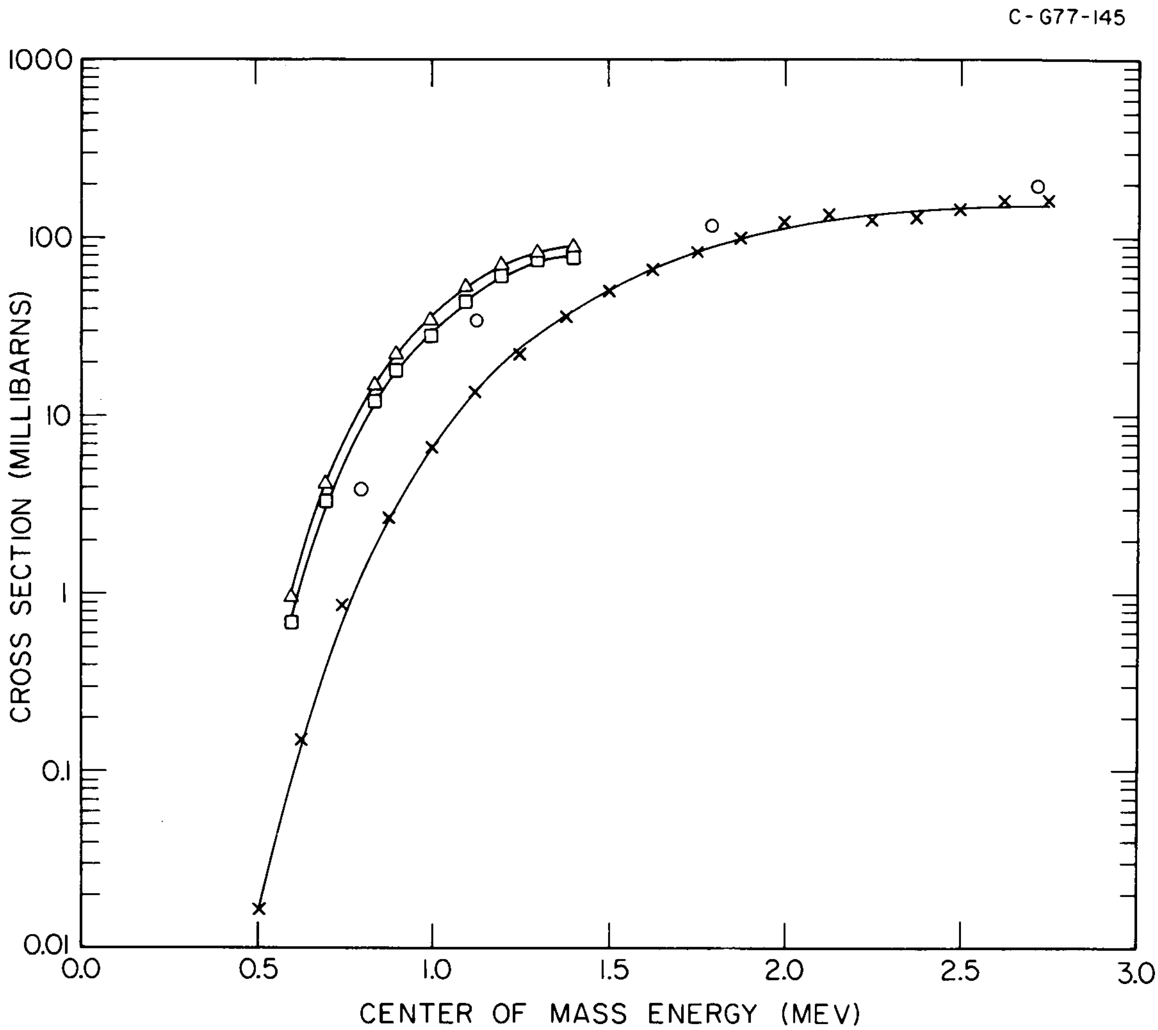

Figure 2. $^6Li + {}^6Li$ Cross Sections

△	$^6Li + {}^6Li \rightarrow {}^7Li^* + \alpha + p$	$^7Li^*$ in .478 MeV state
□	$^6Li + {}^6Li \rightarrow {}^7Be^* + \alpha + n$	$^7Be^*$ in .431 MeV state
○	$^6Li + {}^6Li \rightarrow {}^7Be + \alpha + n$	^7Be radioactivity
×	$^6Li + {}^6Li \rightarrow 3\alpha$	High energy peak

the data given by Teplova.[2] The round ^7Be points are from recent measurements by L. Ruby et al. They measured the 53-day radioactivity of ^7Be. Their cross sections, which are for the production of the two lowest states in ^7Be, should be larger than those of McGrath which are for the excited state only. This discrepancy is a reflection of the difficulties of making accurate measurements of low energy charged particle cross sections.

The α-particle spectrum from the $2^6Li \rightarrow 3\alpha$ reaction shows a large high energy peak that corresponds to the entire Q value, 21 MeV, of the reaction being divided between two of the three α particles.[4] The cross sections shown in Figure 2 include only that part of this reaction that puts α particles into the high energy peak. The uncertainty is about ± 20% at the higher energies and perhaps as much as ± 40% at the lowest energies.

Table 1 shows that neutrons are included among the products in a number of cases. Only one of these, $^{11}C + n$, results in neutrons appreciably above 1.0 MeV. The cross section for the various levels in ^{11}C can be estimated by assuming that they are equal to the cross sections for corresponding levels in the mirror nucleus ^{11}B which have been measured at 2.0 MeV (CM).[5] Using the Q value as roughly the energy of the neutron, cross sections,for the three highest energy groups are 2.3 mb for 9.5 MeV, 0.6 mb for 7.5 MeV, and 1.7 mb for 5.2 MeV. These may be compared with 120 mb in Figure 2 for $^7Be + \alpha + n$. The cross sections for producing $^{10}B + p + n$ and $2\alpha + {}^3He + n$ have not been measured, but I would guess the two together to be 200 mb at the same energy. In summary, the ^6Li-^6Li reaction produces a lot of low energy neutrons but very few with energy above 1.0 MeV.

^7Be is a radioactive material with 53-day half life which could constitute a health hazard. This nucleus is also formed by several other reactions, ^6Li-D, ^6Li-^3He, and ^{10}B-P. It would be desirable to know the cross sections for reactions that would consume ^7Be inside of the reactor. Since ^7Be has a high cross section for thermal neutrons, it could be burned out by collecting it in a region with a large thermal neutron flux.

^{10}Be is also radioactive but with a half life of more than 10^6 years. No measurements have been made for the reaction $^6Li + ^6Li \rightarrow ^{10}Be + 2p$. Nuclear physicists would very much like to have gram quantities of ^{10}Be available for use as targets. There are a number of interesting experiments that cannot be done until ^{10}Be is available.

I would like to raise the question as to what would happen to Be and B reaction products. Would they end up as a coating on the first wall?

The production of $^{12}C + \gamma_6$ is effectively zero, less than a few microbarns at several MeV. This is fortunate because high energy gamma photons heat the containing walls without heating the plasma, and ^{12}C has a high Z that would increase bremsstrahlung losses.

The rapidly rising ^6Li-^6Li cross section could have interesting effects on the temperature dependence of the reactivity of a plasma containing ^6Li along with lower Z fuels. The use of pure ^6Li as a fuel seems unlikely because of the small cross section at low energy.

III. $^3He + ^6Li$ Reaction

The reaction $^6Li + ^3He \rightarrow 2\alpha + p + 16.9$ MeV puts all of its 16.9 MeV into charged particles. It has not been considered seriously as a fusion fuel because the published cross sections have been so small. The actual cross sections are probably three to ten times larger. Figure 3 shows the origin of the uncertainty. Each of these graphs show the number of protons as a function of proton energy for one location of the proton detector and a single ^3He beam energy. The two peaks at the high energy end have been the object of many careful studies. The cross section for the reaction is proportional to the total number of protons, but the published cross sections are based only on the protons in the two peaks. The experiments lost the low energy protons in the foil that stopped the scattered beam. A complete proton spectrum would show two more peaks at the low energy end and a broad continuum in the middle region. The two low energy

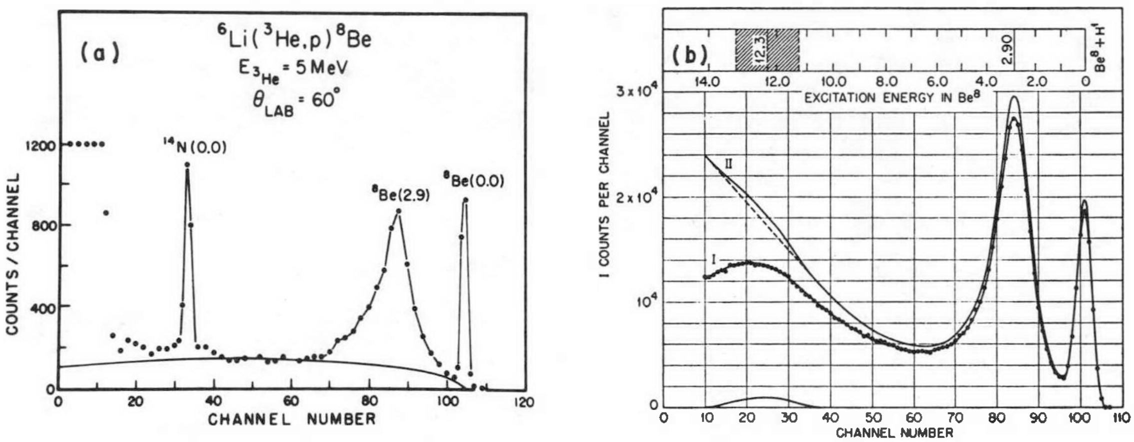

Figure 3.

(a) Proton spectrum. Stopping foils prevented ^3He and α particles from reaching the detector. The lower line is a fit of the three-body phase space expression to the proton continuum.

(b) Pulse height spectrum in NaI for $^6Li(^3He,p)^8Be$ taken at 45° with respect to the incident beam. ^3He beam energy 1.25 MeV. I. Data points normalized to unit pulse-height interval and plotted to arbitrary scale. II. Data points normalized to unit energy interval.

peaks[7] contain at least as many protons as the high energy peaks.[8] This correction, by itself, would multiply the cross sections by a factor of two. The big unanswered question is the size of the continuum in the middle. Gould and Boyce[8] (Figure 3a) assumed that it could be inferred by fitting the high energy end of it with a three-body phase space formula. From our studies of three-body final states at Iowa, we have found that this is not a reliable procedure. The continuum does not look very large in their figure where the beam energy was 5.0 MeV. They claim that the continuum looks even smaller at their lowest energy of 3.0 MeV. Figure 3b suggests that the continuum may be more important at even lower energies.[9] This graph is essentially the same as the one on the left except that the ^3He beam energy was 1.25 MeV.

The production of low energy protons can be studied by looking at high energy α particles. Preliminary work at the University of Iowa shows a large broad peak in the high energy part of the α spectrum, but quantitative work has not yet been done.

The only other positive Q-value reaction is $^{6}Li + {}^{3}He \rightarrow {}^{7}Be + {}^{2}H + 0.1$ MeV. This reaction has been studied from .3 to .9 MeV (CM) by Aleksic et al.[10] The cross section increases with energy and is up to 120 mb at .9 MeV.

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Recent $^{11}B(p,3\alpha)$ Cross Section Measurements†

by

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ABSTRACT

The renewed interest in using the $^{11}B(p,3\alpha)$ reaction in advanced fuel fusion has made it important to have accurate nuclear reaction cross-section data for feasibility evaluations. The purpose of this note is to present the most recent (and reliable) data for the $^{11}B(p,3\alpha)$ cross section and for its $\langle\sigma v\rangle$.

†Work supported in part by the National Science Foundation (Grant PHY76-83685) and by the Electric Power Research Institute (Contract TPS 77-708).

I. Introduction

The combination of a large nuclear reaction cross section and t[redacted] absence of neutrons as primary reaction products combine to make $^{11}B(p,3\alpha)$ a nearly unique choice for advanced fuel fusion. Weaver et al. have provided a detailed discussion of $^{11}B + p$ and its secondary reactions in laser pellet fusion,[1] and Dawson has recently given a parallel treatment for a magnetically confined plasma device.[2] In both cases it is clear that the cross sections used were of uncertain reliability — which prompted our desire to improve the experimental situation. The existing literature for $^{11}B(p,3\alpha)$ yields little consensus[3]; therefore, the expression for $\langle\sigma v\rangle$ in the recent compilation by Fowler et al.[4] was also suspect.

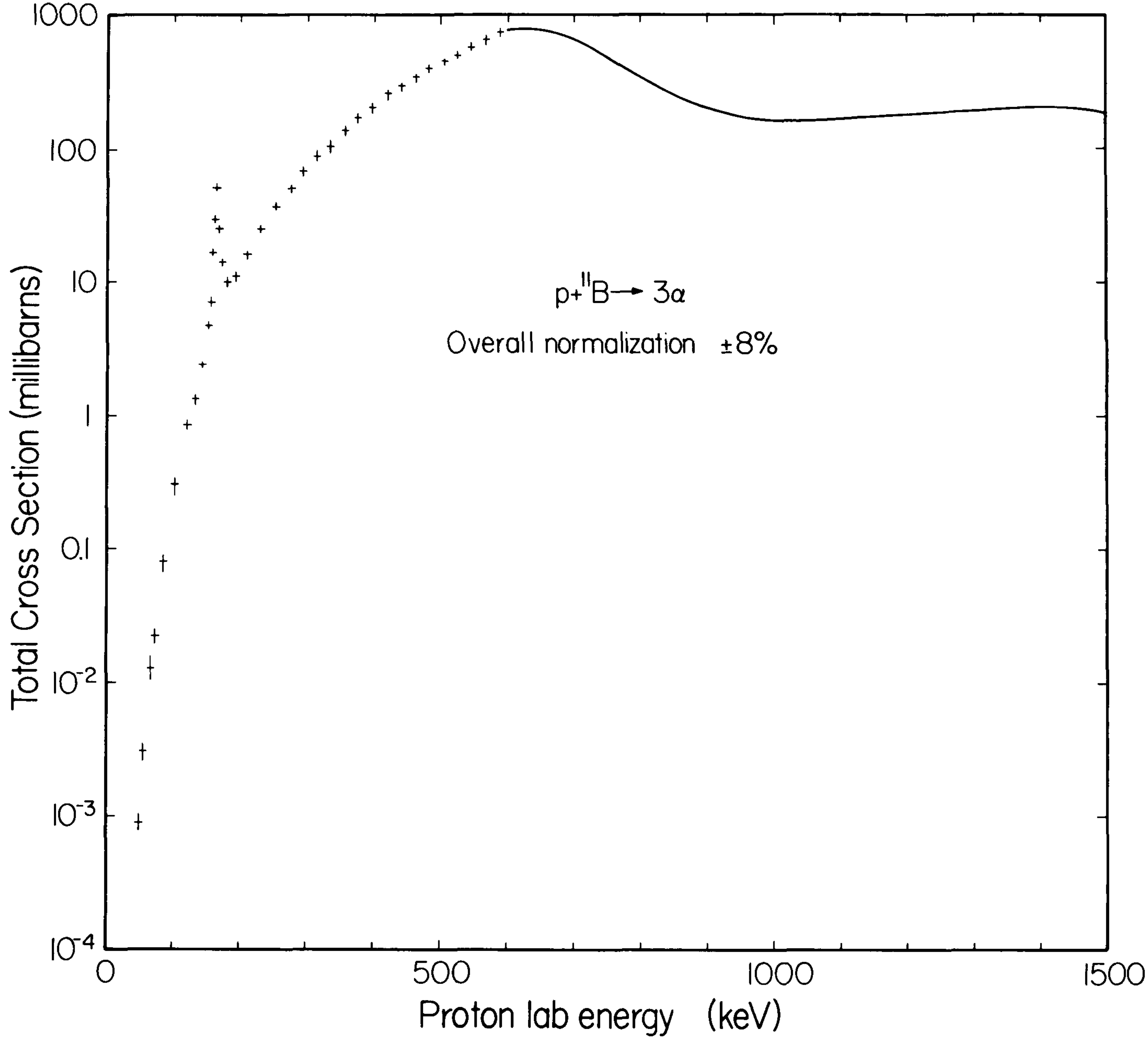
Previously, workers in our laboratory[5] had obtained data for $^{11}B(p,3\alpha)$ over the laboratory energy range $0.15\ \mathrm{MeV} \le E_p \le 1.5\ \mathrm{MeV}$. These data had an estimated overall accuracy of ±20%, which made it difficult for Weaver et al.[6] to decide whether a break-even situation with regard to bremsstrahlung losses could be attained. This note is not meant to replace a more detailed presentation[7]; however, the extent of current interest in $^{11}B(p,3\alpha)$ makes it desirable that accurate cross section and $\langle\sigma v\rangle$ results be available in preliminary form.

II. Reaction Data

The total cross-section values are given in Fig. 1 versus the laboratory proton energy. The data points are from the present work; the solid curve is drawn through the higher energy data of Lowry et al.[5] The overall, absolute uncertainty of the present data is ± 8%. Though the Lowry et al. data had an estimated uncertainty of ± 20%, the agreement with the new data is much better than that original estimate.

The total cross sections shown are obtained from angular distributions taken down to a proton energy of 60 keV. At this point the cross section is nearly isotropic in the laboratory system; thus, the corrections at lower energies for the deviation from isotropy are truly negligible. At the present time data have also been taken down to $E_p = 35$ keV; and the extension of the measurements to higher energies is in progress.

One should note that the rapid change of the cross section at low



Total Cross Section (millibarns)
1000
100
10
1
0.1
10^{-2}
10^{-3}
10^{-4}
$p+{}^{11}B \rightarrow 3\alpha$
Overall normalization ±8%
0
500
1000
1500
Proton lab energy (keV)

Fig. 1

energy requires a very accurate knowledge of the beam energy, the target thickness and the target composition. The determination of these quantities is at least as time consuming as the cross-section measurements themselves, but in their absence the data would be virtually useless.

In Fig. 2 we present the S-factor as a function of the center-of-mass energy. This presentation eliminates most of the variation in the cross section that arises from the Coulomb repulsion between the proton and the ^{11}B target — leaving (to a first approximation) the variation due to specifically nuclear phenomena, in this case the resonances at 150 keV and 540 keV. For this figure:

$$\sigma(E) = \frac{S(E)}{E} \exp(-2\pi\eta),$$

where E is the center-of-mass energy and $\eta = (Z_1 Z_2 e^2/\hbar v)$. (The incident ion has charge Z_1, the target nucleus has charge Z_2 and v is their relative velocity.)

In Fig. 3 the reaction rate for $^{11}B(p,3\alpha)$ is averaged over the Maxwell-Boltzmann distribution and plotted versus kT (in keV). The dashed curve comes from the compilation of Fowler et al.[4] The curves agree well-enough at low temperatures, but the divergence at temperatures above 100 keV is striking. Since Dawson's calculations[2] were based on the dashed curve, $^{11}B(p,3\alpha)$ actually offers a somewhat less optimistic picture than he estimated.

III. Conclusion

The new data presented for $^{11}B(p,3\alpha)$ are of sufficient accuracy to allow reliable feasibility estimates of its role in advanced fuel fusion. Though we shall continue our work to improve the range and precision of the cross-section data, the plots given here for σ, S and $\langle\sigma v\rangle$ are so much better than any that have been used previously that we strongly urge their use not only in all future calculations but also in the recalculation of laser pellet and magnetic confinement devices.

IV. Acknowledgments

The author thanks A. J. Sierk, J. M. Davidson and H. L. Berg for their permission to use this note as a way of making their data available quickly in preliminary form.

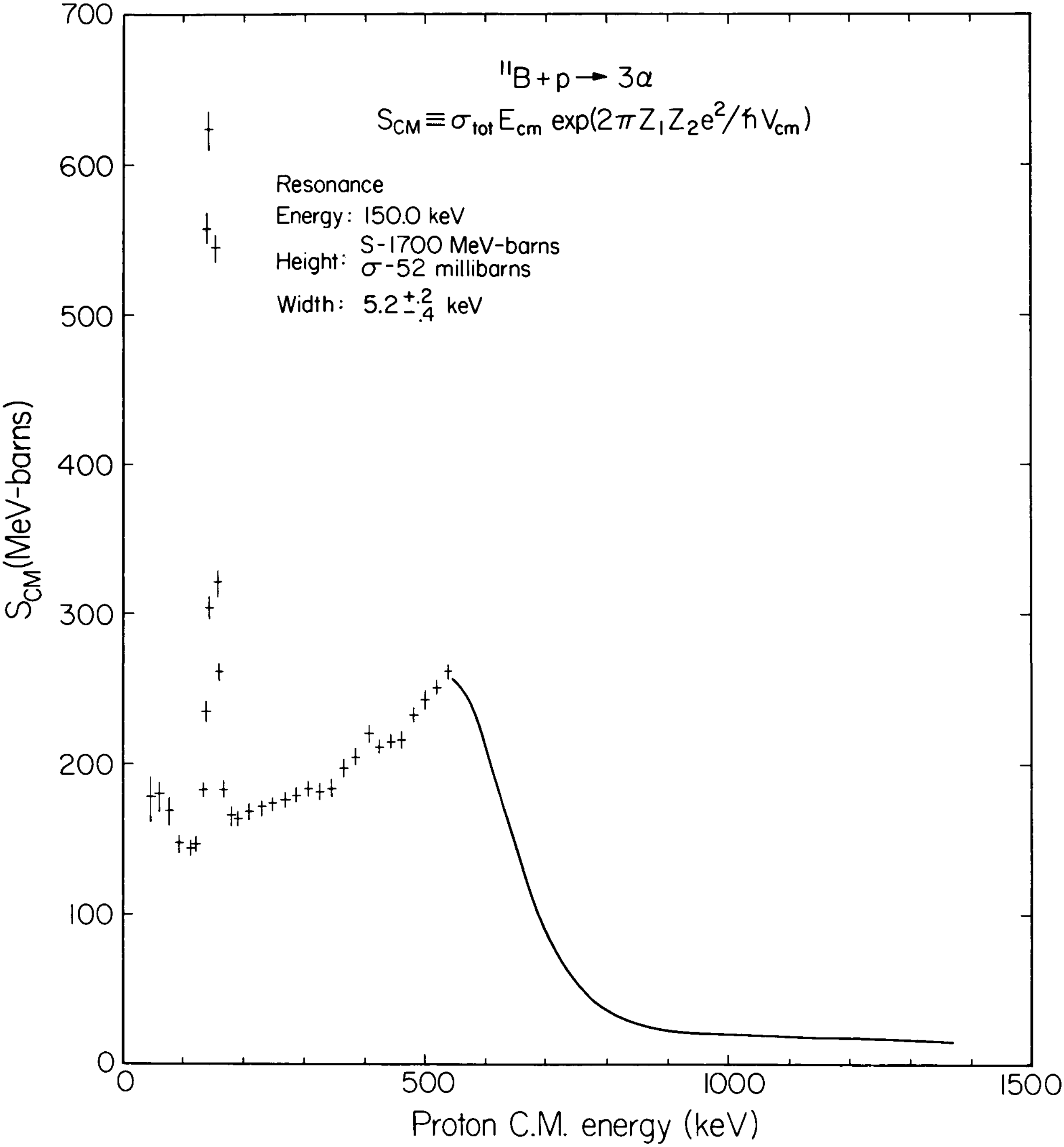
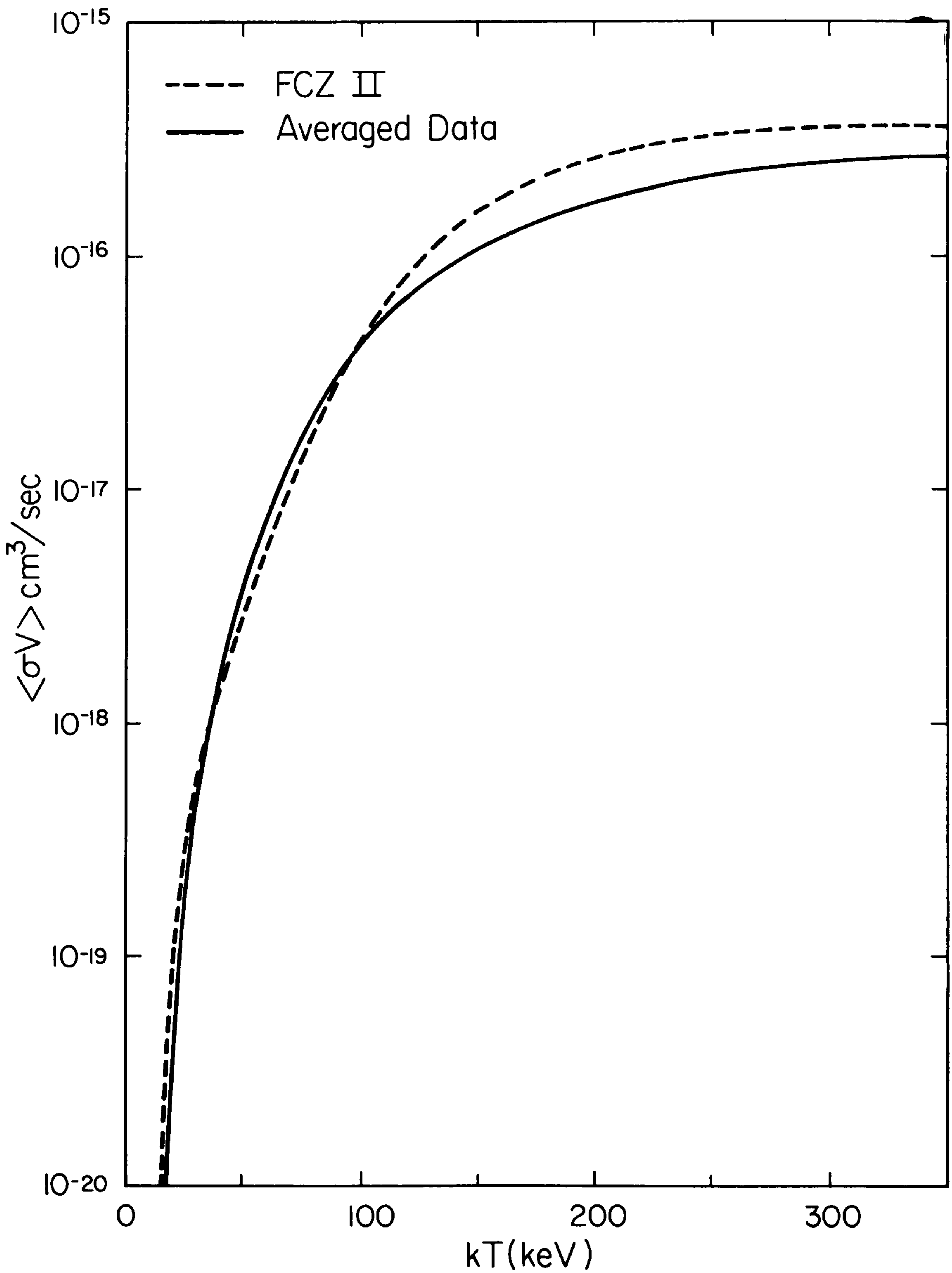

$^{11}B + p \rightarrow 3\alpha$
$S_{CM} \equiv \sigma_{tot} E_{cm} \exp(2\pi Z_1 Z_2 e^2/\hbar V_{cm})$
Resonance
Energy: 150.0 keV
Height: S-1700 MeV-barns
σ-52 millibarns
Width: $5.2^{+.2}_{-.4}$ keV
S_{CM}(MeV-barns)
Proton C.M. energy (keV)
0
100
200
300
400
500
600
700
0
500
1000
1500

Fig. 2



FCZ II
Averaged Data
10^{-15}
10^{-16}
10^{-17}
10^{-18}
10^{-19}
10^{-20}
$\langle \sigma V \rangle \, cm^3/sec$
0
100
200
300
kT(keV)

Fig. 3

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Figure Captions

Figure 1. The total cross section (in millibarns) for the $^{11}B(p,3\alpha)$ reaction as a function of the laboratory proton energy (in keV). The error bars shown give the relative errors of the points; in addition, there is an overall $\pm 8\%$ uncertainty in the absolute scale. The solid curve is drawn through the data of Lowry et al.[5] (The Lowry et al. data had an estimated $\pm 20\%$ uncertainty, and their excellent agreement with the present data is well within these limits.)

Figure 2. S(E) for $^{11}B(p,3\alpha)$ as a function of the center-of-mass proton energy. As in Fig. 1, the data points are from the present work; the solid curve is from ref. 5. (The definition of S(E) is given in the text.)

Figure 3: The reaction rate $\langle\sigma v\rangle$ (in cm^3/sec) for $^{11}B(p,3\alpha)$ as a function of kT (in keV). The solid curve is obtained from the present data plus those from ref. 5. The dashed curve is that given in the recent compilation of Fowler et al.[4]

Advanced Fuel Bumpy Tori

by

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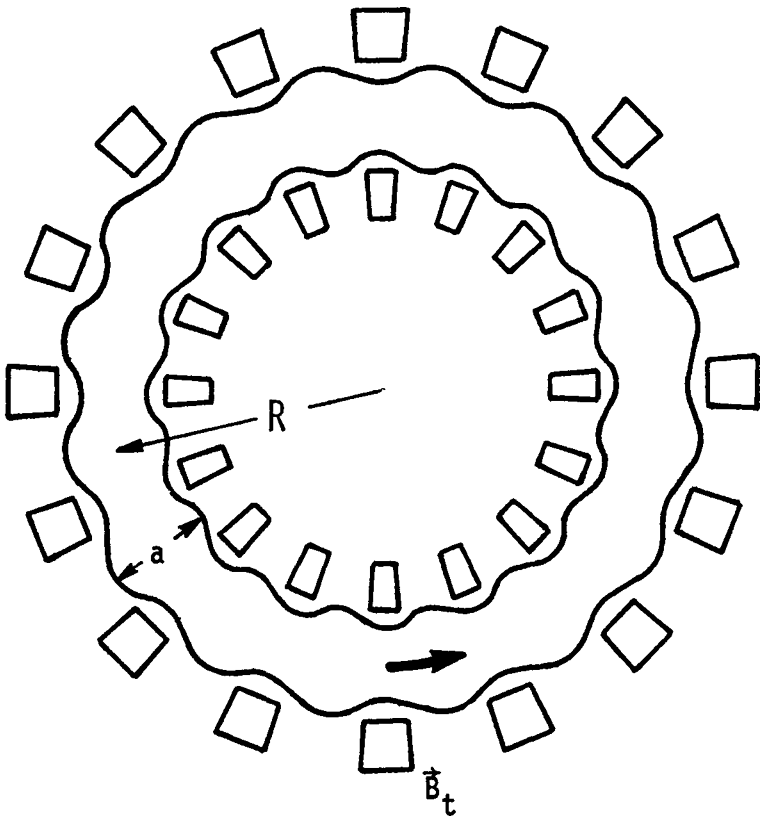
ABSTRACT

Using neoclassical bumpy torus scaling and theoretical predictions of high β bulk plasma stability, the feasibility of the relativistic electron ring bumpy torus has been studied as an advanced fuel reactor. Preliminary results indicate that modest size (600 MW_e to 1200 MW_e) D-^{3}He reactors are achievable under these assumptions with high Q_e (>20) and low neutron wall loading ($\leq$0.05 MW/m^2). The latter result coupled with the steady state nature of the device should result in high availability.

Bumpy tori (figure 1) appear to have many features which lend themselves to an attractive reactor concept utilizing advanced fusion fuels. Many of the uncertainties in the scaling of bumpy tori into the high β temperature regime should be answerable in the early '80s with the EBT II experiments.[1] Hence, advanced fuel bumpy tori studies are especially timely and might even point the way to bypassing DT systems altogether. The problems associated with DT of high energy neutron flux and wall damage, and the necessity to breed tritium are largely eliminated with the advanced fuels. The advanced fuel system may be more attractive, even at the initial stages of fusion power generation.

Bumpy tori are attractive as advanced fuel reactors for three reasons (figure 2): high β, favorable $n\tau$ scaling with temperature and large aspect ratio. High β is especially important for use of advanced fuels because their power density is roughly a factor of fifty lower than that for DT in a device of the same β and magnetic field B. Since power density, P_d, is proportional to β^2B^4, going to high β's and magnetic fields have a dramatic effect on power density and hence the economics (assuming the highest power density minimizes the cost per kilowatt). Maximum power densities, however, cannot be achieved by DT bumpy tori because the neutron wall loading becomes excessive, whereas Cat. D could achieve twice the power density of DT and D-^{3}He a factor of 20 times the DT power density before the neutron wall flux becomes a problem. Hence, it should be possible to design an advanced fuel bumpy torus that is limited only by the wall radiation loading with respectable power density and with a much longer 1st wall lifetime.

Since the minimum confinement requirement near peak power densities for Cat. D and D-^{3}He is at about an ion temperature of 50 keV, it is important that the confinement scaling of bumpy tori increase with temperature (or at least not decrease too fast). The present EBT I experiment has confinement times equal to collisional neoclassical[2] values at temperatures of 100's of eV. At high temperatures the neoclassical theory predicts that $n\tau$ should increase as $T^{3/2}$. While this scaling presents a problem of thermal stability it would give great flexibility in design (and the device could be run slightly subignition to facilitate control of thermal runaway). The uncertainty as to whether this scaling will continue to hold at higher temperatures should be answered in the early '80s with the EBT II experiments.



BUMPY TORUS GEOMETRY

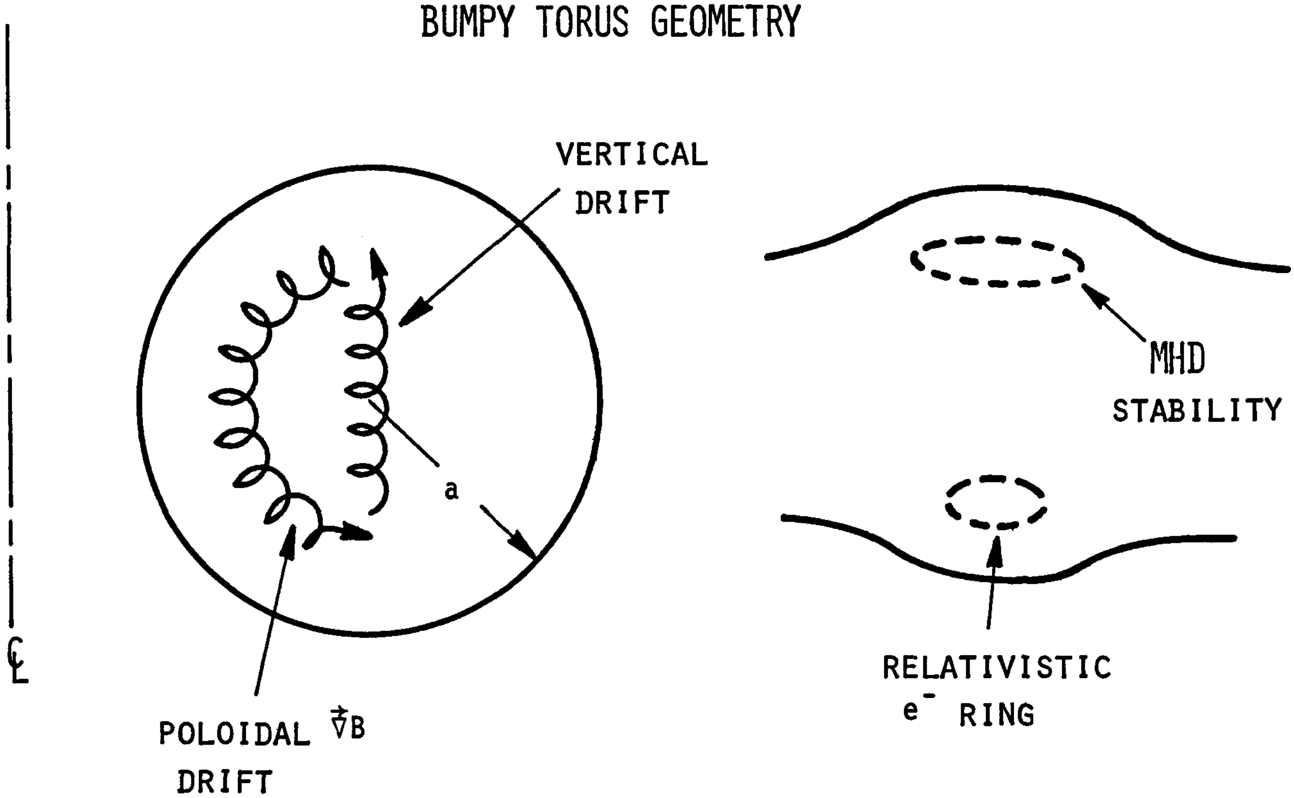


Figure 1

HOW BUMPY TORI MEET PROBLEMS OF ADVANCED FUELS

PROBLEM:	BUMPY TORI SOLUTION:
1) LOW POWER DENSITY $P_d \propto \frac{\beta^2 B^4 <\sigma v> Q_f}{T^2}$	1) HIGH β ($\beta \to 0.5$) THEORETICALLY POSSIBLE
2) HIGH TEMP. AND $n\tau$ NECESSARY	2) NEOCLASSICALLY $n\tau \propto T^{3/2}$ EBTI $\tau \approx$ NEOCLASSICAL
3) LARGE LEAKING PARTICLE POWER $\to$ DIRECT CONVERSION DESIRABLE	3) LARGE ASPECT RATIO CONVENIENT FOR DIVERTOR + EXPANSION REGION

Figure 2

Finally a high aspect ratio is desirable for allowing ease in maintenance and promotes a modular design. Additionally, utilizing advanced fuels allows a thinner blanket and shield. This leads to a better use of the magnet volume and a lower ratio of maximum field to the confining field for a given reactor power level. Since the bumpy torus needs a high aspect ratio for confinement and has a much weaker field between coils it appears quite simple to install a toroidal-bundle divertor (figure 3). This could allow a highly advantageous attachment of direct convertors to the bumpy torus. In view of the large fraction of power carried by leaking charged particles with advanced fuels and the potential for high ion temperature operation with the bumpy torus, attaching a direct convertor could yield a very attractive high efficiency fusion power plant.

With these advantages in mind, a series of advanced fuel bumpy tori reactors were designed using a steady-state computer code devised for use in earlier Tokamak studies.[3,4] In the present designs, the power density was maximized and the thermal power minimized with the following constraints

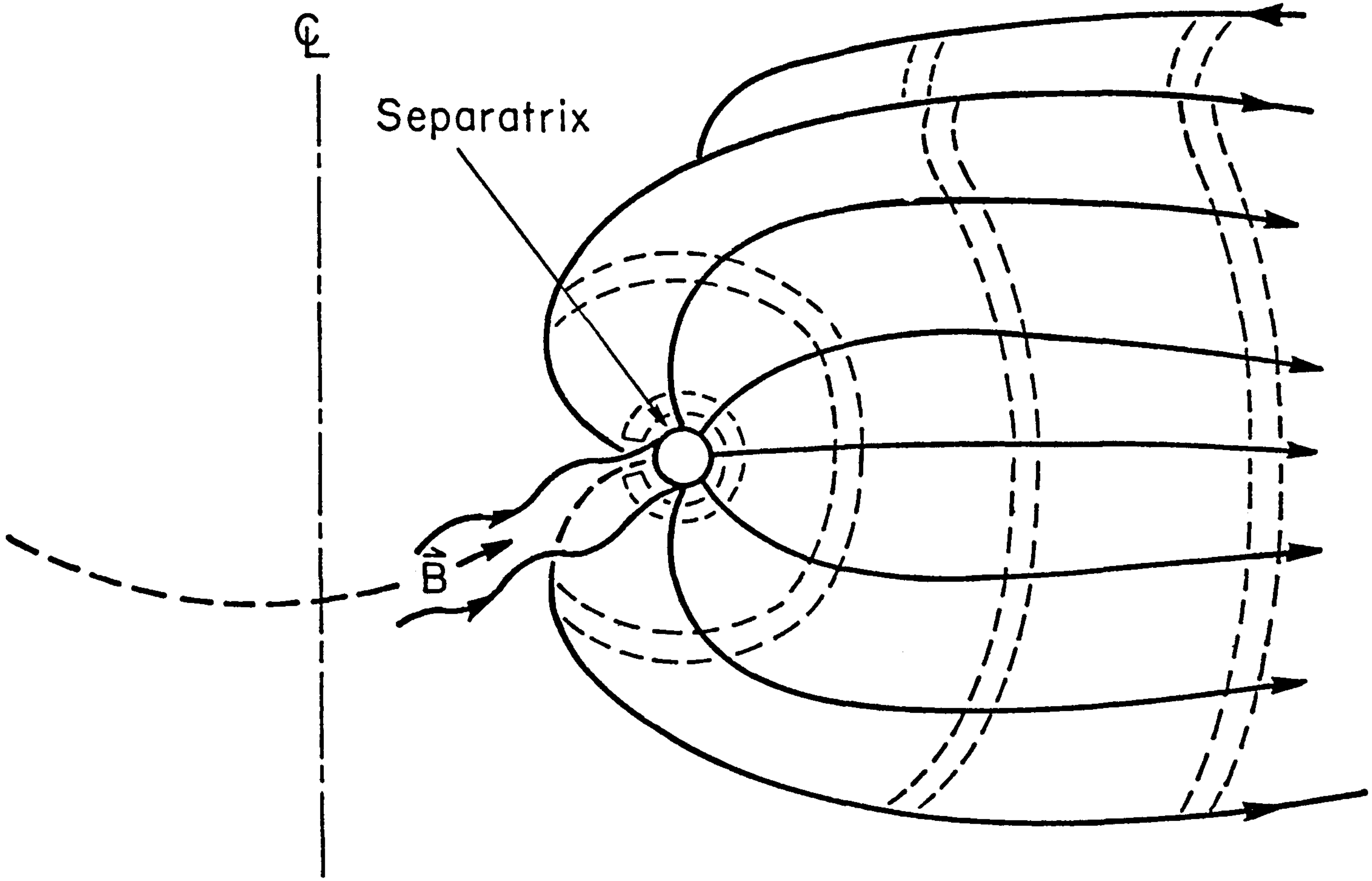
a) $\beta_{Bulk} \simeq \beta_{Annulus} \leq 0.5$

b) B_{max} $\leq$ 110 kG NbT
$\leq$ 165 kG Nb_3Sn

c) Microwave frequency $\leq$ 120 GH_z in annulus

The results for some representative reactors are shown in Table I where neoclassical scaling is assumed and it is assumed that the microwave power sustains electron energy losses in the annuli by electron diffusion and synchrotron radiation. In table I the results are compared with an advanced fuel Tokamak and a DT bumpy torus[5] (figure 4 also). A few points are worth noting. 1) The power densities of the advanced fuel bumpy tori are much greater than the advanced fuel Tokamak and can even be made greater than that of a DT bumpy torus before wall loading limits are encountered. 2) Because neoclassical scaling depends strongly on the coil radius and because thinner blankets are possible for D-^{3}He fueled reactors, one can design a much smaller D-^{3}He fueled device with a correspondingly more modest thermal power (figure 5). A preliminary survey leads to the belief that viable reactor concepts are possible down to about 500 MW_{th} with the economic optium probably occurring at a thermal power of about 1,500 MW.

*Work supported by the Electrical Power Research Institute.



Yin-Yang Coils - - - - -
Magnetic Field Lines ——

Figure 3

Toriodal-Bundle Divertor for
Electron ring Bumpy Torus Reactor

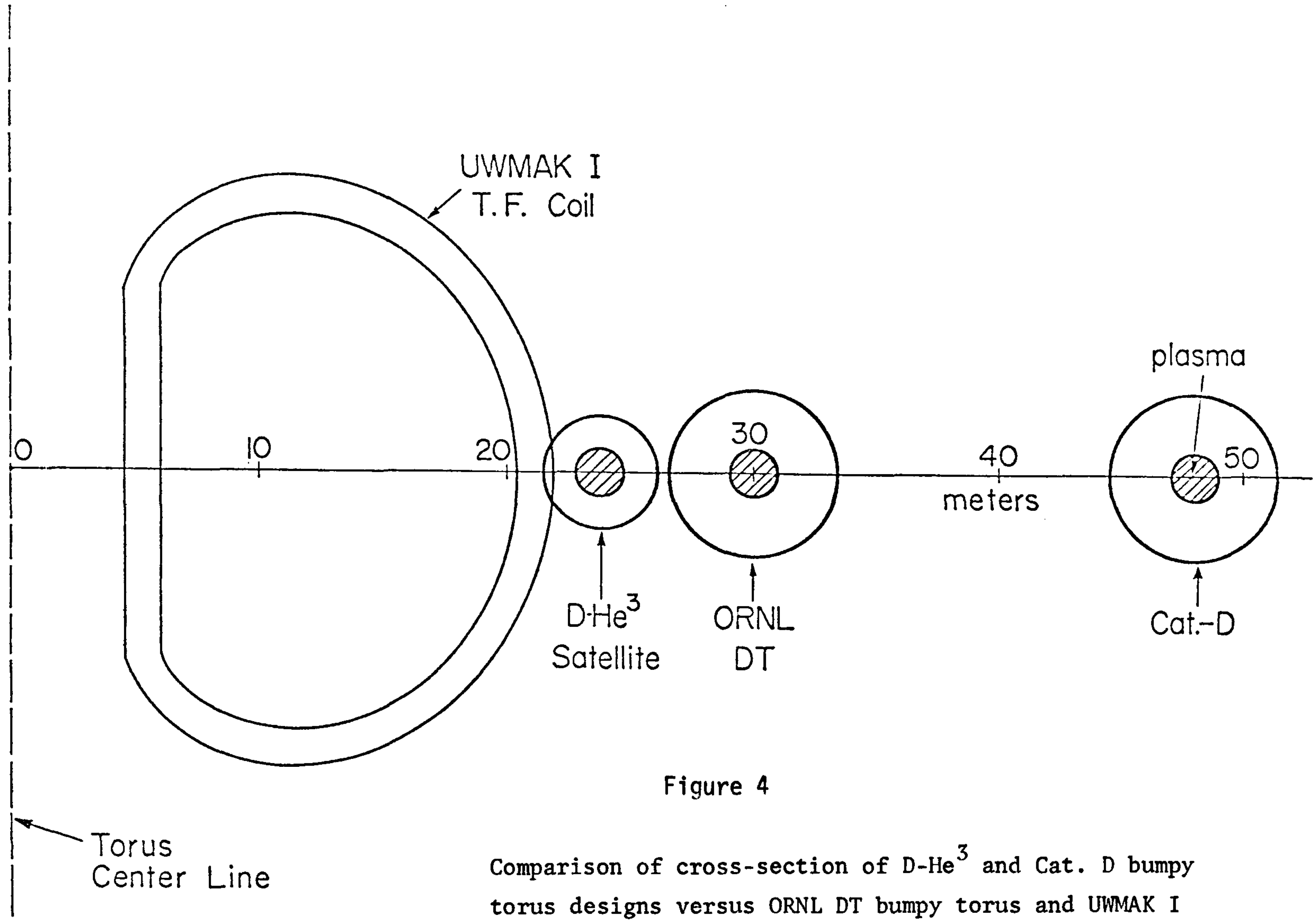


Figure 4

Comparison of cross-section of D-He^3 and Cat. D bumpy torus designs versus ORNL DT bumpy torus and UWMAK I DT tokamak.

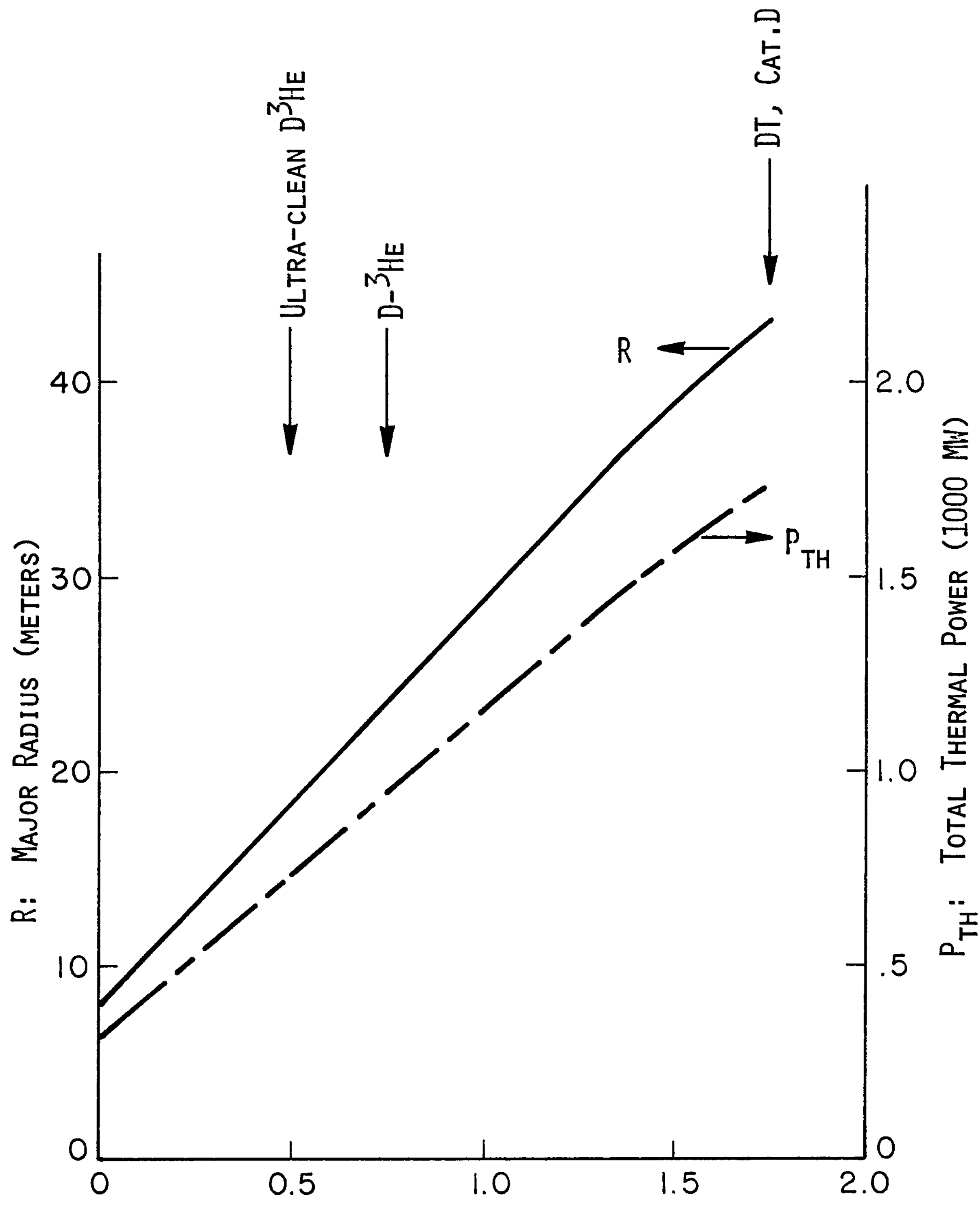


Figure 5

BLANKET THICKNESS

(METERS)

$R_M = 1.8$, $a = 1m$, $\beta = 0.5$, $T = 80$ keV, $\overline{B} = 50$ kG

Comparison of Advanced Fuel Bumpy Tori to the D-T Bumpy Torus (EBTR) and a D-^{3}He Fueled Tokamak

Table 1

	D-^{3}He Tokamak[4]	DT Bumpy Torus[5]	Cat. D Bumpy Torus	D-^{3}He Bumpy Torus
β	0.12	0.25	0.50	0.50
T_i(keV)	45	15	45	80
P_{th}(GW)	2.0	1.78	7.3	0.95
R(m)	8.2	30	55	23.6
a(m)	2.7;4.3	1.0	1.0	1.0
$n_e\tau_E$(cm^{-3}sec)	$4.2\cdot10^{15}$	$3.0\cdot10^{14}$	$3.2\cdot10^{15}$	$1.3\cdot10^{15}$
P_{wn}(MW/m^2)	0.04	1.0	1.1	0.019
P_d(MW/m^3)	1.02	3.37	6.8	2.04
B_{max}(kG)	138	73	141	110
ω_μ(annulus) (GHz)	-	55	120	83.3
^+P_e(GW)/Waste (GW)	1.05/0.95	0.800/0.98	4.0/3.3	0.62/0.35
$^+Q_e\left(\frac{\text{elect. power out}}{\text{elect. power in}}\right)$	∞	66	18.6	19.3
f_b(fractional burnups)				
D	0.079	0.11	0.083	0.055
^{3}He	0.055	-	0.12	0.048
T	0.53	0.11	0.74	0.266

$^+$ Assumes microwave tube efficiency of 0.35 of electrical power into tube. Of this microwave power, it is assumed that 1/3 is absorbed by the e-ring, so the overall microwave system efficiency is 0.12. The thermal efficiency is 0.45 and the direct conversion of the leaking particle is taken to be 70% efficient.

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Start-Up of Advanced Fuel Tokamaks

by

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ABSTRACT

The start up of advanced fuel tokamaks is analyzed to determine if there are any specific challenges related to the start-up of these tokamaks (Cat. D, D^3He) that are not found in a DT tokamak. High neutral particle power and neutral beam penetration have been envisioned as such challenges. The results indicate that using an advanced fuel enriched with tritium the plasma can first achieve D-T ignition and runaway to high temperature. Thus using this 'match head' technique and low intial density and size ignition of the advanced fuel tokamak can be achieved with about the same external neutral beam energy and power requirements as a D-T device. However unless care is taken in the refuelling sequence very high transient neutron wall loadings may result.

I. Introduction

A study of the start up of the advanced fuel tokamaks[1] is necessary to ensure there are no difficulties that would place these reactors at a severe economic penalty with respect to DT tokamak reactors of the UWMAK type. Indeed the large size, density and temperatures encountered in the steady state advanced fuel designs would make the startup of these reactors difficult and expensive especially in the area of external power 0.5-2.0GW. If neutral beam heating was contemplated, neutral beam energies on the order of an MeV would be necessary to achieve proper beam penetration and hence a large extension of current technology. However if first a low density D-T core plasma is ignited where the resulting excess of charged fusion product energy is then used to heat cold fuel to build up the plasma density and/or the plasma size these requirements can be drastically reduced becoming equivalent to those of a DT device.

While the start of both the low-β and high-β reactors involves DT thermal runaway[2] at some stage (also called the 'match head' effect), the very large size for the low-β reactor also requires the expansion of the DT core plasma while the density of the core is being increased by adding cold fuel to balance the excess energy in the alpha particles produced by the fusion reactions in the core. The expansion is accomplished by MHD limited burn propagation.[3] In this process the excess energy in the ignited core region is rapidly conducted to the periphery of the plasma, by MHD turbulence, where it heats cold fuel. In this study, the mean density is increased with minor radius while the plasma current is increased as a^2 (expansion with constant q). The expansion is such that the temperature of the ignited core remains constant. The burn propagation, as described here, is driven entirely by the D-T ignited core.

II. Start-Up Scenarios

Specific start-up scenarios for both the low and high beta Cat.-D reference reactors are described in this section. They represent a rough optimum in the trade offs between external power and neutral beam energy requirements, transient neutron wall loading and start-up time where the reduction external power requirements have been regarded the most crucial.

The low beta reactor start-up sequence is basically a six-step process:

1) Low Density Plasma formation and ohmic heating of a small mirror radius D-T core
2) Ignition of this core by 20MW of neutral beam power
3) Circular Burn Propagation
4) Transition to a non-circular shape and thermal runaway to 45keV
5) Noncircular Burn Propagation to Cat.-D n_e, T, I_p
6) Decay of excess tritium by reaction and leakage

However, the transient neutron wall loading, P_{nw}, at the end of the non-circular burn propagation phase would be too extreme ($\sim$36MW/m^2) if the fuel mixture is not changed before and during this step. To maintain the condition, $P_{nw} < 10$MW/m^2, some of the excess tritium is allowed to decay (being replaced by deuterium) initially before the burn propagation starts and about 21 seconds later when the mirror minor radius reaches 3.9 meters where another 2.0 second decay is imposed. Since this point is at a β limit no thermal runaway can occur and it is assumed that the plasma will not disrupt. An overview of this process is shown in figures 1 and 2 with a spatial view of the burn propagation and with an ($n_e\tau_E$, T_i) trajectory respectively.

The actual parameters of the start-up scenario are plotted on Figure 3. Note especially the neutron wall loadening which increases so rapidly two 'refueling' steps must be taken.

A comparison of the results of this scenario with that of 'full size start-up' is shown in Table I.

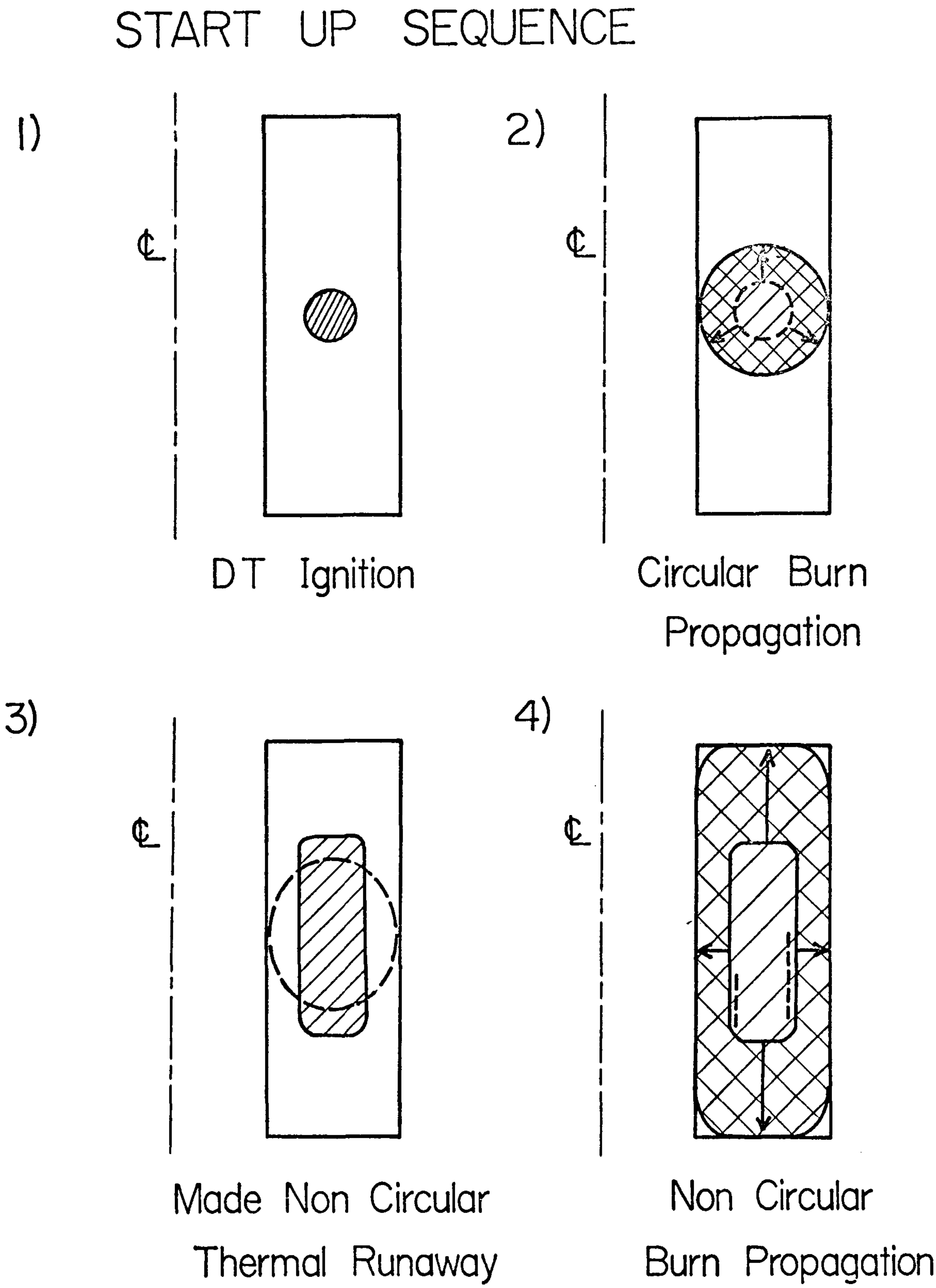
TABLE I

D-T Core Strategy (Matchhead startup) makes ignition of D-D/D-^{3}He Tokamaks feasible

	Conventional ignition	Matchhead startup
External power requirements (.05 GW) +	0.5-2.0 GW	0.05 GW
Beam Penetration Factor na (na - $0.2 \cdot 10^{17}$) + Neutral Beam Energy	$0.6\text{-}1.0 \cdot 10^{17} cm^{-2}$ $\geq$ 1 MeV	$0.2 \cdot 10^{17} cm^{-2}$ 200-300 MkeV
Transient Neutron Wall Loading	0.4-0.7 MW/m^2	10 MW/m^2
Start-Up Time	20-30 sec	70-90 sec

Conclusion: Power and injection requirements reduced to that of conventional D-T reactor

+ Values for "typical" D-T reactor



START UP SEQUENCE
1)
℄
DT Ignition
2)
℄
Circular Burn
Propagation
3)
℄
Made Non Circular
Thermal Runaway
4)
℄
Non Circular
Burn Propagation

Figure 1.

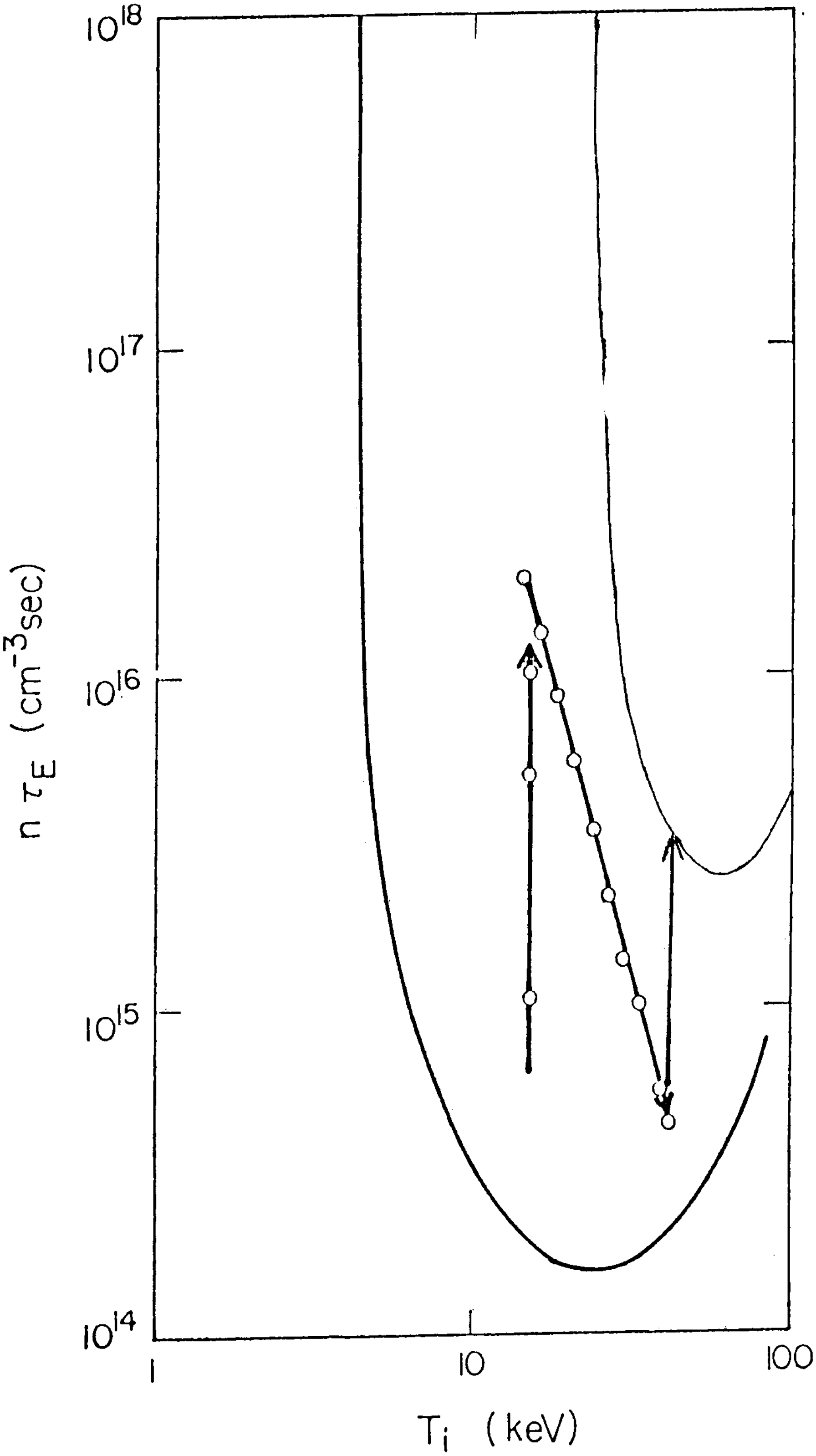


Figure 2. Low β Cat. D Start-Up Sequence

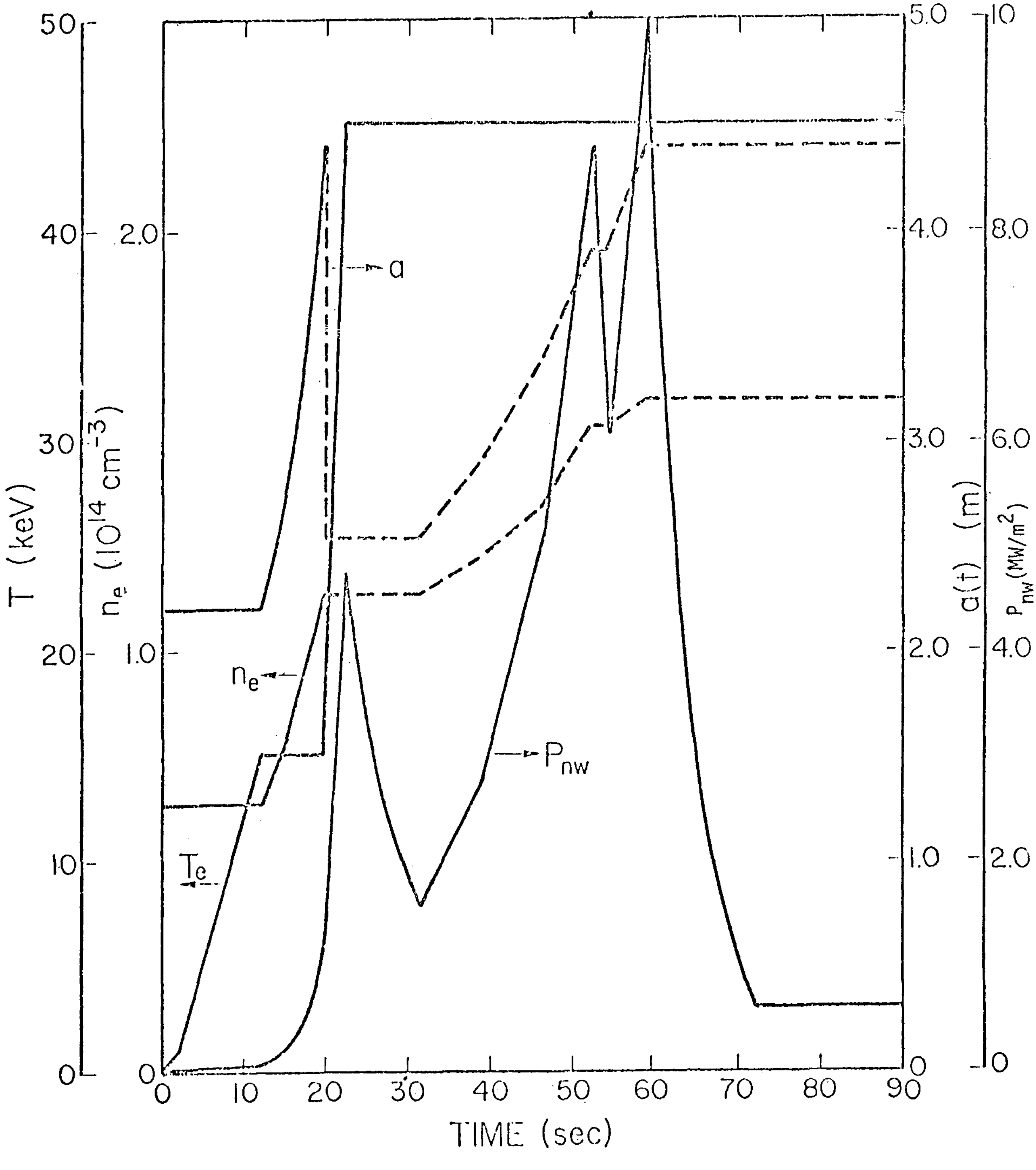


Figure 3: Start-Up of Low-β Cat.-D
REFERENCE TOKAMAK REACTOR

As can be seen by TABLE I substantial reductions in external power are possible and at roughly $1/watt this is significant. However, the blanket and 1st wall must now be designed to take a large transient thermal stress although still within the limits of feasibility.[5] The resistive volt-seconds using the ignited core start-up are also reduced from that of full size start up but still are so small with respect to the inductive volt-seconds (which depend only on the final state of the system) the much optimization is not possible here.

The high beta Cat.-D Tokamak would be relatively easy to start-up with respect to the low beta reactor. The start-up process for the former would consist of five steps:

1) Ohmic heating - breakdown phase
2) Low density ignition of tritium enriched Cat.-D fuel
3) Burn limited density build-up
4) Thermal runaway at high density
5) Decay of tritium

This process is shown is Figure 10 and TABLE I. The initial tritium percentage is held constant until the final decay step and chosen so the peak neutron wall loading was never above $10 MW/m^2$.

A comparison with the full size non-enriched start-up of the high beta reactor is shown in TABLE I.

Here again the external power requirements can be greatly reduced, but with increased transient neutron wall loading.

Thus it appears neither Cat.-D Tokamak requires any significantly greater external power requirements, neutral beam energies, or start-up times than those of D-T tokamak during start-up if the 'match head' effect is used. However, the transient neutron power for this approach can be large which may restrict the flexibility in blanket design somewhat.

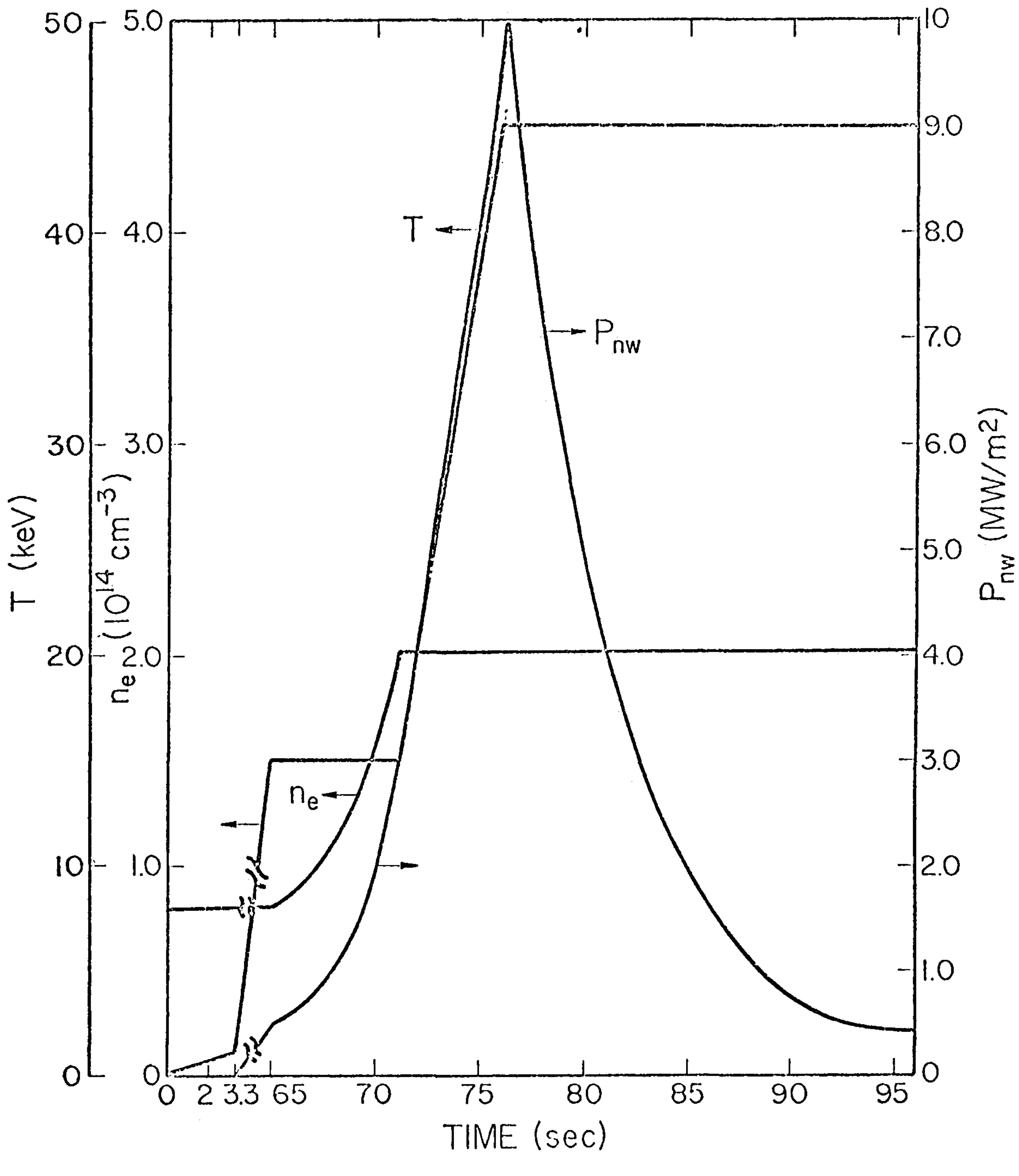


Figure 10: Start-Up High β Cat.-D

REFERENCE TOKAMAK REACTOR

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Fusion Reactors Based on $^6Li(p,\alpha)^3He$

by

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ABSTRACT

Preliminary estimates are given of the parameters required for energy breakeven and for 100 MW_e power production in a thermonuclear reactor based on the catalyzed p-^{6}Li fusion reaction. Energy yields are also calculated for this reaction in a two-component reactor, using a crude proton slowing-down formula.

I. INTRODUCTION

Alternates to the d-t-^{6}Li cycle for fusion reactors deserve careful attention, in view of the manifold problems associated with neutron activation, first wall damage, and tritium inventory management. Reactions involving nuclei of lithium, berillium, and boron have received but sporadic attention in the past since they require higher operating temperature and Lawson number $n\tau$ than does d-t. But the bonus they offer, of the near absence of neutrons and tritium, with most of their reaction energy residing in charged particles, has of late attracted considerable attention. Reactors fueled with light metal ions may also offer methods of plasma production and confinement not requiring the large ubiquitous external magnetic fields common in tokomak designs, thereby reducing synchrotron radiation losses and capital costs. This paper will summarize our preliminary studies of some of the possibilities using these advanced fuels, particularly lithium-6. A recent review by McNally[1] is a valuable overview of the problem.

II. CROSS SECTIONS AND REACTION RATES

The reaction of primary interest in our studies is ^{6}Li(p,α)^{3}He, i.e.

$$p + {}^6\mathrm{Li} \rightarrow {}^3\mathrm{He} + {}^4\mathrm{He} + 4.02\ \mathrm{MeV}.$$

Spinka, Tombrello, and Winkler[2] have compiled the available total cross section data, and fitted it to a modified Gamow formula

$$\sigma(E) = 3.0\ E^{-1} \exp\left[-2.758\ E^{-1/2} - 0.44\ E\right] \text{barns} \qquad (1)$$

where E is the center-of-mass energy in MeV. When the reactants have a Maxwellian distribution at a temperature T, the reaction rate is given by[3]

$$\langle\sigma v\rangle = \frac{8\pi}{M^{1/2}}\left(\frac{1}{2\pi kT}\right)^{3/2} \int_0^\infty dE E\sigma(E)\exp(-E/kT) \quad . \qquad (2)$$

Inserting (1) into (2) yields

$$\langle\sigma v\rangle = 5.07 \times 10^{-15}\, T^{-3/2}\, I(T)\ \mathrm{cm}^3\ \mathrm{sec}^{-1} \tag{3}$$

where T is the plasma ion temperature in MeV, and

$$I(T) = \int_0^\infty dE \exp\left[-\alpha E - \beta E^{-1/2}\right] \tag{4}$$

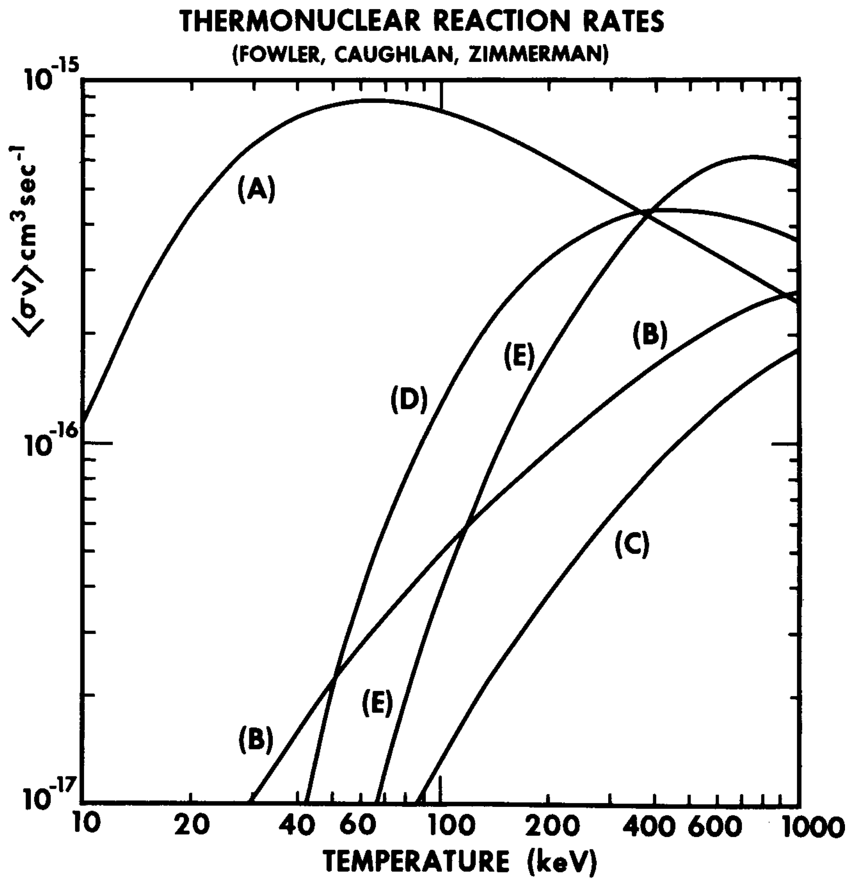
with $\alpha = 0.44 + T^{-1}$ and $\beta = 2.758$. This integral can be approximated as a saddle-point to be $I(T) \simeq 2.78\alpha^{-5/6} \exp(-3.717\alpha^{1/3})$. Values of $\langle\sigma v\rangle$ thus computed are shown in Table I, as a function of T, from 0.1 to 2.0 MeV.

T(keV)	$\langle\sigma v\rangle$ cm^3 sec^{-1} $\times 10^{17}$	
	This Work	Ref. 4
100	1.53	1.33
200	3.60	3.84
300	7.20	6.36
400	9.08	8.76
500	10.5	11.0
600	11.6	12.9
700	12.1	14.5
800	12.6	16.0
1000	13.0	18.2
2000	13.1	-

Table I

For comparison, we also show in Table I the values of $\langle\sigma v\rangle$ given in the compilation of August 17, 1976, prepared by McNally and Sharp.[4] Many of these values are drawn from the work of Fowler, et al.[5] In view of the reasonable agreement between the two sets of values, considering the approximation used in obtaining ours, we shall henceforth use the values from Ref. 4. Fig. 1 shows this reaction rate together with several others.

A companion reaction of great importance is ^{6}Li(^{3}He,p)2α or



(A)- d (t,n) ^4He, 17.6MeV

(B)- { d (d,n) ^3He, 3.3MeV; d (d,p) t, 4.0MeV }

(C)- ^6Li (p,^3He)^4He, 4.02MeV

(D)- { ^9Be (p,d) 2 ^4He, 0.65MeV; ^9Be (p,^4He) ^6Li, 2.13MeV }

(E)- ^{11}B (p,^4He) 2 ^4He, 8.68MeV

Figure 1.

$^{3}He + {}^{6}Li \rightarrow {}^{4}He + {}^{4}He + p + 16.8$ MeV, since its energy yield is so large, and since it rejuvenates the energetic proton required for the primary reaction. The proton acts as a catalyst, and the combination $2^{6}Li \rightarrow 3^{4}He + 20.8$ MeV is called the catalyzed lithium-6 reaction. While the cross section for this reaction is not as well known as that for $^{6}Li(p,\alpha)^{3}He$, it would appear to be high enough in the energy range above 1 MeV for (3) to govern the overall catalyzed reaction, provided the energetic protons can be confined in the reactor. We shall assume this to be so in our subsequent discussions. Thus, even though $<\sigma v>$ for $^{6}Li(p,\alpha)^{3}He$ is the lowest of any of those in Fig. 1 in the energy range below a temperature of 1000 keV, its reaction yield as a catalyzed burn is the highest. As we shall show these factors combine to make the overall lithium-6 cycle rather competitive with other choices.

III. ENERGY BALANCE

We can use the above $<\sigma v>$ values to determine Lawson numbers $n\tau$ for breakeven. Our definition of breakeven is somewhat different from that originally introduced by Lawson,[6] in that we assume that energy can be recovered by direct conversion of the charged particle energy released in the fusion reaction, and by conversion in a thermal cycle of both the plasma heat and the bremsstrahlung. The plasma is taken for simplicity to consist of electrons, of density n_e, and of two species of ions of density n_1 and n_2, and of charge Z_1 and Z_2. Since we fail to include reaction products, the results are limited to low burnup fractions. Thus

$$\begin{aligned} \text{plasma heat} &= \frac{3}{2} \sum neT = \frac{3}{2}\, eT(n_e + n_1 + n_2) \quad \text{joules} \\ &= \frac{3}{2}\, eTn_1(Z_1 + \alpha Z_2 + 1 + \alpha) \equiv An_1 \quad . \end{aligned} \tag{5}$$

The ions and electrons have the same temperature T (volts), and $n_2/n_1 = \alpha$.

The bremsstrahlung loss is[7]

$$\text{brems} = 1.60 \times 10^{-32}\ T^{1/2}(Z_1+\alpha Z_2)(Z_1^2+\alpha Z_2^2)(n_1^2\tau) \text{ joules}$$
$$\equiv B\ n_1^2\ \tau. \tag{6}$$

The thermonuclear energy released is

$$E_{tn} = n_1 n_2 \langle\sigma v\rangle \tau e Q \text{ joules}$$
$$= \alpha\langle\sigma v\rangle eQ(n_1^2\tau) \equiv C(n_1^2\tau) \tag{7}$$

where Q is the energy yield of the fusion reaction in volts.

We assume that the thermonuclear energy can be recovered at an efficiency η_{dc} by direct conversion, and that the plasma heat <u>and</u> the bremsstrahlung can both be converted in a thermal cycle with an efficiency η_{th}. The energy balance condition then is

$$An_1 + Bn_1^2\tau = \eta_{th}(An_1 + Bn_1^2\tau) + \eta_{dc}Cn_1^2\tau$$

or

$$n_1\tau = \frac{A(1-\eta_{th})}{\eta_{dc}C - (1-\eta_{th})B} \quad . \tag{8}$$

We see that breakeven is impossible unless

$$\left(\frac{\eta_{dc}}{1-\eta_{th}}\right)C > B \quad . \tag{9}$$

This latter condition translates into

$$10^{13}\ \frac{\langle\sigma v\rangle Q}{T^{1/2}} > \left(\frac{1-\eta_{th}}{\eta_{dc}}\right)\frac{(Z_1+\alpha Z_2)(Z_1^2+\alpha Z_2^2)}{\alpha} \tag{10}$$

so that an optimum value of $\alpha = \alpha^* = (Z_1/Z_2)^{3/2}$ exists for each combination (Z_1,Z_2) which minimizes the right-hand side. For Li, Be, and B, respectively, the values of α^* are 0.192, 0.125, and 0.0895, for reactions with protons $Z_1 = 1$.

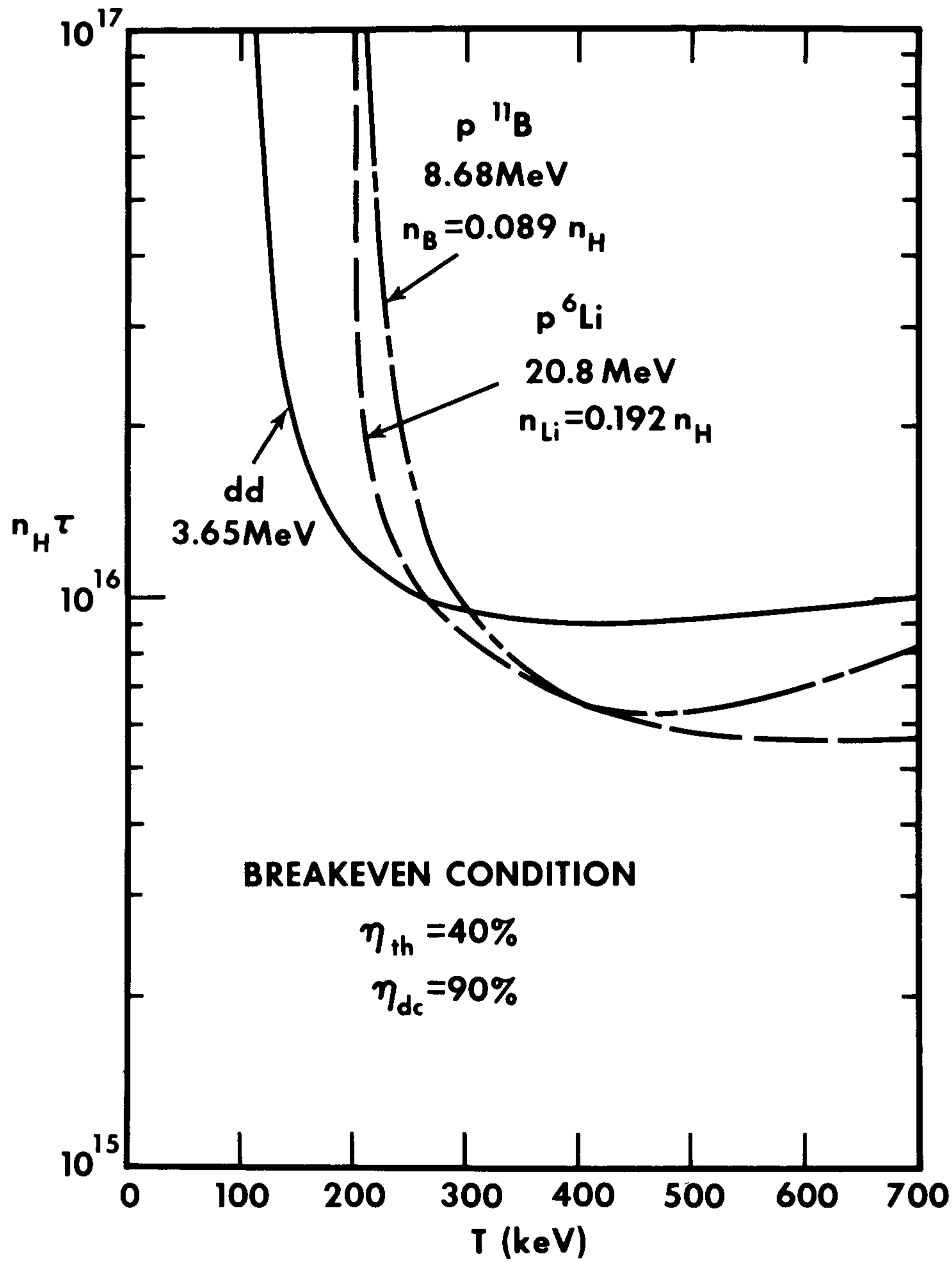

10^{17}
$p\ ^{11}B$
8.68MeV
$n_B = 0.089\ n_H$
$p\ ^{6}Li$
20.8 MeV
$n_{Li} = 0.192\ n_H$
dd
3.65MeV
$n_H \tau$
10^{16}
BREAKEVEN CONDITION
$\eta_{th} = 40\%$
$\eta_{dc} = 90\%$
10^{15}
0 100 200 300 400 500 600 700
T (keV)

Figure 2.

Figure 2 shows the results of applying this procedure to the catalyzed p-^{6}Li reaction (yield 20.8 MeV) as well as to the reactions $p + {}^{11}B \rightarrow 3\alpha + 8.68$ MeV and $d + d \rightarrow t + p + 4.0$ MeV, $n + {}^3He + 3.3$ MeV using reaction rates from Fig. 1. For these examples we have taken $\eta_{dc} = 0.90$ and $\eta_{th} = 0.40$. For p-^{6}Li uncatalyzed (i.e. yield of only 4.02 MeV) or for the reactions involving beryllium from Fig. 1 we found that breakeven was unachievable for the efficiency values chosen. Examination of Fig. 2 shows that the three reactions shown all require $n_H\tau$ values of about 10^{16} cm^{-3} sec, with temperatures of 260 keV required for d-d and p^{6}Li (cat), and 300 keV for p^{11}B. Recent improvements in cross section measurements for p^{11}B reported by Tombrello suggest that the situation may be even less favorable than indicated here, since the reaction rate we used is about a factor of two too high. Operation of d-d in a catalyzed cycle lower both the required ion temperature and the required $n_H\tau$ value. One would also find lower values for the breakeven parameters if the electron temperature were chosen to be a fraction of the ion temperature (we have taken equal temperatures), since the bremsstrahlung depends upon electron temperature.

IV. SAMPLE REACTOR PARAMETERS

Here we provide some numerical values of parameters for the plasma of a 100 MW$_e$ reactor based on the catalyzed p-^{6}Li reaction. Ideal ion confinement is assumed and plasma losses other than bremsstrahlung are ignored. The net electrical peak power generated is

$$P_{peak} = \frac{\mathrm{vol}}{\tau}\left[\eta_{dc}\, E_{tn} - (1-\eta_{th})(\text{plasma heat + brems})\right] \qquad (11)$$

and a duty cycle is adjusted to yield an average electrical output of 100 MW. Table II lists the parameters, for plasma temperatures (electrons and ions equal) of 300 and 600 keV. Plasma volume was taken to be one liter.

T	300 keV	600 keV
$n_H\tau$	$10^{17}cm^{-3}sec$	$10^{16}cm^{-3}sec$
n_H	$10^{18}cm^{-3}$	$10^{18}cm^{-3}$
τ	100 msec	10 msec
n_{Li}	$2\times10^{17}cm^{-3}$	$2\times10^{17}cm^{-3}$
volume	1 liter	1 liter
peak elec power out	13,000 MW	19,000 MW
duty cycle	7.8×10^{-3}	5.3×10^{-3}
period	13 sec	2 sec
average elec power out	100 MW	100 MW

Table II. p^6Li Thermonuclear Reactor Sample Parameters

V. TWO COMPONENT REACTOR

In this approach to the conceptualization of a fusion reactor,[8,9,10] an energetic neutral beam is injected into a target plasma where it is ionized and slows down by Coulomb drag on the target ions and electrons. During the slowing down there is a finite probability p that a fusion reaction will occur. One defines an "F-value"

$$F = p \frac{Q}{E_o} \tag{12}$$

where E_o is the initial projectile energy, and Q the yield of the fusion reaction. The F-value required for energy breakeven depends upon details of the reactor cycle, but values larger than unity are usually sought. For the d-t reaction, values of about 4 were found[8] in the initial calculation with an initial deuteron energy of about 300 keV, incident on a cold triton-electron plasma. Our calculation is the most optimistic one, assuming perfect confinement for target, projectile, and reaction product ions. It further assumes $T_e = \infty$ and $T_i = 0$ for the target plasma so as to maximize the projectile slowing-down time.

We use the slowing-down rate[11]

$$\frac{dE_1}{dt} = -\sum_n \frac{4\pi Z_1^2 Z_n^2 e^4 \ln\Lambda}{v_1 M_n} \; n_H H(v_1/v_n) \tag{13}$$

where $E_1 = \frac{1}{2} M_1 v_1{}^2$ is the kinetic energy of the projectile, n_n, v_n, Z_n, and M_n are the number density, thermal velocity, ionic charge and reduced mass of the target specie, the summation is over plasma species (electrons and ions) and

$$H(x) = 2\pi^{-1/2}\left[\int_0^x d\xi e^{-\xi^2} - \left(1 + \frac{M_n}{M_1}\right) x e^{-x^2}\right] \quad . \tag{14}$$

The probability for occurrence of a fusion reaction is

$$p = n_{n'} \int_0^t dt \sigma v_1(t) \tag{15}$$

where $n_{n'}$ is the number density of target fusile ions. Substituting (13) into (15) gives

$$p = \frac{n_{n'}}{2\pi e^4 \ln\Lambda M_1 Z_1^2} \int_{\frac{3T}{2}}^{E_o} \frac{dE_1 E_1 \sigma(E_1)}{\sum_n n_n Z_n^2 M_n^{-1} H(v_1/v_n)} \quad . \tag{16}$$

For $T_e \to \infty$, $T_{Li} \to 0$, $H(v_1/v_n) \to 1$, and for $M_2/M_1 Z_1^2 Z_2^2 = 2/3$ and $\sigma(E_1)$ given by (1) we find

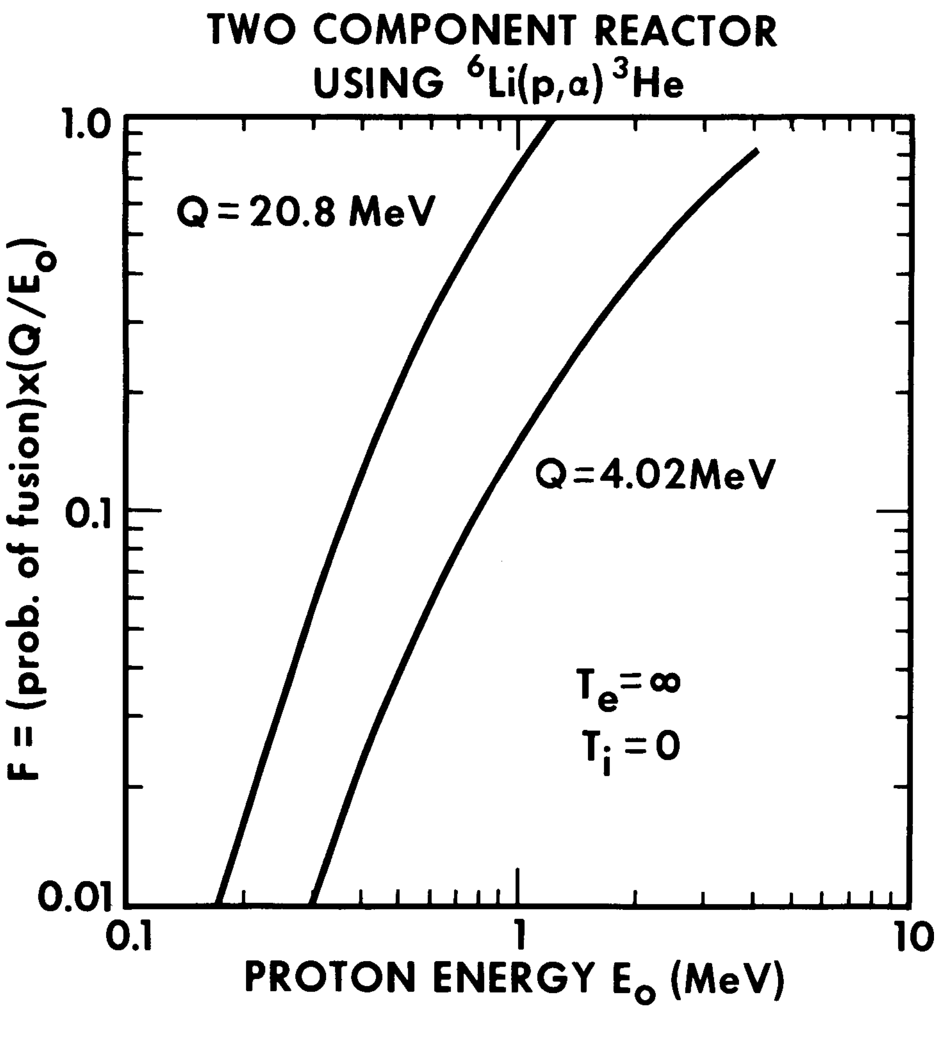

TWO COMPONENT REACTOR
USING $^{6}Li(p,\alpha)^{3}He$
1.0
0.1
0.01
0.1
1
10
Q = 20.8 MeV
Q = 4.02MeV
$T_e = \infty$
$T_i = 0$
F = (prob. of fusion)x(Q/E_o)
PROTON ENERGY E_o (MeV)

Figure 3.

$$p = 15.4(\ell n\Lambda)^{-1} \int_0^{E_0} dE_1 \exp\left[-2.758E_1^{-1/2} - 0.44\, E_1\right]$$

where E_1 and E_0 are in MeV.

Over the energy range of interest, the $E_1^{-1/2}$ term in the exponent of (17) dominates. Using the substitution $u = 2.758\ E^{-1/2}$ transforms the integral in (17) to

$$2(2.758)^2 \int_{u_0}^{\infty} du u^{-3} e^{-u} = (2.758)^2 \left[-Ei(-u_0) + e^{-u_0}\,(u_0^{-2} - u_0^{-1})\right] \quad (18)$$

where $u_0 = 2.758\ E_0^{-1/2}$. We have taken $\ell n\Lambda = 10$ and show, in Fig. 3, the F-values calculated for both the primary (Q = 4.02 MeV), and catalyzed (Q = 20.8 MeV) p-^{6}Li reactions. As can be seen, the primary reaction alone never achieves a unity F-value, while the catalyzed reaction requires a proton energy of greater than about 1 MeV for a unity F-value.

VI. ACKNOWLEDGMENT

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Optimization of Plasma Profiles for Ignited Low-Beta Toroidal Plasmas Utilizing "Advanced Fuels"

by

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ABSTRACT

The radial density and temperature profiles of ignited plasmas utilizing non-DT fuels can be optimized to maximize fusion power density and minimize required $\bar{n}\tau_E$ under the constraint of a maximum average plasma pressure. Strong axial temperature peaking <u>or</u> strong density peaking is advantageous, according to whether the fusion reactivity increases faster or more slowly than quadratically with temperature, respectively. For tokamak plasmas with maximum beta limited to 10% or less by MHD instability, optimal profile tailoring allows ignition of catalyzed D-D or D-^{3}He fuels in devices of reasonable size ($R_0 \lesssim 8$ m), and with first-wall power loadings exceeding 1 MW/m^2.

I. INTRODUCTION

The plasma conditions for equilibrium ignition are markedly more severe for "advanced fusion fuels" than for deuterium-tritium.[1,2] Recent studies using a global (i.e., zero-dimensional) analysis[3] indicate that utilization of advanced fuels in tokamak plasmas results in economically interesting reactors only when the plasma beta (β = plasma pressure/magnetic field pressure) is of the order of several tens of per cent, a value which is forbidden by MHD stability considerations.[4]

It is known that the $n\tau_E$ requirement for ignition of a D-T plasma can be reduced markedly from that of a uniform plasma, if the density (n) and temperature (T) profiles have strong axial peaking.[5] This result follows from the fact that the fusion reaction rate is proportional to $n^2\langle\sigma v\rangle$, with $\langle\sigma v\rangle$ increasing faster than T, so that $\int n^2\langle\sigma v\rangle dV$ is increased substantially even for the same average value of nT. A similar result can be shown for TCT-type plasmas with axially peaked temperature and beam-deposition profiles.[6] Now the beta limitation in tokamak plasmas, which is determined by MHD instability, refers only to the average plasma pressure, whereas the beta in a local region such as the plasma center can be many times larger.[4] Thus by profile tailoring, one can set up a small high-beta plasma region which produces most of the fusion power; this productive region is surrounded by a plasma of much lower beta, so that the volume-averaged plasma pressure remains within the MHD limit.

In this study, we examine the effects of profile shaping on the requirements for obtaining ignited plasmas with non-DT fusion fuels. We find that in general it is important to decouple the radial dependences of n and T. By strong temperature peaking and relatively weak density peaking — as in experimental tokamak operation — it is possible to achieve ignition in D-D and D-^{3}He plasmas of reasonable size, with realistic beta-values.

II. PROFILE OPTIMIZATION

1. Fusion Power Density. Consider the two simple plasma pressure profiles shown in Fig. 1. Each plasma has the same average pressure p_0, and the same average beta, $\bar{\beta} = \bar{p}/B^2$, provided

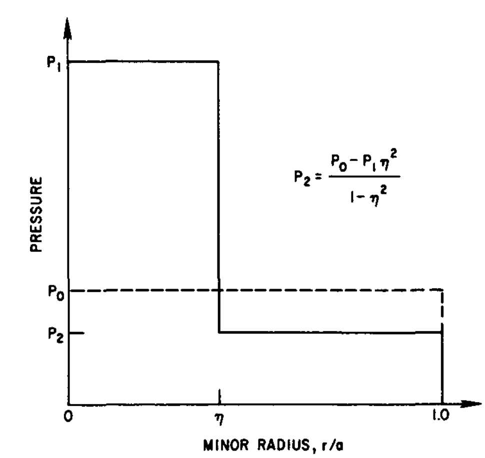

Figure 1. Two plasma pressure profiles with the same spatially-averaged pressure. (773476)

that $$\eta^2 p_1 + (1-\eta^2)p_2 = p_0 \tag{1}$$

where $\eta < 1$. As indicated in Fig. 2, the fusion reactivities in the temperature range of interest can be fitted reasonably well by simple formulas of the type $\langle\sigma v\rangle \propto T^s$. (See also Table 1.) Using this relation and $p = 2nT$, the ratio of the spatially-averaged fusion power densities for the two pressure profiles of Fig. 1 is

$$R_f = \eta^2 \left(\frac{p_1}{p_0}\right)^2 \left(\frac{T_1}{T_0}\right)^{s-2} + (1-\eta^2)\left(\frac{p_2}{p_0}\right)^2 \left(\frac{T_2}{T_0}\right)^{s-2} . \tag{2}$$

The maximum value of R_f occurs when $p_1/p_0 = \eta^{-2}$ and $p_2 = 0$; the outer region is now a cold dense plasma which still must carry a large fraction of the plasma current. Then $R_{fmax} = \eta^{-2}(T_1/T_0)^{s-2}$. For example, if $\eta = 1/2$ and $s = 2$, then the average fusion power density is increased by a factor of 4, when the plasma pressure is quadrupled at $r/a < 1/2$, and made nearly zero at $r/a > 1/2$ — for the same average pressure as in the uniform case.

Now if $s > 2$, it is clear from Eq. (2) that the increase in pressure $[\Delta p \propto \Delta(nT)]$ is obtained most favorably by increasing T_1 rather than n_1. Thus if $\Delta T_1 \propto p_1$ with $n_1 = n_2$, we have $R_{fmax} = \eta^{2-2s}$. Table 1 shows the maximum possible gains in average fusion power density, for several important reactions.

On the other hand, if $s < 2$, it is evidently more advantageous to increase n_1 rather than T_1, in order to produce the desired pressure increase. If $T_1 = T_0$, then R_{fmax} has the same value as for the $s = 2$ case. Of the more practical fusion reactions, only D-D has $s < 2$, and here s is sufficiently close to 2 so that temperature peaking does not lead to a marked reduction from the maximum possible R_f.

The advantage of temperature peaking with a uniform density profile is especially marked for reactions with $s \gg 1$, even when

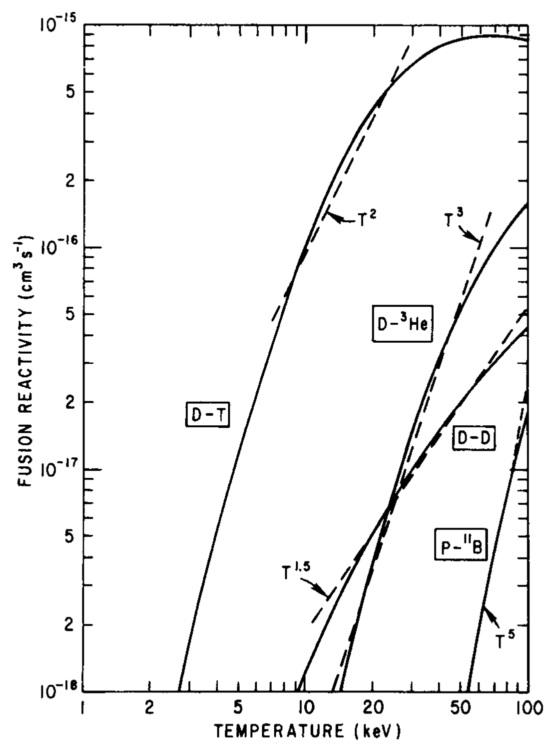

Figure 2. Solid lines are fusion reactivities for Maxwellian ion velocity distributions. Dashed lines are linear fits (on the log-log scale). (773473)

Table 1. Effect of Spatial Peaking on Power Densities

Reaction	Temperature Range (keV)	Temperature Dependence of $<\sigma v>$	Fusion Power Density	Maximum Increase In Average Power Density* $\eta = 1/2$	$\eta = 1/3$
D-D	15-70	$T^{3/2}$	$\beta^2 T^{-1/2}$	4	9
D-T	8-25	T^2	$\beta^2 T^0$	4	9
D-^{3}He	15-60	T^3	$\beta^2 T$	16	81
p-^{11}B	55-95	T^5	$\beta^2 T^3$	256	6561

*Two-level pressure model. See Fig. 1.

$p_1/p_0 < \eta^{-2}$. For example, consider a p-^{11}B plasma (s = 5) with $\eta = 1/2$ and $p_1/p_0 = 2$. The increase in pressure at $r/a < 1/2$ is best obtained by doubling T; to keep $\bar{p}$ the same, T is reduced by 1/3 at $r/a > 1/2$, with $n(r)$ = constant. Then $R_f = 8.0$, even though the plasma at $r/a > 1/2$ contributes essentially no fusion power. However, the outer region does serve to enhance energy confinement, as well as to reduce the spatially-averaged pressure.

2. Confinement Parameter. Defining the energy confinement time τ_E' to include all plasma transport and radiative losses, the overall plasma power balance at equilibrium ignition is

$$\frac{\int 3nTdV}{\tau_E'} = \bar{P}_{f+} \times \text{(plasma volume)} \tag{3}$$

where $\bar{P}_{f+}$ is the average fusion power density including only charge fusion-reaction products, and we have assumed $T_e = T_i$. Thus

$$\bar{n}\tau_E' = \frac{3}{2}\,\bar{n}\,\frac{\bar{p}}{\bar{P}_{f+}} \propto \frac{\bar{n}}{R_f} \tag{4}$$

where R_f is given by Eq. (2). (The last operation assumes that one operates a given plasma at the maximum possible $\bar{p}$.) If $\bar{n}$ does not change markedly when going to peak temperature profiles, as is the usual case, then the required $\bar{n}\tau_E'$ is inversely proportional to R_f, whose maximum value is shown in Table 1.

III. NUMERICAL RESULTS

1. Profiles. The conclusions of the previous section are qualitatively valid for the smoothly varying pressure profiles that are encountered in practice. This section presents the results of numerical evaluations of fusion

power densities for D-D and D-^{3}He plasmas, as a function of realistic spatial profiles. Our assumptions are as follows:

(a) $T_i(r) = T_e(r) = T(r) = T_c(1-r^2/a^2)^x$.

(b) All ion species in a given plasma have the same density profile as the electrons: $n(r) = n_c(1-r^2/a^2)^y$.

(c) All tritium and helium-3 reaction products are burned up at the same rate as produced (catalyzed D-D operation[2]). The tritium concentration is neglected. (It's actually $\lesssim$ 0.5%.) The equilibrium ^{3}He concentration is calculated for each temperature profile.

(d) The concentrations of ^{4}He and H reaction products are neglected.

(e) Synchrotron radiation loss is neglected.

Figure 3 shows radial profiles of n, T and P_f for x = 3 and y = 1. Evidently, only a small fraction of the plasma volume contributes to the fusion power production — although the entire plasma contributes to energy confinement, if $\bar{n}\tau_E \propto \bar{n}a^2$.

2. Ignition Criteria. The equilibrium ignition condition is

$$\bar{n}_e\tau_E = \frac{\bar{n}_e \frac{3}{2}\overline{nT}}{\bar{P}_{f+} - bZ_{eff}\overline{n_e^2T_e^{1/2}}} \tag{5}$$

Here the "bar" over a symbol represents the spatially averaged value; $n = n_e + n_i$, where n_i is the total ion density; and the bremsstrahlung constant $b = 3.65\times10^{-15}$, with T_e in keV and n_e in cm^{-3}. It is most useful to plot $\bar{n}_e\tau_E$ versus the density-averaged temperature, <T>, where

$$\langle T\rangle = \frac{\int_0^a n(r)T(r)2\pi r dr}{\int_0^a n(r)2\pi r dr} = \frac{\bar{p}}{\bar{n}} \tag{6}$$

Thus plasmas of the same $\bar{n}$ and <T> have the same beta.

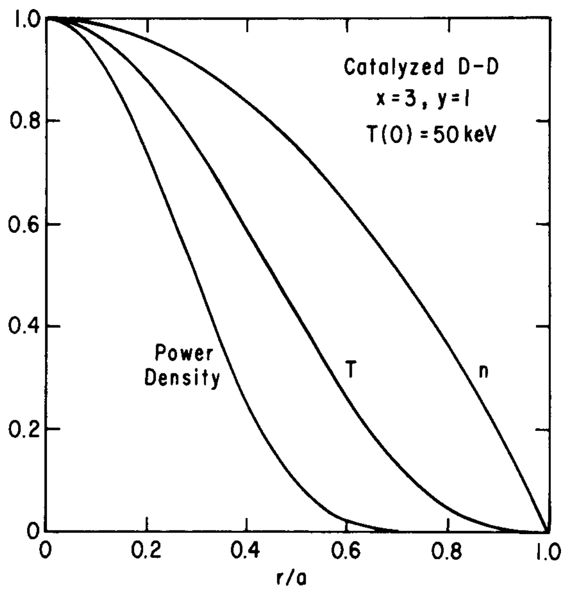

Figure 3. Profiles of plasma density (n), temperature (T), and fusion power density for x = 3 and y = 1. (773491)

Figures 4 and 5 show the ignition criteria calculated numerically from Eq. (5) with the fusion reactivities taken from Ref. 7. The required $\bar{n}\tau_E$ for D-D when x = 3 is reduced by a factor of 2.5 to 10 from the value for a uniform plasma. This reduction corresponds to a rapid increase in fusion power density, and is in the range predicted in Table 1. Furthermore, ignition can now be obtained with <T> as low as 16 keV. (Analogous results were obtained previously[5] for D-T).

Figure 5 shows that factor of 10 reductions in $\bar{n}\tau_E$ can be obtained for x = 3 in a 1:1 mixture of D-^{3}He (Z_{eff} = 1.67). The gain in power density is not as dramatic as indicated in Table 1, because D-D reactions are still important, especially at smaller <T>. Whereas the catalyzed-D plasma contains an equilibrium concentration of 10 to 20% ^{3}He, and 37% of the fusion power is produced in fast neutrons, for the 1:1 D-^{3}He mixture only 6 to 10% of the fusion power is produced in fast neutrons, depending on <T>.

3. Power Density. Figure 6 shows the spatially-averaged fusion power density (including neutron production) in catalyzed D-D plasmas with a fixed

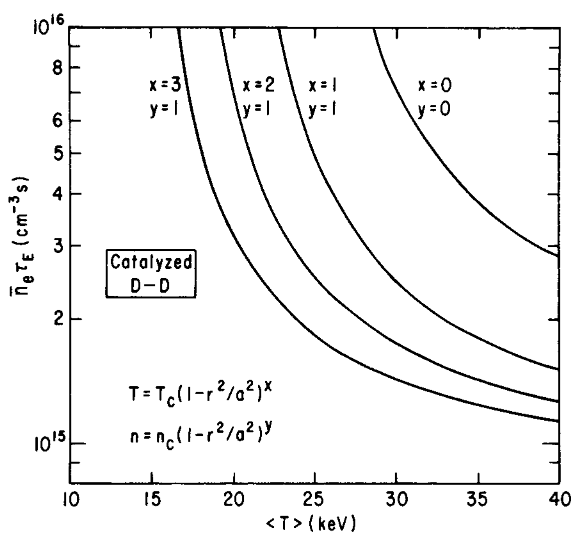

Figure 4. Ignition criteria for catalyzed D-D fuel, for various temperature and density profiles, with $T_e(r) = T_i(r)$. The ^{3}H and ^{3}He are burned up at the same rate as produced.

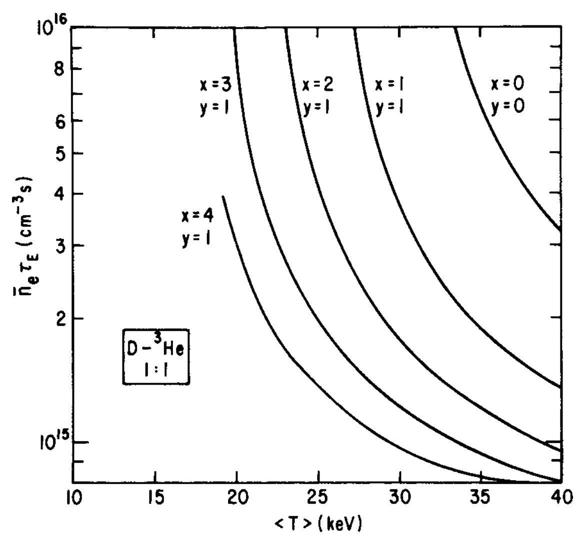

Figure 5. Ignition criteria for D-^{3}He fuel, for various temperature and density profiles, with $T_e(r) = T_i(r)$. Equal concentrations of D and ^{3}He. (773473)

total pressure of 2.1 J/cm^3, corresponding to $\bar{\beta}$ = 8% at B_t = 8.0 T. (The pressure of decelerating charged fusion-reaction products[8] increases the total pressure and $\bar{\beta}$ by 20 to 25%.) Whilst $\bar{P}_f$ increases as the temperature profile narrows, the gain in $\bar{P}_f$ at a given <T> is not as strong as the reduction in $\bar{n}\tau_E$; as can be seen from Eq. (5), the different dependence is due to the bremsstrahlung radiation. Evidently, power densities exceeding 1 W/cm^3 are achievable when $x \gtrsim 2$. The corresponding first-wall power loading would exceed 1 MW/m^2 in tokamaks with plasma half-widths of 2 m or greater.

4. Proton-Boron Reaction. We have found it impossible to ignite a p-^{11}B plasma with uniform density, but using temperature peaking as large as x = 7. (Synchrotron radiation loss was arbitrarily set equal to the bremsstrahlung loss.) For T_i > 100 keV, however, the temperature dependence of $<\sigma v>$ is much less steep than T^5 (see Fig. 2), so that density peaking becomes increasingly preferred. Although ignition of p-^{11}B may always be unattainable,[9] the advantages of spatial peaking would be retained in a beam-driven mode of operation.[6]

IV. MINIMUM SIZE ADVANCED-FUEL TOKAMAK REACTORS

1. Restrictions. The results of Section III have been used to determine the minimum size of D-D or D-^{3}He reactors that produce a first-wall power loading ϕ_W > 1 MW/m^2. The following restrictions were applied:
(a) The maximum $\bar{\beta}$ determined by stability against MHD "ballooning" modes varies nearly inversely with plasma aspect ratio.[10] At R/a = 3, $\bar{\beta} \lesssim 10\%$.
(b) The degree of temperature peaking is limited to $x \lesssim 3$ by the maximum current-density peaking allowed for kink-mode stability, as well as for attaining the

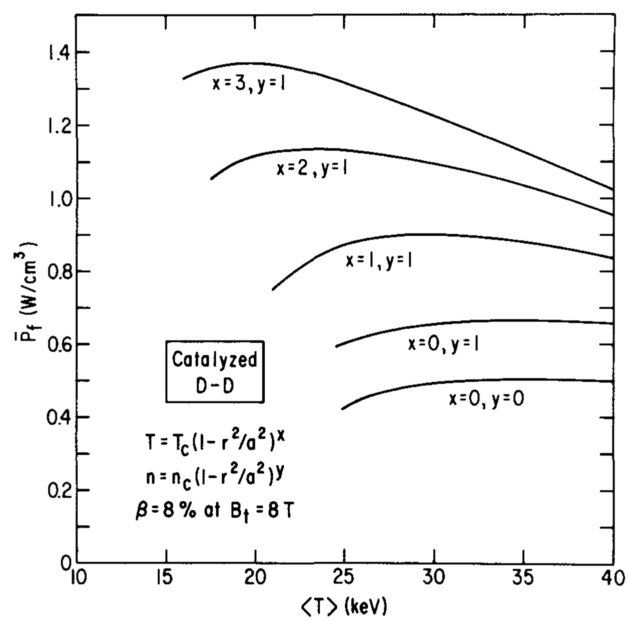

Figure 6. Spatially-averaged fusion power density for various temperature and density profiles, with a constant spatially-averaged plasma pressure: 2.1 J/cm^3, or $\bar{\beta}$ = 8% at B_t = 8.0 T. (Energetic ions not included in $\bar{\beta}$.) The minimum ignition temperature is at the left end of each curve. (773490)

highest possible beta.[4]

(c) Synchrotron radiation loss[1] may be intolerable at temperatures much exceeding 25 keV.

(d) High-$\bar{n}$ tokamak operation is achieved with relatively flat density profiles ($y \lesssim 1$), as in recent PLT experiments with ^4He-D mixtures.[11]

(e) When impurity radiation loss is relatively unimportant, the empirical energy confinement scaling[12] is $\tau_E \simeq 3\times10^{-19} q^{1/2} \bar{n}_e \langle a^2 \rangle$. In the following examples, this scaling of τ_E has been divided by 2 to allow for enhanced radiation loss. Then the required plasma size is $\bar{n}_e^2 \langle a^2 \rangle = 6.7\times10^{18} q^{-1/2} \bar{n}_e \tau_E$.

(f) The maximum practical magnetic field at the coil windings is 16 T, with Nb_3Sn conductor. Considering the relatively small neutron wall loading with advanced-fuel plasmas, a total thickness of 1.0 m between the first wall and the coil windings is adequate.

[Effects (b) and (c) can be ameliorated if it is possible to operate with $T_i > T_e$ in the hot central region. But decoupling of T_i and T_e is not easily achieved in an ignited tokamak plasma, where one needs large $\bar{n}$ to achieve large $\bar{n}\tau_E$.]

2. Examples. Table 2 gives plasma parameters for ignited D-D and D-^{3}He reactors with fusion power productions 77% and 62% respectively of that of a reference high-field, high-density D-T reactor.[13] The physical size of these reactors is quite practical, although the plasma current must be approximately four times that of the D-T plasma. The advanced-fuel plasmas have size and power outputs similar to those determined previously for plasmas of uniform profile[3]; but the present examples require a spatially-averaged beta only one-third as large, and $\langle T \rangle$ only 60% as large, as the uniform-profile cases.

The first-wall power loadings are only about one-quarter that of the D-T reactor. This economic disadvantage must be weighed against the well-known advantages of the advanced fuels that result from the elimination of tritium breeding: a huge reduction in tritium inventory, the absence of lithium-fire hazard, and especially the flexibility in choosing blanket composition for minimization of long-term activation. The latter problem is relieved in any event because of the reduction in neutron fluence for a given production of fusion energy. Further reduction in reactor size — or an increase in fusion power output — can be anticipated if even larger magnetic fields eventually become available.

Table 2. Low-Beta Ignited Tokamak Reactors

	D-T	D-D	D-^{3}He
Major radius (m)	6.0	7.7	7.7
Plasma radius (m)	1.2	2.6	2.6
Vertical elongation of plasma	1.6	1.6	1.6
B_t at plasma (T)	7.1	7.8	8.0
B_t at coils (T)	13.1	15.4	15.8
Plasma current (MA)	6.8	26	27
<T> (keV)	8.0	25	25
Peak temperature (keV)	12.4	62 [a]	62 [a]
$\bar{n}_e$ (cm^{-3})	3.4×10^{14}	2.6×10^{14}	2.7×10^{14}
Peak density (cm^{-3})	5.7×10^{14}	5.2×10^{14} [b]	5.4×10^{14} [b]
$\bar{n}_e\tau_E$ ($cm^{-3}s$)	4.0×10^{14}	2.0×10^{15}	2.2×10^{15}
$\bar{\beta}$ including energetic charged reaction products	0.046	0.09 [c]	0.09 [c]
$\bar{P}_f$ (W/cm^3)	9.2	1.18	0.96
Total fusion power (MW)	1995	1540	1230
First-wall power loading (MW/m^2)	4.7	1.4	1.13
Neutron wall loading (MW/m^2)	3.75	0.53	0.09

[a] $x = 3$

[b] $y = 1$

[c] Energetic ions account for 10% of the total plasma pressure.

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The Use of a Hot Sheath Tormac For Advance Fuels

by

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ABSTRACT

The use of hot electrons in a Tormac sheath is predicted to improve stability and increase $n\tau$ by an order of magnitude. An effective $n\tau$ for energy containment is derived and system parameters for several advance fuels are shown. In none of the advance fuels cases considered is a reactor with fields greater than 10 Wb or major plasma radius of more than 3 m required for ignition. Minimum systems have power output of under 100 MW thermal. System parameters for a hot sheath Tormac have a wide latitide. Sizes, magnetic fields, operating temperatures can be chosen to optimize engineering and economic considerations.

The use of a D-T fuel for a fusion reactor has the advantage of being the lowest temperature and lowest $n\tau$ of any fuel so far considered. On the other hand, fuels, such as D-D, D-He^3, and P-^{11}B, offer advantages over D-T either in reduced radiation, less waste heat or simpler handling. The problem is to find a reactor system which can meet the more stringent requirements of an advance fuel system by sustaining the higher temperatures and longer containment times. In this paper, the operation of a hot sheath Tormac is discussed as an answer to this problem.

Tormac[1] (toroidal magnetic cusp) is a cusp magnetic field configuration which has an absolute minimum-B geometry. The advantage of absolute minimum B is that it is mhd stable at high-β. The importance of this to advance fuel systems is that the magnetic field within the plasma can be reduced to very small intensities so as to minimize synchrotron radiation. In Tormac, the local $\underline{\beta}$ can be made as high as four without degrading the containment time. Higher $\underline{\beta}$ values are possible with only a small cost in containment time.

In Tormac a high-β plasma is held on the surface by a cusp magnetic field. A four pole line cusp is illustrated in Fig. 1 with a pressure profile shown in Fig. 2.

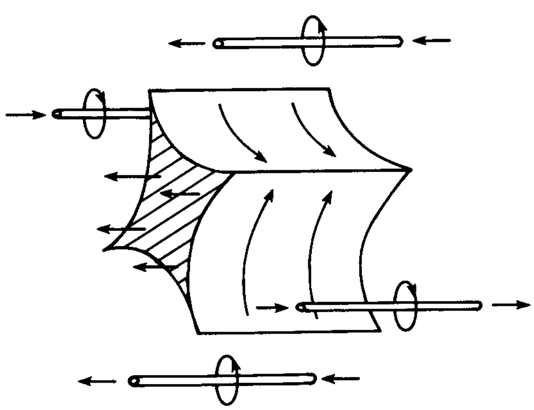

Fig. 1: Tormac

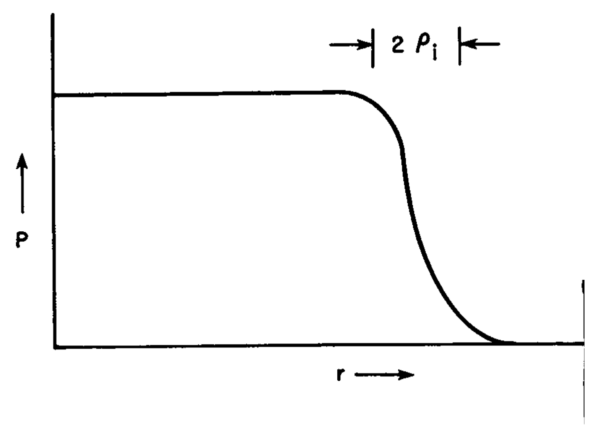

Fig. 2: Pressure versus radius

The absolute minimum-B geometry implies (Jukes theorem)[2] that the plasma pressure is held on open field lines. If we define the sheath as the surface region over which the pressure change occurs then this region must contain open magnetic field lines.

A crucial feature of Tormac is the containment of particles on open field lines. In the sheath region, particles are trapped between collisions by a canonical invariant,[3] and an adiabatic invariant.[4] These invariants combine to give mirror like trapping in the sheath. The only way a particle can escape from Tormac is along open an open field line so that the loss rate is determined in the sheath. In the ideal case, the sheath loss is due only to interparticle collisions. These collisions scatter particles into the loss cone. Since the probability of loss is about equal to the probability of cross field diffusion the sheath thickness is the order of ion gyroradius, r_i. The containment time for such a system is then given by

$$n\tau = 0.1\left(R_p/r_i\right)\tau_{ii}n \tag{1}$$

where τ_{ii} is the ion collision time, R_p the plasma characteristic minor radius. The factor 0.1 includes geometric factors and implies a cusp mirror ratio of about 1.5. For most reactor designs R_p/r_i is near 100.

If collisions are classical τ_{ii} is large enough for most reactors. On the other hand, microturbulence in the sheath could lead to an effective τ_{ii} much lower than classical. Recent 2XII results[5] have indicated a degredation of τ_{ii} by about five; however, in Tormac magnetic field shear is thought to reduce this factor.[6]

A second effect that could increase the particle loss rate from the sheath is the drift of particles from the main plasma into the sheath region. This drift can be prevented by including, inside the region of open field lines, a twist in the magnetic field lines. This rotational transform can be accomplished by including in the plasma a toroidal current and a resulting poloidally closed magnetic field component.

The implementation of the internal magnetic field can best be discussed in terms of a specific example. In the current Tormac experiments, the toroidal bicusp[7] shown in Fig. 3 is used. In this shape, the radius of curvature of

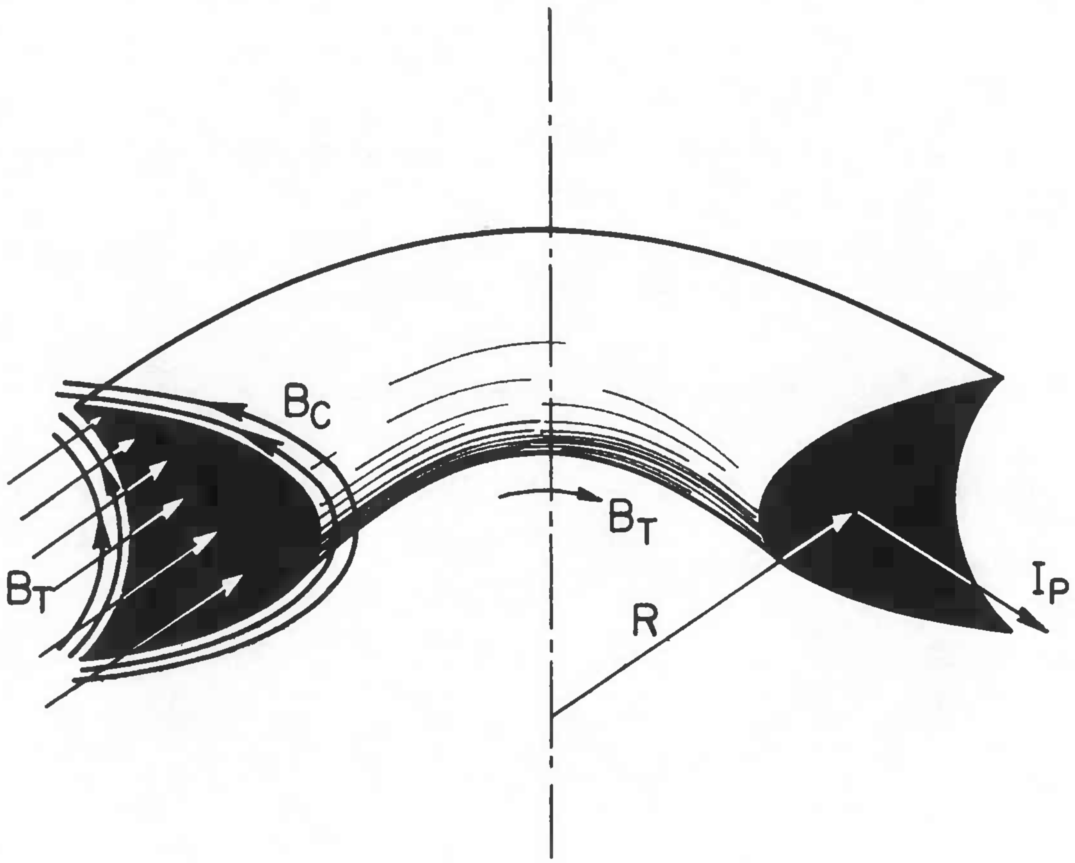

Fig. 3: Toroidal bicusp

the magnetic field lines in the surface of the plasma is negative as required for an absolute minimum-B geometry. This is accomplished by having the magnetic field lines at small radius dominated by the toroidal component of field;and,the intensity of the field lines at the outer toroidal radius dominated by the poloidal component of field.

Limiting the number of cusps to two,as shown in the bicusp, has several advantages over shapes with a larger number of cusps. Experimentally, fewer cusps reduces the complexity of the device and makes it easier to produce the required magnetic field shaping without bringing the coils too close to the plasma. Theoretically, the bicusp reduces the particle loss cone[3] and optimises the volume to surface ratio.

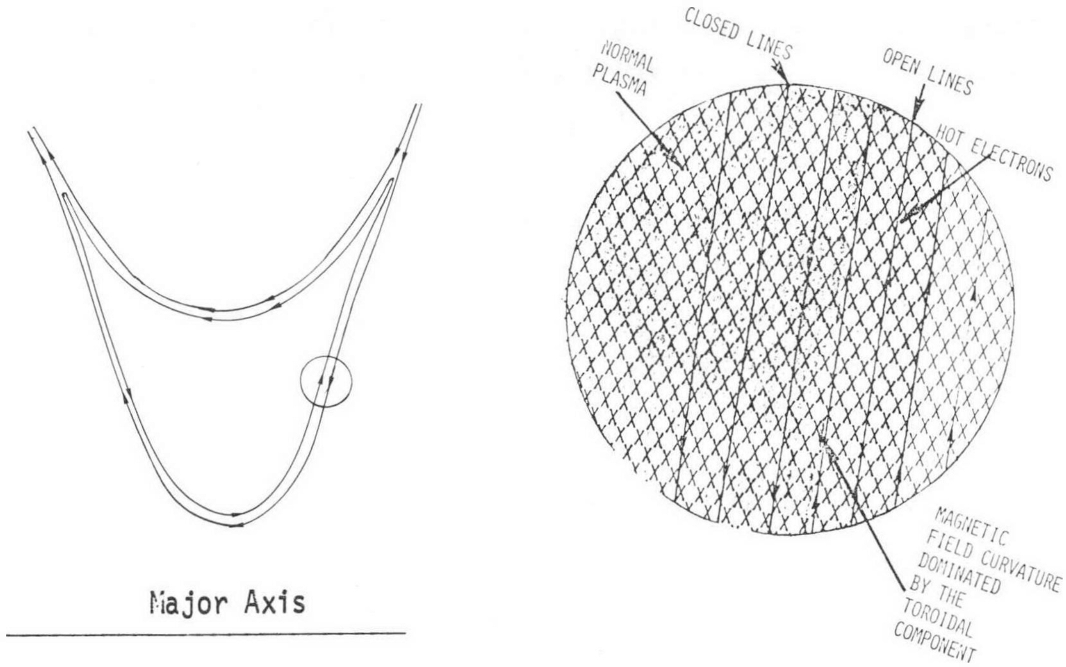

Fig. 4: Cross sections of bicusp with internally closed poloidal field.

A possible poloidal configuration for a bicusp is given in Fig. 4. There is indicated in Fig. 4 a poloidal field reversal in the sheath. It should be remembered in considering this diagram that the toroidal magnetic field is present everywhere. What appears as a region of field reversal is only a region of mild field shear. Thus, while the field reversal region appears as a critical region in this drawing, theory indicates a positive stability of this region. In particular, if the radius of curvature of the field line is negative over the pressure surface, the flow of material in this region goes at the diffusion rate.[8]

The closed poloidal magnetic field line in figure 4 exhibits bad curvature in the cusp region. Stability in this region depends on an average minimum-B, averaging the short cross over distance in the cusps with the good curvature over the rest of the device.[9]

Fortunately, the bad curvature region is small and the curvature, dominated by the toroidal magnetic field, is large. The gradient in pressure is the order of the cusp distance from the plasma. The width of the region is a few gyroradii. To estimate the stability of this region of unfavorable curvature in the cusp one can use the relationship $B\ell^2 < RS$,[10] where ℓ is the connection length; R is the radius of curvature of the magnetic field line; and S is the pressure gradient length. These parameters assume about half the plasma pressure is supported on closed magnetic field lines. In the cusp region the closed poloidal magnetic field line extends about $R_p/2$ from the plasma. If this field supports one half the plasma pressure, $S \sim 4R_p$. Estimating the strength of the toroidal and poloidal component strength the connection length from one side of the cusp to the other is about $R_p/2$. The curvature is then about R_p A/2 where A is the aspect ratio.

Thus stability requires $\beta < 8A$. This calculation should be averaged over an ion radius so that the inequality is satisfied for all reasonable values of β. The use of closed poloidal magnetic field lines to support part of the plasma pressure in the bicusp is a way to limit particle drifts and to improve the $n\tau$. However, the use of such a field opens even more exciting possibilities.

As mentioned above a most serious question about Tormac is the sheath stability against microturbulence. In particular, the drift modes with frequencies near $\sqrt{\omega_i \omega_e}$ are predicted to be mildly unstable. In the 2XII experiment these instabilities were damped with cold ions. Part of the problem is that because of the negative potential found in mirror contained plasmas, all low energy ions are promptly lost from a plasma.

One method of curing this problem is to heat the electrons so that their loss rate matches that of the ions. Without an internally closed poloidal magnetic field to control electron thermal conductivity this would be impossible. This internal reverse field Tormac makes it possible to hold hot electrons in the sheath.

The ability to maintain a stable, hot, non isotropic electron density in the presents of hot ions has been experimentally demonstrated[11] and a method of producing such plasmas is currently under development. Hot electrons have previously been proposed as a method of reducing electron thermal conductivity along magnetic field lines and the tolerance of the sheaths to a neutral gas background.[12] These are not seen as severe problems in a conventional Tormac. However, hot electrons do have a dramatic affect on the classically predicted sheath containment time.

To calculate the time constant for a hot electron sheath, consider the case of a D-D plasma with a temperature of T_i.

The electrons in the sheath are maintained by microwaves at 24 T_i so that the sheath time, $\tau_s = \tau_{ii} = \tau_e$. The sheath pressure is $n_e (T_i + T_e) = 25 n_e T_e$. One representative solution to the magnetic profile in the sheath is given by[4]

$$B = B_c \tanh^2 (x_\perp / \lambda) \;, \tag{2}$$

where B_c is the cusp field intensity just outside the plasma surface. So that for equilibrium

$$n_e (T_e + T_i) \sim (B_c^2 - B^2)/8\pi \;. \tag{3}$$

This implies that in the region where $T_e = 24 T_i$ the density, n_e is 1/12.5 the value it would have if $T_e = T_i$. Correspondingly, the sheath time constant τ_s is 12.5 times as large as it would be for a cold sheath Tormac. Thus

$$(n\tau)_p = 5\, n\tau_{ii}\, Rp/r_i \qquad \text{or}$$

$$(n\tau)_p = 3.6 \times 10^{13}\, R(m)\, B(w)\, T(keV)\ \text{sec/cm}^3 . \tag{4}$$

Eq. 4 neglects synchrotron radiation. For a hot sheath Tormac synchrotron radiation gives

$$(n\tau)_s = 7.1 \times 10^{20}\, R(m)\, B(w)\, T_i^{-5/2}\, (keV)\ \text{sec cm}^{-3} \tag{5}$$

The combined $(n\tau\, T_i)$ due to the sum of both particle losses and synchrnotron radiation is shown in Fig. 5.

Curves are drawn for R B = 1 and R B = 10. Also plotted are the ignition curves for various reactions. **The curve intersections represent operating parameters for an ignition reactor.**

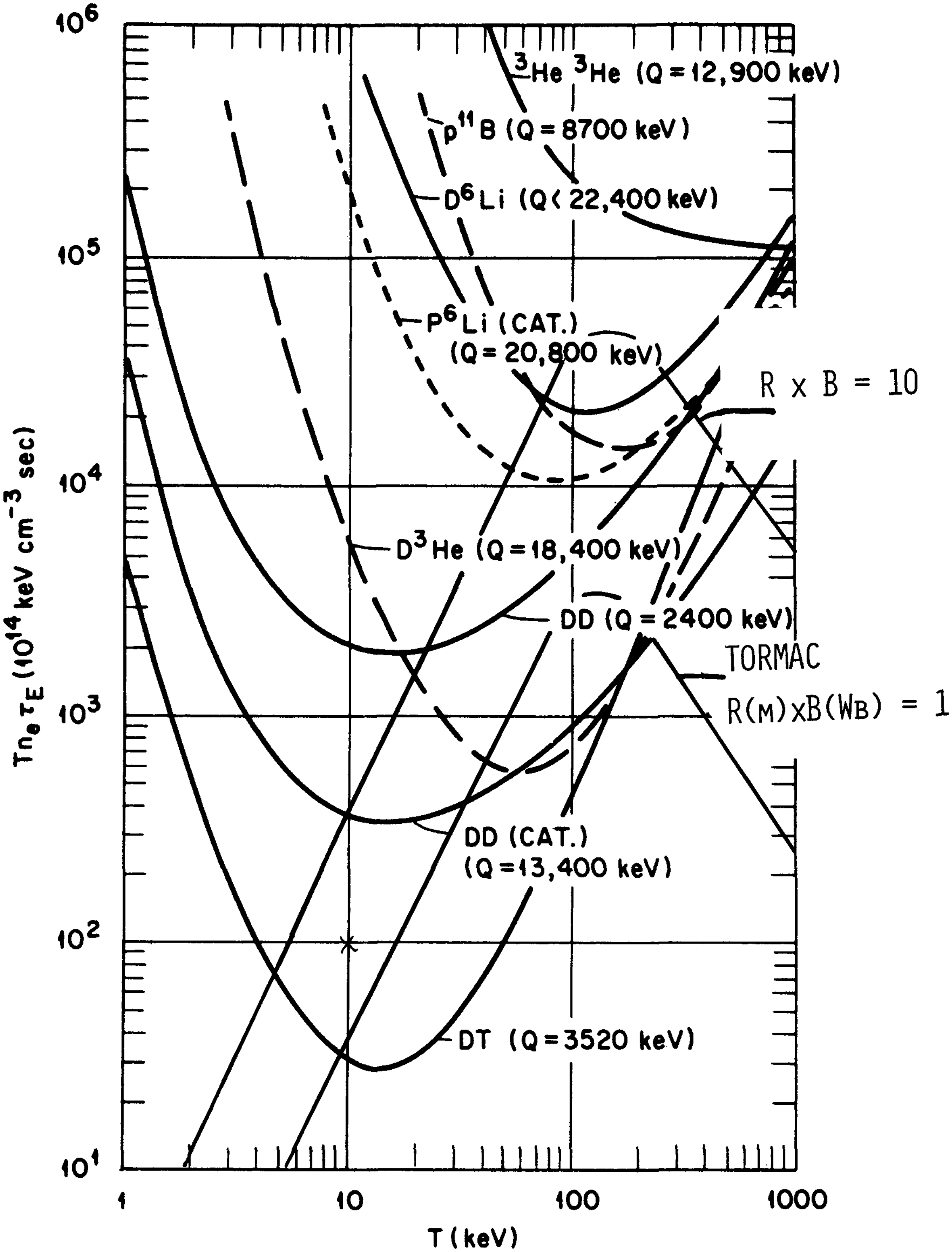

Fig. 5: Tormac energy loss parameter ($T\ n_e\ \tau_E$) for R B = 1 and R B = 10 plotted on top of ignition curves as given by J. R. McNally, Jr.

The net power P, produced in charged particles for such systems is given by

$$P = \frac{2\,\pi^2\ A\ R^3\ B^4}{2\ n\ T_i\ \tau} \tag{6}$$

Typically for D-T, $R_pB = 1$, the total thermal output including neutrons is about 70 MW thermal. For P-^{11}B, $R_pB = 5$ only about 2 MW is produced. Other reactions give intermediate power output.

It must be pointed out that the long time constants and low reaction rates predicted for these minimum systems require vacuum pressures surrounding the plasma that may be difficult to realize in practice.

The use of heated electrons both stabilizes the sheath and improves the $n\tau$ by about an order of magnitude. Thus, the hot sheath Tormac can support almost any of the known reactions at ignition with a very modest size systems.

In conclusion, it might be stated that systems parameters for a hot sheath Tormac have a wide latitude. Sizes, magnetic fields, operating temperatures can be chosen to optimize engineering and economic considerations.

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Prospects for a DD Tandem Mirror

by

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ABSTRACT

The possibility of burning advanced fusion fuels in a tandem mirror is considered for a catalyzed DD cycle, in which the T and He^3 reaction products from DD burn in both the solenoid and plugs are reinjected for complete burnup: $3D \rightarrow p + He^4 + n + 21.6$ MeV. Classical radial transport of the He^4 ash determines the steady state alpha fraction in the solenoid. Synchrotron radiation losses are minimized at high beta, such that charged particle fusion power recovered in a direct converter exceeds radiation losses by a factor greater than two. An overall system Q = 4.5 is found for one reactor example but the power output is large (3 GW(e) net) due to the low power density in the solenoid. Optimizing recirculating power cost (Q) against plug/solenoid density ratio (power density) should result in much smaller reactor size and cost.

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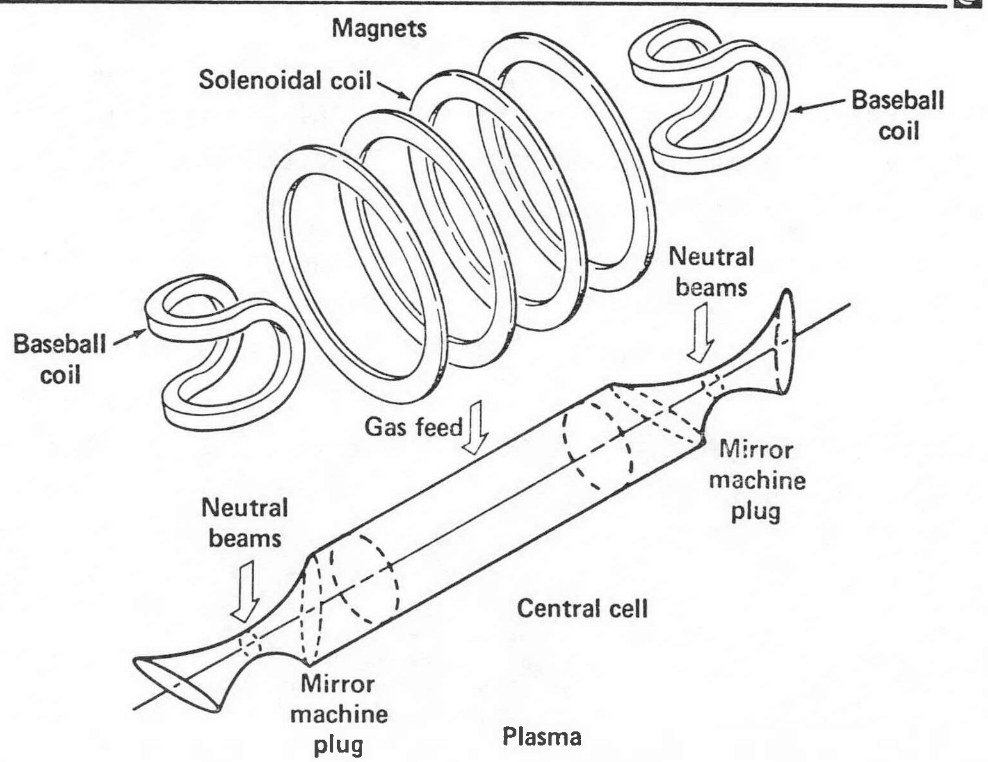
I. INTRODUCTION

The tandem mirror concept[1,2] has first been evaluated as a DT burning reactor in recent LLL studies.[3,4] Use of advanced fusion fuels in a tandem mirror is now being considered for the following reasons: In a tandem mirror (Fig. 1) electrostatic confinement of ions in the solenoid scales as $n\tau \propto T_i^{3/2}(n_p/n_c)^{T_e/T_i}$, where T_i and T_e are the solenoid ion and electron temperatures, respectively, and where n_p/n_c is the plug to solenoid electron density ratio. As classical mirror machines, ion confinement in the plugs scales as $n\tau \propto E_p^{3/2}$, where E_p is the mean plug ion energy. Since $n\tau$ in both the plugs and in the solenoid increases with ion energy, a tandem mirror should be suited for burning advanced fusion fuels such as DD, which require higher temperatures for adequate reaction rate, and also higher $n\tau$ than with DT as well. Development of efficient, high energy neutral beams is progressing rapidly at Livermore and Berkeley, and one may expect that neutral beams, especially those based on negative ion acceleration, would be especially useful in fueling and heating both the plugs and the solenoid of an advanced fuel tandem mirror in steady state. Because

FIGURE 1.

TANDEM MIRROR MACHINE



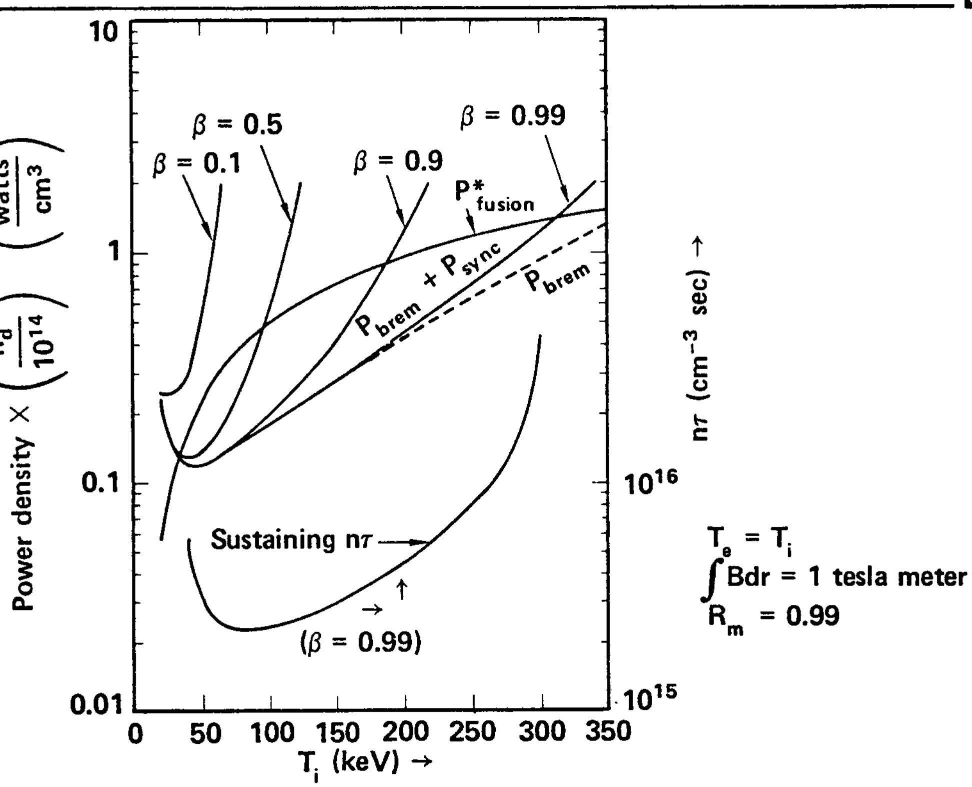
of the linear magnetic geometry in a tandem mirror, and because of the minimum - $|B|$ character of the high field regions in the plugs, the plasma beta, defined as the ratio of plasma pressure to appplied magnetic field pressure, is expected theoretically to be very high, of order unity, in both the plugs and in the solenoid. In the single mirror cell 2XIIB experiment, plasma betas up to values just short of field-reversal have been achieved with quasi-steady neutral beam injection; i.e., the field inside the 2XIIB plasma is near zero. Such high betas would allow operation at high T_e in an advanced-fuel tandem mirror, since synchrotron-radiation losses would be greatly reduced by the low fields in the plasma. The open-ended magnetic field geometry, and the fact that ion losses are nearly monoenergetic due to a high ratio of plasma potential to ion temperature, makes direct conversion of plasma losses particularly suitable to a tandem mirror. Direct conversion is especially important with advanced fuels since a higher fraction of fusion energy appears as charged particles. Finally, because the plasma losses are primarily along field lines and not across field lines, it is possible to distribute neutral injection sources such that the radial deuterium density gradient is flat over most of the plasma volume. This keeps the average field in the plasma low, and makes possible alpha ash removal by classical radial transport via deuterium-alpha collisions.

II. Cat-DD FUEL CYCLE FOR TANDEM MIRRORS

Consideration of advanced fuel use in a tandem mirror has thus far been limited to the catalyzed DD fuel cycle. Other fuel cycles may be considered in the future. In the present case, tritium and He^3 reaction products generated by DD reactions in both the solenoid and in the plugs of a tandem mirror are reinjected until complete burnup is obtained; $3D \rightarrow p + He^4 + n + 21.6$ MeV, of which 64% is charged particle energy and 36% is neutron energy. Typically we have found a tritium burnup fraction $f_{bt} \simeq 60\%$, so we could, as a future option, allow the escaping tritium to convert to He^3 by beta decay before reinjection. The subsequent burn of the extra He^3 could then raise the charged particle fraction of fusion energy to as high as 80%, depending on f_{bt}. A disadvantage of this scheme is the long tritium conversion time (12-year half life), and the consequent large inventory of stored tritium required in steady state. In the present calculation, the deuterium plugs of the tandem mirror breeds an extra amount of the He^3 for burning in the solenoid approximately equal to the escaping tritium from the solenoid. The tritium recovered in the direct converter is promptly reinjected and burned to minimize the tritium inventory.

As yet, confinement in the solenoid for the cat-DD cycle has not been found to be sufficient at practical density ratios ($n_p/n_c < 30$) to achieve ignition. Therefore the solenoid requires a continuous input of energy to be energetically sustained in steady state. Because we wish to maximize charged particle fusion power relative to radiation losses, a DD tandem mirror requires $T_i \gtrsim T_e$ in the solenoid (see Fig. 2), rather than $T_e < T_i$ as in the case of the DT tandem mirror.[3] Heating ions rather than electrons in the solenoid is therefore required. An attractive and very economical solution is to arrange for the very energetic deuterium ion losses from the plugs to escape preferentially into the solenoid by either tilting the plug mirrors (make the outer mirrors stronger) or by unbalancing the plug

CAT DD FUSION POWER DENSITY AND RADIATION LOSSES - FIGURE 2.



ambipolar potential inward. Normally, the outward ambipolar potential drop in the plugs is much larger than the drop to the solenoid, such that it is very difficult to get the plug ions to pass preferentially into the solenoid. However, addition of low field, low power auxiliary plugs beyond the main plugs, and maintained at densities just greater than the solenoid density, makes the solenoid potential drop slightly larger, such that all the main plug ions escape into the solenoid even with equal mirror fields in the plugs. In this way a large fraction of the neutral beam injection power required to maintain the plugs is re-used to sustain the solenoid. Tritium and He^3 bred in the plugs are injected automatically into the solenoid. The tritium escaping from the solenoid and recovered in the direct converter is reinjected via the plug neutral beams, since tritium scattering rates are lower than for deuterium, and make the plugs more efficient. Escaping He^3 is re-injected at low energy directly into the solenoid. Since in general the required deuterium injection current in the solenoid is much greater than the plug injection current, supplementary low energy neutral beams are used to make up the solenoid deuterium losses, and to control the steady state density in the solenoid.

III. COMPUTATION OF SOLENOID PARAMETERS

The energetic reaction products, principally the 14 MeV protons and 3.5 MeV alphas, contribute a large fraction of the energy input to the solenoid, and to the plasma pressure. The relative rates of energy input to the plasma thermal ions and electrons by the slowing down reaction products determines the ratio T_e/T_i which in turn strongly affects the thermal ion confinement time in the solenoid potential well. Most of the reaction products are initially magnetically confined by the large mirror ratio ($\simeq$ 20) in the solenoid. Some are scattered into the loss cone while slowing down, but most survive to thermalize with the electrostatically confined ions at temperature $T_i < \phi_i$, the potential well depth in the solenoid.

To compute the energy-exchange, mutual-scattering loss and fusion among the five ion species D, T, He^3,α,and p in the solenoid, a multi-species, two-dimensional Fokker-Planck computer code has recently been developed at Livermore by Marvin Rensink and Art Mirin for advanced-fuel tandem mirror calculations. The code computes the time evoluti of the distribution functions $f(v, \theta, t)$ for each of the five ion species until steady state is achieved. The electron energy balance is calculated by an analytic rate equation, including drag computed for each ion distribution, relativistic synchrotron and bremsstrahlung radiation loss, and electron energy $\phi_e + T_e$ carried out with each ion charge lost. The confining electron potential ϕ_e is computed by requiring a balance of ion and electron loss rates at each time step. The ion confining potential barrier ϕ_i is computed from $\phi_i = T_e \ln (n_p/n_c)$ at each time step. The density ratio n_p/n_c is an input parameter. Fuel ion losses by fusion burnup in DT, DD_p, DD_n and DHe^3 reactions appear as isotropic sources for the reaction products at their appropriate energies. Radial transport of thermalized alphas is treated by including a loss term $-f_\alpha/\tau_{\alpha r}$ in the alpha-component Fokker-Planck equation. The alpha radial loss time constant $\tau_{\alpha r}$ is given by[4]

$$\tau_{\alpha r} = 3.9 \times 10^{11} \frac{\sqrt{\bar{E}_0} \, \langle r_c \, B_{ci} \rangle^2}{n_0 \, \ln \Lambda_{D\alpha}} \text{ sec} \tag{1}$$

where $\bar{E}_D$ is the mean deuterium energy in keV, n_D is the deuterium density in cm^{-3}, and $\langle r_c \, B_{ci} \rangle = r_c \, B_c \sqrt{1 - \beta_c}$ is the product of solenoid radius and internal field in cm-tesla. The parameter $\langle r_c \, B_{ci} \rangle$ appearing in $\tau_{\alpha r}$ and also in the synchrotron radiation loss (which depends on the radial size of the plasma) is the only way the radial dimension of the system appears in the calculation. All plasma densities and temperatures except for the alphas are assumed to be kept uniform across the solenoid cross section by appropriate injection profiles. Since an outward flux of alphas requires an equal charge flux of deuterium inward,[4] an auxiliary deuterium source term = $2 \, f_\alpha/\tau_{\alpha r}$ is included in the deuterium component of the Fokker-Planck equation to represent the inward deuterium flux. The He^3 sources and sinks by fusion burn are both uniform across the solenoid, so that no radial gradient and transport occurs for He^3.

IV. COMPUTATION OF PLUG PARAMETERS

The parameters of the plugs are determined by the given density ratio n_p/n_c, by the condition that no **flow of electron energy** occurs between the plugs and the solenoid, and by conservation of magnetic flux

$$\pi r_c^2 B_{ci} = \pi r_p^2 B_{pi} \tag{2}$$

where r_p and B_{pi} are the plug radius and internal field, respectively. In future work, the plug parameters can be computed more generally by an additional 2D Fokker-Planck calculation for the plug ions, coupling ion and electron energy losses between the solenoid and the plugs internally in the code. For the present we use a plug confinement formula

$$(n\tau)_p = 3.3 \times 10^{10} E_p^{3/2} \log_{10} \left[\frac{R_m}{1 + \frac{\phi_i}{E_{inj}}} \right], \tag{3}$$

which is a best fit to several previous 2D Fokker Planck runs for the plugs. The plug mirror ratio $R_m = R_{vac}/(1 - \beta_p/2)$ is enhanced by the plug beta according to an empircal formula based on 2XIIB experiments.[3] The ambipolar potential seen by the main plug ions with the auxiliary plugs present is ϕ_i, the solenoid potential barrier. E_{inj} is the plug injection energy (all units in keV). The average plug ion energy $\acute{E}_p$ is determined from the condition of electron energy balance in the plugs:

$$\frac{n_p^2}{(n\tau)_p} (\phi_e + \phi_i + T_e) + P_{radp} = \frac{n_p^2 (E_p - 3/2\ T_e)}{(n\tau)_{drag}} , \tag{4}$$

where P_{radp} is the bremsstrahlung and synchrotron radiation loss in the plugs. The deuterium plug ion cooling rate on the electrons $(n\tau)_{drag}$ is given by

$$(n\tau)_{drag} = 10^{12} T_e^{3/2} \tag{5}$$

where T_e is in keV.

The plug injection energy E_{inj} is determined by ion energy balance the plugs:

$$\frac{n_p^2 (E_p - 3/2T_e)}{(n\tau)_{drag}} + \frac{n_p^2}{(n\tau)_p} <E_L>= \frac{n_p^2}{(n\tau)_p} E_{inj} + P_{fp}^* \tag{6}$$

where $<E_L>$ is the mean escaping plug ion kinetic energy, and P_{fp}^* is the charged fusion power generated and deposited in the plugs. Using the relation $<E_L> \simeq 0.42\ E_{inj}$ obtained from appropriate Fokker-Planck test calculations, Eq. (6) can be solved for E_{inj} together with Eq. (4) for E_p, using the value of T_e, ϕ_i and ϕ_e obtained in the solenoid Fokker-Planck run.

The rest of the tandem mirror parameters are obtained as follows: Selecting a maximum plug external field B_p and beta β_p, the maximum plug density n_p can be computed, having E_p from Eq. (4). The solenoid electron density n_c is then obtained from the density ratio n_p/n_c, and the solenoid external and internal fields B_c and B_{ci}, respectively, calculated from a chosen beta β_c and pressure balance:

$$\beta_c \frac{B_c^2}{8\pi} = n_c T_e + \sum_{i=1}^{5} n_i \quad <E_{i_\perp}> \quad . \tag{7}$$

The solenoid radius is then given by

$$r_c = \ <r_c \quad B_{ci}> / B_{ci} \ . \tag{8}$$

Conservation of magnetic flux Eq. (7) then gives the plug radius r_p. The plugs are assumed to be equivalent to uniform spheres of radius r_p at density n_p, in computing the total neutral beam injection power for the main plugs. The auxiliary plugs are assumed to be spheres of density $n_{paux} = n_c$ and radius r_{paux} determined by flux conservation at the same beta $\beta_{paux} = \beta_p$. Finally, the solenoid length is determined by the requirement that the plug ion energy loss escaping from the plugs into the solenoid I_p $(<E_L> + \phi_i)$ match the required input power to the solenoid, minus a nominal amount for the supplementary low energy neutral beams .

V. REACTOR EXAMPLE

An example set of reactor parameters are given in Table A for one particular set of chosen input parameters indicated. A power flow diagram for this case is shown in Fig. 3. Although the average injection energy per deuteron in the solenoid $\bar{E}_{injs}$ has not yet been varied enough to optimize Q, the Q obtained in this case is quite adequate for a viable reactor energy balance. Note that the large majority of electrical power produced comes from the direct converter, which has a single stage efficiency

$$\eta_{Dc} = \frac{\phi_i + \phi_e}{\phi_i + \phi_e + T_i} \simeq 0.9. \tag{9}$$

Such a simple and efficient direct converter can bring about substantial economies in the energy conversion. Because the direct converter is the dominant energy conversion system in this reactor, the efficiency requirements for thermal energy recovery in the solenoid and plug blankets are of secondary importance. Indeed, energy recovery in the blankets may not be necessary at all if economies in the blanket design can thereby be obtained. For the case in Table A and Fig. 3, a modest efficiency $\eta_{th} = 0.33$ was assumed for low-temperature water coolant (350 - 400° C), believing this will be cost effective. A thick cast aluminum structure should make a cheap and low activation blanket, so that 4.65 MeV decay energy of neutron capture in the aluminum is assumed to obtain the blanket multiplication factors M in Fig. 4.

As a very rough preliminary cost estimate, the water cooled aluminum blanket and 20 kG solenoid magnet would cost about \$1M per meter, or \$2 x 10^9 total. At \$250 per kilowatt handled in the direct converter, and \$300 per kilowatt consumed by the plug neutral beams, the recirculating power cost would be about \$1 x 10^9, half as much. Thus a reduced reactor cost may result if a smaller density ratio $n_p/n_c < 30$ is chosen, increasing power density and decreasing solenoid cost at the expense of a higher recirculating power fraction. The overall size and power output would also be reduced, since both the solenoid and plug radius would shrink, the plug power would decrease, and the matching solenoid fusion power = Q x injection power would decrease. Future

TABLE A. EXAMPLE REACTOR PARAMETERS

Solenoid

$n_c = 2.5 \times 10^{13} cm^{-3}$

$T_e = 141$ keV

$n_D = 1.53 \times 10^{13}$cm-3

$\overline{E}_D = 287$ keV

$n_T = 3.53 \times 10^{11} cm^{-3}$

$\overline{E}_T = 355$ keV

$n_{He^3} = 2.03 \times 10^{12} cm^{-3}$

$\overline{E}_{He^3} = 298$ keV

$n = 1.36 \times 10^{12} cm^{-3}$

$\overline{E} = 453$ keV

$n_p = 2.57 \times 10^{12} cm^{-3}$

$\overline{E}_p = 1641$ keV

$B_c = 2T$

$\beta_c = 0.99$

$B_{ci} = 0.2$ T

$r_c = 485$ cm

$L_c = 1.96$ km

$I_c = 1510$ A

$\phi_i = 480$ keV

$\phi_e = 973$ keV

$(n\tau)_D = 4 \times 10^{15} cm^{-3}$sec

Plugs

$n_p = 7.5 \times 10^{14} cm^{-3}$

$T_e = 141$ keV

$n_D = 6.3 \times 10^{14} cm^{-3}$

$n_T = 5.8 \times 10^{13} cm^{-3}$

$n_{He^3} = 7 \times 10^{12} cm^{-3}$

$n_\alpha = 3.6 \times 10^{12} cm^{-3}$

$n_p = 4.1 \times 10^{13} cm^{-3}$

$E_p = 2040$ keV

E_{inj} - 2400 keV

$B_p = 18T$

$\beta_p = 1.5$

$B_{pi} = 4.5$ T

$r_p = 104$ cm

$I_p = 494$ A

$R_{vac} = 1.07$

$(n\tau)_p = 1.7 \times 10^{15}$

Auxiliary Plugs

$n_{paux} \gtrsim 2.5 \times 10^{13} cm^{-3}$

$T_e = 141$ keV

$E_{paux} = 2040$ keV

$E_{injaux} = 2400$ keV

$B_{paux} = 3.2$ T

$B_{piaux} = 0.82T$

$\beta_{paux} = 1.5$

$R_{vac} = 1.07$

$(n\tau)_{paux} = 1.5 \times 10^{15}$

$P_{trapped} = 19.5$ MW

$P_{incident} = 92$ MW

$r_{p2} = 243$ cm

Input Parameters

$n_p/n_c = 30$

$\langle E_D \rangle_{input} = 400$ keV
(in solenoid)

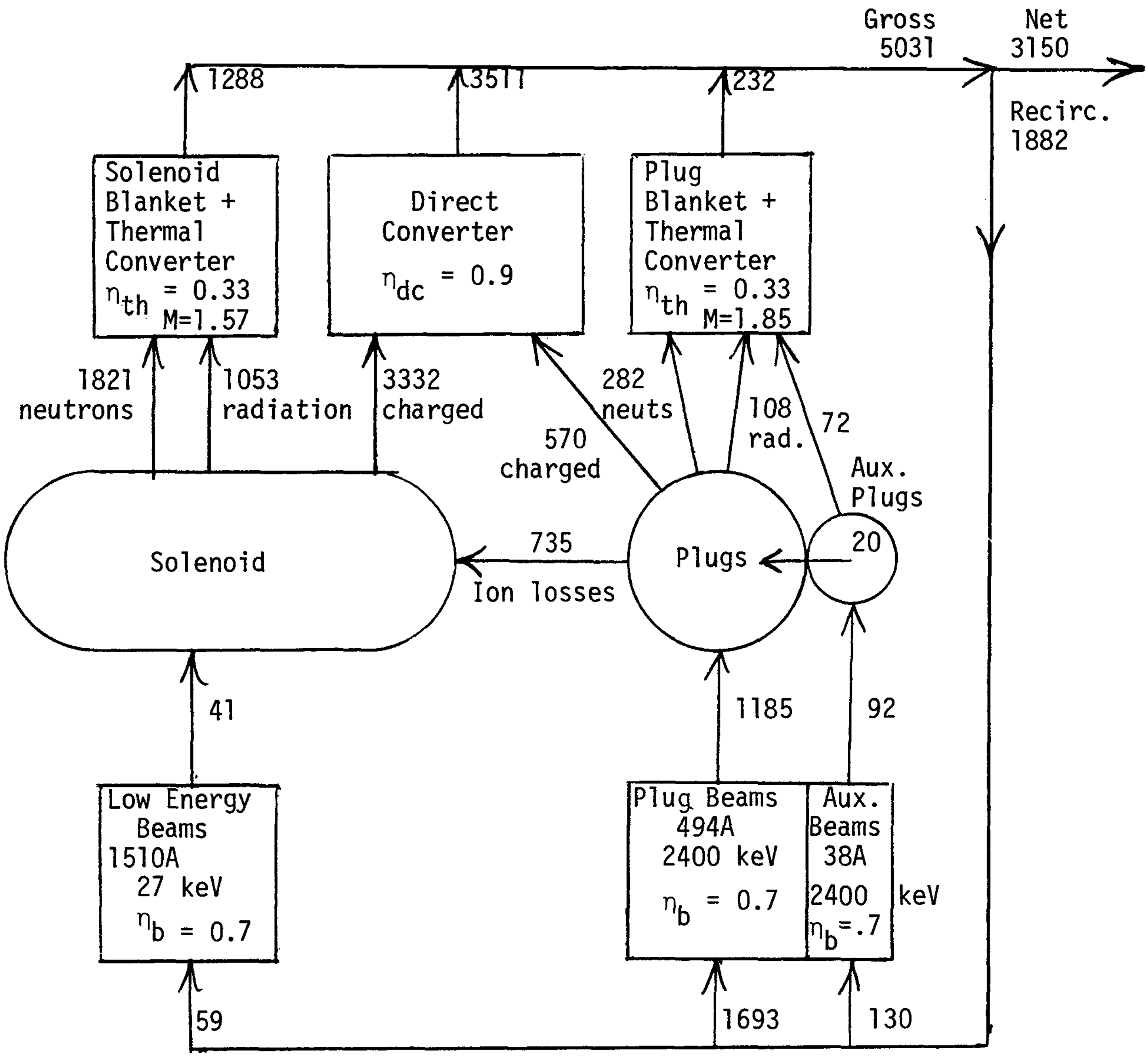
$\langle r_c B_{ci} \rangle = 100$ cm T

$\beta_p = 1.5$

$\beta_c = 0.99$

$B_p = 18$ T

$R_{vac} = 1.07$
(in plugs)



$$\text{Overall } Q \equiv \frac{\text{total fusion power}}{\text{injection power}} = 4.5$$

$$\text{Direct Converter } Q \equiv \frac{\text{charged fusion power-radiation}}{\text{injection power}} = 2.0$$

$$\text{Recirculating power fraction } f_r = \frac{1882}{5031} = 37\%$$

Figure 3. Power Flow Diagram (in MW units)

work will be concerned with optimizing the trade off between Q and power density to reduce reactor size and cost.

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2. T. K. Fowler and B. G. Logan, Comments on Plasma Physics and Controlled Physics and Controlled Fusion Research, Vol. II, NO. 6, 167 (1977).
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AUTOBIBLIOGRAPHY ON ION-LAYER PRODUCTION, PROPERTIES, NEEDS, AND APPLICATIONS

by

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ABSTRACT

An autobibliography on advanced fusion fuels and the Ion-layer plasma electromagnetic fusion configuration is presented. This was prepared at the request of G. H. Miley for distribution at the EPRI Review Meeting and was presented as part of the poster session.

1. "Proposal to Demonstrate a Thermonuclear Plasma of High Density," J. R. McNally to E. D. Shipley, December 8, 1957, unpublished.

Discussed 7.7-MeV HD^+ injection using third mode of ORNL 86-in. cyclotron to form a self-pinching ion current ring analogous to the "smoke ring" concept of Hartland Snyder.

2. "Supplemental Comments on . . .," J. R. McNally to E. D. Shipley, December 25, 1957, unpublished.

Discussed philosophy of multi-MeV injection buildup in a magnetic mirror and properties of "negative magnetic mirror" due to high ion ring current (an Ion-layer or reversed field mirror).

3. "On the Energy Dependence of the DCX Type of Sherwood Device," J. R. McNally, Jr., 1958, unpublished.

Evaluation of DCX prospects using various molecular ion injection energies (46 pages).

Abstract: A qualitative survey is made of the energy dependence of the many trapping and loss rates in a deuterium-fed DCX type of Sherwood device. It appears that no self-sustained nuclear reaction can be obtained with either carbon arc or plasma breakup at an injection energy of 600 keV. Considerations of high energy injection with plasma breakup (1-mA, 9.6-MeV D_2^+ injection) suggest the possibility of a self-sustained reaction wherein the nuclear reaction rate is substantially equal to the trapping rate ($>10^{16}$ nuclear events per second). Some general conditions to be met for igniting a self-sustained thermonuclear reaction are proposed. The concept of a negative magnetic mirror is introduced as a stability criterion for a thermonuclear reactor.

4. "On the Possibility of Charge Exchange Losses in the Carbon Arc," J. R. McNally, Jr., 1958, unpublished.

Proposed a serious loss mechanism due to the carbon arc used in DCX experiments (σ_{CX} due to carbon arc estimated as $\sim 3 \times 10^{-19}$ cm^2 for protons on arc carbon ions).

5. "The Direct Current Experiment (DCX) and High-Temperature Measurements in the Carbon Arc," J. R. McNally, Jr., in Optical Spectrometric Measurements of High Temperature, P. J. Dickerman, ed., Univ. of Chicago Press, Chicago, Illinois, 1961.

Discussed the DCX approach to achieving a fusion reactor, residual charge-exchange losses in the carbon arc, and the unusually high ion

temperatures in the DCX carbon arc ($T_i/T_e \sim 20$). Note: later in much longer arcs $T_i/T_e \to 100$!

6. "Speculation on the Attainment of a High Density and a Large Circulating Current of Very Hot Protons in a Magnetic Mirror Device," J. R. McNally, Jr., pp. 89-93 in ORNL-3392, Oak Ridge National Laboratory (1962).

   Discussed physics requirements for 6-MeV H_2^+ injection into a 60-kG/30-kG magnetic mirror and prospects for Lorentz trapping and/or exponentiation to high proton densities so as to lead to a reversed field mirror configuration.

7. "Further Speculations on Very High Energy Injection into a Magnetic Mirror," J. R. McNally, Jr., Mozelle Rankin, and E. D. Shipley, pp. 119-120 in ORNL-3472, Oak Ridge National Laboratory (1963).

   Discussed the severity of proton energy losses to cold electrons by Fokker-Planck treatment of density buildup evaluations of previous reference. [Note: later treatments suggest a solution to this problem by electron cyclotron heating (ECH) of the "cold" electrons.]

8. "Conjectures on Fusion Chain Reaction Cycles and the I-Layer Configuration," J. R. McNally, Jr., August 6, 1964, unpublished.

   Discussed (1) the possibility of using trapped MeV protons to "ignite" a fusion chain reaction in view of the observation that ions in an energetic, 6-m-long, magnetically confined carbon arc are 100 times "hotter" than the electrons due to special quantum atomic processes and (2) the elucidation of properties predicted for the Ion-layer reacting configuration.

9. "Fusion Chain Reactions," J. Rand McNally, Jr., Bull. Am. Phys. Soc. 11, 849 (1966), text available.

   Abstract: The concept of charged-particle fusion chain reactions will be discussed in terms of (1) ^{6}Li or ^{6}LiD as the basic fuel material and (2) protons, deuterons, tritons, ^{3}He, and alpha particles as chain centers. The charged-particle chain reactions may be of importance to the controlled fusion research program, although it is recognized that severe technological difficulties exist at present (e.g., the synchrotron radiation problem and the more dilute plasma).

10. "A Novel Concept for Start-Up of Controlled Fusion Reactions," J. Rand McNally, Jr., Bull. Am. Phys. Soc. 14, 726 (1969), text available.

    Abstract: Injection of 10-mA, 4-MeV H_2^+ into a 40/20-kG magnetic mirror at $<10^{-7}$ torr will lead to a plasma density of 10^9 H^+/cm^3 at 1 MeV, limited primarily by stopping power losses to cold electrons. Addition of electron cyclotron heating will grossly reduce the energy drain on the fast protons and the density will exponentiate to 5×10^{12} at $T_e \sim 400$ keV. Closing of the magnetic mirror occurs at about 5×10^{12} H^+/cm^3 ($T_{e,+} \sim 1$ MeV) producing an I(ion)-current layer. ^6Li fuel fed to the hot proton plasma will ignite charged particle fusion chain reactions. The fuel feed (including ^3He, D, T) would then be programmed to lower the ion temperature and increase the ion density and hence the reactivity for DT. Eventually the fusion chain reactions must be replaced by the DT reactions for economic reasons.

11. "A Method for Start-Up of Controlled Fusion Reactions," J. Rand McNally, Jr., AEC-sponsored meeting on Fusion Reactor Technology, Madison, Wisconsin, April 1, 1970, text available.

Abstract: Injection of 4-MeV H_2^+ into a 40/20-kG magnetic mirror at $<10^{-7}$ torr will lead to a plasma density of 10^9 H^+/cm^3 at 1 MeV, limited primarily by stopping power losses to cold electrons. Programmed addition of electron cyclotron heating (ECH) at 5.5 mm will grossly reduce the energy drain on the fast protons and the density will gradually exponentiate to 5×10^{12} at $T_e \sim 1$ MeV. Closing of the magnetic mirror occurs at about 5×10^{12} H^+/cm^3 ($T \sim 1$ MeV) producing an I(ion)-current layer. ^6Li fuel fed to the hot proton plasma will ignite charged particle fusion chain reactions. The feed (including ^3He, D, T) could be programmed to lower the ion temperature and increase the ion density and hence the reactivity for DT, thus permitting a reduction in ECH and beam power. This approach appears to offer a technologically feasible test of the scientific feasibility of fusion start-up by injection-accumulation techniques involving a magnetic mirror and MeV energies.

12. Prospects of a Multi-MeV H_2^+ Injection-Accumulation Experiment, J. Rand McNally, Jr., Bull. Am. Phys. Soc. 15, 1440 (1970); ORNL/TM-3207, Oak Ridge National Laboratory, (November 1970).

Abstract: Crocker, Blow, and Watson discuss fusion reactors at $\sim$200 kG, ion temperatures $\sim$1 MeV, and lithium and deuterium as primordial fuel. The possibility of fusion start-up at MeV temperatures and more modest fields was discussed earlier. The prospects of a 2-mA, 4-MeV H_2^+ injection-accumulation experiment at 10^{-8} torr H_2 in a 40/20-kG mirror field with 5.5 mm ECH (electron cyclotron heating) will be presented.

Neglecting instabilities one should obtain an $n\tau \sim 5 \times 10^{14}$ sec/cm^{-3} before nuclear elastic scattering losses limit the exponential buildup. Such an experiment would test the basic principle of plasma exponentiation, which has yet to be demonstrated in injection-accumulation experiments.

13. "Fusion Chain Reactions — I," J. Rand McNally, Jr., Nucl. Fusion 11, 187 (1971).

    Outlined some of the reactions involved in prospective chain reactions using ^{6}Li and/or ^{6}LiH fuel.

14. "Fusion Chain Reactions — II," J. Rand McNally, Jr., Nucl. Fusion 11, 189 (1971).

    Continuation of Ref. 13 with emphasis on ^{6}LiD fuel.

15. "Fusion Chain Reactions — III, The Production of MeV Plasmas," J. Rand McNally, Jr., Nucl. Fusion 11, 191 (1971).

    Updated version of Refs. 11 and 12.

16. Nuclear Fusion Resonance Reactions of Possible CTR Interest, J. Rand McNally, Jr., ORNL/TM-3233, Oak Ridge National Laboratory (January 1971).

    Abstract: Some speculations are presented on possible nuclear fusion resonance reactions which may be of importance to the development of "clean" controlled thermonuclear reactors of either toroidal or mirror type.

17. "Speculations on the Configurational Properties of a Fusioning Plasma," J. Rand McNally, Jr., Nucl. Fusion 12, 265 (1972).

    Discussed the Ion-layer, E-core reacting plasma for *in situ* acceleration of ionized "cold" fuel ions to the plasma core and reactivity sustainment of the Ion-layer, E-core electromagnetic configuration.

18. Prospects for Alternate Fusion Fuel Cycles at Ultra-High Temperatures, J. Rand McNally, Jr., ORNL/TM-3783, Oak Ridge National Laboratory (1972).

    Abstract: Recent experiments and theory give support to the idea of developing a closed magnetic mirror by the self-field of a trapped ring current. Since the mirror reactor may be able to operate at much higher burning temperatures than toroidal reactors, the viability of charged particle fusion chain reactions is thereby increased and may permit the chain reaction burning of cheap Li or Be nuclear fuel. Reaction kinetics studies in partially closed mirrors show MeV energies for the light fusion reaction products which can act as catalysts or chain centers for the propagation of such reactions. The MeV particles may also permit sustainment of both the ring current configuration and the burning temperature. Numerous problem areas associated with this approach are tabulated and will require extensive research.

19. "Fusion, Nuclear," J. Rand McNally, Jr., p. 481 in Encyclopedia of Chemistry, 3rd ed., Van Nostrand Reinhold Company, 1973.

    Discussed the concept of nuclear fusion chain reactions.

20. "Nuclear Fusion Chain Reaction Applications in Physics and Astrophysics," J. Rand McNally, Jr., Nuclear Data in Science and Technology 2, 41 (1973).

    Abstract: Nuclear-fusion chain reactions have been proposed for supernovae and controlled fusion reactors. The concept of fusion chain reactions will be traced with emphasis on possible applications in physics and astrophysics. Reaction-kinetics calculations of D-T and D-^{3}He fusioning plasmas reveal the presence of suprathermal particles as the end products of the reactions. These suprathermal end products can be active chain centers for the propagation of energy-releasing fusion chain reactions with fuels having $Z \geqslant 3$. A ^{6}LiD-fueled fusion reactor has a calculated Lawson number, $n\tau$, lower by a factor of three compared to a pure-deuterium-fueled fusion reactor. The first ^{6}LiD-fueled fusion reactor will be sub-critical with the electrons sustained at MeV temperatures by electron cyclotron heating. If net power producers using ^{6}LiD-fuel are eventually feasible, they would be large in size, well reflected, and have a high beta ($\beta = 8\pi nkT/B^2$) in order to minimize synchrotron radiation losses. Closed magnetic configurations would also be required to ensure the propagation of fusion chain reactions. Higher-powered dc accelerators would be essential to exploit this field of controlled fusion reactors, although a scientific feasibility experiment may be executed with existing accelerators ($\sim$2-mA, 4-MeV H_2^+). Present-day astrophysical calculations appear to be inadequate to account for the rapid nuclear processes in astrophysical explosions inasmuch as such calculations neglect the presence of suprathermal chain centers and multiplying chain reactions. The 55-day light decay of Type-I supernovae may possibly be explained by the production of ^{7}Be (53.61 d) in chain-reaction burning of neon and oxygen followed by ^{7}Li-burning in the residual star. Evaluation of these new prospects will require more accurate data on and broader coverage of the pertinent nuclear reaction cross-sections for fuels Li-S and n, p, d, t, ^{3}He, α-particles as suprathermal (E up to 20 MeV) chain centers.

21. Reactivity of Advanced Fusion Fuels, J. Rand McNally, Jr., ORNL/TM-4647, Oak Ridge National Laboratory (July 1974).

    Discussed prospects for advanced fusion fuels in high β plasmas.

22. "Reactivity of Closed Fusion Reactor Systems for Advanced Fuels," J. Rand McNally, Jr., R. D. Sharp, R. H. Fowler, and J. F. Clarke, Nucl. Fusion 14, 579 (1974).

    Discussed preliminary theoretical results of burning requirements for the advanced fusion fuels DD and D^{6}Li.

23. Fusion Chain Reactor Prospects and Problems, J. Rand McNally, Jr., ORNL/TM-4575, Oak Ridge National Laboratory (July 1974).

    Abstract: Recent major developments in CTR research give support to the idea of developing a fusioning plasma in a magnetic mirror which is closed by the self-magnetic-field of a trapped ring current of ions (I-layer), with the electrons sustained at MeV energies principally by electron cyclotron heating (ECH). The viability of charged particle fusion chain reactions is thereby increased and sub-critical nuclear chain reaction burning of cheap ^{6}LiD nuclear fuel in a radiologically cleaner system may ensue. Reaction kinetics studies of D-^{3}He fueled plasmas in partially closed mirrors ($R = 10^4$) give MeV energies for the average energies of the light fusion reaction product ions which are the necessary catalysts or active chain centers for the propagation of such fusion chain reactions. In a closed magnetic mirror or I-layer the energetic charged reaction products may sustain both the I-layer configuration as a result of nutational motion and burning conditions, and may also lead to the sub-critical start-up of supplemental linear or toroidal multiple mirror reactors in which the electron temperature is also sustained by ECH in small systems. Larger systems would be required for net power producers if such are feasible. Some of the numerous problem areas associated with this approach are discussed and these will require extensive research and development.

24. "Advanced Fuels for Nuclear Fusion Reactors," J. Rand McNally, Jr., Proc. 3rd Conference on Application of Small Accelerators, p. 233 (1975).

    Discussed applications of high β reactors, such as the proton E-layer of Christofilos (1957) or the Ion-layer, to burning advanced fuels DD and D^{6}Li.

25. *Simplified Approach to Attaining a Proton E-Layer*, J. Rand McNally, Jr., ORNL/TM-4965, Oak Ridge National Laboratory (July 1975).

Abstract: A simpler approach than proposed by Christofilos for producing a large proton E-layer is presented. Such a high-beta closed-magnetic mirror configuration might permit burning the advanced fusion fuels DD and D^6Li in an economic, radiologically and chemically safer, structurally simple, reliable, steady-state, I-layer plasma-magnetic system.

26. *A Double Quantum Jump in CTR*, J. Rand McNally, Jr., ORNL/TM-4967, Oak Ridge National Laboratory (July 1975).

Reviewed the need for a double quantum jump in CTR to burn the advanced fusion fuel DD with emphasis on the MeV injection approach to demonstrate scientific feasibility of achieving large $Tn\tau$ values. Appendices present information on ameliorating the negative mass instability.

27. *DT and Advanced Fusion Fuels Computer Code*, R. D. Sharp and J. Rand McNally, Jr., ORNL/TM-5013, Oak Ridge National Laboratory (September 1975).

Summary: A simulation model has been developed to explore the prospects of using advanced fuels (D + D, D + ^{6}Li) for fusion reactors. The Etzweiler, Clarke, and Fowler (ECF) zero-dimensional code has been adapted to look at the reactivity of such fuels. The model consists of a system of simultaneous first order differential equations depicting particle densities and ion and electron temperatures. This model does not pretend to be the most efficient one possible, but it has given consistent and explainable results. The code can easily be modified to expand its range of reactions (e.g., ^{3}He + ^{3}He, P + ^{11}B, etc.).

28. "Advanced Fusion Fuels," J. Rand McNally, Jr., *Proc. 6th IEEE Symposium on Engineering Problems of Fusion Research*, p. 1012 (1976).

Presents results of parametric studies of the reactivity of DD and D^6Li advanced fusion fuels and discusses important control features.

29. "Advanced Fuels for Nuclear Fusion Reactors," J. Rand McNally, Jr., *Proc. Conf. on Nuclear Cross Sections and Technology*, NBS Special Publication 425, Vol. 2, p. 683, Boulder, Colorado, 1975.

Abstract: Should magnetic confinement of hot plasma prove satisfactory at high β ($16\pi nkT/B^2 > 0.1$), thermonuclear fusion fuels other than D•T may be contemplated for future fusion reactors. The prospe[illegible] of the advanced fusion fuels D•D and ^{6}Li•D for fusion reactors is quite

promising provided the system is large, well reflected and possesses a high β. The first generation reactions produce the very active, energy-rich fuels t and ^3He which exhibit a high burnup probability in very hot plasmas. Steady-state burning of D•D can ensue in a 60-kG field, 5-m reactor for $\beta \sim 0.2$ and reflectivity $R_\mu = 0.9$ provided the confinement time is about 38 sec. The feasibility of steady-state burning of $^6Li \cdot D$ has not yet been demonstrated but many important features of such systems still need to be incorporated in the reactivity code. In particular, there is a need for new and improved nuclear cross section data for over 80 reaction possibilities.

30. "Some Fusion Perspectives," J. Rand McNally, Jr., ERDA-ORAU Symposium on Energy Sources for the Future, CONF-760744 (April 1977).

    Abstract: This paper presents a review of fusion concepts including possible fuel cycles for both magnetic and inertial confinement. A discussion of mirror reactor possibilities is included.

31. "Mirrors: Past and Future," J. Rand McNally, Jr., Physics Today (August 1976).

    Discussed significant improvements in mirror plasma properties and prognostications for future devices such as the proton E-layer.

32. <u>The Ignition Parameter $Tn\tau$ and the Energy Multiplication Factor k for Fusioning Plasmas</u>, J. Rand McNally, Jr., ORNL/TM-5766, Oak Ridge National Laboratory (April 1977).

    Abstract: This paper presents some novel interpretations of fusion plasmas which may be of interest to both fission and fusion scientists and engineers. A new fusion ignition parameter $(Tn_e\tau_E)_I$ is proposed which is proportional to β^2B^4 and inversely proportional to the fusion power density (P_{Fusion}) of a reacting plasma. Curves are given for many potential nuclear fusion fuels. The energy utilization factor in existing devices is defined as $f = P_{Fusion}/P_{Loss} = (Tn_e\tau_L)/(Tn_e\tau_E)_I$; in experimental plasmas, f has increased by about two orders of magnitude in the past decade and now exceeds 10^{-4} (a "nearest" f^* exceeds 10^{-3}). The f factor is also analogous to its fission counterpart in the four-factor neutron multiplication factor $k = f\eta\varepsilon p$, where f is the neutron thermal utilization factor. Past, present, and future fusion experiments are discussed briefly in this context.

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The D-^{3}He Field-Reversed Mirror As A Minimum-Size Satellite

by

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ABSTRACT

Calculations indicate that D-^{3}He fueled Field-Reversed Mirror (FRM) reactors may be possible with power levels in the megawatt range using plasma volumes of tens of liters. Units could be stacked for higher powers. A bulk of the output power would be in the form of charged-particles and radiation with neutrons carrying < 7% of the power. This, combined with the elimination of tritium breeding requirements makes the design of a long-life, very low-radioactivity blanket possible. Consequently, the D-^{3}He FRM potentially offers a unique satellite (to a D-D or D-T ^{3}He generator) that could be placed in a variety of locations near small load centers such as manufacturing plants and community centers.

I. INTRODUCTION

While most conventional approaches to magnetic fusion lead to large-size power plants, the field-reversed mirror (FRM) potentially offers an attractive small size unit.[1] We have explored the possibility of capitalizing on this feature through development of a D-^{3}He-fueled FRM that would serve as a satellite to larger D-T or cat-D reactors.[3,4] To facilitate our study, we extended the plasma model of Condit et al[5] to include an explicit treatment of the "circulating" injected ions, a diamagnetic current estimate, and a pressure instability limited diffusion rate.

Interest in FRMs has intensified with promising high-β results from the 2X-II experiment at Livermore. The concept is not new but the approach is. As illustrated in Fig. 1, in the FRM, internal plasma currents, rather than single gyroradius high-energy electron or ion beams, maintain the magnetic field reversal. This makes initiation of reversal by neutral-beam injection at sub-MeV energies conceivable. The resulting closed field-line region provides improved confinement, while the surrounding

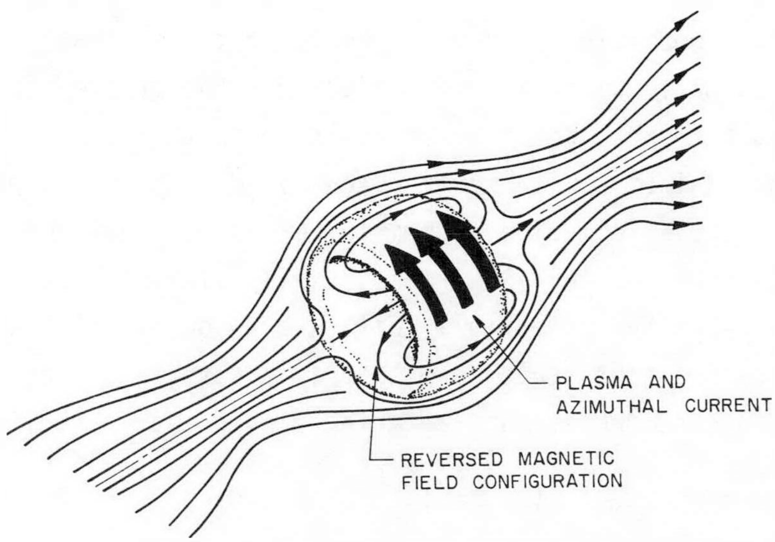

Figure 1. Field Reversal.

mirror field is compatible with electrostatic direct energy conversion. It is this compatibility with direct conversion processes that makes the FRM so attractive for burning advanced fuels. Thus, as shown later, good overall plant efficiencies can be achieved with a modest energy multiplication factor ($Q_p \sim 1$-5).

II. THE MODEL

To provide a consistent check on the field-reversal requirement it is necessary to calculate the net plasma current. The contribution from the injected ions is found by evaluating the contribution of contained orbits associated with superthermal ions prior to randomization by collisions. In addition to creating a current, these ions contribute to fusion by beam-target reactions. A diamagnetic current created by cross-field diffusion also contributes to field reversal, and the total current is the sum of this component plus the injected component. The ratio of the total current to the injected current is defined as γ_B, which typically ranges from 1 to 3. Because there remains an uncertainty about the type of diffusion to be expected in the FRM, a factor γ_c is also included, defined as the ratio of the diffusion coefficient employed to that for classical diffusion. A rapid turbulent diffusion is assumed to occur when the local β exceeds pressure balance requirements.

A key aspect of this modeling is the use of the Hill's vortex field (Fig. 2) to represent the reversed-field configuration.[6,7] While this analogy is not perfect, it is thought to be adequate for the study of injected orbits, diffusion currents, etc. that are of importance to conceptual design studies. The very important advantage this offers is that a simple analytic representation of the three-dimensional field of Fig. 2 is possible, and this allows rapid survey-type calculations.

For example, the Hill's vortex model has been used to classify injected-ion orbits which in turn enables an estimate of their effective contribution to the current causing reversal.[7] This theory relys on

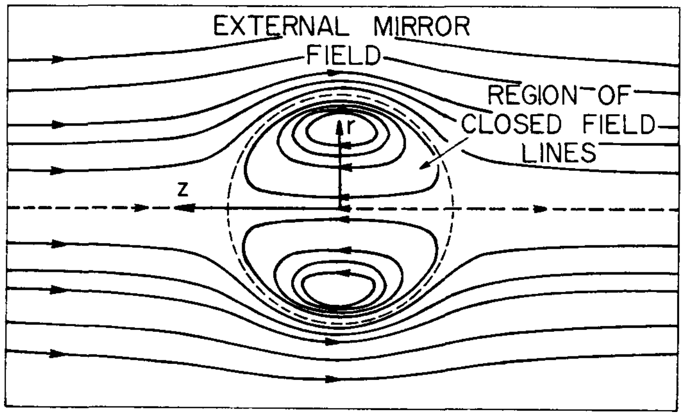

Figure 2. Hill's Vortex representation of field reversal.

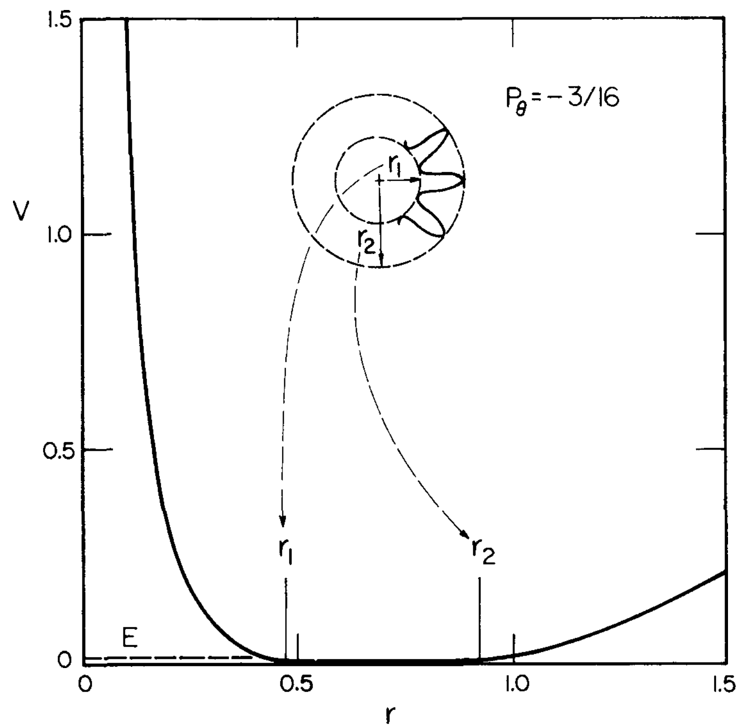

Figure 3. Orbital motion for particles trapped in the potential well at Z=0 with P_θ = -.1875.

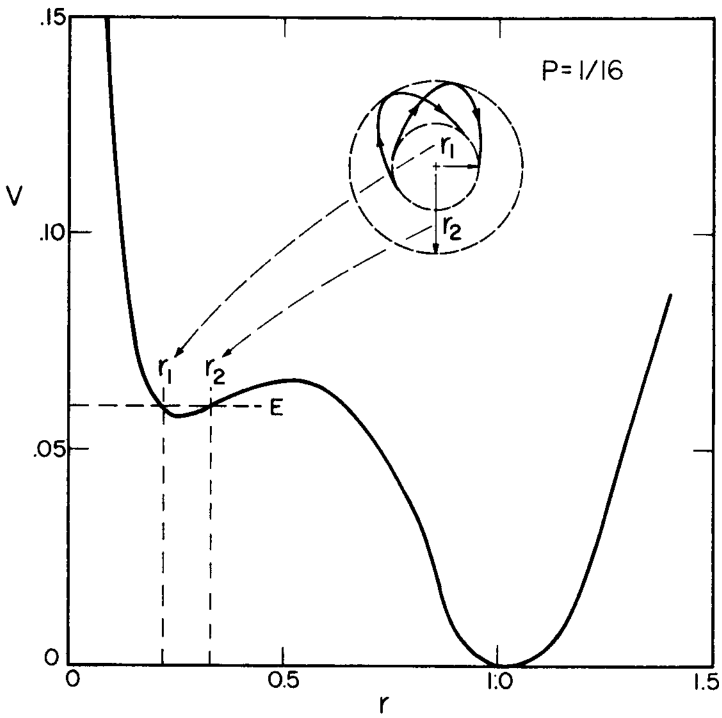

Figure 4. Orbital motion for particles trapped in the potential well at Z=0 with P_θ = .0625.

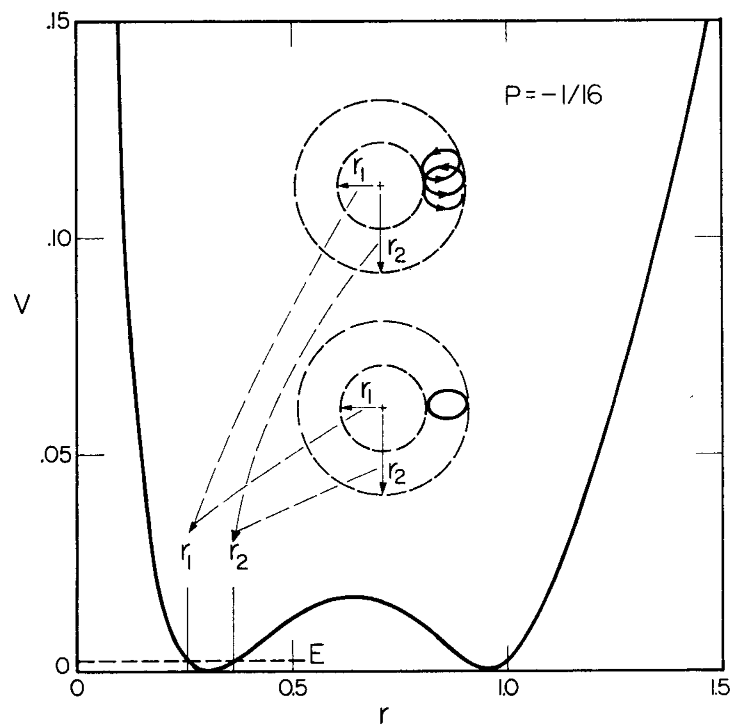

Figure 5. Orbital motion for particles trapped in the potential well at Z=0 with P_θ = -.0625.

an effective potential which can be derived from the vortex field. Then, as shown in Figs. 3-5, depending on the angular momentum P_θ and energy of the ions, the potential "well" they are trapped in changes as does their orbital motion. In addition to their contribution to the current causing reversal, these ions, with their large orbits, are responsible for stabilizing the FRM plasma.

III. NON-FUSION PRODUCT HEATED FRMs

Initial calculations have employed a "pessimistic" model in that no credit is taken for fusion-product heating. While it is thought that a reasonable faction (>>10%) of the fusion-product (3.5-MeV alpha and 14-MeV proton) energy will be retained despite the small size of the FRM,[8] the details of the heating profile and product build-up in the plasma have yet to be worked out. As shown later, the added heating can have a very significant beneficial effect; however, in view of the uncertainties involved, it is felt that the "pessimistic" no-heating model is of interest to establish lower limits.

Table 1 summarizes reactor parameters for three separate designs. Cases A and B illustrate the effect of injection energy, while cases B and C represent different reactor sizes. Case A is the most efficient with ~46% overall efficiency. This, however, is achieved using a relatively high injection energy of 600 keV. Case B (300-keV injection) gives an attractive reactor, but the efficiency falls to ~36%. The preceding designs have quite modest sizes, ~120 litre. (These calculations assume a field configuration is used such that a 3/1 elongation of the base Hill's spherical vortex is achieved.) Case C shows that even smaller units, ~50 litre, are possible, but at the sacrifice of efficiency.

Figure 6 shows the plasma output power split vs injection energy for Case C. A striking feature is the large fraction of output power carried by charged particles. The sum of leakage and charged fusion-power output is always $\gtrsim$70%, making it possible to obtain a reasonable overall efficiency despite the relatively low Q_p. The increased charged-particle power at

TABLE 1

Reactor Parameters for Three FRM Designs

	A	B	C
Vacuum Field (kG)	60	60	60
Injection Current (amps)	3.1	14.0	7.5
Injection Energy (keV)	600	300	300
Ion Temperature (keV)	102	68	77
Electron Temperature (keV)	65	48	48
Background Density (cm^{-3})	4.6×10^{14}	7.7×10^{14}	6.6×10^{14}
Circulating Density (cm^{-3})	5.7×10^{13}	6.4×10^{13}	8.8×10^{13}
γ_B	1.7	2.2	2.1
γ_C	21	35	31
Confinement Time (sec)	3.2	1.0	.72
Plasma Volume (litre)	118	118	50
Neutron Wall Flux ($cm^{-2}sec^{-1}$)	2.4×10^{13}	4.4×10^{13}	2.8×10^{14}
Fusion Power (MW)	1.7	2.9	1.1
Energy Multiplication (Q_p)	1.4	1.0	.72
Net Power Output (MW)	.79	1.05	.22
Overall Efficiency* (%)	46	36	20

*Assumes injection, direct conversion, and thermal effs of 80, 85, and 50% respectively.

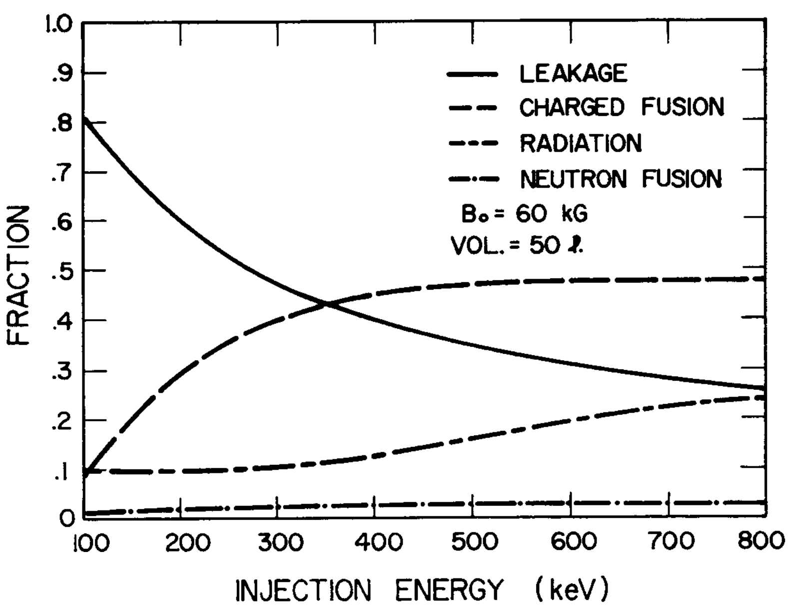

Figure 6. Plasma power output split vs. injection energy.

higher injection energies generally accounts for the increased overall plant efficiency for Case A of Table 1.

With neutron powers generally $\lesssim$7%, radiation emission becomes the dominant energy flux on the first wall. This, plus freedom from tritium breeding requirements, makes possible the use of a high-temperature helium-cooled graphite blanket. This, in turn, permits good thermal efficiency with minimum danger of radiation damage.

IV. FRM CONCEPTS WITH FUSION PRODUCT HEATING

Many of the high-energy fusion products produced in the FRM will have orbits that pass through the external open field lines. Some of these will naturally turn and pass back through the closed field line plasma, depositing a fraction of their energy there. Others will drift and scatter into the loss cone associated with the external region. This loss can be reduced to some extent by use of a larger external mirror ratio, e.g. rough calculations show that the fraction of fusion product energy retained can be more than doubled (base case ~10%) by increasing the mirror ratio from slightly over 1 to 2. Another consideration is that colder plasma in the outer field-line region may result in more deposition there.

Further, the orbits and fractional energy retained may be significantly different if units are stacked or if the closed field line region is elongated. These problems are now under study; however it is clear that some fusion product heating should occur, giving improved performance compared to the previous "pessimistic" case. If, in fact, methods to increase fusion-product retention such as increased external mirror ratios are found to be practical and are employed, preliminary estimates are that as much as 60% of the fusion product energy might be retained. Some results to illustrate the significance of this to the reactor design are given in Table 2.

Table 2: Fusion-Product Heated FRMs*

	Case D	Case E
Minor Radius, a, cm	10	15
Injection Energy, E_I, keV(a)	550	800
Injection Power, P_I, MW	0.26	0.64
Ion Temp., T_i, keV	60	53
Electron Temp., T_e, keV	50	47
Vacuum Field, B_o, kG	80	80
Plasma, Vol., V_p, ℓ(b)	4.2	14
Energy Multiplication, Q	3.1	4.0
Net Power Out, P_{Net}, MW(c)	0.18	0.70
Fractional Energy Release:		
Radiation, f_R, %	60	62
Chg. Particles f_{cp}, %	39	37
Neutrons f_n, %	1	1

*Assumes 60% of fusion product energy retained; 10% going to ions, the remainder to electrons.

(a) subscript I refers to injected beam: T_i and T_e to background plasma.

(b) based on a spherical vortex (non-elongated) volume.

(c) assumes, η'_{DC} = 60%, η'_I = 80%, and η_{th} = 40%.

Two designs are shown, Cases D and E, differing by mirror radius (10 and 15 cm, respectively) and by injection energy (550 and 800 keV, respectively). Also, an 80-kG field is used here, compared to 60 kG in the earlier designs.

Several important points are immediately observed. First, considerably higher energy multiplication factors (Qs) are obtained here, namely 3.1 and 4.0 for the two designs. This helps leviate concern about the large recirculation fractions associated with the "pessimistic" cases. A second observation, however, is that now the fractional power released with radiation is considerably higher. This mainly occurs because the fusion product heating increases T_e, which causes a rapid increase in cyclotron emission. This is not bad, but the fraction of the energy processed by direct conversion is decreased and the main route to high efficiency is shifted to obtaining a blanket design that effectively handles the radiation. A third difference, which is somewhat arbitrary at this stage, is that a purely spherical vortex plasma shape is assumed here, whereas the earlier results used an elongated shape. Consequently, the present volumes, hence absolute power levels, are even lower than for the "pessimistic" designs.

V. REACTOR CONCEPTS

The D-^{3}He reactor would closely resemble designs suggested by LLL workers[5] for D-T fueled FRMs. To provide an indication of the scale of the systems involved, a single cell lay-out is shown in Fig. 7.

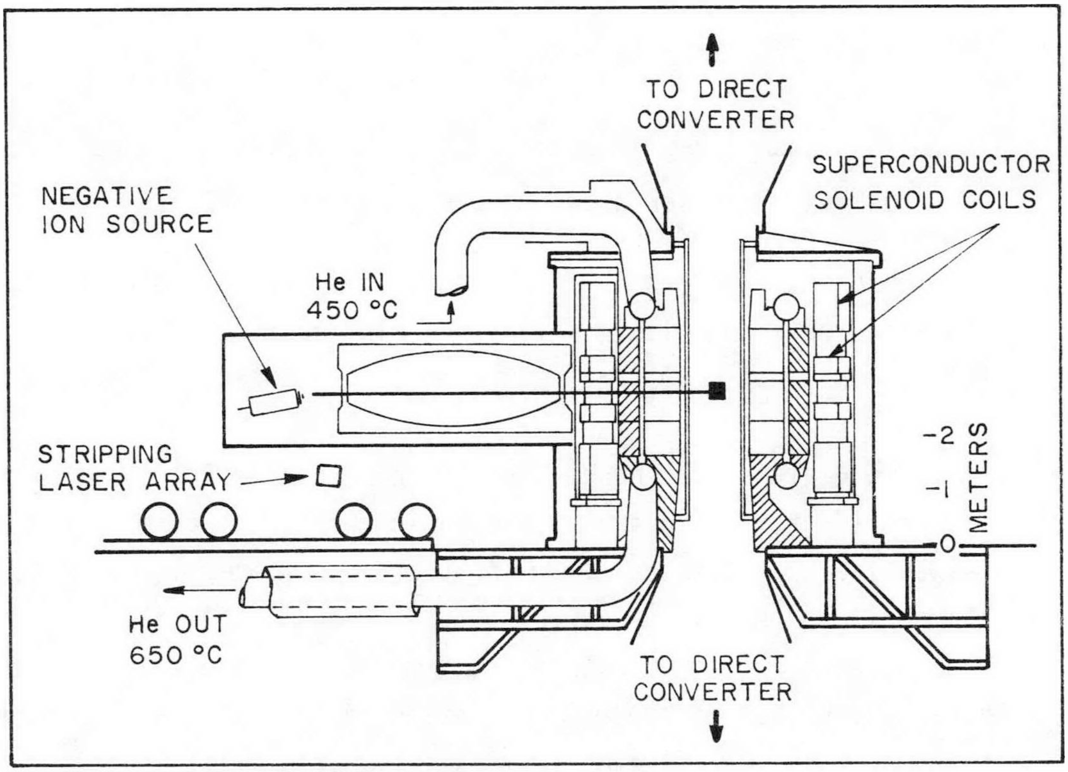

Figure 7. Conceptual FRM reactor design, based on Ref. 5.

Larger power outputs could be obtained by simply stacking such cells (see the 12-cell sketch in Ref. 5).

The key differences between the D-^{3}He and D-T designs are: 1) D-^{3}He requires higher injection energies, hence may be forced to use negative-ion injection techniques, 2) use of D-^{3}He places a premium on retention of a significant portion of the fusion product energy, thus may employ special features such as a higher mirror ratio, i.e. larger magnet structures, and 3) the D-^{3}He blanket can be greatly simplified by elimination of tritium breeding but needs to be designed to effectively handle large cyclotron and bremsstrahlung radiation loads.

VI. CONCLUSIONS

While field reversal has not yet been demonstrated experimentally, both 2X-II experiments and plasma simulation studies are very encouraging. The present conceptual design does not represent a large extrapolation in size, fields, etc. compared to next-generation mirror experiments. Consequently, if reversal is indeed demonstrated, concepts such as those proposed here must be viewed as realistic near-term objectives.

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Comments About p-^{11}B Ignition

by

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ABSTRACT

Ignited (or nearly-ignited) operation appears to be essential for a favorable energy balance when using p-^{11}B. Calculations show that even with optimistic cross sections and the elimination of cyclotron radiation, ignition requires an electron-ion temperature ratio < 1/5 and T_i > 120 keV. Such calculations are sensitive to both detailed numerical methods and the plasma modeling; consequently, it is suggested that interested workers set up a benchmark problem so that various approaches can be intercompared.

I. INTRODUCTION

The p-^{11}B reaction:

$$p + {}^{11}B \rightarrow 3\alpha + 8.7\ \text{MeV} \tag{1}$$

perhaps represents the "ideal" fusion reaction since the fuel is plentiful and neutrons are largely eliminated. [Some neutron and gamma-radiation are still obtained from "side" reactions, but yields are low, < 1 per 10^3 reactions [1].] In view of its attractiveness, there is a strong motivation to find ways to ignite p-^{11}B. However, a central difficulty is the increased radiation emission due to the high Z of boron. With the enhanced bremsstrahlung that occurs, any confinement system that results in significant cyclotron emission probably should be ruled out. Thus, Weaver et al.[2] originally suggested laser-pellet approaches while Dawson[3] has proposed surface-field confinement such as achieved in a Surmac. Another approach, first explored by Bathke et al.,[4] that uses two-component fusion with a cold ^{11}B target plasma to suppress radiation looks less promising because the short slowing-down time of protons in the target limits the energy multiplication.

In summary, while there is no assurance that an attractive confinement scheme can be found for burning p-^{11}B, some important approaches have been suggested. The two most attractive, pellet microexplosions and surface confinement, rely on ignited or near-ignited operation. Consequently, a thorough understanding of the physics of p-^{11}B ignition is basic to its continued development. Due to the high-Z, high-temperature plasma plus possible non-thermal effects, ignition calculations are considerably more complicated than for normal D-T plasmas. The present discussion will concentrate on an idealized model where plasma losses, i.e. details of the confinement geometry are neglected. It is proposed that such a model be employed by workers in the field to intercompare calculational techniques before more complicated device-oriented studies are undertaken.

II. ENERGY BALANCES AND THE RADIATION PARAMETER ψ_R

The parameter of interest for ignition calculations is the *radiation parameter* ψ_R.[5] It represents the fraction of the charged fusion-product power deposited in the plasma that *remains* after radiation losses are accounted for, i.e.,

$$\psi_R \equiv (1-\chi_R)f_c \qquad (2)$$

where f_c is the fraction of the fusion energy originally carried by charged particles ($f_c \sim 1.0$ for p-^{11}B) and χ_R is the fraction of this energy radiated away by the plasma.

With this definition, ignition against radiation losses is simply described by

$$\psi_R \equiv 0. \qquad (3)$$

Then, the charged fusion-product power just balances bremsstrahlung plus cyclotron losses (as noted earlier, the present discussion ignores additional losses due to leaking plasma, thermal conduction, etc. -- i.e., this can be viewed as a minimum criterion for ignition).

One reason ψ_R is so convenient for discussing ignition is that it enters naturally into overall energy balances. Thus, as shown in Ref. 6, the overall fusion plant efficiency η_o can be written as:

$$\eta_o = (1-\psi_R)\eta_{th} + \psi_R\eta'_{DC} - \frac{1}{Q_p}\left(\frac{1}{\eta'_I} - \eta'_{DC}\right) \qquad (4)$$

where η_{th}, η'_I, and η'_{DC} are the thermal recovery, injector, and direct conversion efficiencies, respectively (primes indicate inclusion of bottoming cycle efficiencies),[6] and Q_p is the plasma energy multiplication factor. In this formulation, ψ_R and Q_p are functions of the type of fuel and confinement system while the η's depend on external energy handling systems.

III. EVALUATION OF ψ_R

To simplify matters, we assume steady state operation such that:

$$\psi_R \sim \left(1 - \frac{P_R}{P_C}\right)f_c \qquad (5)$$

where P_R and P_C are the radiation and charged particle powers, respectively. Using the nomenclature of Ref. 5, this can be written as

$$\psi_R \sim \left[1 - \frac{cT_e^{1/2} + d\,\frac{1-\beta}{\beta}\,K_cT_e^2}{b\langle\sigma v(T_i)\rangle_{jK}(E_c)_{jK}}\right] f_c \qquad (6)$$

where the term involving $T_e^{1/2}$ accounts for bremsstrahlung radiation while the term with T_e^2 accounts for cyclotron emission. In the latter,

$(\frac{1-\beta}{\beta})$ corrects for exclusion of the confining magnetic field by the plasma and K_c gives the portion of the emitted radiation that is actually absorbed in the wall after reflection and reabsorption are accounted for. In subsequent graphs, $(\frac{1-\beta}{\beta})K_c$ is treated as a parameter. Note that for values of $(\frac{1-\beta}{\beta})K_c < 10^{-3}$ cyclotron emission is small, between 10^{-3} to 10^{-1} emission is normal, and $> 10^{-1}$ it is large.

The fusion power enters through the term involving $<\sigma v(T_i)>$ which brings in the fusion cross section σ. Since fusion rates involve T_i, but radiation depends on T_e, the ratio T_e/T_i becomes an important parameter.

Calculated values of ψ_R for p-^{11}B are shown in Fig. 1. (For comparison, an equivalent plot for D-T is included as Fig. 2.) These calculations, done in 1975(5), used optimistic p-^{11}B fusion cross section, labeled set A in Fig. 1, from Ref. 3, and more pessimistic values from Ref. 7 for set B. Fortunately, as discussed in the present meeting, recent measurements(8) are in better agreement with Set A than B. Consequently, Fig. 1 suggests that ignition with p-^{11}B may be possible, but under quite restrictive conditions, namely:

- $T_e/T_i < 1/5$ (7a)
- $120 < T_i < 400$ (7b)
- $(\frac{1-\beta}{\beta})K_c < 10^{-3}$ (7c)

In other words, with $T_i > 120$, not only must cyclotron emission be suppressed but the electron temperature must be maintained well below T_i. Consequently, it must be concluded that ignition is possible but very difficult to achieve.

It should be stressed, however, that the calculations of Fig. 1 contain several key assumptions. First, both the ion and electron distributions are treated as Maxwellians. In fact, with the high-Z ^{11}B plasma, strong radiation emission, and MeV-alpha population involved, deviations from Maxwellian behavior can be anticipated. Further, relativistic corrections to radiation emission can be important at the temperatures involved. Consequently, an accurate evaluation of ψ_R requires a plasma model that, among other things, imploys the following:

- accurate fusion cross sections
- allowance for non-Maxwellian populations

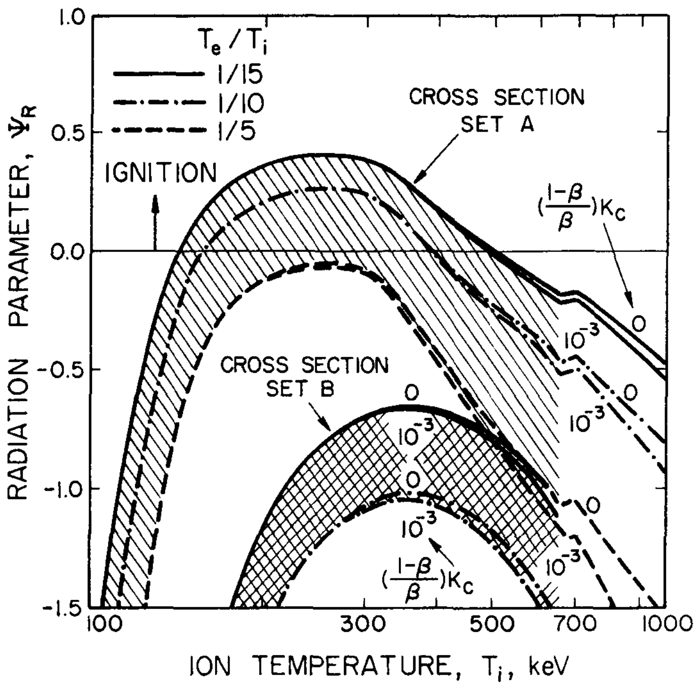

Figure 1. Radiation Parameter vs. T_i for p-^{11}B

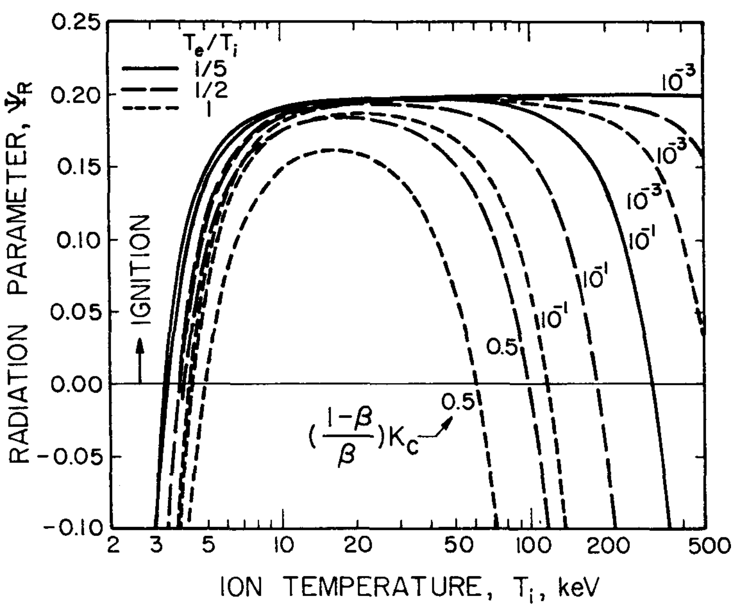

Figure 2. Radiation Parameter for D-T.

- allowance for non-thermal effects, including nuclear-elastic collisions
- relativistic corrections

IV. SUB-IGNITION OPERATION

If all of the output energy for a fusion plant is recirculated to drive injectors and heat the fuel, $\eta_o = 0$. Thus solving Eq. (4) for Q_p with $\eta_o = 0$ represents the minimum energy multiplication $(Q_p)_{min}$ possible with a given set of η's. This corresponds to

$$(Q_p)_{min} = (\frac{1}{\eta_I'} - \eta_{DC}')/\left[(1 - \psi_R)\eta_{th} + \psi_R \eta_{DC}'\right] \equiv (\frac{1}{\eta_I'} - \eta_{DC}')/\Delta \quad (8)$$

which is plotted in Fig. 3 for various combinations of η's.

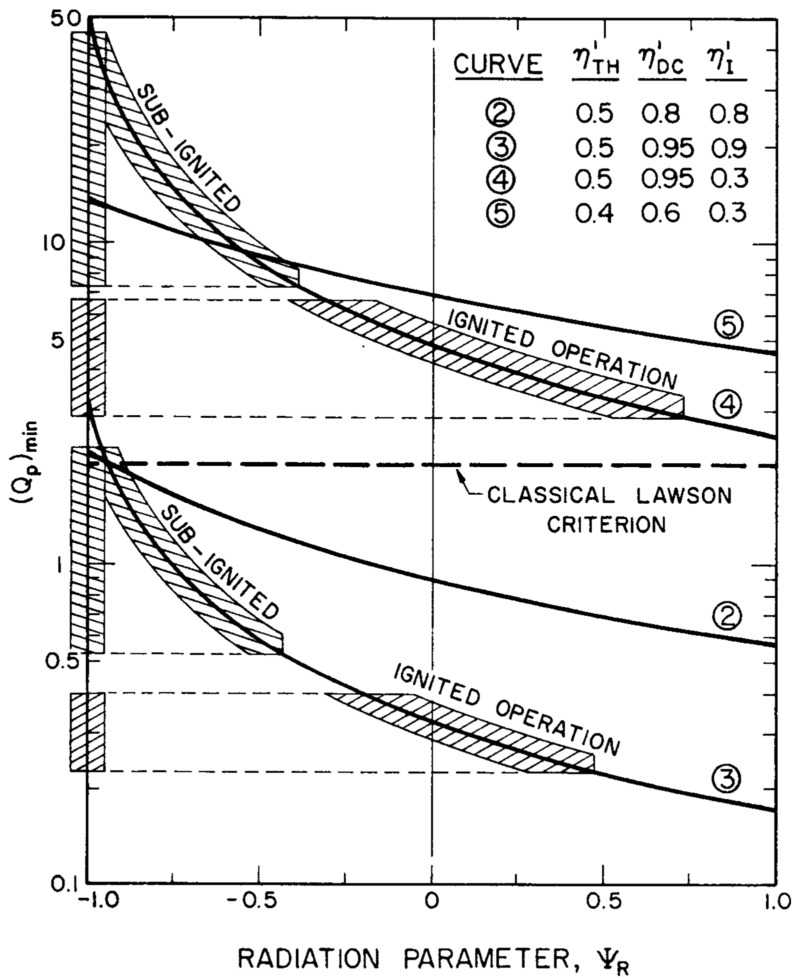

Figure 3. Variation in the minimum Q_p-value with ψ_R

For reference note that $(Q_P)_{min} = 2$ corresponds to the classical Lawson criterion ($\eta'_{DC} = \eta_{th} = 1/3$; $\eta'_I = 1$). Consider, for example, curve 3 in Fig. 3 which corresponds to relatively high η'_{DC} and η'_I. For ignited operation ($\psi_R = 0$) we see that $(Q_P)_{min} \sim 0.3$. However for sub-ignition where $-0.5 > \psi_R > -1.0$, we see that $3.0 > (Q_P)_{min} > 0.5$. Not only is this a significant penality in Q_P requirement, but the components required to handle the large recirculating powers are likely to be costly. [Note that for $\eta_o > 0$, the Q_P-value must be raised above $(Q_P)_{min}$ by the factor $(1- \eta_o/\Delta)^{-1}$. Also, for *actual* ignition where added losses such a plasma leakage occur, it will be necessary to achieve $\psi_R > 0$.] A similar argument holds for other situations, e.g. curve 4, with lower "injector" efficiency, better represents laser-pellet fusion. Then $(Q_P)_{min} \sim 5$ for $\psi_R = 0$ but $(Q_P)_{min} > 8$ for $\psi_R < -0.5$.

While these values of Q_P may seem modest compared to D-T requirements, it is difficult to achieve values of $Q_P > 0.1$ for p-^{11}B without resorting to Maxwellian-type fusion in near-ignited configurations. For example, Bathke, et al.(4) showed that beam-target (two-component fusion) systems would require ^{11}B target plasmas with $T_e > 1$ keV to achieve $Q_p > 0.1$. Such hot targets, however, result in serious radiation emission problems.

V. SUMMARY

In conclusion, p-^{11}B operation in a near-ignition condition seems imperative. The feasibility of ignition can be understood from ψ_R-plots, but these calculations are very sensitive to both cross-section data and calculational techniques. Since preliminary results indicate the prospects for ignition are marginal, it is important to further refine calculational methods. To do this, tests using a benchmark problem were proposed by the author during the EPRI review meeting.

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Reactor Technology - Power Conversion Systems and Reactor Operation and Maintenance

by

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ABSTRACT

The use of advanced fuels permits the use of coolants (organic, high pressure helium) that result in power conversion systems with good thermal efficiency and relatively low cost. Water coolant would significantly reduce thermal efficiency, while lithium and salt coolants, which have been proposed for DT reactors, will have comparable power conversion efficiencies, but will probably be significantly more expensive. Helium cooled blankets with direct gas turbine power conversion cycles can also be used with DT reactors, but activation problems will be more severe, and the portion of blanket power in the metallic structure will probably not be available for the direct cycle, because of temperature limitations.

A very important potential advantage of advanced fuel reactors over DT fusion reactors is the possibility of easier blanket maintenance and reduced down time for replacement. If unexpected leaks occur, in most cases the leaking circuit can be shut off and a redundant cooling curcuit will take over the thermal load. With the DHe^3 reactor, it appears practical to do this while the reactor is operating, as long as the leak is small enough not to shut down the reactor. Redundancy for Cat-D reactors has not been explored in detail, but appears feasible in principle. The idea of mobile units operating in the reactor chamber for service and maintenance of radioactive elements is explored.

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I. POWER CONVERSION SYSTEMS

The coolant outlet temperature (∿400°C) for the DHe^3 reactors corresponds to a gross power conversion efficiency of ∿37% if a standard steam cycle is used, comparable to that projected for the HWOCR (Heavy Water Moderate Organic Cooled Reactor). Lewis [(1)] has calculated that a gross cycle efficiency of 44% is achievable by an additional stage of reheat during the steam expansion, with a top organic temperature of 410°C and more efficient design of the heat exchange circuit. Net cycle efficiency (exclusive of power requirements for the refrigerators for the magnet system and the OH coils) would be ∿1% lower. The Lewis estimate corresponds to a cycle that achieves 80% of Carnot efficiency. While achievable, the cost effectiveness of such a cycle has to be evaluated, and this requires detailed cost estimates of the cycle as a function of efficiency, as well as cost estimates for the reactor.

The helium outlet temperatures for the Cat-D reactor blankets range from 800°C to 1000°, which would give a net cycle efficiency of ∿38-40% with a standard steam cycle. The inlet helium temperature was 400°C for all cases studied. With an inlet temperature of 800°C and outlet temperature of 1000°C, which appears feasible, a direct cycle He gas turbine powe conversion system can be used, comparable to that proposed for the HTGR.[(2] The net cycle efficiency is essentially the same, ∿40%, but there are important cost and environmental advantages. The cost of the direct cycle conversion machinery and heat exchange surface should be substantially less than that for the steam cycle. The intermediate heat exchanger between helium and steam could be eliminated. A high temperature regenerative heat exchanger is required for the direct cycle but the helium working fluid is inert.

The He circuit with its turbine, compressor, and heat exchanger can be essentially completely free of radioactivity, since no tritium is released to the helium circuit and the SiC tubes do not significantly activate, so that no radioactive corrosion or erosion products can be carried along by the helium.

The high reject temperature of the direct cycle (maxium reject temperature is ∿180°C) allows the use of compact, cheap dry cooling towers the reject heat is dumped at the power station. If desired, an additional $\sim$10% efficiency (i.e., a combined cycle efficiency of $\sim$50%) could be obtained with an organic bottoming cycle operating on the reject heat. However, a better mode of use for the reject heat is probably as a high-grade hot water source for process heat and district heating/cooling applications. Earlier detailed studies of district heat from fusion reactors [3] indicate that substantial cost benefits result if waste heat is used for district heating. Reject heat temperature of $\sim 100^{o}C$ are suitable for hot water district heat; the portion of reject heat between 100 and $180^{o}C$ could be used for various process heat applications and district cooling with absorption type air conditioners.

In summary, the use of advanced fuels permits the use of coolants (organic, high pressure helium) that result in power conversion systems with good thermal efficiency and relatively low cost. Organic coolants cannot be used for DT reactors. Water coolant would significantly reduce thermal efficiency, while lithium and salt coolants, which have been proposed for DT reactors, will have comparable power conversion efficiencies, but will probably be significantly more expensive. Helium cooled blankets with direct gas turbine power conversion cycles can also be used with DT reactors, but activation problems will be more severe, and the portion of blanket power in the metallic structure will probably not be available for the direct cycle, because of temperature limitations.

II. REACTOR OPERATIONS AND MAINTENANCE

A very important potential advantage of advanced fuel reactors over DT fusion reactors is the possibility of easier blanket maintenance and reduced down time for replacement. The blankets described in a comparison paper should last for the 30 year plant life without scheduled replacement. If unexpected leaks occur, in most cases the leaking circuit can be shut off and a redundant cooling circuit will take over the thermal load. With the DHe^3 reactor, it appears practical to do this while the reactor is operating, as long as the leak is small enough not to shut down the reactor. Redundancy for Cat-D reactors has not been explored in detail, but appears feasible in principle.

If two circuits in the same module leak, which seems to be very unlikley, or if a module is damaged by an unexpected plasma dump, module replacement will be necessary. Such events are expected to be rare, i.e., a few times a year at most. The replacement of failed modules can be carried out by an operator guided mobile service unit which enters the toroidal vacuum chamber and removes the module in a short time, i.e., a day or so.

The idea of mobile units for service and maintenance of radioactive elements of a nuclear plant is not new. A mobile service unit (known as the Beetle) [4] was constructed for servicing ANP (Aircraft Nuclear Propulsion) reactors. The Beetle weighed 85 tons, due to the high level radiation shielded cab for the operator, and the very high power propulsion unit. The operator had 12 inches of lead shielding, and 2 foot thick lead windows. The vehicle was a track laying machine (i.e., a caterpillar tractor) with ability to turn, go straight ahead, or revolve on its own axis, as required.

In the advanced fuel reactors considered in this study, the blanket is modular, with the modules either being graphite blocks with internal coolant tubes, or aluminum plates with coolant passages. To replace modules which have failed, the MSU (Mobile Service Unit) would enter the toroidal plasma chamber, disconnect the cooling tubes, remove the modules, and transport them out of the toroidal plasma chamber. The MSU would also transport in replacement modules, reinstall and reconnect all cooling lines, test them, and then withdraw from the plasma chamber so that operation could resume.

The Beetle was a rather massive machine, (85 tons); the MSU for advanced fuel fusion reactors should be considerably lighter. For advance fusion reactors, the operator's cab will require only modest shielding, since the radiation level in the shutdown plasma chamber is relatively low. Based on 1D calculations using a 100 group ANISM P_3S_8 model with ENDF IV-B cross sections and the activation decay chain code WEB (a BNL Code), exposure rates for Cat-D reactors with a graphite blanket and 0.1 ppm each of the activating impurities (Mn, Fe, etc.) will result in doses to unprotected personnel of $\sim$100 mr/hour. Two to three inches of lead shielding should attenuate radiation sufficiently to allow unlimited working time inside the cab. The reduction from 12 inches of shiel[illegible] to 2 inches will greatly reduce the cab weight. For DHe^3 reactors with aluminum blankets, the impurity activation will cause a dose of ∿30 mr/hour to unprotected personnel inside the plasma chamber if the aluminum impurity level is kept at ∿0.2 wppm each of the principal activating elements (Mn, Ti, Fe, Zr. etc.). This assumes that the Na^{24} activity has decayed, which will require a shut down period of ∿10 days before the plasma chamber can be entered. The dose from Al^{26} would be small until after a few years of reactor operation. After 30 years of operation, the dose from Al^{26} activation in DHe^3 reactors will be ∿100 mr/hour. The proposed MSU could service both catalyzed DD and DHe^3 reactors.

The dose rates are low enough with advanced fuel fusion reactors to permit limited direct (i.e., unshielded) access to the plasma chambers if necessary. Non-routine or unexpected operations can be carried out directly by the operator who could leave the shielded cab for extended intervals. Assuming that the operator received 1 R/year (20% of the maximum dose permitted by 10CFR20), an operator could work outside the cab for approximately 10 hours/year. Such direct access capability could be very important for solving unexpected problems. With DT fueled reactors with activating structure, the radiation dose would be too high to permit any direct access to the plasma chamber. In fact, the dose rates would probably be too great to permit even shielded cabs to operate inside the toroidal vacuum chamber, and blanket repair/maintenance work would have to be done fully remotely. This, combined with the much more frequent blanket replacement required for DT reactors, should result in considerably higher plant factors for advanced fuel reactors.

An important feature of the MSU is the weight of the unit. The physical dimensions must be small enough to fit into a 10 ft wide by 30 ft high toroidal vacuum chamber. This limits the width of the unit to about 8 ft, and with the cab or payload in an elevated position, makes it vulnerable to tipping. The maximum module weight to be handled by the MSU appears to be about 5 tons. For stability, it would appear that a vehicle weight of say 25 tons would be necessary in order to prevent tipping by an offset weight. The other operations to be handled by manipulators, such as unscrewing tubing, cutting or repairing tubing, unscrewing manhole and flange studs, inserting locking pins, and the like, require light loads which would not disturb the main vehicle. The MSU will, of necessity, have a fork-lift feature which apparently would have to overhang the wheels or tracks, and thus tend to tip the unit over unless there were adequate counterbalancing weight. The fork lift feature should be able to lower onto its own floor a top or side section of the blanket, or to reach beyond the track area and pull up a section of the blanket floor. The tracked vehicle feature of the Beetle appears well suited to the MSU, since the load on the blanket floor would be spread over a large area. If the two tracks of the MSU were 10 ft long by 2 ft wide, the bearing pressure on the blanket floor, assuming uniform distribution, would be only 10 psi which should not damage the blanket.

The weight of the MSU can be reduced from the 85 ton Beetle weight by thinner shielding, and substitution of hydraulic motors for the 550 Hp self-contained power unit used in the Beetle. The power needs of the MSU for the advanced fuel reactors can be met by hydraulic transmission through a flexible spooled line of approximately two inches ID. This line would connect the MSU with the hydraulic pumps and power source outside the blanket/shield assembly. Air and electrical connections could be also spooled to the MSU through flexible lines. A 25 ton weight for the MSU seems readily achievable, given the thinner shielding and external power source.

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Bundle Divertor Designs for Attaching Direct Convertors to the "ILB" Advanced Fuel Tokamaks*

by

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ABSTRACT

Extensive parametric surveys of bundle divertor scaling have been conducted. These have resulted in attractive divertor designs for attaching direct convertors to the "ILB" CAT-D and D-^{3}He fueled Tokamak reactors.

Three conductor types were examined: warm copper, cryogenic aluminum, and Nb_3Sn superconducting. In general, the cryogenic aluminum bundled divertor designs were the most attractive, resulting in low coil power consumption, smaller size, and minimal disturbance of the bulk plasma. It appears that, for advanced fuels, the bundle divertor is a sensible alternative to other divertor types.

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I. INTRODUCTION

Of the various approaches to fusion, Tokamaks appear to hold the potential for the highest Q values. In considering the use of advanced fusion fuels, direct conversion of the large fraction of power carried by charged particle seems an appropriate means to achieve high efficiency reactors. Coupling direct converters to a Tokamak is, however, a non-trivial task.

The ILB[1,2] CAT-D and D-^{3}He fueled Tokamak reactors achieve ignited operation. A divertor is proposed to carry the leaking charged particles from the plasma chamber to an expansion chamber leading to the direct convertors. A natural candidate for this divertor function is the "bundle divertor".

The bundle divertor provides vacuum pumping and fuel recovery requirements (in conjunction with the expansion chamber) while providing the connection for a direct convertor. Confidence in this concept is bolstered through the experimental successes of bundle divertor operation on DITE at Culham.[3]

II. BUNDLE DIVERTOR SCALING

The bundle divertor is a "local" divertor as illustrated schematically in Figure 1. This is the reason why it is a natural choice for use in connecting a Tokamak to a direct convertor. This is well illustrated by Figure 2 which shows a poloidal divertor and direct collector. A bundle divertor and direct collector are illustrated in a companion paper.[4] In this role, the bundle divertor serves as one of five major components:

(1) the Tokamak, where the fusion reactions occur,
(2) the bundle divertor, which directs some of the field lines out of the Tokamak,
(3) the toroidal field nulling coils, which extend the diverted field lines out beyond the toroidal field coils,
(4) the expansion chamber, where the charged particles' energy becomes more monodirectional, and

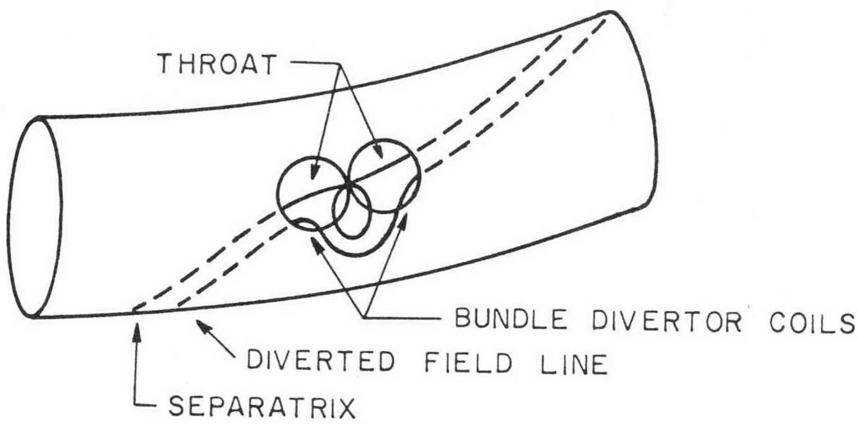

Figure 1. A Schematic of a Bundle Divertor

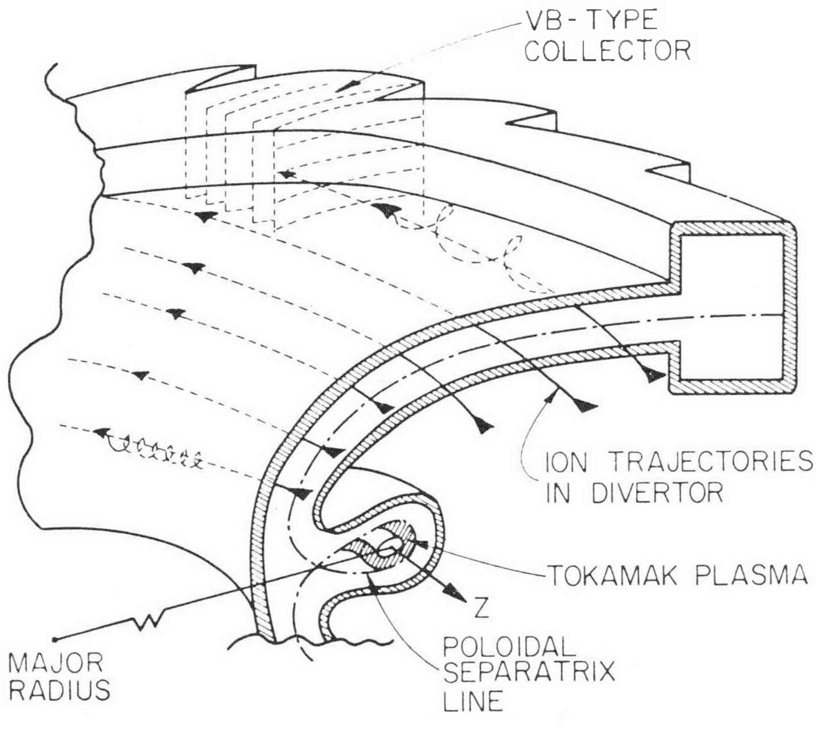

Figure 2. A poloidal divertor/direct convertor from Miley[7] based on concepts initially suggested by Yoshikawa[8]. The toroidal field coils are not shown nor is it clear how they can be integrated.

(5) the direct collector, where the charged particles' kinetic energy is converted into electricity.

Not pictured in Figure 1 are the components 3, 4, and 5. When these 3 components are added, the loop of the diverted magnetic field line shown is greatly extended.

Figure 3 illustrates how the magnetic field lines of the bundle divertor coil add to the field lines in the Tokamak to produce the result of diverting some lines out of the Tokamak. In practice, component 3, the toroidal field nulling coils, would create an additional magnetic field which has not been illustrated.

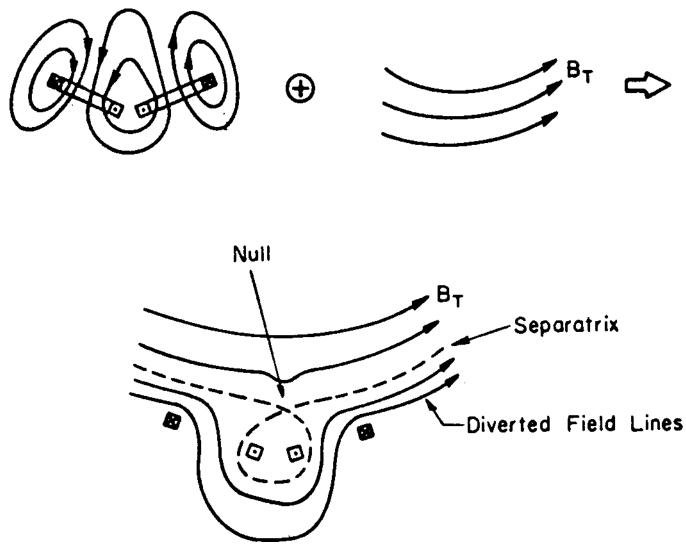

Figure 3. The resulting magnetic field obtained by adding the field of the divertor coils plus the field inside a Tokamak.

There is little previous work on bundle divertor scaling. Extending earlier work at the University of Illinois[5] and Culham Laboratory[6], a model of the bundle divertor was developed. This model was then used to study three types of divertor: warm copper, Nb_3Sn superconducting, and cryogenic aluminum. For simplicity, the effects of components 3-5 were not considered in the model.

It was realized that a major constraint would be shielding thickness since neutrons would be very detrimental to all the coil types in terms of heating as well as changing the current-carrying properties. The importance of minimum shield thickness is that the cross-section of the two divertor coils must fit inside the tear-drop shaped separatrix (the dashed line in Fig. 3). Actual coil cross-sections are bigger than is shown in Fig. 3 since the B-field to be diverted is quite large (> 50 kG). Therefore, an auxiliary study was conducted[7] so that minimum shield thicknesses for each case could be estimated.

Other important constraints were:

1) power consumption (for normal conductors),
2) ripple on the plasma centerline,
3) redundancy,
4) maximum field strengths less than 15 Tesla, and
5) coil size small enough to fit between toroidal field coils.

Power consumption consisted mainly of ohmic losses in the warm copper case and refrigeration in the cryogenic aluminum case. Ripple is shown in Figure 3, but it is not known how much can be tolerated. One or two percent seems quite reasonable, however. Although one divertor is clearly the optimum number (in terms of minimizing power consumption and total materials, for example), more than one might be needed in practice. Perhaps three bundle divertor/direct collection systems would be needed on a reactor that would normally use two of them. Finally, 15 Tesla was felt to be the technological limit that would be applicable.

III. REFERENCE DIVERTORS

Using the model previously discussed, optimum divertor designs were drawn up for five ILB reactors. The reactors chosen were the three D-^{3}He reactors and the two Catalyzed D reactors. Table I shows parameters for the two cases which bracket the five reactors: HHβ and CLβ. These reactors bracket the others since CLβ has the most neutrons and highest magnetic field while HHβ is the approximately

Table I-A. Single* Bundle Divertor Parameters For ILB-HHβ Reactor

Conductor Type	Angle Between Coils	Coil Radius (cm)	Cross-Sectional Area (cm^2)	Required Shielding (cm)	Power Consumption (MW)	Ripple	Surface Field (T.)	Scrapeoff Techniques (cm)	$\eta_{D.C.}^{(c)}$	$\eta_{th}^{(d)}$
Warm Copper	170°	70	1100	20	245	1.0%	12	24	37%	40%
Superconducting(a)	120°	186	20,000	40	-	7.8	15	16	49%	40%
Cryogenic Aluminum	160°	90	2300	30	58(b)	2.0	14	22	46%	40%

Table I-B. Single Bundle Divertor Parameters For ILB-DLβ Reactor

Conductor Type	Angle Between Coils	Coil Radius (cm)	Cross Sectional Area (cm^2)	Required Shielding (cm)	Power Consumption (MW)	Ripple	Surface Field (T.)	Scrapeoff Techniques (cm)	$\eta_{D.C.}^{(c)}$	$\eta_{th}^{(d)}$
Warm Copper	170°	125	4300	30	542	1.0%	15	25	41.4%	40%
Superconducting(a)	160°	261	30,000	60	-(a)	6.3%	15	17	46%	40%
Cryogenic Aluminum	170°	160	7200	40	110(b)	2.0%	15	22	45%	40%

* assumes reactor is operated with one bundle divertor in use

(a) assumes a current density of 2000 A/cm^2 and that refrigeration power is negligible

(b) assumes refrigeration power of 100 watts/watt

(c) η_{DC} ≡ efficiency with direct collection assuming that total direct collection efficiency is 62% and that thermal conversion efficiency is 40%:

$$\eta_{DC} \equiv \frac{(0.62)P_{CP} + (0.4)/(P_{th} - P_{CP}) - P_{div}}{P_{th}}$$

where P_{CP} ≡ power of leaking charged particles; P_{th} ≡ reactor thermal power; and P_{div} ≡ power consumed by divertor (column 5).

(d) η_{th} ≡ efficiency without direct conversion. (Note: if η_{th} is lowered, both the last two columns are lowered, but the last column goes down more.)

Table II-A. Single* Divertor Parameters For ILB-HLβ Reactor

Conductor Type	Angle Between Coils	Coil Radius (cm)	Cross-Sectional Area (cm^2)	Required Shielding (cm)	Power Consumption (MW)	Ripple	Surface Field (T.)	Scrapeoff Techniques (cm)	$\eta_{D.C.}^{(c)}$	$\eta_{th}^{(d)}$
Warm Copper	170°	77	1500	20	342	1.0%	15	22	30%	40%
Superconducting(a)	150°	195	21,000	40	-	8.8%	15	14	47%	40%
Cryogenic Aluminum	160°	101	2400	30	84(b)	2.0%	15	19	43%	40%

Table II-B. Dual** Bundle Divertor Parameters For ILB-HLβ Reactor

Conductor Type	Angle Between Coils	Coil Radius (cm)	Cross-Sectional Area (cm^2)	Required Shielding (cm)	Power Consumption (MW)	Ripple	Surface Field (T.)	Scrapeoff Techniques (cm)	$\eta_{D.C.}^{(c)}$	$\eta_{th}^{(d)}$
Warm Copper	170°	77	1700	20	612	1.0%	15	15	16%	40%
Superconducting(a)	150°	194	21,000	40	-	8.7%	15	10	47%	40%
Cryogenic Aluminum	160°	101	2600	30	155(b)	2.0%	15	14	39%	40%

**Assumes reactor is operated with two bundle divertors in use. (Note (Note: Columns 1-4 apply to <u>each</u> of the two.)

[For other footnotes, see Table I.]

opposite. Table II gives the same parameters for the cases of 1 divertor and 2 divertors for the HLβ reactor, as an example of how redundacy requirements affect the parameters.

Table III summarizes the comparison between the material choices. From Tables I and II, it is clear that copper coils consume too much power and superconducting coils cause an inordinate amount of perturbation of the field. The cryogenic aluminum coils serve to provide the most desirable features within the given constraints. Therefore, the reference divertors which are summarized in Table IV, are cryogenic aluminum.

IV. CONCLUSIONS

The conclusions drawn from the scaling studies conducted are:

1) Bundle divertors with copper coils raise the overall plant efficiency insignificantly (or lower it) since they consume too much power.
2) Superconducting coils cause a great deal of distortion in the undiverted field and are largest.
3) Cryogenic aluminum coils may work well in terms of size, power consumption, and induced ripple.

A word of caution is in order regarding the conclusion ruling out superconductors. There are three mitigating factors that were not considered in the model on which the conclusions were based:

1) The assumptions of the model are conservative. Recent work on relaxing the assumptions to more realistic cases has confirmed this fact.
2) Current densities substantially higher than 2000 A/cm^2 may be possible and ease the bundle divertor requirements considerably.
3) The addition of the effect of the toroidal field nulling coils (shown in Figure 4) **eases the bundle divertor requirements.**

It should be noted that 1 and 3 apply to all the material types and, thus, cryogenic aluminum divertors should be even more attractive than the tables present.

Table III

Bundle Divertor Results - Comparison of Materials

	Cu warm	Cryogenic Aluminum (16°K)	Superconducting Nb_3Sn
R*	∿1	2-7	>>10
Reduction in Waste Heat/MW_e	-	5-15%	12-20%
Technological Requirements and Cost	Low	Medium	High

*R ≡ Ratio of change in plasma power to divertor power consumed

Table IV. Reference Bundle Divertor Paramters

	ILB-CLβ	ILB-CHβ	ILB-HLβ	ILB-HHβ	ILB-HHβV
Coil Radius, m.	1.6	1.3	1.0	0.9	0.9
Throat Diameter, m. Field, T.	2.1 15	1.4 12	1.2 15	1.0 14	1.1 14
Conductor Area, m^2.	0.72	0.52	0.24	0.23	0.25
Angle Between Coils	170°	160°	160°	160°	160°
Shielding (Minimum), m.	0.40	0.40	0.30	0.30	0.30
Scrapeoff Thickness, cm.	22	27	20	22	22
Field Ripple+	2.0	2.0	2.0	2.0	2.0
Current, MA.	34	23	22	19	19
Refrigeration Power, MW	110	52	84	58	56
Charged Particle, MW	3300	980	630	811	770
Plasma Power Recovered,** MW	730	220	140	180	170

† - on plasma centerline

* - assuming 100 w/w

** - assuming overall direct conversion efficiency of 62% and thermal conversion efficiency of 40%.

TOROIDAL FIELD NULLING COIL

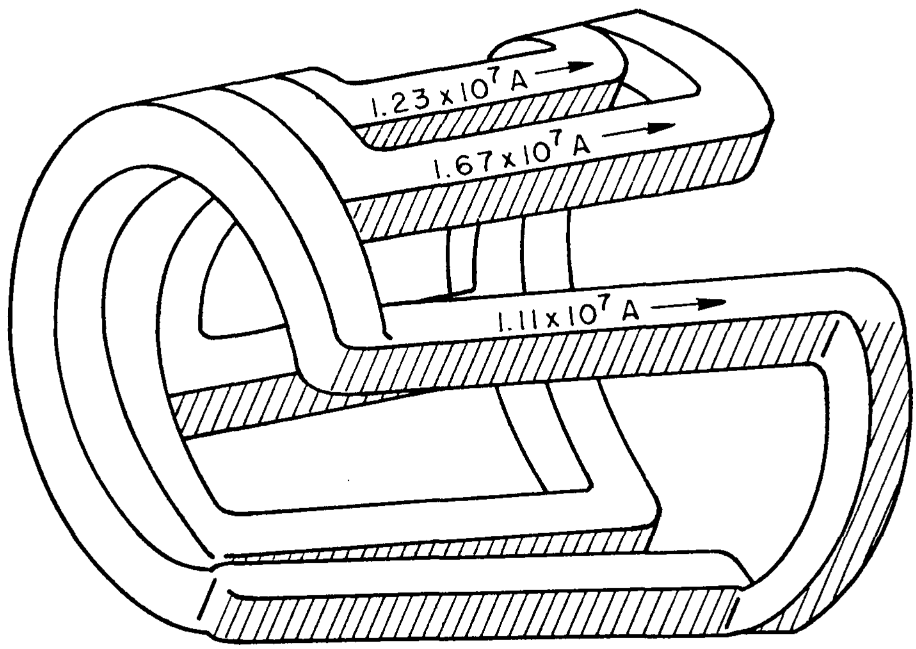

Figure 4. The toroidal magnetic field nulling coils developed by Livermore.[4] Adding in their field should, as a byproduct, ease the bundle divertor requirements.

Another point is important in evaluating the significance of the reference divertors. The direct conversion efficiency used, 62%, is a fairly low value due to our goal of achieving a conservative estimate of the gains. If a higher value is chosen, the attractiveness of using a bundle divertor/direct collection system is improved. Similarly, the thermal conversion efficiency used, 40%, might be high and, if lowered, adds to the gains in conversion efficiency noted previously.

In summary, the advanced fuel IBL reactors can gain substantially in efficiency through the use of bundle divertors and direct collection, a fact which does not apply to D-T reactors because of the lower charged particle power and the massive neutron shielding requirements.

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Catalyzed Deuterium Fueled Tokamak Reactors*

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ABSTRACT

Catalyzed deuterium fuel presents several advantages relative to D-T. These are, freedom from tritium breeding, high charged particle power fraction and lowered neutron energy deposition in the blanket. Higher temperature operation, lower power densities and increased confinement are simultaneously required. However, the present study has developed designs which have capitalized upon the advantages of catalyzed deuterium to overcome the difficulties associated with the fuel while obtaining high efficiency.

*This work is supported by EPRI (RP-645-1)

I. INTRODUCTION

A. Advantages of Catalyzed Deuterium Fusion Fuel

Catalyzed deuterium (CAT-D) has the highest power density of the advanced fuels at modest temperatures. The present study, therefore, chose this fuel to examine the potential for utilizing CAT-D as the fuel for Tokamak power reactors.[1]

Many other advantages are apparent with CAT-D. These are

1) Deuterium is a cheap, abundant, non-radioactive fuel.
2) No breeding is required in the blanket.
3) Blanket flexibility allows higher efficiency, thinner blanket.
4) Lower neutron wall loading at the same gross thermal power allows longer life first walls.
5) Neutrons which are produced are "free" and can breed ${}^{3}_{1}T$ or fissile fuel, if desired.
6) The high fraction of charged particle power allows the advantageous use of direct conversion, even in an ignited device.
7) Relatively modest ignition requirements permit 5 GW_e reactors under the most conservative confinement scaling estimates and about 1.5 GW_e under the "best estimate" of Tokamak confinement scaling. This is simultaneous with wall loading and sizes that appear economical.
8) CAT-D is a natural for "match-head" ignition procedures; startup is, therefore, only as difficult as for D-T.

B. Disadvantages of Catalyzed Deuterium Fuel

Because of the substantially lower power density of CAT-D relative to D-T, higher magnet technology (Nb_3Sn) must be used to achieve the desired power densities. It still appears possible, however, with maximum fields of about 15 Tesla at the conductors, to achieve economical power densities.

CAT-D requires the recirculation of unburned ^{3}He. Innovations in fueling ^{3}He are necessary or high energy injection will be required which could limit the plasma Q value to 5-10.

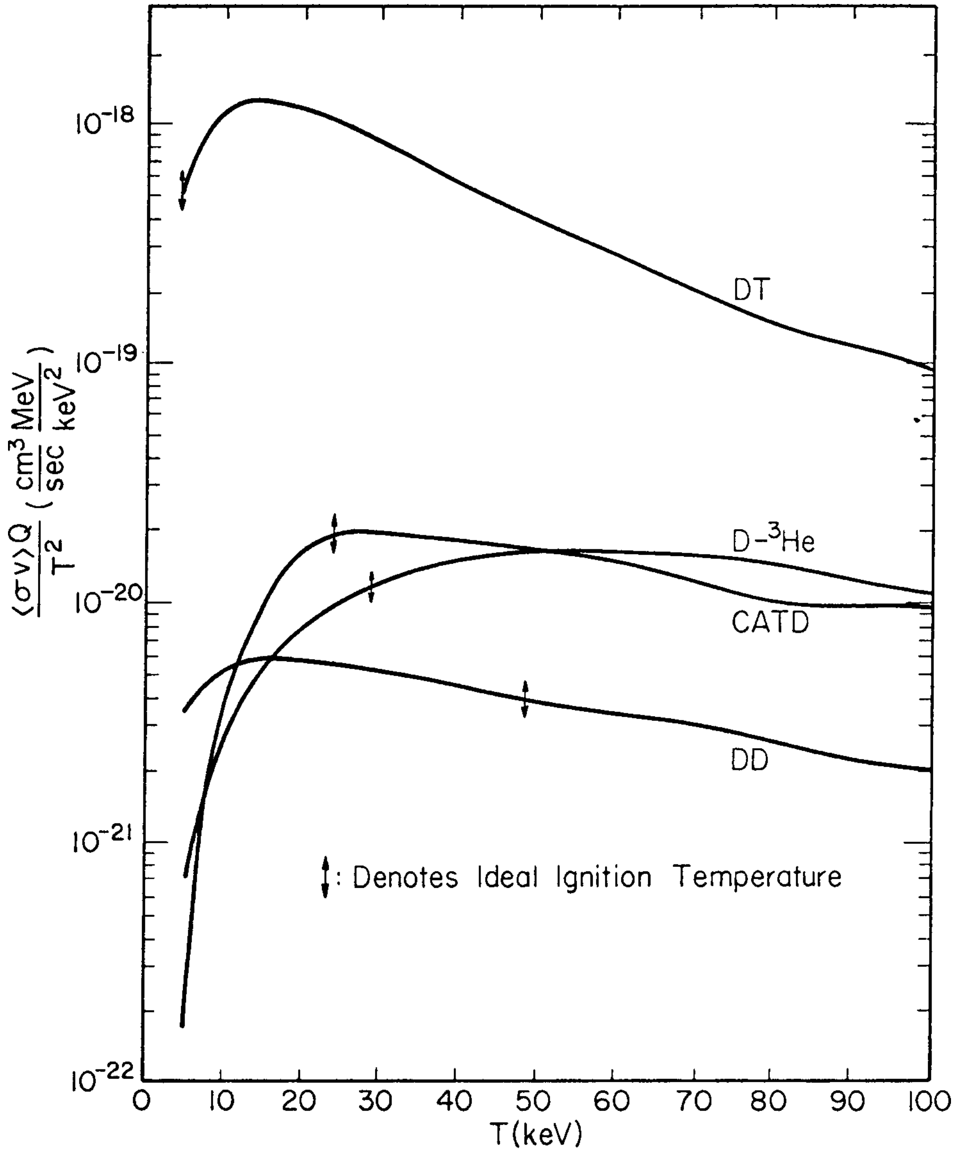

Figure 1. Relative Power Densities of the Prime Fusion Fuels

Lastly, approximately the same <u>number</u> of neutrons per unit power is produced with CAT-D as with D-T; care in design must be exercised to minimize activation. The present designs achieve this and even allow for limited direct maintenance.

II. PLASMA ENGINEERING FOR CAT-D FUEL

A global code was written which solved the particle (eight species) and energy balance equations for CAT-D.[1,2] Due to the broad range of parameters which it was necessary to survey, many features were incorporated to facilitate the search for an attractive ignited case.

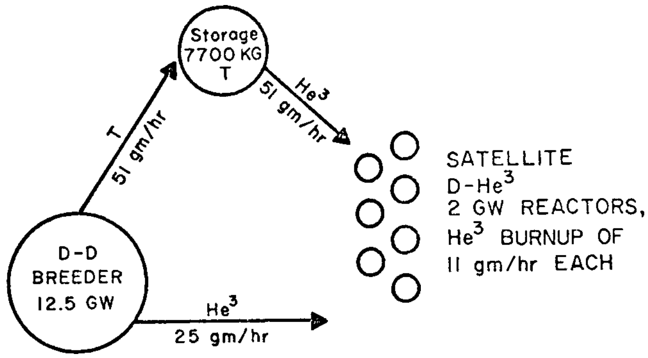

Figure 2. Catalyzed D Fuel Cycle

The independent variable of the code is fractional fuel burnup. Each iteration specifies a slightly higher burnup until ignition is obtained. The assumed plasma temperature profile was flat. This is because a high edge temperature was desired for utilizing direct conversion and to broaden the current profile. Additionally, it appeared that cyclotron radiation losses could be very severe with strongly peaked temperature profiles. The assumption of flat temperature profiles then led to a simple average for calculating the losses in the energy balance equations.

One of the outputs of the code was the required energy confinement time. Then, *a posteriori*, the obtained energy confinement time with the resultant machine was checked. In all cases, the required confinement time was below the empirical scaling for Tokamak confinement[3] and usually met the minimum theoretically predicted confinement scaling.[1]

Further simplification was possible in that a preliminary survey revealed that the suprathermal fusion products would only contribute 1-3% to the bulk plasma pressure. It was therefore deemed reasonable to ignore this contribution for the sake of computational speed. This comes about as a minor benefit of the low power density for a given fuel density of CAT-D relative to D-T. (At the same fuel density and temperature, the suprathermal fusion product pressure would be 30-50 times higher than with D-T.)

The specified parameters in the global code, CATD, were

P_{th}: gross thermal power

B_T: toroidal magnetic field

q: safety factor

A: aspect ratio

κ: plasma non-circularity

ρ_W: first wall resistivity and

F_H: fraction of holes in the first wall (used in the calculation of cyclotron losses)

F_I: fraction of ions which are impurities (carbon assumed)

T_I: ion temperature

M: $Te = MT_i$

and R: $\tau_E = R\tau_p$

Several values of each of these variables were input, typically, and the code stepped through the fractional burnup until a desired Q value was obtained (e.g. ∞ for ignition). The resultant output is

Q: plasma multiplication

R: major radius

a: minor radius

τ_E: energy confinement time

n_e, n_D, etc.: particle densities

I_p: plasma current

f_B: fractional burnup of each fuel species

S_R: recirculation rates of ^3T and ^3He

ϕ_{wn}: uncollided neutron flux at the first wall

P_e: electric power using assumed efficiencies

P_{Lp}: total power carried by leaking charged particles

P_N: total neutron power

P_{INJ}: required injection power

$P_{cyc\text{-}walls}$: cyclotron radiation power absorbed by the first wall

$P_{cyc\text{-}holes}$: cyclotron radiation lost through wall penetrations

P_{br}: total brehmsstrahlung power

B_{max}: maximum field at the toroidal coils

and all of the first wall power loadings.

In attempting to achieve the highest feasible wall loading, it was discovered that the brehmsstrahlung and cyclotron radiation were usually limiting factors. A radiation wall loading of 0.7 MW/m^2 was the practical maximum that was found.[4] The leaking plasma is recovered by a high efficiency divertor thereby eliminating it as a load factor on the first wall. The corresponding gross wall loading (P_{th}/wall area) is about

$2MW/m^2$. This was then set as the desired gross wall loading.

The parameter surveys were guided by the simple scaling relations between B_{max}, R, P_w and P_{th}. The relation between these parameters for a 3:1 noncircular Tokamak plasma using CAT-D at 45 keV and conventional beta constraints[1] is illustrated in Fig. 3.

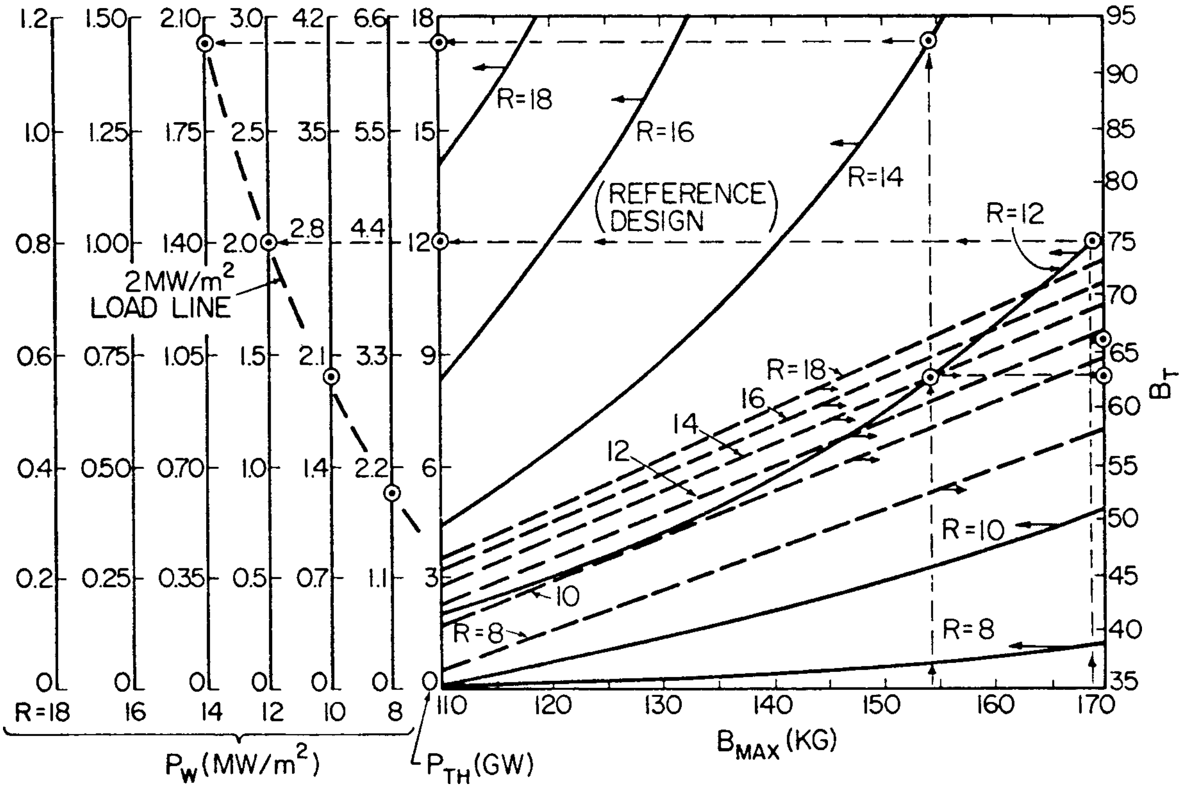

Figure 3. CAT-D Scaling for a 3:1 Noncircular Tokamak at an Ion Temperature of 45 keV.

Examination of Fig. 3 reveals that a total thermal power of about 12 Gw is necessary in order to achieve the desired wall loading at maximum fields of about 15 tesla. The power splits for this size reactor, illustrated in Fig. 4, indicates that the ion temperature of 45 keV appears optimal for utilizing direct conversion by maximizing the charged particle power flow.

III. REFERENCE REACTOR DESIGNS

Utilizing the scaling of the previous section, the CAT-D fueled noncircular cross section Tokamak reactor satisfies each of the constraints (maximum wall loading, turbulent confinement scaling, less than 170 KG maximum B-Field, etc.) at a size of 13.2 m. (major radius) and

Table 1. Ignited Cat-D Fuel Tokamak Reactors

ILB-CHβ

Plasma Dimensions m

R: 10.61
a: 3.54 A: 3
b: 5.66 κ: 1.6

Plasma Parameters

q 2.1
beta 0.3
$T_i;T_e$ 45; 45
τ_e 17.7 sec.
$n_e\tau_e$ 3.6 x 10^{15} sec-cm^{-3}
fractional D;T;^{3}He
burnups: 0.069; 0.7; 0.10

Reactor Parameters

Thermal Power-GW: 3.0
B_T 47.0 kG
B_{MAX}- 109 Kg
Inboard Blanket and
Shield Thickness-m 2.0
First Wall Resistivity 2x10^7Ω-m
First Wall Hole Fraction 0.1
Plasma-Wall Separation-m 0.5
Plasma Current-MA 23

Power Splits GW

Leaking Particles 0.977
Brehmsstrahlung 0.759
Cyclotron-wall 0.027
Cyclotron-holes 0.052
Neutrons- 1.18
Gross Electric- 1.4
η_{DC}=0.6,η_{th}=0.40

Wall Loadings-MW/m^2

Cyclotron 0.0099
Brehmsstrahlung 0.28
Neutron 0.436

Uncollided First Wall Neutron Flux
cm^{-2}sec^{-1}

2.45 MeV: 1.55 x 10^{13}
14.1 MeV: 1.55 x 10^{13}

ILB-CLβ

Plasma Dimensions m

R: 13.2
a: 4.9 A: 3
b: 13.2 κ: 3

Plasma Parameters

q 3
beta = 0.12; β_p = 2.01
$T_i;T_e$ 45;45
τ_e 20.3
$n_e\tau_E$ 4.4 x 10^{15}
fractional D;T;^{3}He
burnups: 0.083;0.74;0.12

Reactor Parameters

Thermal Power-GW: 12.5
B_T- 77.3 kG
B_{MAX}- 162 kG
Inboard Blanket and
Shield Thickness-m 2.0
First Wall Resistivity 2x10^7Ω-m
First Wall Hole Fraction 0.1
Plasma-Wall Separation-m 0.5
Plasma Current-MA 94

Power Splits GW

Leaking Particles 3.27
Brehmsstrahlung 3.14
Cyclotron-Wall 0.522
Cyclotron-holes 0.812
Neutrons- 4.76
Gross Electric- 5.64
η_{DC}=0.6,η_{th}=0.40

Wall Loadings-MW/m^2

Cyclotron 0.085
Brehmsstrahlung 0.51
Neutron 0.774

Uncollided First Wall Neutron Flux
cm^{-2}sec^{-1}

2.45 MeV: 2.75 x 10^{13}
14.1 MeV: 2.75 x 10^{13}

Illustrated in a companion paper in these proceedings[6] is the plasma cross-section of the ILB-CLβ plasma, including the location of the field shaping coils. A low activity graphite blanket[4] is used with helium coolant. The first wall is pyrographite operating at a temperature of 1700°C. This first wall is felt sufficient to provide a highly reflecting (at the cyclotron wavelengths) surface but also minimizes sputtering and impurity induced radiation in the plasma.

IV. Conclusions

If high-β operation of Tokamaks is achieved, a Cat-D fueled reactor could be a straightforward extension of D-T reactor technology. In some respects it would be far simpler than the D-T reactor (e.g., in the blanket design).

Even with low-β scaling (as is presently considered feasible), operation with CAT-D appears possible. The relatively large power levels of this reactor (ILB-CLβ), however, would probably make the reactor more attractive as a breeder (e.g., for fueling D-^{3}He reactors[7]).

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D-^{3}He Fueled Tokamak Reactors*

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ABSTRACT

D-^{3}He offers the highest power density of the advanced fusion fuels at elevated temperatures. As the fuel for a Tokamak reactors, the advantage of reduced neutron production (vs. CAT-D or D-T) can lead to significant simplification in blanket and shield design and possibly, in siting. A small size ($\sim$8 m major radius) Tokamak reactor using D-^{3}He seems an attractive and feasible potentiality. Providing the ^{3}He fuel appears to be the major difficulty, but a D-D breeder/D-^{3}He satellite may fulfill this need.

*This work is supported by EPRI(RP-645-1)

I. INTRODUCTION

$D-^3He$ fuel offers many advantages over D-T. Principally,

1) The fuel itself is non-radioactive.
2) Very low neutron production minimizes activation and simplifies maintenance. Urban siting might be possible due to the greatly reduced biological hazard potential.
3) Non-breeding blankets are possible allowing thin, high efficiency blanket designs.
4) The very low neutron production allows blanket designs with 30 year lifetimes.
5) The large component of power carried by charged particles allows the greatest capitalization on high efficiency direct conversion.
6) The power density is largest of the advanced fusion fuels at temperatures above 45 kev. Thus, it appears feasible to have a Tokamak reactor fueled with $D-^3He$ operating within Nb_3Sn superconductor constraints.
7) $D-^3He$ has the most modest ignition requirements of the advanced fuels.

As with any of the advanced fuels, however, there are strong disadvantages. The main disadvantage is that the power density is substantially lower than that for D-T. The advantages, above, mitigate this but overall more severe requirements on the confining magnets seems unavoidable.

Additionally, there is no ready source of ^3He. It appears that ^3He must be bred, by D-D reactions or by decay of tritium which has been bred in a D-D reactor blanket. First generation $D-^3He$ reactors might even consider using excess tritium produced in the blankets of D-T reactors. Injecting the ^3He fuel into the reactor is also a severe problem. At present, **fuel injection is an unsolved problem although edge injection** at low energy or other techniques might be envisioned.

II. PLASMA ENGINEERING FOR $D-^3He$

Figure 1 illustrates the power density of the advanced fuels relative to D-T. $D-^3He$ enjoys a respectable power density at elevated temperatures. (To obtain the power density in watts/cc, multiply the values in Fig. 1 by $99\beta^2B^4$.) Coupling this scaling with the geometry of a Tokamak results

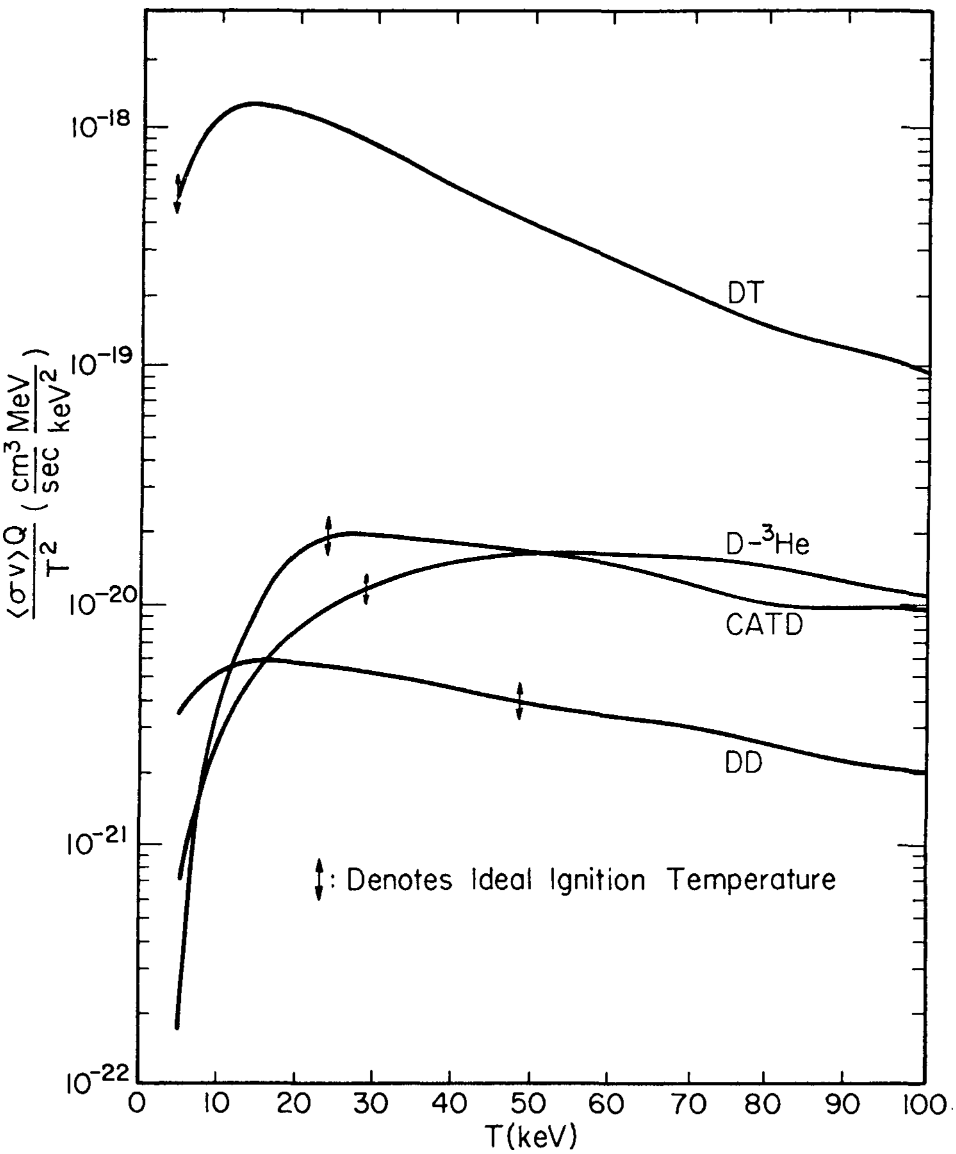

Figure 1. Power Densities of the Prime Fusion Fuels

in the scaling illustrated in Figure 2.

A gross wall loading of about 1 MW/m^2 results in a radiation wall loading of about 2/3 MW/m^2. This is about the maximum loading possible within materials constraints for the D-^{3}He fueled Tokamaks discussed here.

Supplying the ^{3}He fuel may be achieved by a breeding blanket on a CAT-D reactor.[1] This scenario is illustrated in Figure 3. This introduces the complexity of breeding blankets with CAT-D but, the flexibility of "clean" D-^{3}He satellite reactors may still be sufficiently attractive

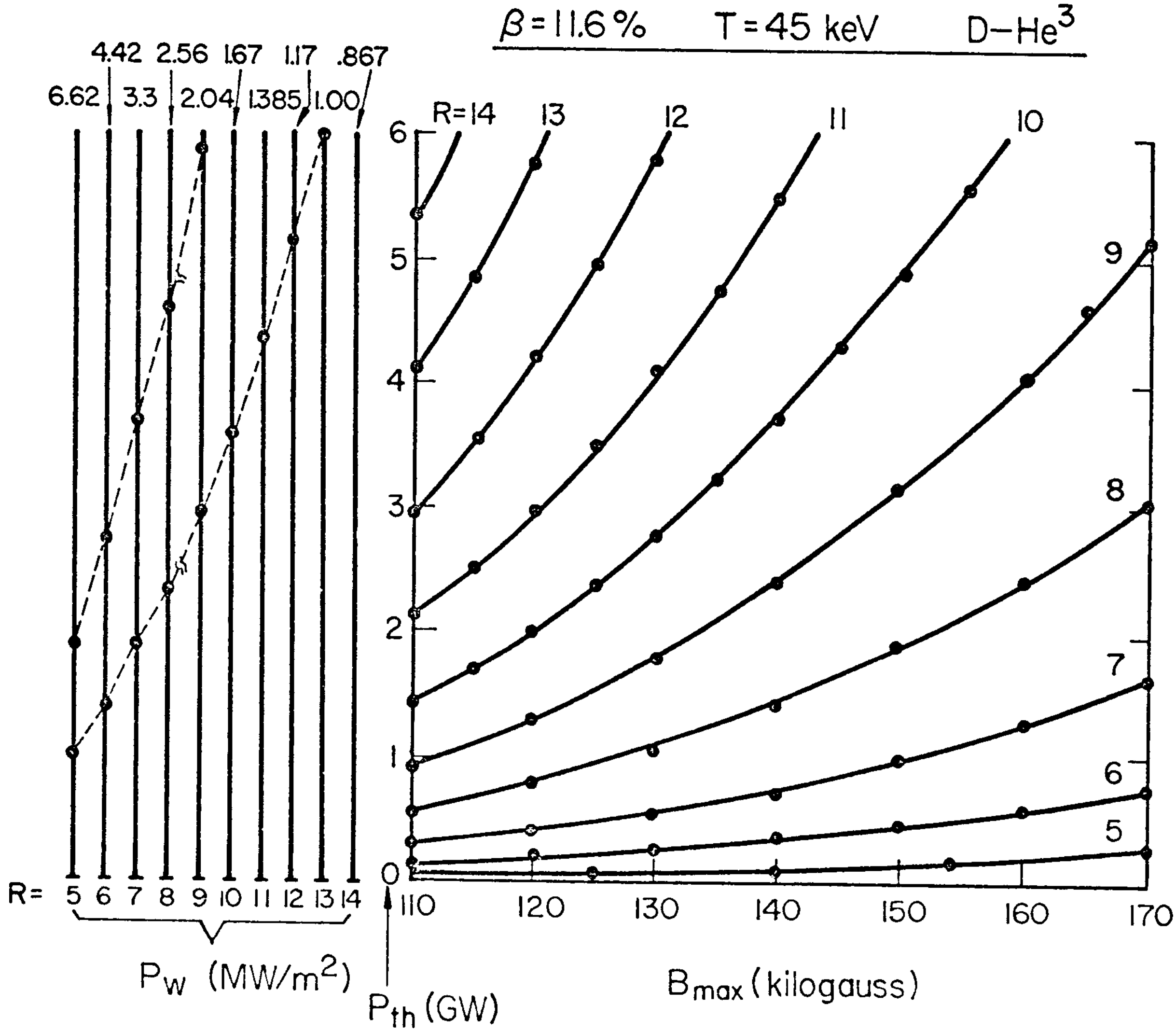

Figure 2. D-^{3}He Tokamak Scaling Dashed Lines Give Constant Wall Loading Scaling.

to warrant this complication.

The confinement requirements for D-^{3}He are roughly the same as that for CAT-D. For an aspect ratio of 3 in a Tokamak, the confinement requirements are illustrated in Figure 4. These requirements were calculated using a code similar to the one used for CAT-D.[1]

The predicted beta limit of 12% for such a Tokamak allows ignited operation for a 2000 MW_{th} reactor in the ion temperature range 38-65 KeV. Should high beta (∿0.3) be possible, this operating range can be extended to about 95 KeV (assuming, pessimistically, that $T_e = T_i$).

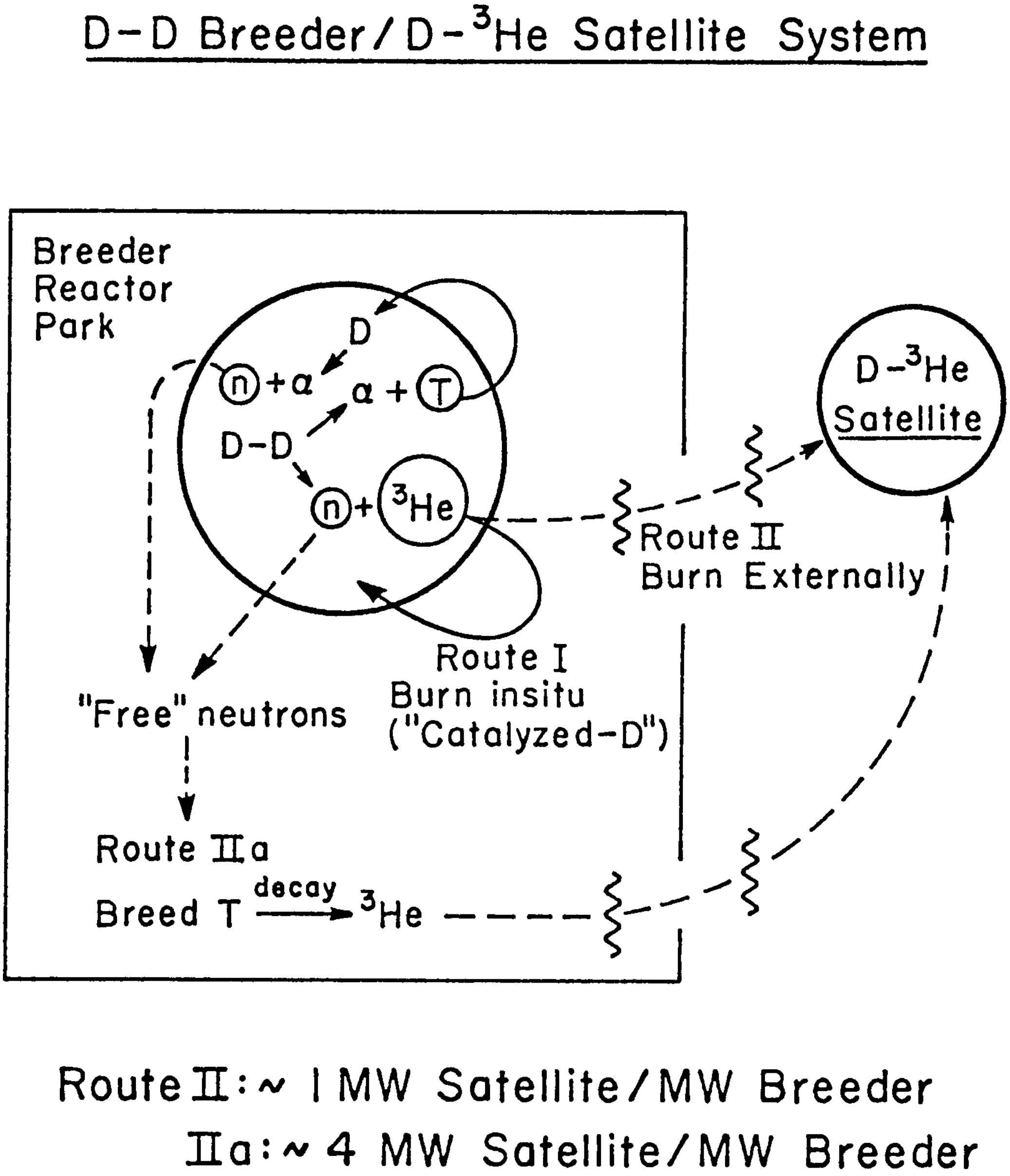

Figure 3. Scenario For Fueling D-^{3}He Reactors

The "cleanliness" advantage of D-^{3}He over other fusion fuels is illustrated in Figure 5. With the low beta plasma, a neutron power roughly 6% of the total power is obtained. At high beta, it is possible to run deuterium "lean" and still maintain ignition. This results in a factor of two reduction in the neutron power. This characteristic of D-^{3}He allows thinner blankets and shields. Alternative blanket and

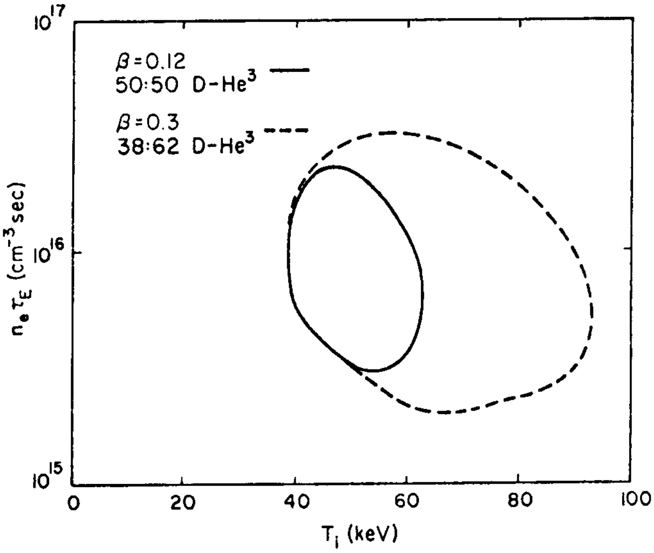

Figure 4. Confinement Requirements For a $2GW_{th}$ D-^{3}He Ignited Plasma

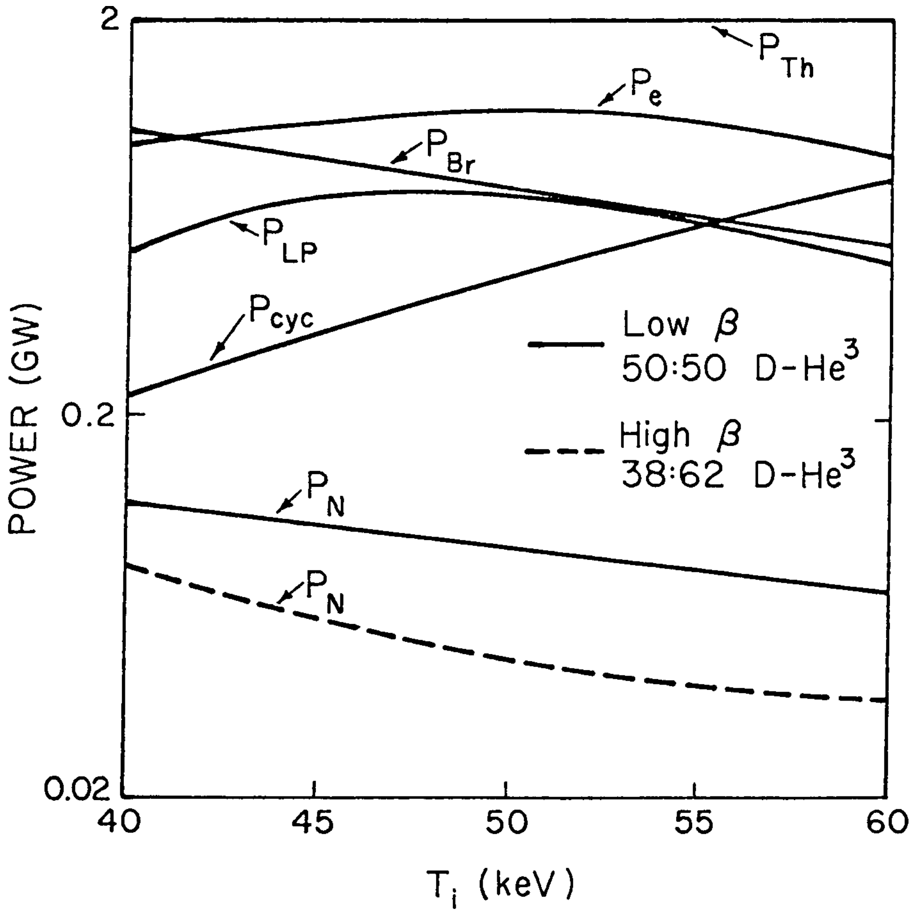

Figure 5. Power Splits For $2GW_{th}$ D-^{3}He Tokamak Plasmas

coolant materials may also be considered with low neutron production.

III. REFERENCE D-^{3}He REACTORS

Three reference D-^{3}He reactors[2] were obtained ILB-HLβ (University of Illinois, Livermore, Brookhaven - Low Beta D-^{3}He Reactor), ILB-HHB (High Beta, D-^{3}He Reactor) and ILB-HHβU (High Beta, Ultra-Clean D-^{3}He Reactor). These designs are summarized in Tables 1-3.

The required confinement in these reactors ranged from four to twenty times that time predicted for turbulent scaling from WASH-1295. This appears to be reasonable, however, in view of the conservation and uncertainties regarding trapped particle scaling.

Several other features are notable. Because of the low neutron production the inner blanket and shield are thinner making possible a smaller machine with fuller utilization of the toroidal field. These devices are considerably smaller than the CAT-D[1] reactors, for example, because of this factor. Figure 6 illustrates the relative sizes of the Low Beta CAT-D design (ILB-CLβ) and ILB-HHβ. This comparison would be less dramatic if the High Beta CAT-D design (ILB-CHβ) were shown. ILB-HHβ and ILB-HLβ, however, are nearly the same size.

In contrast to the CAT-D reactors, the D-^{3}He reactors utilize a terphenyl-cooled aluminum blanket[3] (with a graphite liner). This organic coolant would not be feasible with D-T or CAT-D because of severe radiolytic decomposition. With D-^{3}He the organic coolant suffers only a minor radiolytic and pyrolytic decomposition and allows a low pressure heat removal in a "low technology" blanket.

The large fraction of power carried by charged particles in the D-^{3}He reactors permits advantageous utilization of direct conversion. As with CAT-D, a bundle divertor-direct convertor tie-on is envisioned.[4]

IIV. CONCLUSIONS

The ILB series of D-^{3}He reference Tokamak reactors demonstrates a strong potential utility for D-^{3}He fuel as a highly environmentally acceptable power plant.[5] Should high beta operation be feasible in Tokamaks, then extremely clean, deuterium "lean," D-^{3}He reactors appear feasible.

Even with conventional low beta scaling of Tokamaks, it appears possible to have clean, high efficiency Tokamak reactors. It is

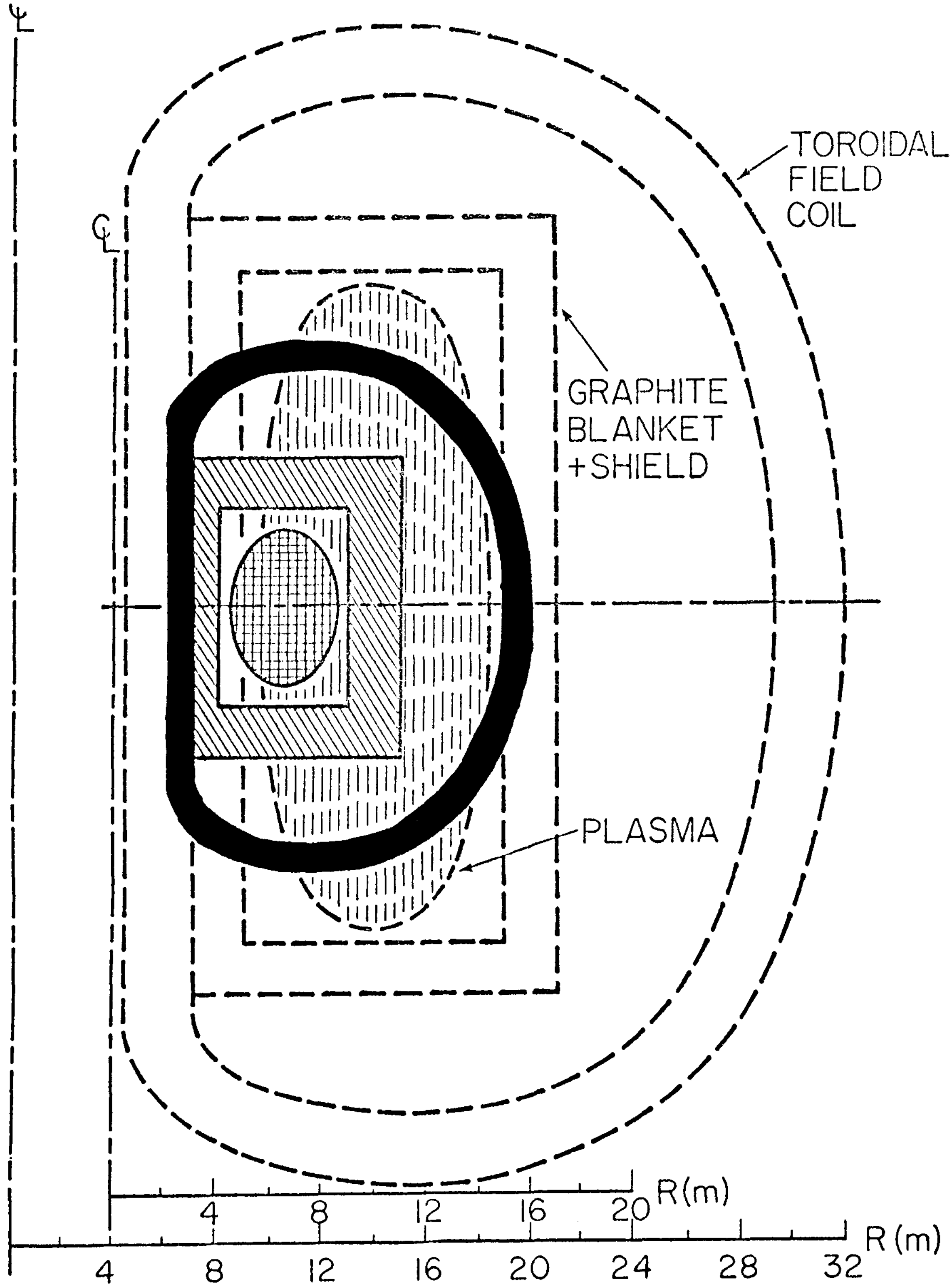

Figure 6. Relative Sizes of ILB-CLβ (Cat-D) and ILB-HHβ (D-^{3}He) Reactors.

Plasma Dimensions m.

R: 8.24
a: 2.75 A: 3
b: 8.24 k: 3.0

Plasma Parameters

q:	3.0
β:	0.12, β_p = 2.01
T_i; T_e:	45; 45 keV
τ_E:	21.2 sec.
$n_e\tau_E$	4.3×10^{15} $cm^{-3}sec$
fractional burnups:	D - 0.079 T - 0.53 ^3He - 0.055

Reactor Parameters

Thermal Power-GW:	2.0
B_T-kG	70
B_{MAX}-kG	13.8
Inboard Blanket and Shield Thickness-m	1.0
First Wall Resistivity	2×10^{-7} Ω-m
First Wall Hole Fraction	0.1
Plasma-Wall Separation-m	0.25
Plasma Current-MA	53

Power Splits GW

Leaking Particles	0.627
Brehmsstrahlung	0.916
Cyclotron-wall	0.125
Cyclotron-holes	0.12
Neutrons-	0.103
Gross Electric-	.938

η_{DC} = 0.62, η_{th} = 0.40

Wall Loadings-MW/m^2

Cyclotron	0.0502
Brehmsstrahlung	0.369
Neutron	0.0416

Uncollided First Wall Neutron Flux $cm^{-2}sec^{-1}$

2.45 MeV: 2.90×10^{12}
14.1 MeV: 1.34×10^{12}

Table 1. Reference "Low" Beta D-^{3}He Tokamak

Plasma Dimensions m.

R: 7.58

a: 2.53 A: 3

b: 4.04 k: 1.6

Plasma Parameters

q:	2.1	
β:	0.3	
T_i; Te:	45; 45	keV
τ_E	10.7	sec
$n_e\tau_E$	3.4×10^{15}	$cm^{-3}sec$

fractional burnups: D - 0.063, T - 0.047, ^3He - 0.044

Reactor Parameters

Thermal Power-GW:	2.0
B_T-kG	55
B_{MAX}-kG	112
Inboard Blanket and Shield Thickness-m	1
First Wall Resistivity	2×10^{-7} Ω-m
First Wall Hole Fraction	0.1
Plasma-Wall Separation-m	0.25
Plasma Current-MA	20

Power Splits GW

Leaking Particles	0.811
Brehmsstrahlung	0.926
Cyclotron-wall	0.125
Cyclotron-holes	0.042
Neutrons-	0.098
Gross Electric-	0.978

$\eta_{DC} = 0.62$, $\eta_{th} = 0.40$

Wall Loadings-MW/m^2

Cyclotron	0.0158
Brehmsstrahlung	0.624
Neutron	0.0683

Uncollided First Wall Neutron Flux

$cm^{-2}sec^{-1}$

2.45 MeV: 5.16×10^{12}

14.1 MeV: 2.13×10^{12}

Table 2. Reference "High" Beta D-^{3}He Tokamak

ILB-HHβU
($D:{}^3He$ = 38:62)

Plasma Dimensions m.

R: 7.79

a: 2.60 A: 3

b: 4.15 k: 1.6

Plasma Parameters

q: 2.1

beta: 0.3

$T_i;T_e$: 45; 45 keV

τ_E 12.2 sec.

$n_e\tau_E$ 4.0×10^{15} cm^{-3} sec

fractional burnups: D - 0.074
T - 0.45
^{3}He - 0.040

Reactor Parameters

Thermal Power-GW: 2.0

B_T-kG 55

B_{MAX}-kG 110

Inboard Blanket and Shield Thickness-m 1.0

First Wall Resistivity $2\times10^{-7}\Omega$-m

First Wall Hole Fraction 0.1

Plasma-Wall Separation-m 0.25

Plasma Current-MA 20

Power Splits GW

Leaking Particles	0.77
Brehmsstrahlung	1.09
Cyclotron-wall	0.0295
Cyclotron-holes	0.0451
Neutrons-	0.0653
Gross Electric-	0.969

η_{DC} = 0.62, η_{th} = 0.40

Wall Loadings-MW/m^2

Cyclotron	0.01616
Brehmsstrahlung	0.7204
Neutron	0.043

Uncollided First Wall Neutron Flux $cm^{-2}sec^{-1}$

2.45 MeV: 3.37×10^{12}

14.1 MeV: 1.32×10^{12}

Table 3. Reference Ultra-Clean High Beta D-^{3}He Tokamak

envisioned that these can be sited near load centers and employ a low technology design except for the magnets.

An unsolved problem is fueling (injection would lead to a low Q and pellets seem infeasible). Simpler designs without direct convertors and employing cold gas blanket fueling might be a possible alternative.[4]

Supplying the ^3He fuel can be accomplished through D-D breeder reactors. A major problem here, however, is the large inventory of tritium which must be stored in order to obtain a reasonable flow of ^3He. A storage of 800 kg of tritium is necessary to supply ^3He for a 1000 MW_{th} D-^3He reactor in steady-state.

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SOME NEW IDEAS ON WET WOOD BURNERS

by

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ABSTRACT

We investigate a class of wet wood burners in which the ion temperature is maintained by external heating but the electrons rapidly lose energy due to anomalous processes or by impurity radiation. It is assumed that the ion energy confinement is good while that of the electrons is poor, as has been observed in some fusion experiments. We find that energy multiplications of 10 or greater can be achieved with electron energy confinement times such that $n\tau_E > 2 \times 10^{13}$.

Any thermonuclear device which acts as an energy amplifier we may call a wet wood burner, as energy must be supplied to maintain the reaction. According to the above definition, a number of devices being considered as controlled fusion reactors would be classed as wet wood burners. The conventional mirror machine and the Astron device are examples. The device which has received the most attention recently is the so-called Two-Component Torus,[1] in which an energetic beam of neutral deuterons is injected into a toroidal plasma of hot electrons and cold tritium ions. The neutrals are ionized by the plasma and trapped in the magnetic field. As they slow down in the plasma some of the deuterons react with the tritons producing a significant amount of fusion power. If the neutral beam has energy near the peak of the fusion cross section, and if the electron temperature exceeds 4 KeV, then more fusion energy can be produced than is required to accelerate the beam. At the same time the beam supplies the energy required to maintain the plasma.

This approach is of interest because plasmas of the required conditions (T_e = 4 KeV, $T_i \geq 0$) have been achieved in existing Tokamak discharges and neutral beam technology appears capable of producing the required beams (100 Amps, 200 KeV) with high efficiency although it has not yet done so. At present instabilities which might be produced by the beam appear not to be serious both on theoretical and existing experimental grounds.[2]

Notwithstanding the relatively good prospects for achieving break-even by this approach, the maximum gain one can get with high electron temperatures is of the order of 3 - 4. For a practical reactor one would prefer much larger gains, 10 or greater, to compensate for unavoidable losses and inefficiences in the beam generating system. To this end energy clamping of the energetic beam (accelerating the beam at just the rate it is slowing down) has been considered.[3] By holding the beam at the peak of the fusion cross-section, the gain can be roughly doubled; i.e. energy multiplications of 6 - 8 can in principal be achieved. This gain, however, is only achieved at the expense of increased complexity of the device. In all these devices it is required that the confinement time of the energetic ions be long compared to their loss time.

Here we should like to discuss some new ideas on wet wood burners which appear capable of giving large yield ratios (10 or greater). For a number of fusion experiments it has been found that the energy losses by electrons are anomalously large, while ion losses remain roughly classical. The recent experiment on the French Tokamak TFR[4] show electron energy losses which are hundreds of times larger than neoclassical or even pseudo-classical theory. On the other hand the ion confinement seemed to obey the neoclassical theory. In the ATC experiments[5] the electron energy confinement time is less than that for the ions. In experiments on ion cyclotron heating of the C stellerator[6] ion heating to 500 eV did not seem to enhance the losses, while increased electron temperatures did.

There are a number of reasons why one might expect electron energy losses to be large. First, there is the problem of impurity radiation. If the level of high z impurities exceeds even a fraction of a percent, radiation losses by the electrons are greatly enhanced and the possibility of achieving ignition may be unattainable. Because of their high cyclotron frequency and small Larmor radii, collective effects enhance electron transport sooner than they do ion transport, and may lead to large heat losses by the electrons. Recent computer simulations have shown that electron heat transport can be substantially increased by lower hybrid waves and by plasma wave transport.[7] Heat conduction along field lines due to electrons in open ended devices or in Tokamaks with poor magnetic surfaces (as might be produced by hydromagnetic instabilities) could lead to rapid electron energy loss.

Here we should like to point out that even with high electron energy loss, it is possible to achieve a wet wood burner type operation with yield ratios greater than 10. Calculations by the Wisconsin group[8] on impurity dominated discharges have shown similar results. In these calculations we made the following assumptions:

1. The ion and ion energy confinement times are long compared to the ion electron energy transfer time.
2. A high ion temperature is maintained by direct ion heating.
3. Relatively rapid energy loss by the electrons exists, so that the electron temperature is relatively low. The only heat input to the electrons is from the hot ions and the α particle reaction products. A minimum energy confinement time for the electrons is required, which we will compute, but it is far below that given by the Lawson condition.
4. The major energy loss by the ions is to colder electrons.

If the above assumptions can be realized, relatively large energy multiplications can be achieved. They can be much larger than in a two-component torus with a clamped energetic beam, as the following elementary calculation shows.

The energy which must be supplied to the ions so as to maintain their temperature against energy loss to the electrons is the following (assuming a DT reaction)

$$P_{1i} = \frac{3}{2} \frac{n_i(T_i - T_e)}{\tau_{ei} \text{ (Energy)}} \tag{1}$$

where n_i is the total ion density, T_i and T_e are the ion and electron temperature and τ_{ei}(Energy) is the ion electron energy exchange time. Taking the density of deuterons to equal the density of tritons:

$$n_D = n_T = \frac{1}{2} n_i$$

the fusion power production is

$$P_{1F} = \frac{n_i^2}{4} <\sigma v>_1 w_F \tag{2}$$

where w_F is the energy released per fusion reaction (we may take it to be 17.58 MeV or 22.4 MeV if we include energy from breading tritium; it may even be doubled by using various n - 2n reactions in the blanket and then absorbing the extra neutron in a suitable nucleus which releases a large amount of energy but which does not leave a long lived radioactive byproduct). The energy multiplication is

$$Q_1 = \frac{P_{1F}}{P_{1i}} = \frac{n_i <\sigma v>_1 w_F \tau_{ei}(\text{Energy})}{6(T_i - T_e)} \tag{3}$$

Now for a clamped two component torus reactor we must supply the following power to the beam

$$P_b = \frac{n_b(w_b - \frac{3}{2}T_e)}{\tau_{ei}(\text{Energy})} + \frac{n_b w_b}{\tau_{bi}} \tag{4}$$

where n_b is the density of beam ions, w_b is the beam energy and τ_{bi} is the beam ion collision time. The fusion power by the beam is

$$P_{Fb} = n_b n_i <\sigma v>_b w_F \tag{5}$$

where we have taken the beam to be one type of ion (deuterium) and the background to be another (tritium).

The energy multiplication is

$$Q_2 = \frac{n_i \langle\sigma v\rangle_b \, w_F \, \tau_{ei}(\text{Energy})}{\left(w_b - \frac{3}{2} T_e\right) + w_b \frac{\tau_{ei}(\text{Energy})}{\tau_{bi}}} \tag{6}$$

$$\approx \frac{n_i \langle\sigma v\rangle_b \, w_F \tau_{ei}(\text{Energy})}{w_b \left(1 + \frac{\tau_{ei}(\text{Energy})}{\tau_{bi}}\right)}$$

where the last relation is obtained by assuming $w_b \gg T_e$. The ratio of the two Q's is

$$R = \frac{Q_1}{Q_2} = \frac{\langle\sigma v\rangle_1 \, w_b}{6(T_i - T_e)\langle\sigma v\rangle_b} \left(1 + \frac{\tau_{ei}(\text{Energy})}{\tau_{bi}}\right) \tag{7}$$

where we have taken n_i and T_e equal in the two cases. Now if we take the ion temperature in the first case to be 20 KeV ($\langle\sigma v\rangle_1 = 4.3 \times 10^{-16}\text{cm}^3/\text{sec}$) the beam energy to be 150 KeV (max $\sigma v = 1.5 \times 10^{-15}\text{cm}^3/\text{sec}$) and the electron temperature to be 10 KeV and using the energy loss rates given in Ref. 1, this ratio comes out to be 1.8.

Thus the energy gain is greater than for the clamped beam case. In a way it is like the clamped beam case in that energy is fed into the ions so as to clamp their temperature. Further collisions between ions do not remove energy from the ion distribution, only the energy loss to the electrons must be made up. In this sense it is like a regular thermonuclear reactor, where collisions between the particles do not cause a loss of energy from the plasma but simply redistribute it; in the clamped system with cold ions energy losses to the ions must also be made up. Further, because of the much larger value of w_b than T_i, more energy must be supplied to the beam than to the relatively colder ions, and this largely offsets the factor 6 and the difference in cross-section. We may also note that in the hot ion case some of the fusion energy is deposited directly in the ions (for the above case about 2%), further reducing the power needed from outside and increasing the effective Q. Finally, for both this scheme and the clamped two-component torus the energy must be contained in the hot component for times long compared to the ion electron energy loss time. Thus, the conditions on ion energy confinement are not much different although they might be affected by the ion distribution function.

One of the important quantities is the required electron energy confinement time. This is obtained by dividing the electron energy density by the power input to the electrons. The energy input to the electrons is

$$P_e = \frac{3}{2} \frac{n_i(T_i - T_e)}{\tau_{ei}(\text{Energy})} + \frac{n_i^2}{4} <\sigma V>_1 w_\alpha \tag{8}$$

where w_α is the energy released in α particles. Dividing $\frac{3}{2} n_e T_e$ by P_e gives

$$n_e t_e = \frac{T_e \; n_e \; \tau_{ei}(\text{Energy})}{(T_i - T_e) + \frac{n_i \tau_{ei}(\text{Energy})}{6} <\sigma V>_1 \; w_\alpha} \tag{9}$$

Using equation (3) for Q and taking w_F = 22.4 MeV (18 MeV from the DT reaction and 4.82 MeV from nLi^6 reaction), and equation (9) for t_e and assuming Maxwellian distributions for ions we have computed Q and $n_e t_e$ as functions of T_e for a number of ion temperatures. Plots of these results are shown in figure 1. From these plots we note that with T_i = 20 KeV break even, Q = 1, is achieved for electron temperatures T_e = 3.2 KeV. The corresponding value of $n_e t_e$ from figure 2 is 1.6 x 10^{+12}. If the electron temperature is raised to 10 KeV, Q becomes 8.6 and the corresponding value of $n_e t_e$ is 2.2 x 10^{+13}. In this latter case the actual Q would be about 20% larger because a fraction of the fusion energy (2%) is deposited directly in the ions and need not be supplied from outside. If a substantial fraction of the α energy could be induced to be deposited directly in the ions, then the energy multiplication would be substantially increased. For the above case if roughly half the α energy went directly to the ions it slightly larger $n_e t_e$ are required because less energy is supplied to the plasma.

One can carry out similar calculations for other reactions. We have made such calculations for D, He^3 and these results are shown in figure 2. Figure 2a shows Q vs T_e for a number of ion temperatures and figure 2b shows the equivalent $n_e t_e$. Figure 2c shows $\tilde{Q}$ the gain if it is assumed that all the reaction product energy is deposited in the ions, which is over-optimistic but not too bad at high electron temperatures.

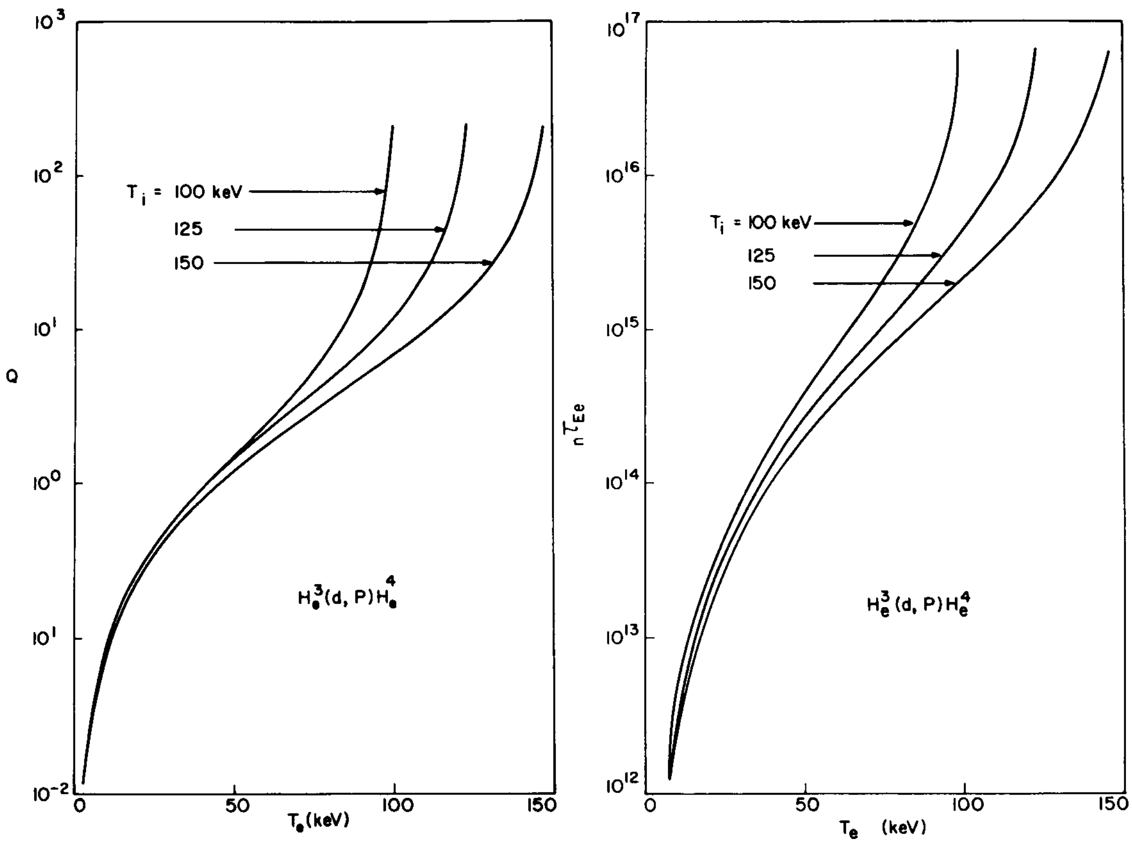

Figure 1
Energy multiplication Q and electron energy confinement time versus electron temperature T_e in the $T(d, n)H_e^4$ reaction.

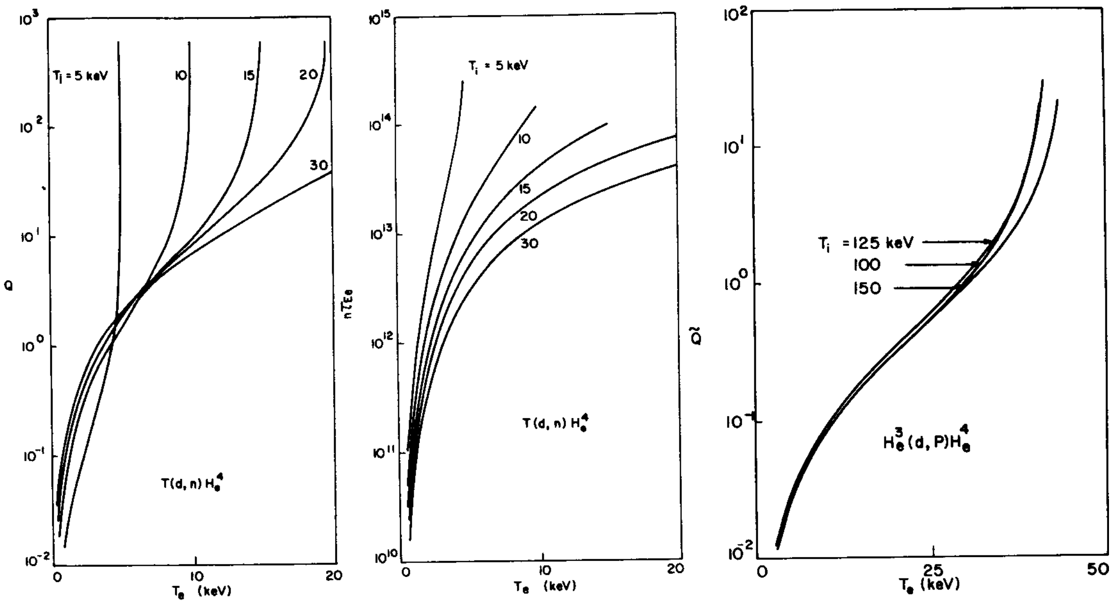

Figure 2
(a) Energy multiplication Q, (b) Electron energy confinement time, and (c) Energy gain $\tilde{Q} = Q/(1-Q)$, versus electron temperature T_e in the He^3 (d, p)He^4 reaction.

Methods of Heating

One of the best methods of heating is the injection of neutral beams at energies several times the ion temperature but lower than $(m_b/m_e)^{1/3}T_e$, the point at which the beam delivers as much energy to the electrons as the ions. Undoubtedly some advantage can be made of the two-component plasma idea of injecting, where the cross-sections are larger, but it appears on balance one will lose if one goes to the peak of the cross-section because the beam then gives too large a fraction of its energy to the electrons. The ideal energy can only be determined by more detailed calculations.

Cyclotron resonant absorption would be another good candidate since it delivers most of its energy to the ions. Any type of wave heating which strongly favors ion heating would be good.

Conclusions

The above calculations show that viable wet wood burners can be achieved even with quite heavy electron energy loss. In fact roughly an order of magnitude below Lawson will suffice for the electron energy containment time. On the other hand the ion energy confinment time must definitely meet the Lawson criterion. This result is consistent with that found by the Wisconsin group for impurity dominated discharges in Tokamak.[8] If it should turn out that Tokamaks exhibit good ion confinement at high ion temperature then such a wet wood burner would be an attractive reactor even in the face of severe impurty problems and anomalous electron heat loss. It is thus essential to pin down the ion energy loss as accurately as possible.

The realization that heavy electron energy loss can be tolerated allows one to consider other types of reactors. For example a long straight machine with cold gas end plugs is cooled mainly by electron heat conduction. Since one can tolerate high electron heat loss if one supplies heat directly to the ions one can operate devices of this type with much more modest dimensions. Further, because a high impurity level is also tolerable, one can use impurities to inhibit heat conduction losses by both electrons and ions. For example, a straight device operating at a density of 5×10^{16} with electrons at 10 KeV and ions maintained at 15 KeV by direct ion heating and an effective Z of 5 would yield an energy multiplication of 10 if it were 500 m long.

Acknowledgement

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MULTIPOLES AS REACTORS

John M. Dawson

The work that I am going to describe started because of a request by Bill Gough of EPRI that I take a look at the possibilities of advanced fuels. Such fuels, particularly those which produce no neutrons or few low energy neutrons could sufficiently simplify the engineering and extend the life times of reactors, that they could have significant impact on the adoption of fusion. In looking at these it appeared that multipoles had quite an advantage here because of the low magnetic field in the region where the plasma was located. Because all advanced fuels must operate at high temperatures, syncrotron radiation is a very serious energy loss from such devices unless the plasma can be located in a low field region and this difficulty is automatically solved in multipoles. Since these early investigations it appears that there may also be ways to make successful multipole reactors using DT which I will touch on if I have time. However, the most exciting aspect is their potential for operating with neutronless producing reactions.

Since this work began a great many people have contributed to it. I have made a partial list of them here; in particular Don Kerst, Bob Conn, Don Sweetman, and the plasma group under Don Arnush at TRW have made quite significant contributions. In fact TRW has made a proposal to make a feasibility study of the concept.

First, to remind you of what a multipole is, a simplified drawing of one is shown in Figure 1. The field is very, very low in the center and the plasma tends to concentrate in the low field region so it tends to be a very good confiner of plasma. Generally multipoles have been dismissed; I think this is largely due to the fact that people could not see how current and cooling could be provided to a ring internal to the plasma. Any leads and their associated magnetic shielding, create losses so that it will be difficult to make it go. One might consider using superconductors but this appears totally impractical in the intense neutron environment of the DT reaction. However, if a reaction can be found which produces no neutron or relatively few neutrons, then it does appear practical to use superconducting coils. Such a coil would be constructed as shown in Figure 2. Most of the plasma energy would be given off as x-ray radiation. This can be absorbed in a thin layer, perhaps 10 cm, of tungsten which would run at about 2000°K. At that temperature it can radiate about 1MW per M^2 and so it radiates away as much energy as it receives from x-rays. Inside this is a layer of high temperature insulation, graphite wool is a good example of that. This would be follwed by perhaps a layer of Li about 20 cm thick that absorbs a lot of heat in being heated up and melted. Inside that there is a layer of super insulation and then the superconductor. The overall radius would be of the order of 1 M. Bob Conn of Wisconsin has made some calculations of how long such a ring could operate and he finds times of the order of two days before they must be taken down and cooled.

The fact that such a reactor gives off most of its energy as x-rays of the order of 100 KeV leads to another advantage. Such x-rays will pass through a thin wall of low z material, as shown in Figure 3. They are preferentially absorbed by high z gases so that they can heat the gas to a very high temperature, perhaps 3 to 5000° K. In principal one can use such hot gas to run a very high efficiency heat engine; one possibility is to use an MHD topping cycle; a second possibility is to use an engine designed by Abe Hertzberg, and for which an example has been built and run; these promise to give thermal efficiencies in the 60 to 70% range.

Let us return to the thermonuclear reaction. Figure 4 shows cross sections for some thermonuclear reactions. Next to DT, P-^{11}B appears to be the best fuel here so I took a rather hard look at this operating in a multipole. This reaction is as follows:

$$P + {}^{11}B \quad 3\,{}^{4}He + 8.7\ \text{MeV}$$

There are a few side reactions producing neutrons:

$$P + {}^{11}B \rightarrow {}^{11}C + n$$

$${}^{4}He + {}^{11}B \rightarrow {}^{14}N + n$$

Estimates of these made by Weaver and Woods indicate they would be many orders of magnitude below the primary reaction.

In investigating this reaction I assumed we would have a driven reactor so that the ions would be maintained at the peak of their reactivity. The electrons must run at a lower temperature than this because at this temperature they radiate more energy than is produced by the reaction just due to Bremsstrahlung. The higher the electron temperature the more strongly they radiate so we would like to run them at as low a temperature as possible. On the other hand the colder electrons cool the ions and too much power will be required to sustain the ion temperature if the electrons are too cold. Since the cooling rate is proportional to $T_e^{-3/2}$ from this point of view we want to run the electrons as hot as possible. We must then compromise between these conflicting requirements.

Figure 5 shows the ratio of the Bremsstrahlung to the thermonuclear output assuming the ions are maintained at the temperature of peak reactivity. ε is the ratio of boron density to proton density. We see that the thermonuclear output balances the Bremmstrahlung at between 100 and about 150 KeV.

The next important thing is how does the thermonuclear output balance against the electron cooling. This is shown in Table 1. For an electron temperature of 100 KeV the cooling is about twice as large as the thermonuclear output. For electron temperatures of 150 KeV it just about balances. Fortunately most of the reaction energy goes directly to the ions at these high electron temperatures so the reaction can just about sustain the ion temperature. If the system is driven with a neutral beam of perhaps 30% of the TN output it should be possible to maintain the high ion temperature while sustaining losses greater than the minimal Bremsstrahlung. In this case energy multiplications of a factor of 3 would be possible and with energy recovery efficiencies of 2/3 circulating powers in the 30 to 50% range should be possible depending on the neutral beam efficiencies.

Of critical importance are the plasma and heat losses. I have estimated the syncrotron losses and for an octupole they came out to be about 3% of the Bremsstrahlung losses assuming 90% reflecting walls. As far as plasma losses go we only have the Wisconsin and GA experiments to extrapolate from; these were done at low temperatures and densities. However, these scaling laws if extrapolated to the thermonuclear range give very good confinement, as is shown in Table II.

P-"B looks touch and go, we should like to have a larger margin. Are there other reactions which might be made to go? In fact there are quite a few. Table III shows some of these. It is not clear whether or not superconducting coils could be used with those reactions which produce neutrons. However, it is possible to operate many of these reactions in a fashion where the neutron production is much below that for DT and further, their spectrum is much softer than the 14 MeV produced there. For example, D^3He can be run ^3He rich and D poor and if it is further run as a wet wood burner with a deuterium beam into a predominant ^3He plasma, somewhere between 2 and 3 orders of magnitude reduction in neutron production occurs and these are mainly DD neutrons of 2 MeV. Only detailed calculations can tell if this is sufficient.

Once one starts to think of multipoles as reactors many possible modes of operation occur which could apply to any fuel including DT. In this short talk it is only possible to touch on these.

First it is possible to put leads to the rods and operate it in a mirror mode. In this case one would introduce a modest toroidal field of say several hundred Gauss and trap the plasmas in the center as in a conventional mirror machine. This is like a mirror machine with the Ioffe fields predominating over the axial field and these fields support the pressure. If the multipolar field is say 30 KG in the bridge, then to escape around the bridge the particle must overcome a 100 to 1 mirror ratio. This is already enough to significantly improve the Q of a mirror. However, most particles going around the rods will not escape, they will re-enter the plasma. In the region around the cooling and current supplies we must divert the plasma away and particles entering these regions could be mirror lost.

A second possibility is to generate the current in the rods without material supports. Two possibilities exist. The side of the rod facing the plasma receives a large amount of radiation from x-rays and neutrons and will be hotter than the rear side which is cooled by radiation. It is theoretically possible to run a generator off of this temperature difference which will drive current in the hoops. I believe an MHD type generator would be best and I have made some rough estimates of whether or not this is feasible and it appears that it is but size may become a problem.

The third possibility is to power the rings with neutral beams. A neutral beam of hydrogen or of deuterium of several 100 KeV would be directed through a thin Be foil. The electron would be stripped off by the foil while the protron continues onto a plate several hundred KeV higher. We thus have a battery which we match to the load by the number of strands of wire making up each loop. For a reactor a few thousand amps of 100 KeV beam should surface. One must run the rings hot, 1500 K to 2000 K so as to get rid of the energy. Again whether or not this is a practical engineering possibility requires a great deal of study, but my preliminary look indicates it is not totally out of the question.

Other possibilities for powering the rings are undoubtedly possible. One would of course not design or build such complicated rings to find out the feasibility of multipole fusion. One can build transient machines which can run for several minutes with either cooled rings or superconducting rings to find out if the physics is okay. Such machines should be relatively cheap and only if these were a success would one design and build complex ring structures. The most exciting thing is the possibilities of operating with neutronless fuels and if these should prove possible the engineering simplifications could easily result in such a reactor coming faster than a complex DT reactor.

PARTIAL LIST OF CONTRIBUTORS

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6.	B. Fried	UCLA
7.	F. Chen	UCLA
8.	A. Hertzberg (Efficient Heat Engine)	University of Washington
9.	G. Miley (Reaction Cross Sections)	University of Illinois
10.	E. Norbeck (Reaction Cross Sections)	University of Iowa

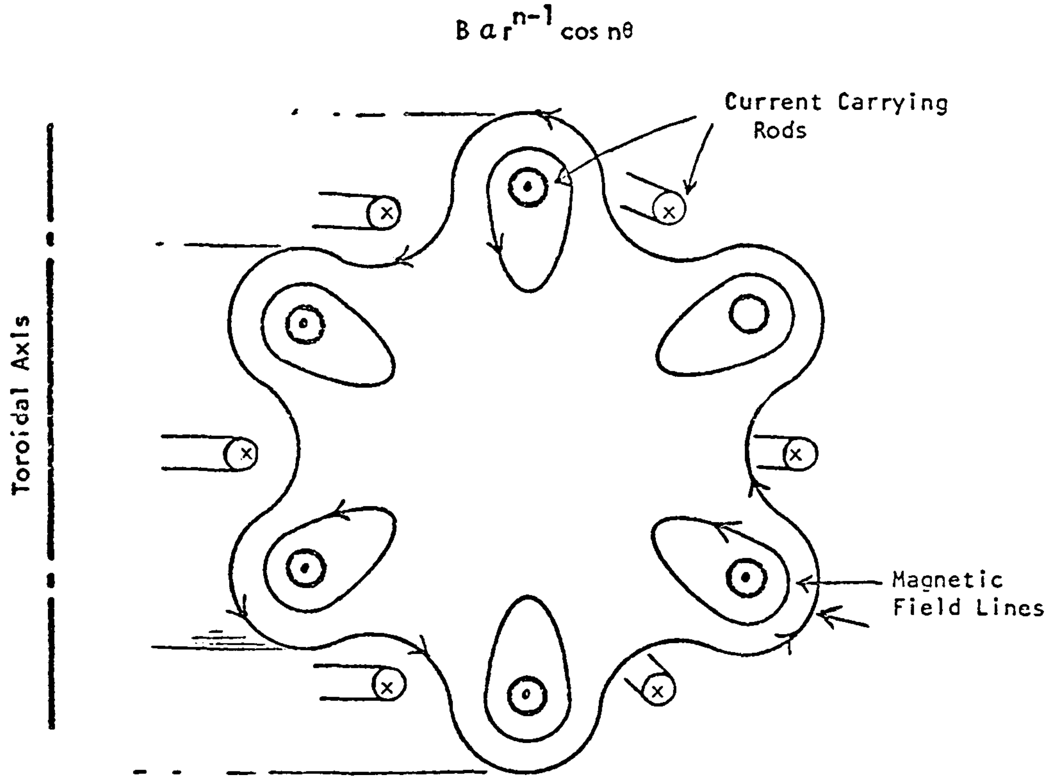


Figure 1 Multipole Configuration

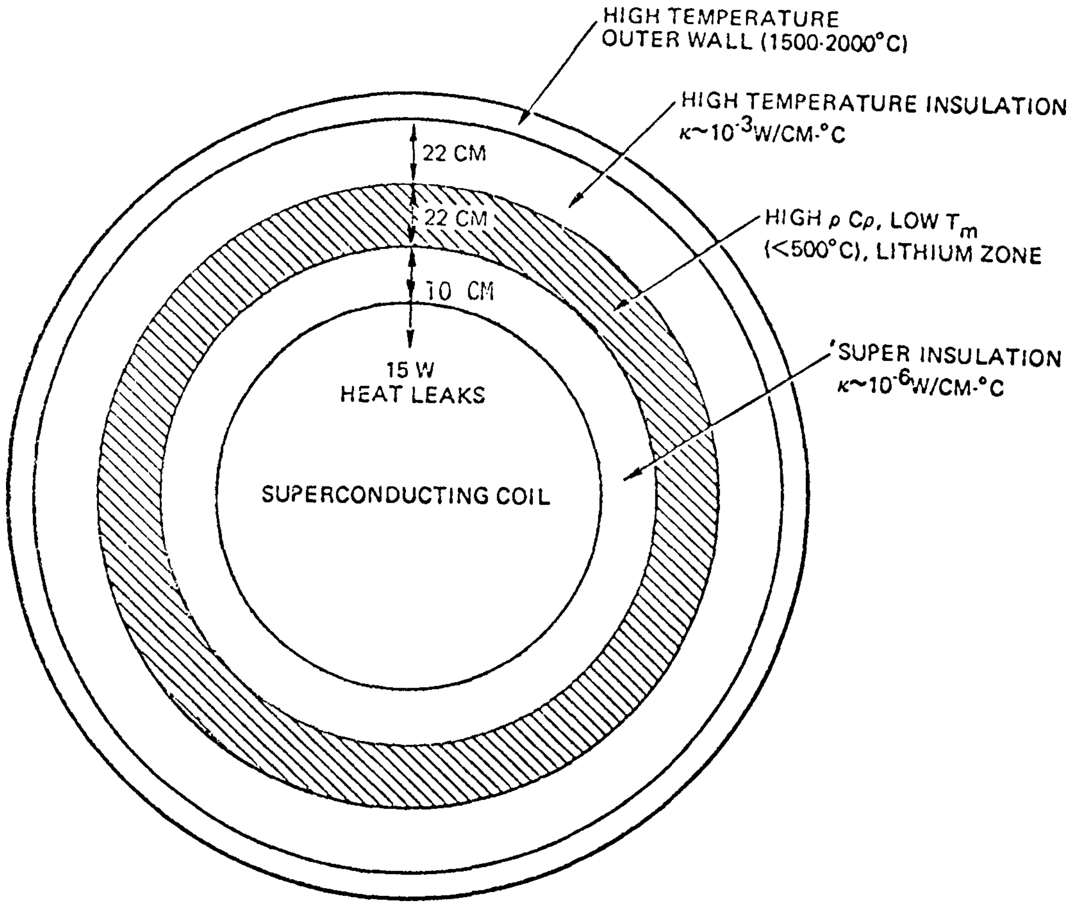


Figure 2   Generic Design for a Levitated Octopole Coil Without External Cooling.

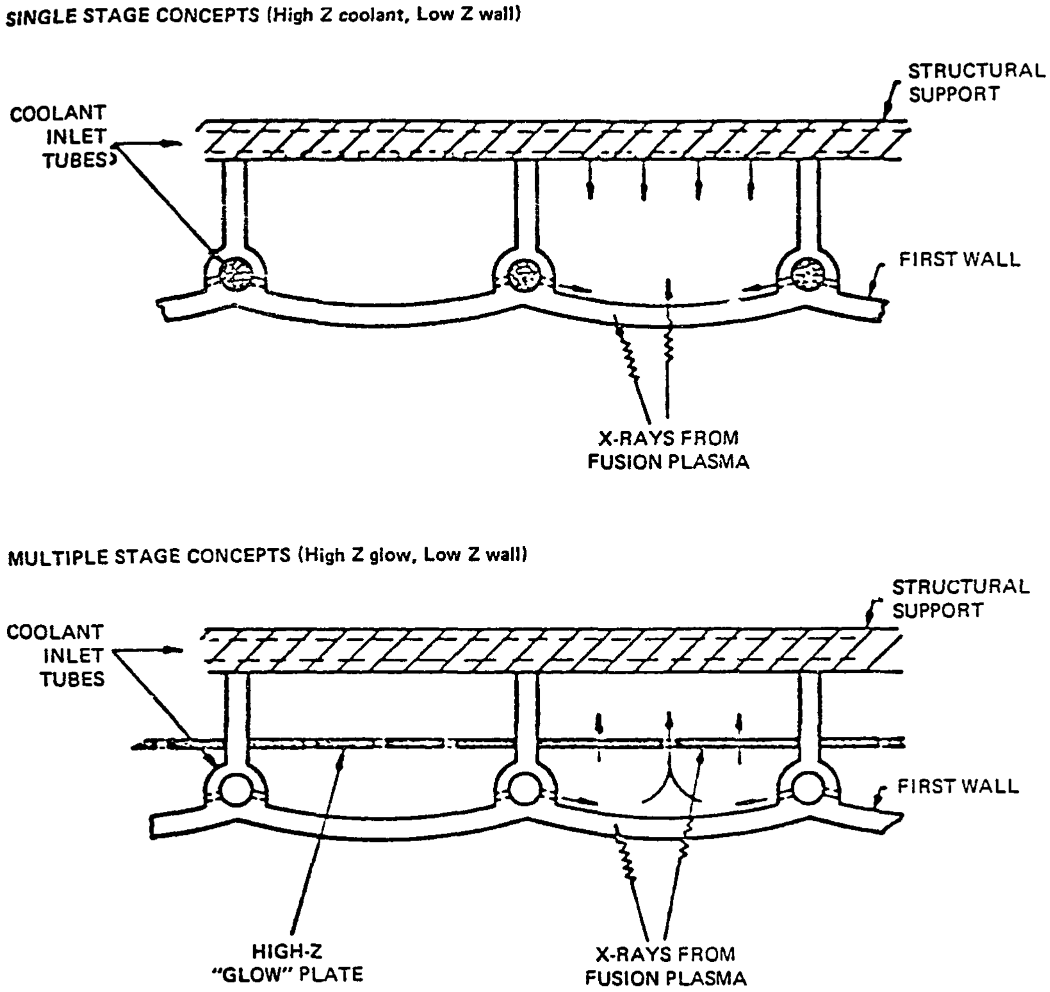


Figure 3 Radiation boiler/fusion reactor first wall concepts

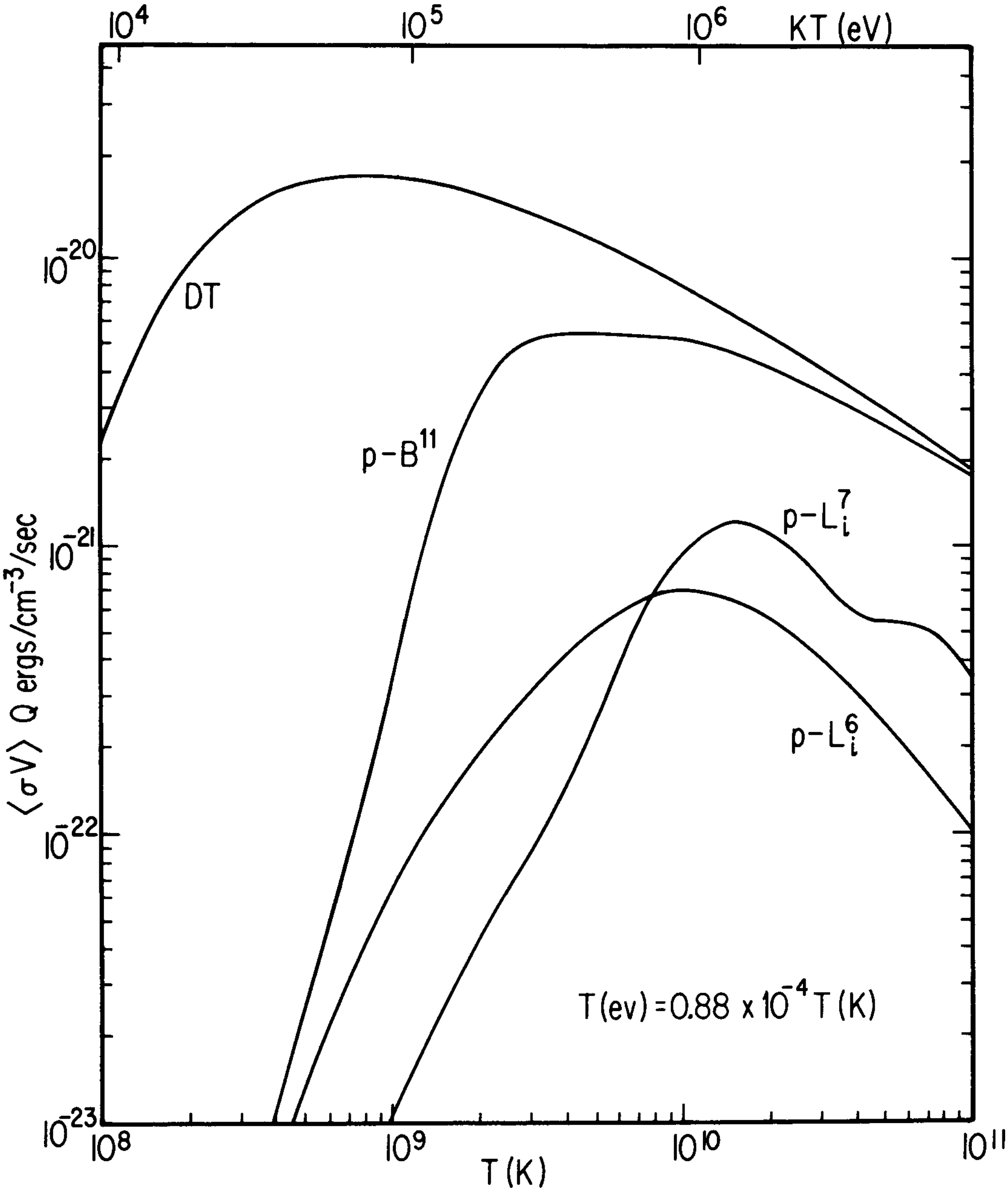


Figure 4. $\langle \sigma V \rangle$ For a number of fusion fuels.

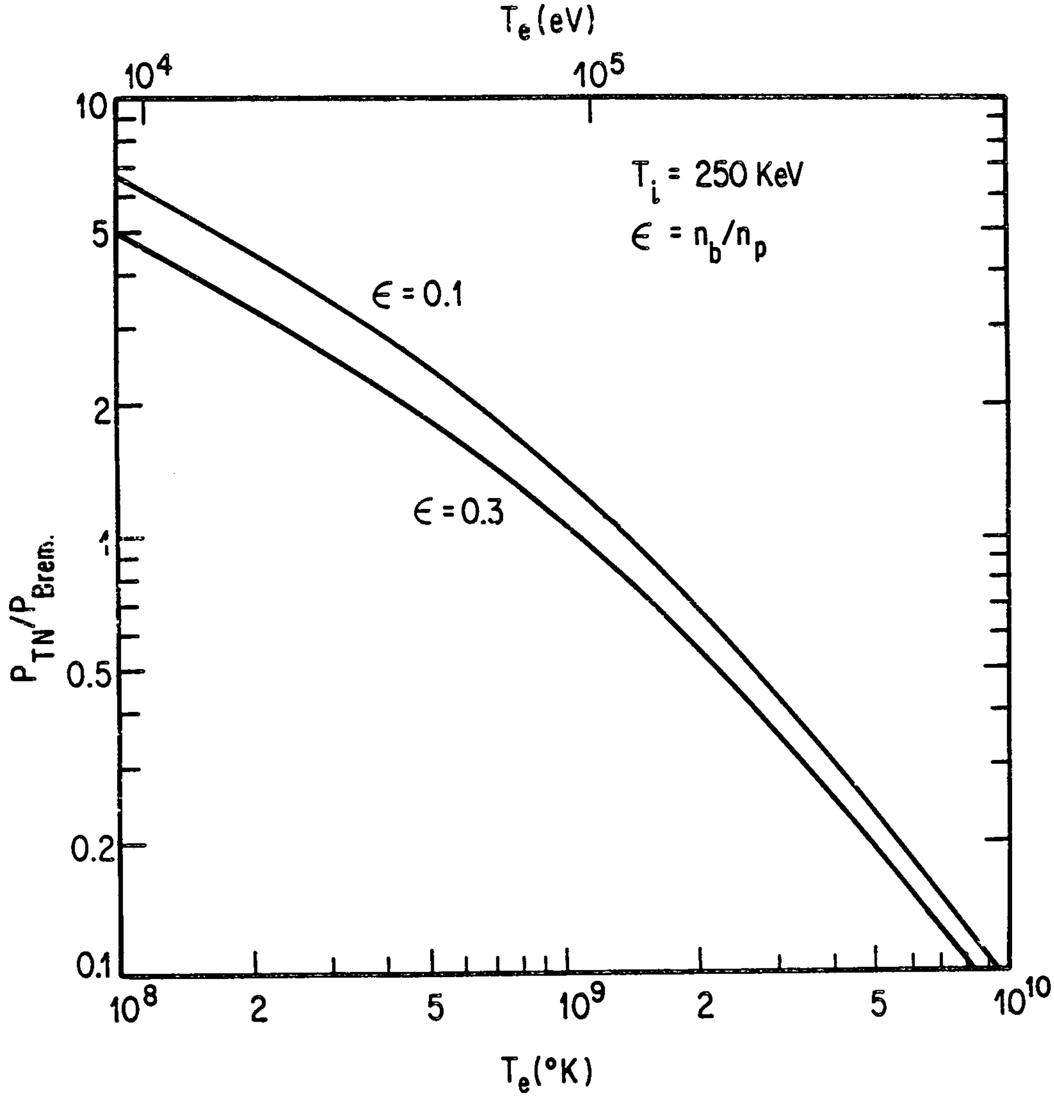
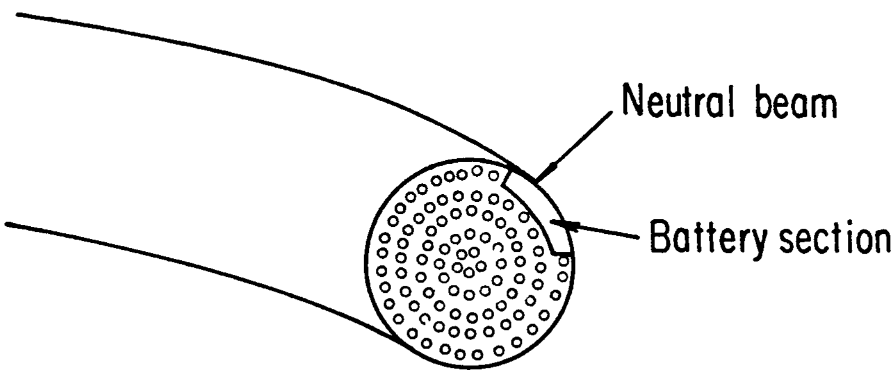

T_e(eV)
10^4
10^5
T_i = 250 KeV
$\epsilon = n_b/n_p$
ϵ = 0.1
ϵ = 0.3
$P_{TN}/P_{Brem.}$
10
5
2
1
0.5
0.2
0.1
10^8
2
5
10^9
2
5
10^{10}
T_e(°K)

Figure 5


Neutral beam
Battery section

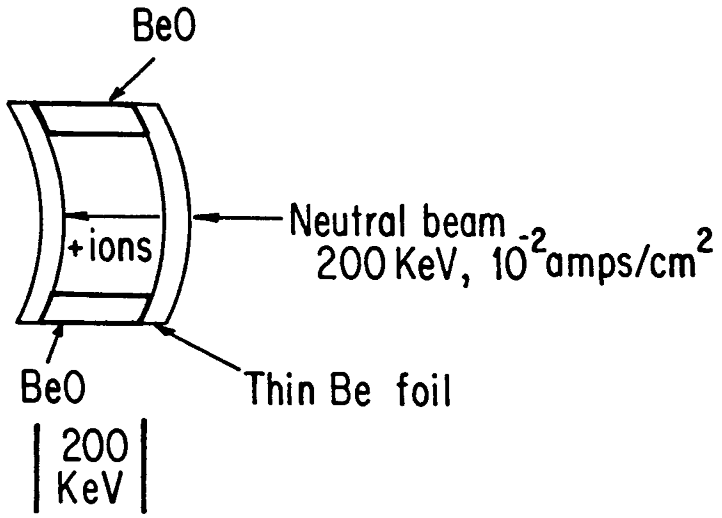

BeO
+ions
Neutral beam
200 KeV, 10^{-2} amps/cm^2
BeO
Thin Be foil
200
KeV

Figure 6

Table I Ratio of Thermonuclear Power to Power Going from Ions to Electrons (P_{TN}/P_{ie})

$$\frac{P_{TN}}{P_{ie}} = \frac{\epsilon}{(1 + 2.27\epsilon)(1 + 5\epsilon)} F(T_i, T_e)$$

T_e	T_i	P_{TN}/P_{ie}	ϵ
100 KeV	300 KeV	0.45	0.2
125 KeV	300 KeV	0.72	0.2
150 KeV	300 KeV	1.10	0.2
100 KeV	300 KeV	0.46	0.3
125 KeV	300 KeV	0.73	0.3
150 KeV	300 KeV	1.13	0.3

TABLE II

Wisconsin Scaling

$$D_{\text{Poloidal Only}} = \frac{2\times10^8\ T_e}{\sqrt{n\ell}} \qquad T_e \text{ in ev, } \ell \text{ in cm}$$

For $T_e = 10^5$, $n = 10^{14}$, $\ell = 2R = 400$ cm

$$D_{\text{Poloidal}} = 3.16 \times 10^2 \text{ cm}^2/\text{sec}$$

$$\tau = \frac{.2\ R^2}{D} = 25 \text{ sec} \qquad n\tau = 2.5 \times 10^{15}$$

$$D_{\text{Toroidal + Poloidal}} \begin{cases} \text{In Private Flux} & = \dfrac{1.2 \times 10^{11}\ T_e^{3/2}}{n\ell} \\[2ex] \text{In Common Flux} & = \dfrac{1 \times 10^{12}\ T_e^{3/2}}{n\ell} \end{cases}$$

Using the Larger $D = 8 \times 10^2$ cm^2/sec

$$\tau = \frac{.2\ R^2}{D} = 10 \text{ sec} \qquad n\tau = 10^{15}$$

G. A. Scaling

$$D = 10^{-3}\ D_{\text{Bohm}} = 6 \times 10^3\ T_e/B$$

For $T_e = 10^5$, $B = 50$ KG $\qquad D = 1.2 \times 10^4$

To get $n\tau = 10^{15}$ requires $R \approx 800$ cm

Determination of scaling in the thermonuclear range important.

	1)	$P + {}^{6}Li$	$\rightarrow$	${}^{3}He + {}^{4}He +$	3.864	MeV	
		${}^{3}He + {}^{6}Li$	$\rightarrow$	$2\,{}^{4}He + P +$	16.6	MeV	
		${}^{6}Li + {}^{6}Li$	$\rightarrow$	$3\,{}^{4}He +$	20.5	MeV	
	2)	${}^{3}He + D$	$\rightarrow$	${}^{4}He + P +$	18.2	MeV	
		Run 90% ${}^{3}He$, 10% D					
	3)	${}^{3}He + {}^{9}Be$	$\rightarrow$	$3\,{}^{4}He +$	18.74	MeV	
		${}^{4}He + {}^{9}Be$	$\rightarrow$	$3\,{}^{4}He + n$	-1.6	MeV	
83% of Energy Gives Off As Charged Products	4)	$D + {}^{6}Li$	$\rightarrow$	${}^{7}Li + P +$	4.9	MeV	
				${}^{7}Be + n +$	3.3	MeV	- Max Neutron Energy 2.89 MeV
				${}^{4}He + T + P +$	2.5	MeV	
				${}^{4}He + {}^{3}He + n +$	1.7	MeV	Max Neutron Energy 1.5 MeV
				$2\,{}^{4}He$	22.0	MeV	
		$P + {}^{6}Li$	$\rightarrow$	${}^{3}He + {}^{4}He +$	3.864	MeV	
		$T + {}^{6}Li$	$\rightarrow$	${}^{7}Li + D +$	.9	MeV	
				${}^{7}Li + n + P$	-1.2	MeV	Low Neutron Energy
		${}^{3}He + {}^{6}Li$	$\rightarrow$	$2\,{}^{4}He + P +$	16.6	MeV	
		${}^{3}He + D$	$\rightarrow$	${}^{4}He + P +$	18.7	MeV	
		$D + {}^{7}Be$	$\rightarrow$	$2\,{}^{4}He + P +$	16.5	MeV	
		${}^{7}Be + {}^{6}Li$	$\rightarrow$	$3\,{}^{4}He + P +$	15.0	MeV	
		$D + D$	$\rightarrow$	$T + P +$	4	MeV	
		$D + D$	$\rightarrow$	${}^{3}He + n +$	3.25	MeV	Neutron Energy 2.4 MeV
		$T + D$	$\rightarrow$	${}^{4}He + n +$	17.4	MeV	Neutron Energy 14 MeV
		${}^{3}He + D$	$\rightarrow$	${}^{4}He + P +$	18.2	MeV	

Table III

CTR Using the P-^{11}B Reaction

By

John M. Dawson

The possibility of achieving useful energy release from the neutronless P-^{11}B reaction in a magnetically confined plasma device is examined. It appears that a real possibility exists although it is not possible at the present time to be definitive about it due to uncertainties in the cross sections, in the degree to which synchrotron radiation can be reduced, in whether or not a sufficiently effective confinement device can be built, and because detailed Fokker-Planck calculations of energy multiplication by an injected energetic hydrogen beam do not exist. If one considers only bremsstrahlung losses and ignores synchrotron and plasma losses, then it appears that one can come close to ignition. By direction ion heating using a 1 MeV proton beam so as to sustain the ions at the temperature giving peak reactivity, it appears that energy multiplications of 3 $\sim$ 4 should be achievable while providing an energy surplus of 30 $\sim$ 40% of the bremsstrahlung to balance synchrotron and other losses. Success depends on reducing the synchrotron emission to a fraction of the bremsstrahlung; an upper bound on the synchrotron emission is estimated and it is shown that at 100 KeV electron temperatures and a β = 1 plasma, the synchrotron emission is not serious, while at 150 KeV and β = 1 it is serious. By reducing the magnetic field in the bulk of the plasma to a low value synchrotron emission should be manageable. It appears that this is possible with multipole devices. These devices are advantageous from the stability and confinement points of view also, and recent plasma confinement measurements indicate they may be adequate. With a neutronless reaction multipoles using floating superconducting rings should be possible. In addition to the absence of neutron, the P-^{11}B reaction has the advantage of great abundance of fuel and the fact that breeding of fuel is not necessary.

By far the easiest fuel to achieve a thermonuclear reaction in is a mixture of deuterium and tritium. It has the largest fusion cross section, the reaction releases a large amount of energy, it can be ignited at the lowest temperature of any mixture (about 5 KeV).[1] This reaction has the disadvantage of giving off most of its energy as 14 MeV neutrons. These cause nuclear activation of the containment vessel causing it to become radioactive and thus generating a radioactive waste problem. Tritium, which is used as one of the fuels, does not occur in nature in significant quantities because of its twelve-year half life. It must be bred from Li in a blanket using the reactions*

$$^{6}\text{Li} + n \rightarrow {}^{4}\text{He} + {}^{3}\text{T} + 4\text{ MeV}$$

$$^{7}\text{Li} + n \rightarrow {}^{4}\text{He} + {}^{3}\text{T} + n - 2\text{ MeV}$$

Because of this a complex tritium breeding blanket of 1 $\sim$ 2 meters thickness must surround the plasma; handling of the radioactive tritium becomes a problem.

The prolific neutron production of such a reactor means that it must be provided with extensive shielding and routine maintenance and operation must be done by remote control. If superconducting coils are to made use of, they must be separated from the plasma by at least one meter of shielding. Thus, despite the great advantages of the DT reaction, its neutron production raises a number of complex engineering difficulties as well as radiological problems. Achieving a reaction which would eliminate these would merit considerable effort. These difficulties would be largely overcome if a suitable reaction could be found which produced only charged products (including all significant secondary reactions), and if it were possible to find a confinement device of sufficient efficiency that it could be made to go.

A search of possible nuclear reactions which might be potential fusion reactions was carried out by Crocker, Blow, and Watson.[2] Among the non-neutron producting reactions which they list are the following:

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* Another possibility is the reaction

$$^{10}\text{B} = n \rightarrow 2\,{}^{4}\text{He} + {}^{3}\text{T} + 367\text{ KeV}$$

$$^{10}\text{B} + n \rightarrow {}^{7}\text{Li} + {}^{4}\text{He} + 2.9\text{ MeV}$$

$$^{7}\text{Li} + n \rightarrow {}^{4}\text{He} + {}^{3}\text{T} + n - 26\text{ MeV}$$

$$P + {}^{6}Li \rightarrow {}^{3}He + {}^{4}He + 1.8\ \mathrm{MeV}, \qquad \sigma_{Max} = 200\ \mathrm{mb}$$

$${}^{6}Li + {}^{3}He \rightarrow 2\,{}^{4}He + P + 16.8\ \mathrm{MeV}, \qquad \sigma_{Max} = 30\ \mathrm{mb}$$

$$\rightarrow {}^{8}Be + P + \gamma + 13.9 + 2.9\ \mathrm{MeV}, \qquad \sigma_{Max} > 60\ \mathrm{mb}$$

$${}^{3}He + {}^{3}He \rightarrow 2P + {}^{4}He + 12.8\ \mathrm{MeV}, \qquad \sigma_{Max} > 30\ \mathrm{mb}$$

Except for the first of these, all of these reactions involve ${}^{3}He$ which also does not occur in an appreciable quantity on earth and would also have to be manufactured.*

All the other reactions listed by Crocker et al. produce neutrons either directly or as side reactions and hence do not offer great improvements over the DT reaction. Unfortunately, the study of Crocker et al. only included reactants up through Li and thus missed probably the best exotic fuel of them all, P,${}^{11}B$:

$$P + {}^{11}B \rightarrow 3\,{}^{4}He + 8.7\ \mathrm{MeV} \qquad \sigma_{Max} \approx 700\ \mathrm{mb}$$

This reaction has recently been discussed by Weaver, Zinnerman, and Wood [3] in relation to laser pellet fusion. They give a very good discussion of the reaction, including side reactions and the abundance of the fuel, and show that if compressions of 10^5 times solid density can be achieved, it would be a feasible laser pellet fuel.

A plot of $\langle \sigma v \rangle$ Q for a number of reactions is given in Fig. 1. Q is the reaction energy given off as charged particles. These curves are computed from formulas given by Fowler.[4] It is clear that $P^{11}B$ is very favorable in terms of its reactivity at relatively low energies. It is also very favorable fuel in terms of its abundance. Its one disadvantage is the relatively high z of Boron which results in a rather high bremsstrahlung. This means that it is relatively difficult to ignite, although it may be possible. There appears to be some uncertainty in the cross section; this is discussed by Weaver et al.[3] Millie gives values of $\langle \sigma v \rangle$ which are about half those shown in Fig. 1 which, if correct, would make the ignition impossible. There seems to be a number of reasons why his values are so low. First he appears to take the pessimistic

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*It appears that in the big bang creation of the universe an appreciable amount of ${}^{3}He$ was produced, and it may be possible to find significant quantities of this material on the outer planets or their satellites, for example Titan. However, this can at best be viewed as a distant potential.

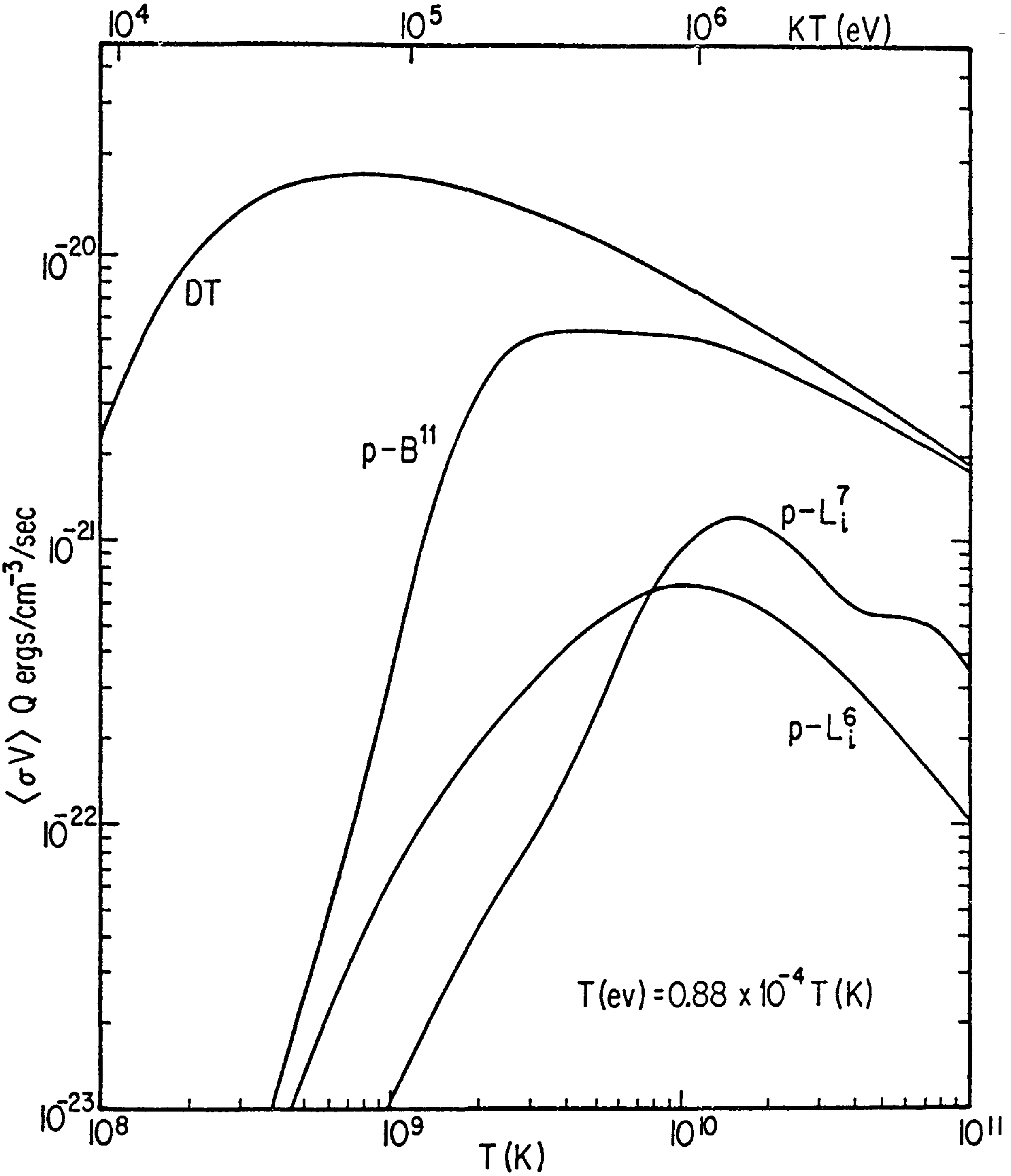

Figure 1. Reactivities of some exotic fuels.

value for the cross section. Second, there are seven resonances out to an energy of 5 MeV. Millie takes the cross sections only out to 2 MeV, but for 300 KeV temperatures there are an appreciable number of particles beyond this energy that contribute significantly to the reaction. Another discussion of the P-B reaction is given by Moreau(6) in a recent thesis. He gives an optimistic and a pessimistic curve for $\langle \sigma v \rangle$. His pessimistic value has a peak value which is also about half that of Fig. 1. His optimistic value has about the same peak value as shown in Fig. 1, but it occurs at higher energy. At 300 KeV it is about 30% lower than Fig. 1. Our value of $\langle \sigma v \rangle$ is very close to that given as the most probable value by Weaver et al.(3) which also peaks at about 300 KeV and is the most up to date result given in the literature.

Even if ignition proves impossible, it should be possible to operate in a wet wood burner mode with relatively large energy multiplication. Such operation would require sustaining electron temperatures of $100 \sim 150$ KeV, ion temperatures of 300 KeV and a hot ion tail of about 1 MeV. The system might be sustained by injection of a 1 MeV neutral hydrogen beam.

The high electron temperature is required to prevent too rapid cooling of the ions and to insure that an appreciable fraction of the reaction energy goes to the ions. Sustaining such a high electron temperature results in the conflicting condition of a high bremsstrahlung rate so one must reach a compromise. If one is to succeed at all, synchrotron radiation must be reduced to a fraction of the bremsstrahlung. This is best achieved by confining the plasma in a region of low magnetic field. Of all magnetic confinement devices, multipoles and Surmace (high order multipoles) appear to offer the best prospects for reducing synchrotron radiation. Unfortunately, the complexity of the magnetic geometry makes accurate calculations difficult.

Multipoles appear also to be very favorable from the plasma confinement point of view. Extrapolation of recent Wisconsin results indicates that it may be adequate, although much more extensive experiments would be required to answer the question. For non-neutron producing reactions, floating superconducting rings can be considered.

It is the purpose of this paper to estimate the feasibility of the PB reaction in magnetic confinement systems. As mentioned, it is difficult to accurately calculate synchrotron radiation for complex magnetic geometries, so we shall only attempt to obtain upper bounds on this radiation. We consider driven systems. Our calculations are not so accurate as to say definitely that it is possible; this would take a considerable larger study, plus better experimental knowledge of reaction cross section and of plasma confinement in multipole. Nevertheless, the results are sufficiently encouraging that it appears that a real possibility exists and, in view of the advantages of the reaction, further investigations are called for. The calculations proceed as follows:

1. We assume that, by driving the system, we can maintain the ion temperature at the value which gives the maximum reactivity (300 KeV). This can be done by injecting 1 MeV neutral hydrogen atoms generated by the acceleration of negative hydrogen ions followed by stripping the excess electron; the neutrals are ionized and trapped by the plasma. At the electron temperatures required, the energetic protons will primarily give up their energy to the ions. The energetic protons will react at an enhanced rate during their slowing down so that using a Maxwellian ion distribution gives an underestimate of the reactivity.

2. The reaction energy is compared to the bremsstrahlung from the plasma for various electron temperatures. It is found that the reaction energy is sufficient to sustain an electron temperature of 125 KeV. At this electron temperature the reactivity falls slightly short (about 20%) of that required to sustain the ions against electron cooling. By driving the system the ion temperature can be maintained, and there will be a surplus energy supplied to the electrons to make up for other losses. These calculations indicate that it may be possible to have a surplus power of 50% of the bremsstrahlung available to make up synchrotron and other losses while obtaining an energy multiplication of close to 3.

3. We calculate the power required to sustain the ions at 300 KeV for various electron temperatures assuming that the major ion energy loss is to the cooler electrons. This requires that we calculate the fraction of the reaction energy which goes directly to the ions. We have done this assuming only long range Coulomb collisions, and we find that roughly 90% of the energy goes to the ions. In fact, elastic nuclear encounters between the reaction particles, and the plasma particles, should enhance this value. Further, these plus large angle Coulomb scattering will enhance the population of the energetic ion tail

enhancing the reactivity. Weaver et al.[3] say they have examined these effects and find that the reactivity would be increased 2 $\sim$ 10%. In the sense that these effects are neglected, our estimate is conservative.

4. We estimate upper bounds for the synchrotron losses from the plasma. To do this we first assume the plasma is permeated by a uniform B field; we assume the electrons have a relativistic Maxwellian distribution. The synchrotron loss depends on the emission and absorption processes; for low frequencies (harmonics) the emission and absorption come to equilibrium and the plasma emits like a black body at these frequencies. At higher frequencies (harmonics) there is little reabsorption and the emission is the sum of the free emission by all the particles at these frequencies. The black body level and the free escape level both are upper bounds on the emission; the situation is as shown in Fig. 2. There is a critical frequency at which the plasma ceases to be optically thick and where the emission makes a transition from the black body level to the free emission level. The value of ν_c is crucial.

We obtain an upper bound on the emission using the following approach illustrated in Figs. 2b and 2c. We calculate the frequency at which black body radiation for lower frequencies is tolerable. This frequency is a function of the size of the plasma, the reflectivity of the walls, the plasma temperature and density, and the magnetic field strength. We assume that, for frequencies higher than this, the emitted radiation freely escapes from the plasma. We calculate in detail this latter loss assuming a uniform B-field inside the plasma. However, it is also relatively easy to make rough corrections to it for non-uniformity of the B-field.

We can determine a plasma size at which synchrotron emission is a fraction of the bremsstrahlung. If this size comes out of the order of a few meters or less, we consider the situation promising.

We find that, for T_e = 100 KeV and β equals one, things are promising; for T_e = 150 KeV and β equals one, they look much more difficult. However, a modest suppression of the synchrotron radiation should make operation at or near T_e = 150 KeV possible. This should be possible since these results are pessimistic for a number of reasons. They are as follows:

(a) They give an upper bound for the synchrotron loss, even for the conditions assumed (uniform B inside the plasma, a relativistic Maxwellian electron distribution, etc.).

(b) It should be possible to construct a plasma with much lower B field throughout most of its volume.

(c) Because of the rapid loss of energy by the very energetic electrons, the Maxwellian tail of the distribution function should be depleted. Since these energetic electrons radiate a disproportionate fraction of the synchrotron radiation, their suppression will reduce this radiation. It will, however, not affect other quantities, such as ion electron energy transfer, very much.

(d) Because the outer regions of the plasma, where B is large, will radiate more strongly than the inner regions where B is low, the outer regions will cool and their radiation rate will drop. The synchrotron radiation will thus be lower than assuming a uniform temperature and will be governed by energy transport to the outer regions of the plasma. Because of the above, we conclude that it is possible to maintain electron temperatures in the 100 to 150 KeV range, where significant energy multiplication is possible for $P^{11}B$.

Other plasma and energy losses must be kept to an acceptable level. Using the recent scaling laws for plasma loss on the Wisconsin[(7)] octupole, we estimate that acceptable levels can be obtained in advanced versions of that device. The extrapolations are large and may not hold up. Further, these experiments give no measurement of energy transport across fields. Thus it cannot really be said that these losses are known. It appears of great importance to extend these measurements to plasma and fields similar to those which would be required for a reactor. Such an extension does not appear to be a formidable task. By simply cooling the rings to liquid nitrogen temperature times can be extended into the second range; by puff gas filling and using the discharge techniques employed in quips devices,[(8)] a target plasma of a few eV at densities of 10^{13} should be obtainable. By heating this plasma with the Berkeley neutral beams, thermonuclear temperatures should be obtainable. (Because of the long confinement times expected, it should be possible to physically remove the filaments used to generate the target plasma before turning on the neutral beams.)

The calculations of radiation loss and burning are general and could be applied to other geometries than multipoles. Since $\beta \geq 1$ inside the plasma wi be required, there are few candidates. One possibility might be a system where the pressure is wall supported and a modest B field only serves to reduce heat transport. Electron and ion beam devices (astrons) in principle have low internal fields. However, here the theoretical studies predict low energy multiplication even for DT systems, and it is difficult to believe they could make it for $P^{11}B$. If tormac proves workable and if it can operate with a very low internal toroidal field, then it would also be a candidate. Of course, the various pellet approaches (laser, electron beam, ion beam) can in principle work as discussed by Weaver et al.(3)

Multipoles have the advantages that they are MHD stable, at least up to some critical β near one at the outside. There are, of course, other types of instabilities: drift wave instabilities, trapped particle instabilities, velocity space instabilities. The average minimum B well can stabilize drift waves if the density gradients are not too steep; particles are trapped in regions of good curvature, tending to make trapped particles a stabilizing influence rather than a destabilizing one; there appears to be no reason why the distribution function needs to be one that leads to velocity space instabilities, at least if supports and hence loss cones can be eliminated. They also have the great advantage that the internal magnetic field can be very small, so as to reduce the synchrotron level.

Details of the Calculations

I. Thermonuclear Reaction Rate

The thermonuclear reaction rate for $P\text{-}^{11}B$ is given by

$$P_{TN} = n_p n_b \langle\sigma v\rangle\, Q \qquad (1)$$

We use the data of Fowler,(4) the equation is given in Appendix 1, $\langle\sigma v\rangle$ is shown in Fig. 1. Using the maximum value of $\langle\sigma v\rangle$ given by Fig. 1, $\langle\sigma v\rangle_{Max} = 3.9 \times 10^{-16}$ and Q = 8.68 MeV:

$$P_{TN}^{Max} = 5.4 \times 10^{-28}\, n_p n_b \text{ watts/cm}^3 \qquad (2)$$

II. Bremsstrahlung

We use semi-empirical formulae for the relativistic bremsstrahlung from a plasma, which are obtained from fitting the results of Maxon.(9)

First, for the electron-ion bremsstrahlung

$$P_{\text{Brem i.e.}} = 9.3 \times 10^{-14}\, n_e^{\,2}\, \bar{z}\, T_e^{1/2} \left[1 + 2\,(T_e/m_e c^2)\right] \ \text{eV/cm}^3\ \text{sec}$$

$$\bar{z} = \Sigma_i\ n_i z_i^{\,2} \,/\, \Sigma\, n_i z_i \tag{3}$$

where T_e is in eV.

This formula has the correct low temperature form. The high temperature value is not asymptotically correct. However, it fits the results given by Maxon[9] in the region of interest fairly well, and even at T_e = 50 MeV it over-estimates the rate by only a factor of 3.

For the electron-electron bremsstrahlung, which becomes important at the temperatures of interest, the following semiempirical formula was used:

$$P_{\text{Brem, cc}} = \frac{2P_i}{\bar{z}} \quad 1 - \left\{\frac{1}{(1 + T_e/mc^2)^2}\right\} \tag{4}$$

which also fits the results of Maxon[9] fairly well.

As one method of optimizing the ratio of boron to hydrogen, we can maximize the ratio of thermonuclear power production to the total bremsstrahlung. Using equations (1), (3), and (4) and the definition of $\bar{z}$, this relation is given by

$$\frac{P_{TN}}{P_{Brem}} = \frac{1.08 \times 10^{13}\ \varepsilon \,\langle\sigma v\rangle\, Q}{(1+5\varepsilon)\ (1+25\varepsilon) \left[1 + \frac{2(1+5\varepsilon)}{1 + 25\varepsilon}\left\{1 - \frac{1}{(1 + T_e/mc^2)^2}\right\}\right] T_e^{1/2}\ (1 + 2T_e/mc^2)} \tag{5}$$

where $\varepsilon = n_b/n_p$. Maximizing (5) with respect to ε, we find:

Table I

$T_e/m_e c^2$	ε_{Max}
0.02	0.089
0.2	0.107
0.3	0.112

T[illegible] roughly, an ε of 0.1 will maximize P_{TN}/P_{Brem}. Associated with this ε is a $\bar{z}$ of 2.33. As we shall see later, other considerations favor a higher ratio of boron. Since the ratio of thermonuclear power to bremsstrahlung is insensitive to the fraction of boron near this density, it is advantageous to increase the boron density to perhaps 0.2 n_p.

We can compute the radiative cooling time for the electrons by equation:

$$P_{Brem}\ \tau = \frac{3}{2} n_e T_e$$

We find

Table II

$\varepsilon = 0.9$	$\varepsilon = 0.2$	
$n_e\tau = 1.23 \times 10^{15}$	7.4×10^{14}	$T_e = 100$ KeV
$n_e\tau = 1/23 \times 10^{15}$	7.4×10^{14}	$T_e = 150$ KeV

The use of $\varepsilon = 0.2$ rather than 0.1 reduces P_{TN}/P_{Brem} by 7 to 10%.

III. Cooling of the Ions by Electrons

Since we cannot sustain the electron temperature at the required ion temperatures because of bremsstrahlung and synchrotron losses, ion cooling by the electrons is unavoidable.

The power lost from the ions to the electrons[10] is

$$P_{ie} = \frac{16\sqrt{\pi}}{3/2} \frac{e^4 n_e \ \ell n\ \Lambda}{T_e^{3/2}} m_e^{1/2} \ \Sigma_i \frac{z_i^2 n_i}{m_i} \frac{3(T_i - T_e)}{2} \tag{7}$$

Applying this to P-^{11}B and taking the ions to have all the same temperature gives

$$P_{ie} = \frac{(n_p + 2.27\ n_b)\frac{3}{2}(T_i - T_e)}{\tau_{pe}} \tag{8}$$

where

$$\tau_{pe} = \frac{3\sqrt{2}}{16\sqrt{\pi}} \frac{T_e^{3/2}\ m_p}{e^4\ n_e\ m_e^{1/2}\ \ell n\ \Lambda}$$

with the subscript p applying to protons. The actual ion cooling time is given by

$$\tau_{ie} = \frac{3}{2} (n_p + n_b) \frac{T_i}{P_{ie}} \tag{9}$$

Taking $\ell n \Lambda = 20$ and $T_i = 300$ KeV, we obtain some numerical values of interest:

Table III

T_e	$n_e \tau_{pe}$	$n_e \tau_{ei}$
100 KeV	3.8×10^{14}	4.5×10^{14}
150 KeV	7.0×10^{14}	1.1×10^{15}

Another criterion for optimizing the boron density to proton density is to maximize the thermonuclear power production to the power required to sustain the ion temperature against electron cooling for a fixed electron temperature. Writing $n_b/n_p = \varepsilon$, this ratio is given by

$$\frac{P_{TN}}{P_{ie}} = \frac{\varepsilon}{(1 + 2.27\varepsilon)(1 + 5\varepsilon)} F(T_i, T_e) \tag{10}$$

where F is the ratio of $<\sigma v>$ Q to the cooling factor in equations (7) and (8). The optimum value of ε for this expression turns out to be

$$\varepsilon = 0.30$$

This is to be compared to the value of about 0.1 obtained when we optimized with respect to bremsstrahlung loss. In reality, we should try to maximize with respect to the sustaining power while self-consistently calculating the electron temperature, including all forms of electron energy loss. We have not done this, but the optimum ε probably would fall somewhere between 0.1 and 0.3; the value 0.2 should not be a bad choice. Using expression (10) and Fig. 1 for $<\sigma v>$, we may compute the ratio P_{TN}/P_{ie}. We find the values given in the following table:

Table IV

T_e	T_i	P_{TN}/P_{ie}	ε
100 KeV	300 KeV	0.45	0.2
125 KeV	300 KeV	0.72	0.2
150 KeV	300 KeV	1.10	0.2
100 KeV	300 KeV	0.46	0.3
125 KeV	300 KeV	0.73	0.3
150 KeV	300 KeV	1.13	0.3

IV. Fraction of the Reaction Power Going Directly to the Ions

Of the power required to sustain the ions, a certain fraction comes directly from the reaction products. If all of it can be supplied by the reaction products, then we have ignition provided the assumed electron temperature can be sustained. The rate of loss of energy by the α particles to the electrons[10] is given by the equation

$$\left[\frac{dW_\alpha}{dt}\right]_e = -\frac{64}{3}\sqrt{\pi/2}\,\frac{e^4 n_e \ln\Lambda_e}{T_e^{3/2}}\,\frac{m_e^{1/2}}{m_\alpha}\,W_\alpha = -\frac{W_\alpha}{\tau_{e\alpha}} \tag{11}$$

The rate of loss of energy to the ion species i is given by [10]

$$\left[\frac{dW_\alpha}{dt}\right]_i = -\frac{2\pi z_\alpha^2 z_i^2 e^4 n_i}{m_i}\left[\frac{2m_\alpha}{W_\alpha}\right]^{1/2} \ln\Lambda_i$$

Because the minimum impact parameter to be used for ion α collisions is considerably smaller than that for electron α collisions (the α ion distance of closest approach as compared with the electron De Broglie wavelength), we have made a distinction between $\ln\Lambda$ for electrons and ions, $\ln\Lambda_i$, is nearly the same for all ions but is about 20% larger than $\ln\Lambda_e$. The total energy going to all ions is

$$\left.\frac{dW_\alpha}{dt}\right|_{\text{All } i} = -\,2\pi\, z_\alpha^{\,2}\, e^4 \left(\frac{2m_\alpha}{W_\alpha}\right)^{1/2} \ln \Lambda_i \,\Sigma_i \,\frac{z_i^{\,2}\, n_i}{m_i} \tag{13}$$

Adding to this, the rate of loss of energy to the electrons gives the total rate of loss energy by the α particles. Dividing the rate of loss energy to the ions by the total rate of loss of energy gives

$$\frac{(dW_\alpha)_{\text{All } i}}{dW_\alpha} = \frac{1}{1 + \dfrac{4}{3\sqrt{\pi}} \left(\dfrac{W_\alpha}{T_e}\right)^{3/2} \dfrac{\ln \Lambda_e}{\ln \Lambda_i} \left(\dfrac{m_e}{m_\alpha}\right)^{1/2} \dfrac{n_e}{m_\alpha \,\Sigma_i\, n_i z_i^{\,2}/m_i}} \tag{14}$$

Integrating (14) from $W_\alpha = 0$ to the birth energy for the α's and dividing by the birth energy gives the fraction of the energy going to the ions. The following table lists the fraction of energy going to the ions for P-^{11}B for various values of T_e and $\varepsilon = n_b/n_p$, assuming the α's are born at 3 MeV.

Table V

$\varepsilon = 0.1$	$\varepsilon = 0.2$	$\varepsilon = 0.3$
T_e = 100 KeV	T_e = 100 KeV	T_e = 100 KeV
f = 88.2%	f = 87.0%	f = 86.0%
T_e = 150 KeV	T_e = 150 KeV	T_e = 150 KeV
f = 93.3%	f = 92.5%	f = 91.9%

These figures, by neglecting the effect of ion temperature, somewhat overestimate the fraction; on the other hand, by neglecting elastic nuclear scattering, they underestimate the fraction. They also do not include the effects of a spectrum of energies at birth. Nevertheless, it is clear that roughly 90% of the reaction energy will go to the ions and subsequently be passed on to the electrons.

With slightly more than 90% of the reaction energy going directly to the ions, we see from Table IV that at a 150 KeV electron temperature the elevated ion temperature can be maintained by the reaction alone. In our bremsstrahlung calculation we found that the reaction energy was sufficient to balance bremsstrahlung at T_e = 125 KeV. Thus it appears that one cannot quite reach ignition (a conclusion reached by Weaver et al.[3]), but it is close, and with luck (slightly higher cross section), winition might be obtained. Further, there are some effects that will help, as for example a slight increase in reactivity (5 $\sim$ -10%) due to nuclear knock ons and large angle Coulomb scattering of ions to energies of 700 KeV or greater.

In any case, the difference in reactivity could be made up by an input power of 20% of the thermonuclear power, giving an energy multiplication of 5. One good way to do this is to inject an energetic neutral hydrogen beam at about 1 MeV. Such beams, particularly for this energy, can be made very efficient. Just as most of the α energy went to the ions, so will most of the proton energy. The energetic protons will have enhanced reactivity and will produce a certain number of reactions as they cool down to the ion temperature. Accurate answers here require detailed Fokker-Planck calculation. However, rough estimates indicate that in slowing down the reactions will produce about 20% of the beam energy. Thus the total power production would be roughly

$$P_{tot} \approx P_{Tn} + 0.2\, P_{Beam}$$

With beam powers of 30% of the thermonuclear power, energy multiplications of 3.5 could be achieved. Whether or not this is sufficient depends on efficiencies of recovering the plasma energy and of producing the beam. Such a beam would supply the plasma with 36% more energy than the electrons are radiating by bremsstrahlung at 125 KeV and more than enough to sustain the ions at 300 KeV, assuming their major energy loss is to the colder electrons. Thus there would be a substantial fraction of the bremsstrahlung power available to make up for other losses.

As mentioned, there are a number of effects that will enhance the thermonuclear yield. On being slowed down by Coulomb collisions, an α particle does not distribute its energy in a Maxwellian fashion among the ions, but rather distributes it more or less according to the law

$$P(\Delta w)\, d\Delta w \propto d\Delta w/\Delta w^2 \qquad (15)$$

where $P(\Delta w)$ is the probability an ion picks up energy Δw; Δw must, or course, be less than the maximum transferable energy. Thus a long energetic tail will be produced. There will be roughly $(\ell n\ \Lambda)^{-1}$ large angle scatterings per α particle, producing protons or boron ions with energy comparable to the α's and ten times this many producing an ion of one-tenth the α energies or greater. Thus, since each reaction produces 3α's, there will be roughly one proton that gains 300 KeV and that will thus be injected into the most reacting 600 KeV region. This will enhance the reactivity and, perhaps, let us operate at lower ion temperature. Rough estimates obtained from the number of reactions expected during an ion-ion collision time give about 5% enhancement in the reactivity. Weaver et al.(3) estimate these effects as increasing the reactivity by 2 to 10%.

A similar effect is provided by elastic nuclear scattering, which is not negligible at these energies. Accurate estimates of this require detailed elastic cross sections and kinetic calculations.

V. Synchrotron Losses

In order for the $P\text{-}^{11}B$ reaction to succeed, we must be able to reduce all other energy losses to a fraction of that of bremsstrahlung. The exact fraction can only be determined by much more detailed calculations, but a value as large as 40% of the bremsstrahlung rate might be acceptable. For magnetically confined systems, a large additional source of energy loss is synchrotron radiation. We shall now try to find an upper bound to this loss. For the low harmonics, the plasma will be optically thick, and these will be emitted as if the plasma were a black body at these frequencies. For the higher harmonics, the plasma is optically thin and there is little absorption of these; they escape freely from the plasma. The situation is as shown in Fig. 2a. The critical frequency depends on the size of the plasma, the strength of the magnetic field, the plasma density and temperature, and the reflectivity of the walls.

Our calculation of the upper bound to the synchrotron radiation consists of two parts: first we calculate the maximum frequency at which the black body radiation is acceptable; second we assume that all radiation beyond this frequency freely escapes and compute whether or not it is acceptable. Adding these two radiations together gives an upper bound to the total radiation as shown in Fig. 2b and 2c. By increasing the size of the plasma, one can always come to

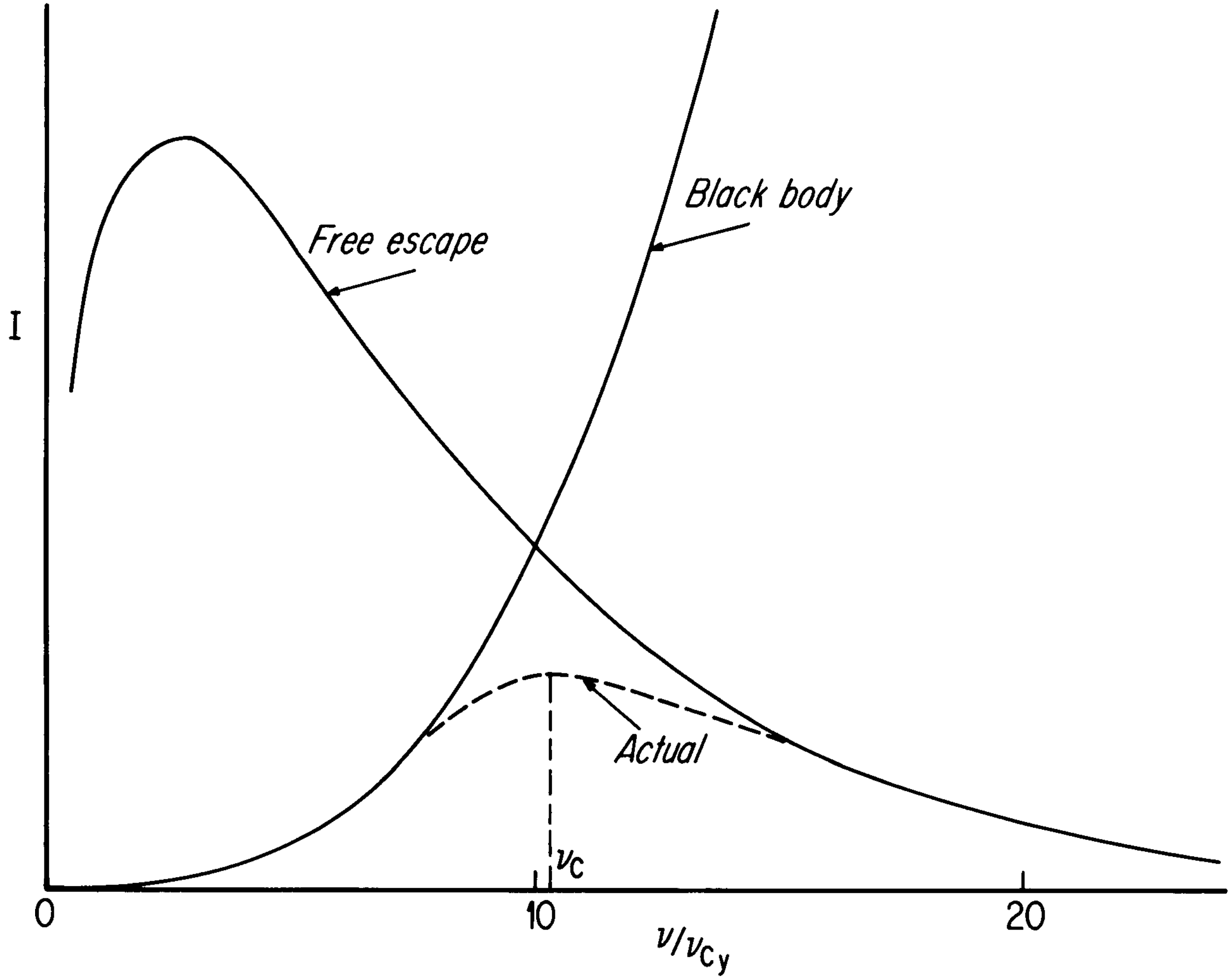

Figure 2a

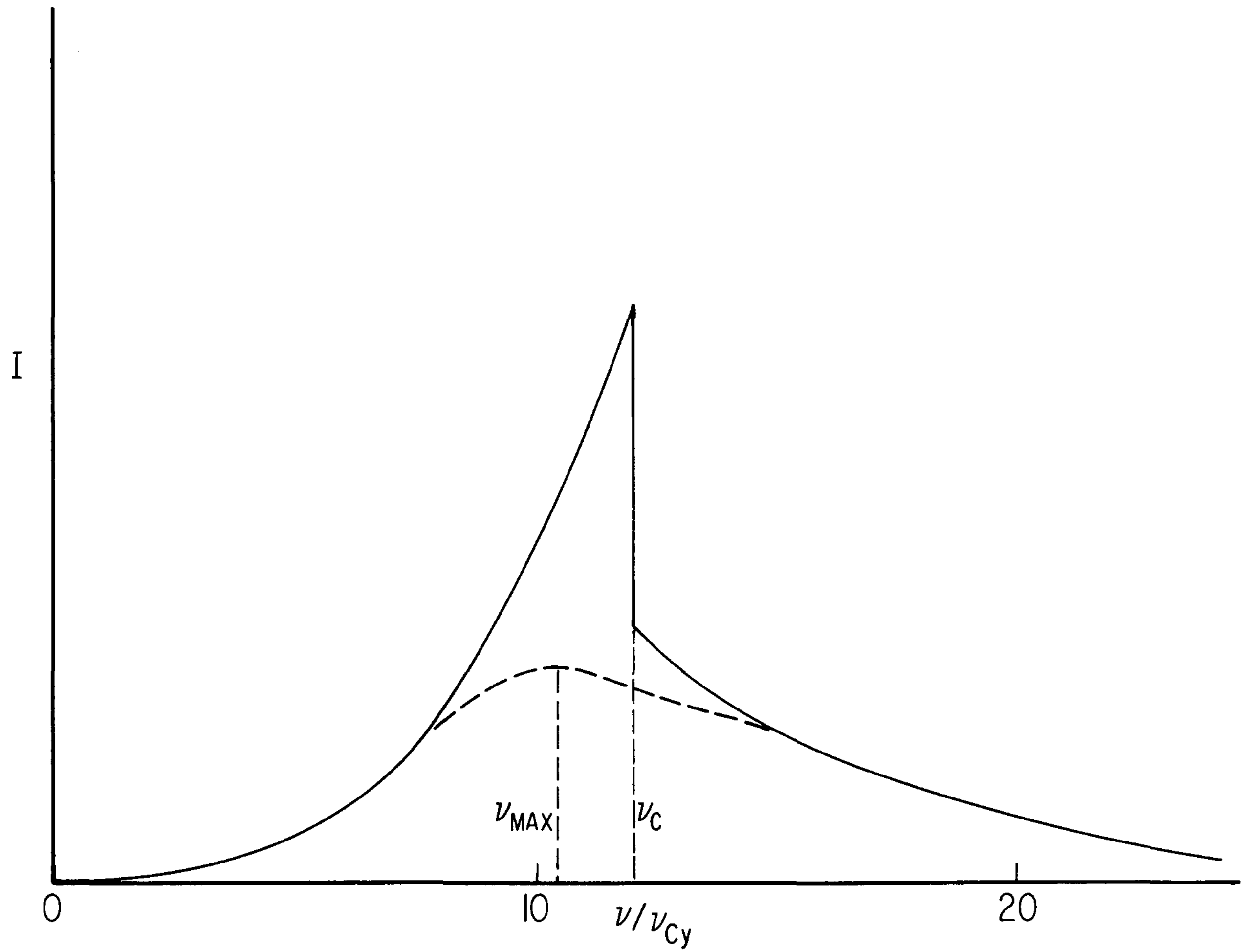

Figure 2b

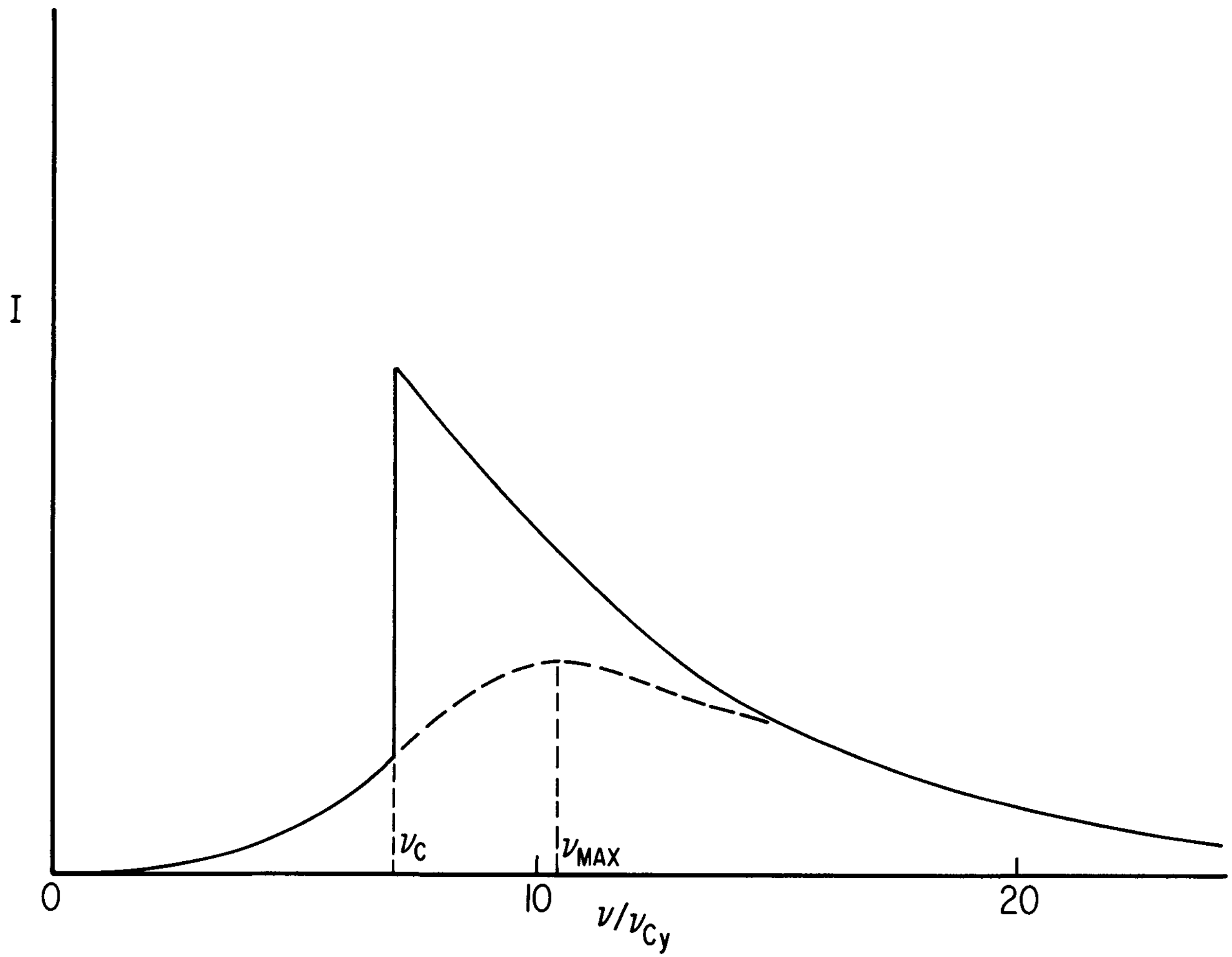

Figure 2c

a size where the synchrotron emission is acceptable. If this size is of the order of meters, then we regard the size as reasonable.

We will assume that the magnetic field is uniform within the plasma, although, as we shall discuss later, it will be advantageous, indeed necessary, to have as low a field in the bulk of the plasma as can be achieved. Thus the field will be nonuniform inside the plasma, rising to a value where $\beta < 1$ at the surface. However, our method of estimating is still valid for obtaining an upper bound.

a. Maximum Acceptable Black Body Level

The black body emissivity per unit area is given by the Rayleigh-Jeans Law

$$I(\nu)\, d\nu = \frac{4\pi K T_e \nu^2 d\nu}{c^2} \tag{16}$$

and the total emissivity out to frequency ν_c is

$$P(\nu_c) = \frac{4\pi}{3} \frac{K T_e \nu_c^3}{c^3} = \frac{4\pi K T_e c}{3\lambda_c^3} \tag{17}$$

Here ν_c is the critical frequency beyond which the plasma must not be black. Taking the plasma to be cylindrical in shape of radius a_o, the electron cooling time due to this radiation is

$$AP(\nu_c)\tau = \frac{3}{2} n_e K T_e V$$

$$\tau = \frac{9}{16\pi} \frac{n_e \lambda_c^3 a_o}{c} \tag{18}$$

If the chamber walls have a reflectivity coefficient R, then τ is multiplied by $(1-R)^{-1}$. Taking this into account, multiplying (17) by n_e and solving for λ_c gives

$$\lambda_c = \left[\frac{16}{9} \frac{n_e \tau c(1-R)}{n_e^2 a_o}\right]^{1/3} \tag{19}$$

In computing the bremsstrahlung, we found that $n_e\tau$ for that process was about 10^{15}. Requiring that this process be weaker by a factor of 4 means $n_e\tau = 4 \times 10^{15}$, and we obtain for λ_c

$$\lambda_c = 8.75 \times 10^8 \left[\frac{(1-R)}{n_e^2\ a_o}\right]^{1/3} = 8.75 \times 10^8 \left[\frac{(1-R)}{n_p^2\ (1+5\varepsilon)^2\ a_o}\right]^{1/3} \tag{20}$$

Taking $n_e = 10^{14}$, $a_o = 200$, and $R = 0.99$ (a reasonable figure for the frequency range of interest) gives

$$\lambda_c = 1.5 \times 10^{-2} \tag{21}$$

which corresponds to the 12th harmonic for a 50 KG field.

Taking the ratio of plasma pressure to magnetic pressure to be β, then

$$\frac{B^2}{8\pi} = \frac{n_eT_e + n_iT_i}{\beta} = \frac{n_p}{\beta}\left[(1+5\varepsilon)T_e + (1+\varepsilon)T_i\right]$$

$$B = 6.3 \times 10^{-6}\left[\frac{n_p}{\beta}\left[(1+5\varepsilon)T_e + (1+\varepsilon)T_i\right]\right]^{1/2} \tag{22}$$

where T_e and T_i are in eV. Using the expression for the cyclotron wavelength,

$$\lambda_{cy} = \frac{\pi\ 10^{-7}c}{B} = \frac{1.05 \times 10^9\ \sqrt{\beta}}{\left[n_p\ \left[(1+5\varepsilon)T_e + (1+\varepsilon)T_i\right]\right]^{1/2}} \tag{23}$$

Dividing (22) by (19) gives the last harmonic at which we can afford for the plasma to be black

$$N = \frac{1.2\ \sqrt{\beta}\ n_p^{1/6}}{\{\ (1+5\varepsilon)T_e + (1+\varepsilon)T_i\}^{1/2}}\left[\frac{(1+5\varepsilon)^2\ a_o}{(1-R)}\right]^{1/3} \tag{24}$$

Because $N_p \propto B^2$, we may also conclude from this that $N \propto B^{1/3}$. As a specific example, take $T_e = 150$ eV, $\varepsilon = 0.2$, $a_o = 300$, $R = 0.99$ (not unreasonable for metallic walls at these frequencies), $\beta = 1$, and $n_p = 10^{14}$. We then find $N = 13.7$.

We must now estimate the radiation at frequencies higher than this. We do this by assuming it is the sum of all the single particle emissions and it all freely escapes. To make this estimate, since N is large, we employ the asymptotic expression for emission of the nth harmonic at large n.[11]

$$I_n = \frac{e^4 B^2 (1 - v^2/c^2)^{5/4}\, n^{1/2}}{2\sqrt{\pi}\, m_e^2\, c^3} \left[\frac{\frac{v}{c}\, e^{(1 - v^2/c^2)^{1/2}}}{1 + (1 - v^2/c^2)^{1/2}} \right]^{2n} \qquad (25)$$

where v is the electron velocity. To find the total emission, we must sum this over all n such that $n > \gamma N$, $\gamma = (1 - v^2/c^2)^{-1/2}$, the γ factor coming from the relativistic shift in the cyclotron frequency for an energetic particle, then multiply by the distribution function for v and integrate over all velocities. In our estimate we have used the two dimensional relativistic Maxwellian

$$f(\gamma)\, d\gamma = \frac{T}{m_o c^2} \exp\left[-(\gamma - 1) \frac{m_o c^2}{T} \right] d\gamma \qquad (26)$$

and have not included the effects of parallel motion which are small. We find the following values:

Table VI

N = 13	$\dfrac{I\,(\nu > 13\nu_c)/m^2c^3}{e^4B^2}$	$\dfrac{\frac{3}{2} T_e}{I\,(\nu > 13\nu_c)} = \tau$
T = 100 KeV	3.0×10^{-4}	$9.9 \times 10^{10}/B^2$
T = 150 KeV	3.1×10^{-3}	$1.5 \times 10^{10}/B^2$
N = 26		
T = 100 KeV	3.6×10^{-5}	$8.2 \times 10^{11}/B^2$
T = 150 KeV	6.4×10^{-4}	$7.1 \times 10^{10}/B^2$

Here τ is the radiation cooling time for the electron if the only energy loss mechanism were synchrotron radiation for frequencies $\nu > N\nu_c$. The values for $N = 26$ are used here to show the effect of reducing the B field in the bulk of the plasma. For a $\beta = 1$, P-^{11}B plasma with $\varepsilon = 0.2$, $n_p = 10^{14}$, $T_i = 300$ KeV, and T_e equals the values given, we get

Table VII

$T_e = 100$ KeV	$B = 46$ KG	$N = 13$	$n_e\tau = 9.4 \times 10^{15}$
$T_e = 150$ KeV	$B = 50$ KG	$N = 13$	$n_e\tau = 1.18 \times 10^{15}$
$T_e = 100$ KeV	$B = 46$ KG	$N = 26$	$n_e\tau = 7.8 \times 10^{16}$
$T_e = 150$ KeV	$B = 50$ KG	$N = 26$	$n_e\tau = 5.8 \times 10^{15}$

These values should be compared to the bremsstrahlung cooling time of $n_e\tau \simeq 8 \times 10^{14}$ for this ε. Thus we see that, at 100 KeV, the emission from $\nu > 13\nu_c$ is an order of magnitude below the bremsstrahlung and should be negligible. At 150 KeV it is about 2/3 of the bremsstrahlung, and this is not negligible. However, if most of the plasma could be located in a low field region, such that $N = 26$, then even here it should be possible to make high harmonic synchrotron emission a small fraction of the bremsstrahlung. At the very least it will be reduced due to the B^2 factor. The lower frequency synchrotron emission would still be limited by the black body level and so should be acceptable for the size plasma and wall reflectivity we have assumed.

<u>Some effects which will reduce the synchrotron emission</u>

There are a number of effects which will tend to reduce the synchrotron emission. First, the more energetic particles emit most strongly, particularly at high frequencies. It will not be possible for the electrons to maintain the tails of the Maxwellian against this loss. There will be accompanying reduction in the synchrotron emissions. If all the synchrotron emission escaped from a $\beta \gtrsim 1$ plasma, electron collisions could not maintain the tail beyond 700 KeV.

One really must solve for the synchrotron radiation and electron distributions together self-consistently.*

A second effect which will take place in a system such as a multipole with weak field in the center and high field towards the outside is radiation cooling of the electrons in the high field region so that their synchrotron emission will be less. The problem then becomes one of heat transport combined with radiation cooling. For multipoles this cooling of the outside should not upset stability from the MHD point of view because the hot plasma tends to reside at the bottom of the magnetic well. The system could be subject to more subtle instabilities such as those associated with an anisotropic distribution function since the synchrotron radiation cools the perpendicular temperature. However, collisions are sufficiently strong to remove these except for rather energetic electrons, $E \gtrsim 700$ KeV. Such instabilities or other anomalous processes could lead to enhanced heat transport. However, the synchrotron emission should be less than that for a uniform temperature plasma.

Multipoles--A Possible Magnetic Confinement System for $P\text{-}^{11}B$

From the previous discussion it appears that the synchrotron emission for a $\beta = 1$ system can be made negligible compared to bremsstrahlung for a 100 KeV electron temperature and a reasonable size plasma. It also appears that, if it can be reduced by a factor of 5-10 or so, it can be made negligible at $T_e =$ 150 KeV, in which case the possibility exists for obtaining useful energy from the $P\text{-}^{11}B$ reaction. The only practical way to reduce the synchrotron emission is by reducing the magnetic field internal to the plasma. It is possible this can be done with a high beta plasma with B rising to a much higher value outside the plasma than it has inside. The most straightforward way to do this appears to be through the use of multipoles or surmacs (high order multipoles) with their intrinsic low internal field. Their strong average minimum B makes them quite stable and should allow them to reach high values of β, possibly even to $\beta \simeq 1$ at the outside. Very good plasma confinement is also required, and the intrinsically good confinement of multipoles is also a great advantage here.

*This is also true, but to a lesser extent, for bremsstrahlung, particularly when relativistic effects are included. Thus it is possible that a small reduction in the bremsstrahlung radiation will also occur.

The magnetic field in a multipole increases as one moves from the center outward towards the conductors. Near the center the field goes as $B \propto r^{n-1}$ $\cos(n\Theta)$, where n is the order of the multipole. Let us take as a model the field going as r^{n-1} from the center to the edge at $r = a_o$ and assume that $\beta = 1$ at $r = a_o$. The field reaches half the $\beta = 1$ value at $r = (a_o/2)^{1/n-1}$. As an example we might take a situation with six conductors. Then the 1/2 field point would be at $(a_o/2)^{1/5}$ and the region inside this would constitute $2^{-2/5}$, or 75% of the volume. The radiation from this internal region would be negligible because both the critical harmonic number is more than twice as large as for a $\beta = 1$ plasma and because of the B^2 dependence of the radiation. Thus only the outer 25% of the plasma will be radiating strongly. Use of higher order multipoles could reduce this somewhat more, but it is probably not practical to reduce the radiating region below 10% of the plasma volume. This, however, should be sufficient to reduce the high harmonics synchrotron radiation to acceptable levels. The black body estimate still provides an acceptable upper bound for the low frequency regime.

Besides the synchrotron and bremsstrahlung loss, there is the problem of plasma and heat conduction losses. The strong stability of the multipole geometry is very favorable in this respect.

Recent measurements at Wisconsin[7] of plasma losses from multipoles indicate a vortex diffusion loss scaling like

$$D = \frac{400\ v_{Ti}}{\sqrt{n\ell}} \tag{27}$$

where v_{Ti} is the ion thermal velocity, n is the density, and ℓ is the length of a line of force. This is independent of B but goes like $n^{-1/2}$. If this formula is extrapolated to a $P\text{-}^{11}B$ reactor, T_i = 300 KeV, $n = 10^{14}$, ℓ = 4000 cm, a_o = 500 cm, then D comes out to be 400 cm^2/sec. Assuming the density gradient is confined to the outer one meter, the diffusion velocity becomes 4 cm/sec, giving a confinement time of

$$\tau = R/2v_o = 60 \text{ sec} \tag{28}$$

This confinement time would give an $n\tau$ of 6×10^{15}, which would be adequate. The measurements were only of plasma confinement and gave no information on heat confinement.

The Wisconsin group found that a small toroidal field could reduce the diffusion by an order of magnitude below equation (27) in the event that the rings were floated. In fact, for this situation they found a confinement proportional to n up to densities of a few times 10^{11}.

Experiments at GA[12] have shown plasma losses given by (27) at low densities, but at higher densities they find the diffusion is 10^{-3} times the Bohm value, i.e.,

$$D = \frac{6 \times 10^3 \, T_e}{B} \tag{29}$$

when T is in eV and B in Gauss. It should be noted that the GA measurements were made with supported rings which Wisconsin found increased the losses.

For a P-B reaction with B = 50 KG and T_e = 150 KeV, equation (29) gives $D = 1.8 \times 10^4$ cm/sec. This would give about 2 sec confinement for the device considered. This is too short. The use of higher field would help (at 100 KG by a factor of 8). The maximum size might also be increased, although it is already getting quite large.

It is rather clear that we really do not know what multipole plasma confinement will be like, particularly at thermonuclear temperature and density. The recent unexpected results on Alcator showing confinement improving proportional to n shows that there can be unanticipated favorable developments and the same could be true in multipoles.

It appears that an experiment is called for with reactor-type plasmas, $n \gtrsim 10^{13} \sim 10^{14}$, $T \gtrsim 10^4$ eV, with times in the second range and with floating rings. Such an experiment could be carried out on an upgraded version of the Wisconsin octupole. By cooling the rings with liquid nitrogen, second long L/R times could be obtained; if liquid He cooling were used, even longer times (several minutes) could be obtained. These times could be further lengthened by modest increases in the dimensions of the device. With scaling like that f[illegible]d at Wisconsin, or even at GA, a low temperature plasma would have many second long confinement. This is sufficiently long that a target plasma could be created by the filamentary discharge method used at UCLA,[8] (plasma with densities up to 10^{13} have been created this way; with puff gas filling the neutral gas load should not be a problem, the filaments could be physically removed, and teh plasma heated to high temperature by intense neutral beams such as are provided by Berkeley sources[13]). Such an experiment should go a long way to proving whether or not multipole confinement is adequate. If multipole confinement proves adequate, more ambitious experiments using P-^{11}B would be called for.

Because of the absence of neutron production by the P-^{11}B reaction, one can conceive of using superconducting rings in the multipole. Such a ring must be shielded from the x-rays and the high heat load. A possible ring design is shown in Fig. 3.

On the outside would be a layer of tungsten which would absorb x-rays and reradiate most of the energy as optical radiation. Inside this would be many layers of reflective material in a vacuum. Next would be a layer of material which melts and absorbs a lot of heat without a temperature rise. This would be followed by more superinsulation and then the superconductor. The superconductor must have a radius of $1/3 \sim 1/2$ of the ring radius so as not to be located in too high a field region. The large size that a reactor must be makes it possible to consider rings of meter radii and of fairly complex structure. It is even conceivable to cool the superconductor between the side of the ring facing the plasma and that opposite to it to operate the driving heat engine. Most of the physics and much of the engineering could be tested out on modest size devices with ordinary conducting rings before one went to such complex structures.

Because the P-^{11}B reaction is so marginal, we must avoid radiation losses caused by impurities. This might be done by coating the walls with boron, which is a high temperature material and below 2000^{0}K has a sufficiently low vapor pressure that it would not unduly load the plasma. The question of the reflectivity of a boron-coated surface to synchrotron radiation is a critical one. At room temperature, boron is a nonconductor. A thin coating of this over a good conductor should not absorb much of the radiation. However, at the high

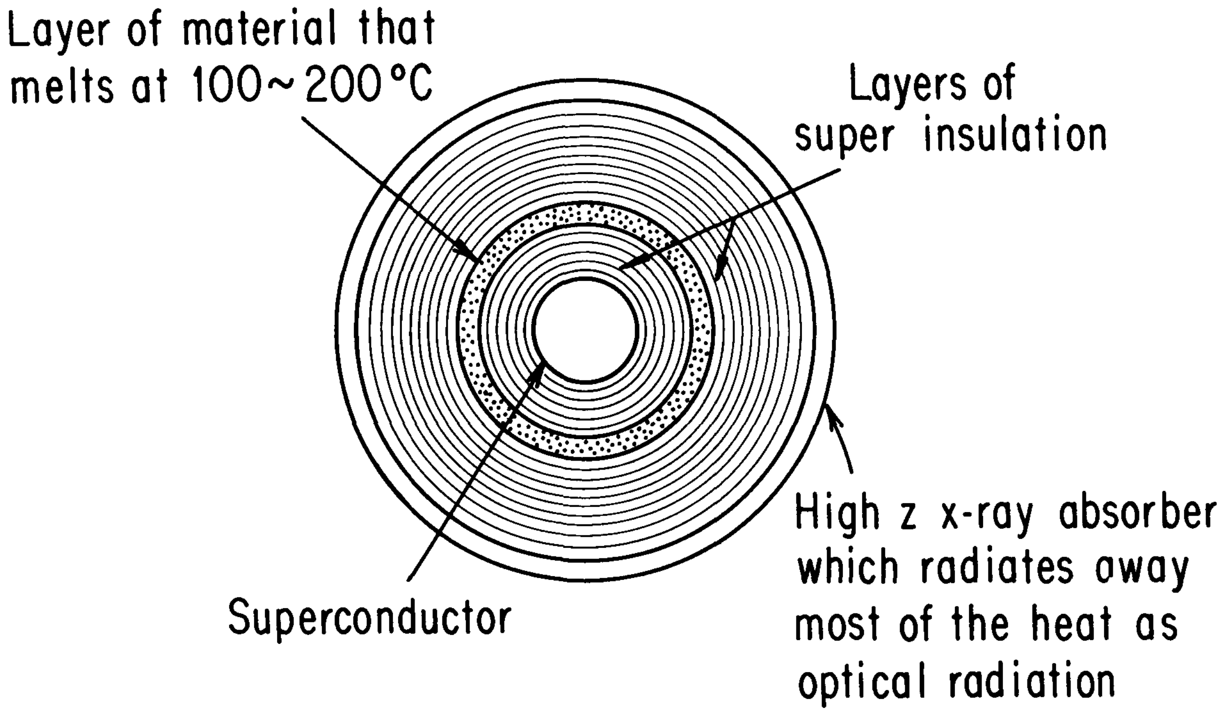
Layer of material that
melts at 100~200°C
Layers of
super insulation
High z x-ray absorber
which radiates away
most of the heat as
optical radiation
Superconductor

Figure 3

perature where the device must operate, and in the x-ray and plasma environment, things may be different.

Acknowledgment

The author is indebted to Dr. F. Chen for valuable discussions and for his help in obtaining the expression for $<\sigma v>$.

Appendix I

The following expression[4] was used to compute $<\sigma v>$ for the P-^{11}B reaction.

$$P + {}^{11}B \rightarrow 3\ {}^{4}He + 8.682\ \text{MeV}$$

$$\begin{aligned} <\sigma v> = {} & 4.301 \times 10^{-13}\ T^{-2/3} \exp\left[-12.095/T^{1/3} - (T/2.02)^{2}\right] \\ & \{1 + 0.035\ T^{1/3} + 1.22\ T^{2/3} + 0.295\ T + 2.15\ T^{4/3} + 1.32\ T^{5/3}) \\ & + 1.229 \times 10^{-17}\ T^{-3/2} \exp\left[-1.733/T\right] \\ & + 1.352 \times 10^{-14}\ T^{-3/2} \exp\left[-7.177/T\right] \\ & + 2.839 \times 10^{-15}\ T^{-2/3} \exp\left[-12.696/T\right] \end{aligned}$$

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SURMACS

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ABSTRACT

Although advanced fuels with low neutron yields are very desirable, only recently are specific confinement configurations proposed which might contain such fuels reacting at much higher temperatures than DT. One axisymmetric confinement scheme is described here which has low synchrotron radiation, optimum confinement times with wall stabilization, good accesibility to beam and RF heating, steady state operation and can be constructed from modular units. The present status of physics leading to our design is summarized together with crucial problems to be investigated.

I. INTRODUCTION

Surmacs stand for surface magnetic confinement devices[1] which contain plasmas in a central, nearly magnetic field-free volume surrounded by a surface layer of magnetic field generated by internal conductors placed near the surface; the thickness of the surface layer is an order of magnitude smaller than the overall diameter. While a large number of surface configurations exist, the configuration of closed surface field lines created by two layers of conductors carrying oppositely directed currents is chosen because of its superior confinement properties. Surmacs are naturally high β confinement devices. The plasma created in the minimum $\beta = 1$ boundary layer.
One reason for using a large number of surface conductors of multipoles of high order is to achieve a rapid decay of the magnetic field away from the surface; hot and dense plasmas created inside would more easily inflate the surface field to the $\beta = 1$ condition near the surface. If multipoles of low order are used, plasmas are generated in a region of significant magnetic fields and inflation cannot be easily achieved. The high β operation of this device requires a minimum magnetic field which together with the lower plasma density and temperature at the surface can keep the energy loss through synchrotron radiation at a tolerable level for advance fuel cycles where electron temperatures exceed 100 KeV.

The Surmac concept is built on rather sound and simple physics footings. In axisymmetric devices plasma equilibrium exists in the presence of surface magnetic fields. All field lines and plasma drift surfaces are closed and do not leave the confinement system. The main plasma volume is stabilized by average, good magnetic field curvature which has been amply proven by experimental data. The central field free region is indeed found to contain uniform and quiescent plasmas as expected from a simple theory.

Axisymmetric toroidal Surmac configurations can now be built with multiple floating superconducting rings. Recent technological advance[2] in our laboratory has produced a relatively simple scheme of levitating many rings simultaneously. These superconducting geometries without internal supports are expected to produce long plasma confinement times and are particularly suited to the containment of advanced fuels[3] such as p-^{11}B, which produce low neutron yields. The radiation shield on the current ring is a fraction of the ring diameter. Other advance forms of remote energy feeds to floating rings are presently investigated which could eliminate the necessity of using superconductors at liquid helium temperatures. In general a large Surmac (5M-10M in diameter) is preferred, which, however, does not necessarily imply a costly device because the main volume is field free and the magnetic configuration is reasonably simple and can be built from modular units.

The surface magnetic configuration makes it simple to heat or refuel Surmacs by neutral beams. Particle beams composed of partially stripped ions (e.g. H_3^+, $^{11}B^+$) with large orbits can be used to inject across the surface layer. Once the beams pass through the surface they are assured of reaching the main volume since there are no internal magnetic fields which would hinder their penetration. No serious instabilities are expected in the main field-free volume.

There appear to be no major insurmountable engineering problems in fabricating superconducting Surmacs to be used for advance fuels with low neutron yields. The uncertainty lies more with the nuclear reaction rates and the extrapolation from low β to high β reactor regimes through several orders of magnitude.

In the high β regime the dominant physics problem is the local ballooning mode in the bridge region behind each internal rod. This instability would have been serious in the high β regime because plasma pressures deform the good curvature region. However our superconducting wall, suitably contoured, will resist changes in the surface field topology. The surface magnetic field layer is expected to stiffen up against high-β perturbations. This is the area where the main thrust of our laboratory investigation will lie.

Our laboratory has recently undertaken the examination of a number of toroidal Surmac designs and the following are being considered because of their simplicity:

1. A toroidal Surmac[4] made up of two concentric layers of rings has been found to give the best symmetry and average good field curvature on all sides. The rings are either levitate superconductors or rings with guarded supports.

2. A triple helix[5] has been built and experimental data indicate a 50% trapping efficiency of Marshall-gun-produced plasmas in the central low B region. The density profiles are uniform and the plasmas are quiet in the main plasma volume where the average magnetic field curvature is favorable.

3. Wall stabilization has enabled us to levitate three rings simultaneously in a small (4" diameter) cryogenic system. We have further improved the system by replacing the wall with passive superconducting loops, greatly simplifying the wall construction.

II. REACTOR EMBODIMENT

A Surmac reactor is visualized as an axisymmetric configuration formed by either multiple floating rings or rings with guarded supports. The cross-section of the toroidal configuration is in general non-circular and an emphasis is on modular construction. The concept has not advanced to a stage where detailed reactor designs have been considered. We present here a preliminary description of our reactor concept as in Figure 1.

1. Advanced Fuels
   A. Confinement Physics - Diffusion coefficients are assumed to be extrapolable from low β to high β regime at reactor temperatures and densities. The energy confinement time which has not yet been measured in great detail is assumed to be nearly the same as the particle confinement time.
   B. The present available data[6,7] on advanced fuels such as $p\text{-}^{11}B$ reactions is assumed to be reasonably accurate. For example, an average reaction rate $\langle\sigma v\rangle Q = 5 \times 10^{-21}$ ergs cm^3/sec is taken which requires $n\tau \gtrsim 10^{15}$.
   C. The β value at which ballooning modes become unstable in the bridge region is computed to be 20% according to MHD theoretical estimates.
2. Technology

   We have looked at various aspects of floating many superconducting rings and have satisfied ourselves in the following:
   A. Superconducting rings with internal dewars can be fabricated. Using super-insulation and heat shield a ring can be floated for 50 hours before recharging.
   B. In the UCLA laboratory multiple rings have been floated stably either by wall stabilization or by discrete feed back loops around the wall. No external sensing and feedback is necessary in this scheme, considerably simplifying the construction.

C. The field errors in large rings (≈5 M diameter) with many windings can be kept to a minimum such that diffusion loss due to field errors is smaller than convective loss.

3. Parameter Regimes

We shall consider p-^{11} reactions and D^3-He with presently available cross-sections for which some preliminary calculations have been carried out. The following set of parameters are selected as typical:

TABLE I

	$p\text{-}^{11}B$	$D\text{-}He^3$
n/cm^3	8×10^{13}	9×10^{13}
T_e KeV	150	100 KeV
T_i KeV	250	100 KeV
B (β = 1) KG	35	25 KG
B (β = .2)KG, bridge region	80	55 KG
R average meter	6	6
τ projected sec	25	25
τ required sec	10	5
Radius of ring meter	0.15	0.15
# Larmor Radii	12 for ^{11}B 40 for p	28 for He^3 50 for p } 50 cm separatrix to hoop
P FUSION (MW)	500	500
Wall Loading MW/M^2	1	1
Plasma Volume M^3	600	600
Surface area M^2	600	600

The required confinement time is determined by the following power balance:

$$\eta \left[P_{Fusion} + \frac{W_{in}}{\tau} \right] = \frac{W_{in}}{\tau}$$

where W_{in} = the amount of energy that must be injected to start the reactor,

$$= \frac{3}{2} nK(T_i + T_e) V.$$

The projected particle confinement time was based on an extrapolation of experimentally observed diffusion coefficient in a purely poloidal field

$$D = \frac{400 v_i}{\sqrt{n l}}$$

where l is the length of a field line in the $\beta = 1$ region. For a combined poloidal and toroidal field configuration in which the level of convective cells can be reduced by an order of magnitude due to better communications between field lines, the projected confinement time is correspondingly longer.

Although the supply of ^3He is limited and must be manufactured through breeding reaction, the D-^{3}He reaction is used as an example of advanced fuels using nuclei lighter than Boron whose supply is abundant. D-^{3}He could at least serve as fuels in demonstration models.

III. PRESENT STATE OF PHYSICS KNOWLEDGE RELATED TO SURMACS

High-β stability and transport scaling rates are the two most crucial areas of physics related to Surmac. There has been much theoretical and experimental work completed on the transport scaling in low β, low order multipoles but very little is known about the scaling and stability at high-β. Table I summarizes transport regimes observed in low β multipoles and lists references to articles in which the experiments are described. Each regime is represented by the appropriate diffusion coefficient scaling law for cases with poloidal field only and for both poloidal and toroidal field.

In the case of poloidal field only, classical transport is observed in the collisional regime, while convective cell-thermal or support and field error losses are observed when the plasma becomes collisionless. When toroidal field is added, Pfirsche-Schulter and neoclassical diffusion have been reported in the collisional regime, while poloidal Bohm

Table 1

TRANSPORT SCALING REGIMES OBSERVED IN TOROIDAL MULTIPOLES AND SURMACS

$B_{poloidal}$ Only

TRANSPORT REGIME	PHYSICS	RELATIVE PARAMETERS	$D_\perp(n,T,B,M)$	REF.
CLASSICAL DIFFUSION	Electron-ion collisions scatter particles across B	high n low T low B	$D_\perp \propto \dfrac{n}{T^{1/2}B^2}$	8 - 13
FIELD ERRORS AND SUPPORT LOSS	Flow along field lines which spiral out due to errors. Direct loss to supports entering confinement volume	low n high T	$D_\perp \propto \left(\dfrac{T}{M_i}\right)^{1/2}$	14 - 20
CONVECTIVE CELLS OR THERMAL FLUCTUATION DIFFUSION	Potential structures across closed poloidal field lines cause $E_T \times B_p$ convection of plasma out of confinement volume	low n high T high B	1. $\varepsilon \lesssim 1$: $D_\perp \propto \dfrac{T}{B}$ 2. $\varepsilon >> 1$: $D_\perp \propto \left(\dfrac{T}{Mn}\right)^{1/2}$	9, 21 - 30
$B_{poloidal} + B_{toroidal}$				
PFIRSCH-SCHÜLTER AND NEOCLASSICAL DIFFUSION	Enhanced classical diffusion due to larger step size (banana orbit) of trapped particles when B_T is applied	high n low T low B	$D_\perp \propto \dfrac{n}{T^{1/2}B^2}$	31
TRAPPED ELECTRON REGIME: POLOIDAL BOHM DIFFUSION	Trapped particle modes produce weak $E_{toroidal}$ turbulence. Plasma $E_T \times B_p$ drifts out of confinement volume	low n high T_e high B	$D_\perp \cong \dfrac{1}{500} \cdot \dfrac{cKT}{16eB_p}$	32, 33
TRAPPED ION REGIME	Trapped Ion modes produce turbulence	low n high T_i high B	? ? ?	33

diffusion is seen when the electrons became collisionless and the trapped electron regime is entered. Transport scaling in the trapped ion regime with collisionless ions has yet to be thoroughly investigated.

A Surmac reactor running on advanced fuel such as p-^{11}B would operate with little or no toroidal field in order to reduce synchrotron radiation losses. Thus, if supports are eliminated by levitation and field errors are made negligible by careful engineering, thermal fluctuation diffusion will determine the transport rate if present day multipole scaling is extrapolated into the reactor regime. Numerically, the diffusion coefficient is found to be[30]

$$D = \frac{400V_{Ti}}{nl}$$

where V_{T_i} is the ion thermal velocity, n is the plasma density, and l is the field line length. Preliminary reactor calculations[7] indicate that the magnitude and scaling of the coefficient may be favorable enough to break-even with p-^{11}B depending upon reaction rates which are not well known at this time. Extrapolation of this scaling from 10^9 - 10^{11} cm^{-3}, 1-10 eV plasma parameters to 10^{14}cm^{-3}, 300 KeV parameters is also uncertain. Clearly, a third generation multipole or Surmac experiment with 10^{13} - 10^{14} cm^{-3} densities and at least kilovolt temperatures is needed to gain confidence in scaling the thermal fluctuation diffusion rate into the reactor regime.

MHD stability against localized ballooning modes in the poor curvature flux segments cannot be extrapolated from present day experiments--no transport and stability studies have been made above β values of $\sim$ .01%. Theoretical estimates[34] of the β limit based on a balance of flute growth rates and the perturbation transit time between stable and unstable flux segments indicate

$$\beta \text{ crit} \quad \frac{\pi^2 aR}{L^2}$$

where a = density gradient scale length, R = field line radius of curvature in the bridge, and L = field line connection length between stable and unstable segments. For physically realizable geometries, this limit is 20-40%. However, the favorable flux segments themselves are not firmly tied down since plasma diamagnetic currents can perturb them as well. This effect can possibly lower the β limit to several percent. Perturbation of the favorable curvature ions and subsequent lowering of β_{crit} can be inhibited by placing a contoured superconducting wall behind the surface field. Changes in the vacuum field due to plasma currents are then offset by image currents induced in the wall. This scheme may also be effective in quenching localized high-β perturbations in the poor curvature bridge if these regions are also bounded by superconducting walls,

The β stability limit is a crucial physics problem. If stable high-β plasma cannot be obtained in the bridge of the surface layer, then higher magnetic fields will be required which will increase synchrotron radiation losses, ohmic losses in the hoops (for normal conductors) and increase the mechanical stress in the rings.

Two Surmac devices are currently under construction which will address these problems. One is a superconducting octopole in which four hoops will be levitated by enclosing them inside a superconducting "cavity" generated by many independent flux conserving superconducting coils. A larger device, Figure 2, consists of six normal conductors suspended vertically by guarded supports inside a normal aluminum cavity. Transport and β limit studies will be made in this device with plasma densities up to $\sim 2 \times 10^{13} cm^{-3}$ and temperature $\leq$ 200 eV. Ballooning mode stablization by superconducting walls is simulated here with normal conductor walls since the growth rate of the perturbations is on a time scale short compared to the skin time of the image wall currents.

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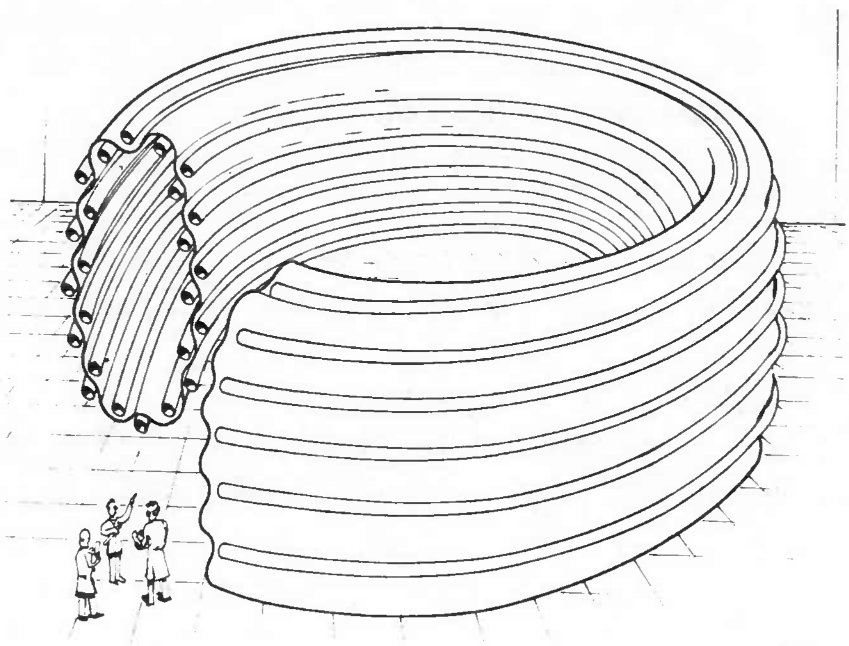


Figure 1(a)

Schematic of Surmac Reactor

TOROIDAL SURMAC

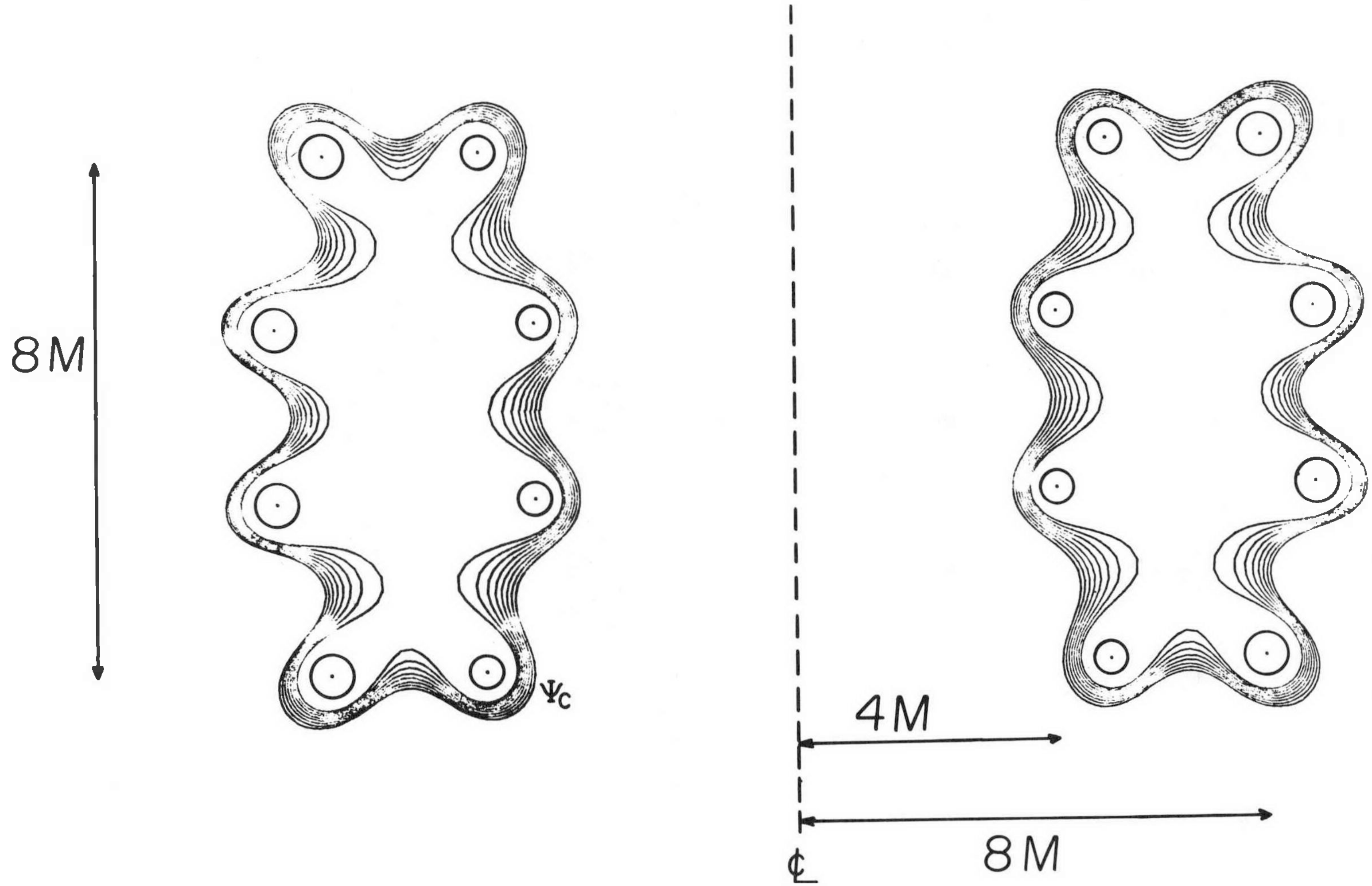


Figure 1(b)

Superconducting Toroidal Surmac configuration with noncircular cross-section. Hoop current is 10 MA and maximum field in the bridge regions is 80 KG. Inner and outer hoop minor diameters are 35 cm and 50 cm.

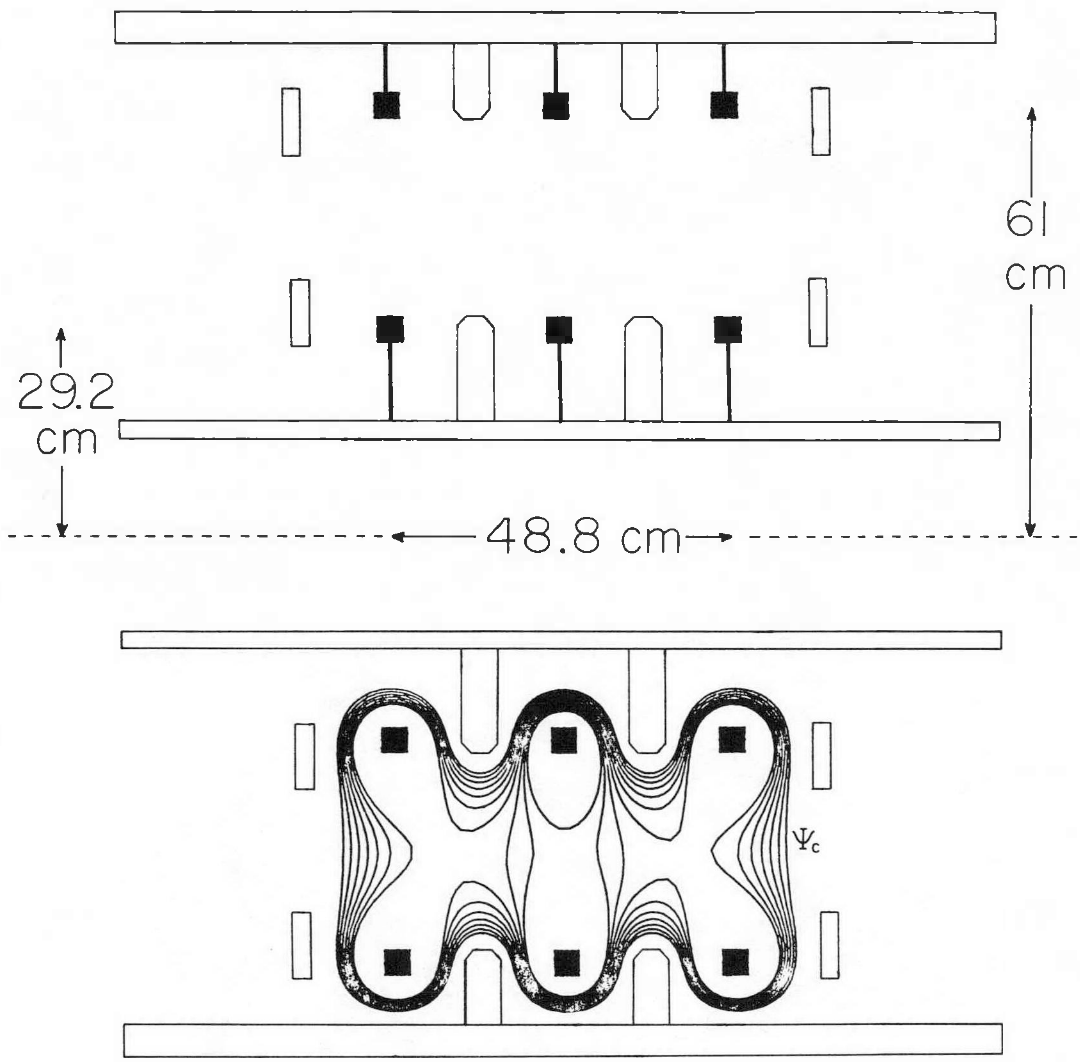


Figure 2

Dodecapole toroidal Surmac with six hoops hung vertically by a single pair of guarded supports per hoop. Return current, field shaping, and force free hoop positions are obtained with aluminum liners and rings surrounding the hoops. Hydrogen plasma of density $\geq 3 \times 10^{13} cm^{-3}$ and temperature $\leq$ 200 eV is confined by surface fields of 2 KG produced by 100 KA in the inner hoops and 50 KA in the outer hoops.

Panel Discussion: Comments on Need and Approaches to Advanced Fusion[†]

Monday, June 27, 1977
4:30 p.m. - 5:30 p.m.

Panel Members: W. Gough, EPRI (presiding)
H. Drew, Texas Utilities Service
B. Maglich, Fusion Energy Corporation
J. Rand McNally, Jr., Oak Ridge National Laboratory
T. Richards, Caterpillar Tractor Company
L. Wood, Lawrence Livermore Laboratory

A) Opening Remarks:

W. Gough: Our first introductory statement will be made by Rand McNally, who has been in the forefront of fusion research nearly from its beginning.

R. McNally: Earlier we have been told that fusion reactors the utilities want would ideally be:

1) simple,
2) cheap, and
3) easily maintained.

I would add a fourth item: steady-state.

The well-known chemical fire triangle is:

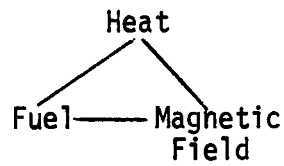
Heat
Fuel —— Oxygen

Without each of these three elements, there is no fire. Controlling one of the three elements controls the burn and you can control it so it burns steady-state. However, fusion reactors may be hard to control since the fusion fire "triangle" becomes

Heat
|
Fuel

For steady-state, if you can get to ignition you get a thermal runaway to a (possibly) stable point. The magnetic field could be thought of as the third element:

[†] Manuscript prepared by Mr. Gary Swift from the tape.



Now controlling the magnetic field allows control of the burn. So, in my personal view, I believe that the magnetic fields will be essential to the development of controlled steady-state reactors. Pellet fusion is not ruled out but I suggest that it must include a magnetic field. Perhaps pellet fusion has its usefulness in igniting a steady-state magnetically confined plasma.

At any rate, in this business we should keep an open-mind (like an open-ended mirror until we learn how to close it) to any option as we pursue the goal of controlled useful fusion power, which is still a great distance away.

H. Drew: When a fusion researcher is making cost estimates, he should do it in terms of dollars per kilowatt. The utilities should be the ones who transform this figure-of-merit into mills per kilowatt-hour because of the factors, e.g. the cost of money to the utility, plant load factors, etc., which must be considered and with which utilities are much more familiar than are scientists.

T. Richards: Caterpillar is interested in the development of small fusion plants. There is an existing world-wide market for a plant of ~1 MW. The total energy of such a plant might be put to use including waste heat. Such approaches as MIGMA or the Field-Reversed Mirror are current prospects for a 1-10 MW fusion plant. Emphases should be on small, clean,and simple. Current small power plants which use petroleum cost around $180-250/per kilowatt; 70%-80% of that cost is in oil. There is not the need in these plants for the long lifetime that utilities demand. A fusion plant could cost considerably more and be in demand due to the fact that advanced fuels should be very cheap.

B. Maglich: There is a need for a uniform criteria for judging the various devices; how far is a particular device from making energy? The P_{in}/P_{out} is the only important criterion. $n\tau$ is often cited forgetting other important parameters such as T. (Note that the $n\tau$ for D-T in a bottle at room teperature is very large, but there are not many fusion ractions. I propose $n\tau T$ as a possibility for this uniform criteria.

If tritium is avoided in the labs now, will we really want it in reactors?

All fusion schemes are currently far from economical power. This should be admitted to the public, the Congress, and the scientific community rather than saying "it's just around the corner." This would get more money for fusion research and invite more ideas.

Developing all the auxiliary technology before proving scientific feasibility is a case of getting the cart before the horse. (The Soviets have more imaginative approach and are far more exploratory).

There is a real need for private industry participation in fusion research. This country's fusion research money goes as 75% to national labs, 10% to universities, and only 15% to private industry (and some of that is for manufacturing). This does not reflect the distribution of research talent. ERDA should be forced to spend 50% of its research money in private industry as the Senate Energy Committee did in the case of Geothermal Research.

L. Wood: Today, I will play the role of the pessimists. Pellet fusion will not be easy using advanced fuels such as D-^{3}He or D-D and I suggest this applies to other approaches, too. It will be very difficult with p-^{11}B. It is one hundred to one hundred thousand times more difficult than using D-T. The qualitative advantages of burning advanced fuels have to be weighed against the formidable technological difficulties of burning advanced fuels.

Almost certainly, classified techniques will be required to get pellet fusion and advanced fuels going. Currently, classified work will probably not be released in the foreseeable future. Progress continues, but the results won't be released.

B) Open Discussion:

J. Powell: How much more than a fission reactor can D-^{3}He fusion reactor cost before it would become unattractive?

H. Drew: That is dependent on a lot of factors; but assuming fuel cost, maintenance availability, and other such factors were comparable to a fission reactor, the capital cost would also have to be comparable.

Utilities might be willing to pay some extra for environmental advantages and siting flexibility.

N. Uckan: I disagree that 50% of research funds should be given to industry or any outfit on a forced basis.

B. Maglich: Research money should be distributed as talent is distributed. There is much research talent in industry. It is interesting to note that most major inventions with the exception of television were developed by private industry and often by small firms on the order of 10-20 persons. (Some exceptions to this notably, the development of the transistor were noted from the audience.)

A. Hertzberg: The problem in industrial research is that industry

1) doesn't take risks, and
2) doesn't plan in the long-term, i.e., for more than about five years.

H. Drew: It is hard for industry to do much in fusion. In fact, industry incentive to do research is turned off by the large government program.

G. Logan: Small reactors do indeed offer advantages by limiting the investment and risks, but tend to imply power amplification rather than ignition. Loss rates will be large to get ignition. Which is more important ignition or small size?

H. Drew: The utilities want something that works and produces more electricity than it consumes.

W. Gough: Assuming that a device has good availability, low cost and small size, then, ignition is not necessary.

R. McNally: I agree except possibly in the case of D-T because 80-90% of energy must be recovered in a thermal cycle. Energy breakeven is 5 times easier than ignition. But you lose a factor of 3 in the thermal cycle so energy breakeven is 3/5 as difficult as ignition in a D-T system.

W. Gough: Point is that you can raise blanket temperature to get higher efficiencies.

D. Kerst: Why are big reactors so bad? Why are transmission costs so important? Can superconducting transmission lines be used?

H. Drew: Superconducting lines would help with the plant size problem, but could bankrupt the country's utilities. High voltage transmission is relatively cheap. Transmission costs, for example, are only about 10% of all costs. Also, large size reactors affect system flexibility and must be highly reliable. (However, fuel choice flexibility for the utilities is diminishing and adding fusion would give an additional option).

W. Gough: Small plants have shorter construction times, and may be useful in developing fusion technology faster. However, utilities would prefer small plants which cost the same, on kilowatt-hour basis, as large plants.

H. Drew: Limited quantities of large plants are acceptable as base-load reactors, but there is a limit to the number of large plants that could be accommodated on any system. That is why utilities would like to see smaller plants developed.

L. Wood: We shouldn't kid ourselves about flexibility in plant siting from advanced fuels. An important point is the public's perception of the lethalness of the plant. Does it produce radioactivity? Yes, different fuels produce different amounts, even p-^{11}B produces radioactivity. It is just a question of magnitudes which is not really going to be very helpful in the public's perception. D-T is 2 orders of magnitude below the light water reactors, and D-D and D-^{3}He are one below that. But even p-^{11}B plants can't be sited in downtown Manhattan, for example. (This got arguments from the audience about the orders of magnitude he stated, about the possibility of underground siting, about the use of low activity materials such as aluminum, about the comparative dirtiness of alternative fuels such as coal, and about the reasonableness of the public.)

Unidentified: Is the ERDA policy of developing a D-T machine first right? Or should we leap from D-T and go directly to advanced fuels?

R. Moir: I am alarmed at the view held by more and more people who perceives D-T fusion to be just another power source rather than the view that its combined virtually inexhaustible resource and environmental characteristics make it qualitatively and quantitatively more desirable than fission or fossile fueled power. Although fusion which avoids tritium and high-energy neutrons is highly desirable, its attainment may be out of reach for a long time to come due to its technical difficulties. For these reasons, I think our priority of first perfecting D-T fusion is a proper priority.

T. Richards: Personally, I don't want to see D-T program stopped (nor is it possible), but alternative concepts should be explored in parallel.

L. Wood: D-T can't be skipped because breakeven is so close in pellet fusion (probably 2 years possibly within 12 months.) The shiva facility will get within a factor of 100 of breakeven and may actually be able to achieve breakeven. A breakeven reactor (pellet or whatever) is only five years away. (The point was made from the audience that the Shiva experiment is not scalable and the needed repetition rates for the lasers haven't been shown.)

W. Gough: D-T is important because we need to obtain a burning plasma as soon as possible, but all options are important and should be explored, not just D-T Tokamak.

J. Turner: ERDA has not narrowed down to just the Tokamak, but unfortunately research money is not unlimited and the more promising approaches get more.

L. Wood: Shiva, in rebuttal to the earlier criticism, can be scaled up an order of magnitude to Shiva Nova and can ignite a commercial (but classified) pellet. Also, the needed repetition rate for glass lasers has been proven feasible. (Some doubts and murmurs were heard in the audience, but the moderator closed the discussion due to the late hour at this point.)

Panel Discussion: Energy Conversion and Applications[†]

Tuesday, June 28, 1977
1:30 p.m. - 2:30 p.m.

Panel Members: R. Scott, EPRI (presiding)
B. Eastlund, Fusion Systems
A. Hertzberg, University of Washington
R. Moir, LLL
J. Powell, BNL

A) Opening Remarks:

R. Scott: The topic for this afternoon's panel discussion will be "Energy Conversion and Applications." Each speaker will be allowed five minutes for opening remarks.

The discussion will then be opened up for comments or questions from the audience.

R. Moir: There are three important points to be considered when dealing with the energy conversion of advanced fuels:

1) Every effort should be made to achieve ignition rather than a beam driven system.
2) It is important to consider fuel cycles with a high charged-particle output and to keep energy from being converted to radiation.
3) Direct conversion should be utilized.

B. Eastlund: I would like to see some emphasis on advanced fuels from basically two points of view:

1) Does one of the advanced fuels allow us to make a significantly better power system from the standpoints of reliability and simplicity of design? Quite clearly, if we can avoid the need for tritium breeding, we can simplify the blanket and further simplify the system.
2) Do some of the unique aspects of advanced fuels lead to simpler design and improved engineering?

[†]Manuscript prepared by Mr. Rick Olson from the tape.

If so, what specific aspects? For example, advanced fuel plasmas may be a strong radiation source. Can this be exploited?

A. Hertzberg: We don't know enough right now to say that advanced fuels (as an engineering package) will replace D-T, but there are significant possibilities that cannot be ignored. We are zeroing in as far as calculations go. For example, recent changes in cross-section measurements haven't moved calculations much. But (and it's an important but), they are just calculations and not hardware. Advances in any of the following areas could significantly influence the situation:

- Improvements in the art of direct collection.
- Development of new thermal cycles. (Thermal cycles show possibility of high efficiency unique to fusion.)
- Development of fuel topping cycles.

We are not yet sure that a fuel topping cycle would be more efficient than a thermal topping cycle, or vice-versa. It would be wise to explore the possibilities.

We can't expect breakthroughs now; progress will be made by "crawl-throughs."

J. Powell: Most energy conversion concepts involving D-T systems can also be extended to advanced fuels. In considering advanced fuels, we would like to take maximum advantage of the following options:

1) Advanced fuels make it possible to design low activity blankets. Materials such as Be, Al, Mg, and ceramics can be used in such blankets to further this advantage.
2) Neutron damage can be reduced by a significant amount (0.1-0.01 of the D-T level).
3) To be realistic, we should not consider using more than one or two power conversion systems in a single power plant.

Comments on some advanced fuels:

Cat.-D: Energy is emitted in the form of 14-MeV neutrons, soft x-rays, charged particles. Direct conversion, and thermal cycles (including energy exchangers or gas turbines) are possibilities for conversion systems.

$D\text{-}He^3$: Charged particles and soft X-rays are present. Direct version and conventional thermal cycles (soft X-rays limit thermal efficiency) are possible energy conversion systems.

$p\text{-}B^{11}$: Most energy appears as hard X-rays. Energy exchangers look good as conversion systems.

B) Open Discussion:

M. Levine: In evaluating devices to use advanced fuels, one must look at some of the unique advantages of the Tormac system.

- $\beta > 5$ in current experiments, leading to low cyclotron radiation levels.
- It is possible to ignite $p\text{-}B^{11}$, but the power density would be extremely low. $D\text{-}He^3$, Cat.-D and D-T all have much better power densities. Consequently, it is difficult to believe that $p\text{-}B^{11}$ would be a useful fuel.
- There are not so many physics constraints to worry about with Tormac.

A. Hertzberg: Driven machines should not be discarded. Beam technology looks good, and the driven systems appear to be relatively simple. Ignition should not be a major criterion. Driven systems present possibilities that look as good as (or better than) ignited systems.

M. Levine: There is only a factor of three difference in $n\tau$ between an ignited systems and a driven system. If we can't achieve ignition, we are unlikely to get a driven system either.

I. Bohachevsky: In evaluating or considering an advanced fuel concept, should we not consider the public and utility acceptance of the system?

J. Powell: Agreed. For example, low activity in the blanket is very important.

R. Scott: We should approach the concept from the standpoint of off-line capability, avoiding utility electric power technology limitations (even though they are important).

A. Hertzberg: Much thought on off-line applications is important.

B. Eastlund: I wonder if utilities are in the fuel-making business.

R. Moir: Considering concepts from an off-line standpoint may be a very small advantage. However, a high fractional operating time is necessary for economics even for off-line applications.

E. Steeve: Fusion machines are extremely capital intensive. From a utility standpoint, we cannot afford to run a fusion plant as an off-line or a peaking device; it must be on-line as often as possible.

A. Hertzberg: Generally, I would agree. However, we will have to go through a learning period. Early fission reactors, for example, were usually off-line.

R. McNally: Advanced fuels in general will have a hard X-ray spectrum. Is this not true for $D\text{-}He^3$ as well as $p\text{-}B^{11}$?

F. Southworth: The wavelengths differ by about a factor of two.

J. Powell: (expanding on his earlier comments) Beryllium is more transparent to X-rays than is aluminum. Aluminum is needed for systems with higher neutron flux.

G. Logan: It is important to note that the cyclotron radiation level is important, perhaps crucial, in advanced fuel systems.

- Wall reflectivities are not known at the desired wavelengths.
- Cyclotron power estimates need to be improved.
- Theory should agree with experimental results.

R. Scott: There has been an EPRI workshop on this topic. Some calculations exist, but the results are not conclusive.

G. Miley: Discussions at the EPRI workshop on cyclotron emission (published EPRI-SR-16) emphasized that there are some important points of disagreement between experimental results and theory. Work must be done to improve the cyclotron radiation theory to the point that it agrees better with the experimental work.

B. Eastlund: Agreed. Cyclotron radiation is definitely an important consideration.

R. Scott: Obviously, this (cyclotron radiation) is an open question.

R. Roth: So far, everything has been discussed in terms of utility needs. What about other applications of advanced fuels, such as space propulstion, aircraft carriers, submarines, etc.? Utility applications are only barely competitive. Perhaps we should direct our efforts towards specialized applications that only fusion devices can handle.

H. Sahlin: I find it encouraging that people are getting together to talk about exciting new ideas such as energy conversion using direct nuclear pumped lasers. But I think there is still a missing idea that will make exotic fuels into something workable. I don't know what that idea is; I can give a couple of examples of old ideas such as muon catalyzed fusion or ionized carbon catalyzed reaction proposed by R. McNally. (See McNally's paper on this meeting.)

R. Scott: What type of climate would be good for initiating new ideas?

F. Southworth: If they expect to initiate new ideas, utilities must look further ahead than five years. Utilities must be willing to make long range investments in very exploratory concepts.

R. Scott: Utilities do have long range goals--that is one of the purposes of EPRI.

A. Hertzberg: The original function of EPRI was to do long-range thinking for the utilities. This thinking, however, is very application-minded.

B. Eastlund: This is an applied project to produce an energy producing product. The purpose of EPRI is not to do basic research, but to apply the research and create a product that will work.

R. McNally: Power is not the only viable goal. For example, a breakeven plant could be used as a fission-fuel breeder.

R. Scott: I'm afraid that we have run out of time. Thank you very much, panel.

SOME QUESTIONS FOR THE WORKSHOPS (Prepared by G. H. Miley)

Four workshops were included in this meeting:

Role of Advanced-Fuel Fusion (C. Ashworth, Chairman)

Approach to Advanced-Fuel Fusion (J. R. McNally, Jr., Chairman)

New Applications Made Possible With Advanced-Fuel Fusion (R. Tarissig, Chairman)

New Energy Conversion for Advanced-Fuel Reactors (B. Eastland/ F. Southworth, Chairman)

As a starting point, some questions for each group were distributed (Tables I-IV). Certainly these questions are not exhaustive, and in some cases they may not even be appropriate. Thus, each workshop group was told to do with their list as they felt fit.

TABLE I

I. Role for Advanced Fuel Fusion

- Can advanced-fuel reactors better meet utility requirements than D-T reactors?
- Is the breeder-satellite approach attractive to utilities?
- Is it desirable to consider leap-froging directly to advanced fuels? Or, do utilities favor a progressive approach?
- Once a D-T economy is developed, under what condition would advanced fuels be phased in?
- What information, studies, etc. do the utilities feel are necessary to evaluate advanced fuels and their role? When would they like this?
- How cruicial is it to avoid tritium breeding? Is the availability of deuterium (or boron) an important advantage compared to using lithium? Is tritium containment a serious problem?
- If you had to single out one key advantage of advanced fuels, what would it be?

TABLE II. APPROACH TO ADVANCED FUEL FUSION

- Should explicit advanced fuel research preceed D-T breakeven? The D-T demo reactor? If so, at what level?
- How near-term are D-D and D-^{3}He systems? p - ^{11}B systems?
- What fuels should be considered?
- What basic data, e.g. cross sections, are needed? What R&D should be undertaken and when?
- What confinement approaches seem best suited to advanced fuels?
- How can (or should) an experimental program be started? When?
- Is the reduced power density of advanced fuels a true disadvantage? Are there ways to overcome this problem?
- Is it possible to capitilize on the reduced neutron wall loading of D-D or D-^{3}He reactors, or is p - ^{11}B essential?

TABLE III. NEW APPLICATIONS WITH ADVANCED FUELS

- What features of advanced fuel reactors are of interest for new applications?
- Can the increased radiation be used, e.g. for chemical processing? Is cyclotron radiation preferred over bremsstrahlung?
- Can the increased charged particle energy be capitalized on by using the plasma exhaust? If so, is it necessary to go to p - ^{11}B in order to entirely eliminate tritium contamination?
- Which "new" applications are not possible with D-T reactors?
- What basic R & D should be undertaken in this area?

TABLE IV. NEW ENERGY CONVERSION

- Can the increased radiation and charged particle outputs be incorporated into new energy conversion techniques?
- Is increased efficiency important in view of the negligible cost of fusion fuels? If so, which factors (e.g. reduced waste heat, reduced recirculation, etc.) are most significant?
- Are there conversion techniques that can compete with direct collection relative to efficiency and cost?
- Can improved blanket thermal cycles be achieved without unduely increasing costs?
- Are multiple conversion processes (e.g. chemical plus electrical) desirable?
- What (if any) experiments or other studies should be undertaken?

REPORT OF WORKING GROUP I:
THE ROLE OF ADVANCED FUEL FUSION IN THE DEVELOPMENT OF FUSION

Definition of Advanced Fuels

"Advanced Fuels" are defined as anything substantially different from 50-50 D-T, such as deuterium rich D-T, all-D, D-He^3, p-B^{11}. It is generally assumed that all-D reactors will be a natural outgrowth of D-T research, although there are utility concerns about whether D-T reactors will be suited for a transition to all-D fuels. All agree that the more exotic advanced fuels such as p-B^{11} require reactor concepts which may be different. Thus, the kind of plasma physics confinements required for burning hydrogen and helium isotopes are seen as being possibly quite similar to D-T whereas the more exotic fuels are seen as requiring different confinement. Any discussion of advanced fuels should recognize this difference.

General Perspectives

Advanced fuel fusion is presently relegated to a post D-T, second generation role. Members of the working group associated with fusion research laboratories expect that the fusion program can be shifted to advanced fuels as the present course of fusion development moves along. Some members of the working group were concerned about whether such a shift can be made without making some effort now to get the program moving in that direction.

Utility members did, however, believe that advanced fuel reactors better meet utility requirements than D-T reactors.

Utility Objectives

U.S. fusion research is currently being funded at the level of a mission oriented program. However, the mission objectives of this program have not been clearly defined from the perspective of the ultimate users. The apparent goal is the attainment of controlled thermonuclear energy. The boundary conditions under which this problem should be solved have not been defined. These boundary conditions include safety, licensing, and cost criteria. It appears that some advanced fuels may have advantages over D-T. If reactors utilizing these fuels can be attained, they will probably better meet utility requirements than D-T reactors.

Fusion research should be pursued within guidelines defined by the ultimate user of the product. The eventual use of advanced fuels must be a critical part of the program goal. Consequently, advanced fuel research must be performed in parallel with D-T research. D-T experiments should be continued since they are easiest to achieve and will give us better understanding of thermonuclear grade plasmas. If energy supply concerns become severe enough, the utilities, with government approval may have to employ D-T reactors for a period of time. D-T reactors are certainly an improvement over fission reactors or no energy. Research on D-T reaction should not, however, be allowed to squeeze advanced fuels out of the fusion program.

Strong feelings were experssed on advanced fuel matters, but there was no unanimity within the working group.

Strong Pro-Advanced Fuel Views

The most pro-advanced fuel position draws a parallel with the last 15 years of history in fission breeder development. Among other things the breeder experience suggests the following:

1. Development programs do not lead to end products different from the ones being pursued. If an approach is intended to serve as a stepping stone to something different, the program must be planned that way.
2. In programs as massive as nuclear reactor development, elevating one approach to a mainline role effectively forecloses or postpones development of alternative concepts.
3. If the mainline approach bogs down enroute to commercialization it tends to drag the alternatives down with it whether they raise the same issues or not. Thus, pursuing a mainline concept which may fail to commercialize can lead to failure or a severe setback for the whole program.

Generally, the strong pro-advanced fuel view is substantially more concerned about the engineering difficulties, the time and cost associated with D-T reactor development, and the difficulty in getting public acceptance.

Thus, the most pro-advanced fuel position would like to see enough work directed specifically toward advanced fuel concepts that gaps in the plasma physics technology can be addressed, most favorable concepts selected for evaluation, and those concepts brought to a level where they can be evaluated and compared with D-T fusion concepts relative to meeting utility needs. This might be considered a parallel, rather than a follow-on, role. Advocates of this role believe the nation can afford it and should pursue it.

Not So Pro-Advanced Fuel Views

Those not so strongly advocating advanced fuel view see considerable virtue in the desirable aspects of advanced fuels but feel that the substantially less difficult plasma physics of D-T fusion and possible near time proof of principle for the mainline tokamak effort more than offset the utility preference disadvantages. The existing course of development is seen as a vehicle which may lead to advanced fuels.

Miscellaneous Working Group Responses

Experiments are needed before advanced fuels and their role can be evaluated. These should be preceeded by analytical studies. Laboratory experiments are an integral component of the evaluation program.

The symbiotic system consisting of fuel producers with separately located satellite power plants received several favorable votes in response to the question "Is the breeder-satellite approach attractive to utilities?" Time did not permit discussion of this approach in reviewing the draft of this statement. The draft position was accepted by the Working Group without comment as follows:

"Breeder-satellite--At first glance this approach appears to satisfy utility needs. The dange exists, though, that a breeder utilizing tritium and fueling advanced fuel reactors will be analogous in the public perception, to a fission reprocessing plant built to satisfy the needs of LWRs or fission breeders. Constructing a single unit which provides for all the needs of the fuel cycle will probably have licensing advantages."

The Working Group generally agreed that it would be desirable to avoid having to breed tritium in fusion reactor blankets. Several of the suggested advanced fuel cycles do not require such breeding. However, costs versus benefits have not been evaluated and the Working Group did not discuss the pros and cons of this to any extent.

The principal reason for wanting to explore advanced fuel fusion in spite of the obviously substantially more difficult plasma physics requirements is to reduce the magnitude of nuclear problems associated with radioactivity. Advanced fuel plants with reduced radioactivity should:

a. reduce risk of public exposure

b. reduce licensing difficulty

c. ease maintenance

d. reduce material cost

For the utility these advantages may remove obstacles in getting needed nuclear energy plants approved, built, paid for, and into reliable operation.

Conclusions

The Working Group feels that effort should be directed toward advanced fuels.

The eventual use of advanced fuels must be a critical part of the program goal. D-T experiments should be continued because they are easiest to achieve and will give us understanding of thermonuclear-grade plasma.

Participants

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D. Arnush
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A. Mense
R. Olson
R. Stambaugh
N. Uckan

REPORT OF WORKING GROUP II: APPROACH TO ADVANCED FUEL FUSION

This workshop included at least 20 members and addressed several questions posed by Professor Miley on the subject. The discussions were wide ranging and vigorously considered with a consensus type response being presented. Individual views were presented as to how to move toward advanced fusion fuel reactor systems but no single choice emerged from among bumpy tori, Ion-Layer (or reversed field mirrors), Tormak, multipoles, ion beam pellet fusion, etc. The following responses reflect the general views of the participants, with minor amplification by the Chairman.

1. Should explicit advanced fuel research precede D-T breakeven? The D-T demo reactor? If so, at what level?

   Response: Advanced fuel research should be in addition to D-T programs, but more directed national emphasis should be given to advanced fusion fuels and potential systems without diminishing D-T programs. It may be essential to ignite advanced fuel burns via the D-T ignition route and the ensuing thermal runaway to the high temperatures and β values necessary for establishing advanced fuel burning.

2. How near-term are D-D and D-^{3}He systems? p-^{11}B systems?

   Response: The present lack of relevant research results do not permit estimating a time scale for any advanced fuel system. Even D-T systems are two to three orders of magnitude away from an ignition condition, though closer to energy break-even. Some of the advanced fuels are expected to be competitive with D-T at $\sim$100 keV while catalyzed D-D is only a factor of ten more difficult to ignite than D-T at 10 keV.

3. What fuels should be considered?

   Response: Emphasis in the short range should be given to D-D (partially or totally catalyzed) and D-^{3}He. In the longer range (and possibly not feasible) are p-^{11}B, p-^{6}Li, and ^{3}He-^{3}He, but physics studies should definitely be continued on these more advanced fuels as well. Intermediate is D-^{6}Li.

4. What basic data, e.g., cross-sections, are needed? What R & D should be undertaken and when?

   Response: There is a need for an updated, basic compilation of σ vs E and $\langle\sigma v\rangle$ vs T for all the essential reactions, preferably undertaken by the nuclear physics community. Among those of special interest are $^{11}B(p, n)^{11}C$, $^{11}B(\alpha, p)^{14}C$, $^{11}B(\alpha, n)^{14}N$, $^6Li(^3He, p)2\alpha$ or ^8Be, $^6Li(^6Li, x)y$. In the still longer range there is a need for nuclear elastic cross-sections since these reactions lead into suprathermal or "beam-plasma" reactions. A continuing basic program should be supported on all the light element fuel reactions with emphasis on priority items in selected energy ranges (e.g., p-^{11}B above 1.5 MeV). Here, the compilation would elucidate lacunae which exist in energy ranges, uncertainty, or reaction types (including even other reactions).

5. What confinement approaches seem best suited to advanced fuels?

   Response: These cannot be selected at this time but must meet the requirements of high β (>10%), high T (>50 keV) and large $n\tau$ ($>10^{15} cm^{-3} sec$). As understanding and devices improve it may be possible to narrow the choices. Developments in one area may flow over into other areas, e.g., the pulsed MeV ion beam technology being developed for pellet fuels may serve as injectors for generation of Ion-Layers or reversed field mirrors.

6. How can (or should) an experimental program be started? When?

   Response: There should be emphasis on intermediate goals of demonstrating physics or technology principles which might then lead synergistically to possible integrated systems, e.g., heating, configurations, direct conversion, refueling, beam technologies, fast vs slow build-up, GeV approach, systems evaluation, etc.

7. Is the reduced power density of advanced fuels a true disadvantage? Are there ways to overcome this problem?

   Response: Power density is a significant disadvantage only if D-T plasmas burn at T<<100 keV; as T increases cat D-D and D-^{3}He become power density competitive at about 100 keV. In addition, even at T<100 keV an increase in $\beta^2 B^4$ (or $n^2 T^2$) for the advanced fuels can make some of them competitive in power density in the range T = 50 - 100 keV. Considerations may be quite different for advanced pellet fuels since they may permit a much smaller containment vessel (down from r $\sim$ 10 m). The environmental (tritium inventory) problem and the restriction to a lithium blanket $\sim$1 m thick mitigate against D-T systems, so some relaxation in power density demands, if real, may actually favor advanced fuels overall.

It is even possible that D-T may be too reactive a fuel at $T \sim 10 - 20$ keV. Advanced fuels permit a wider option in blankets and heat exchanger choices which may lead to cheaper, simpler, environmentally more satisfying systems.

8. Is it possible to capitalize on the reduced neutron wall loading of D-D or D-^{3}He reactors, or is p-^{11}B or another "neutron free" reactor essential?

   Response: A D-D reactor will have the same total neutron wall loading as a comparable powered D-T reactor except that less than half the neutrons are 14 MeV neutrons. Lean-D-rich-^{3}He reactors can have total neutrons < 1/20 and 14 MeV neutrons < 1/100 that of a comparable powered D-T reactor. Other "neutron free" reactor possibilities (p-^{11}B, p-^{6}Li, ^{3}He-^{3}He) should continue to be studied but it was felt that most emphasis should be given to D-D (catalyzed or non-catalyzed) and D-^{3}He advanced fuel reactors. (Little mention was made of D-^{6}Li reactors but these might be considered next of interest after D-D and D-^{3}He.)

9. What other suggestions should be considered?

   Response: A. There will exist a need for a trained group of new graduates in the field of advanced fuel fusion. Their youth, vitality and openness to new ideas as well as the new ideas which they frequently generate should be encouraged by financial support at the university graduate level of a broad based fusion training program covering the needs of and potential systems for advanced fuel fusion.

B. The synchrotron problem is especially significant at the high electron temperatures involving advanced fuels and must eventually be resolved both experimentally and theoretically. There should be a continuing in depth study regarding density, temperature and β (and B) gradient effects, wall reflectivity effects on polarization, as well as actual comparisons of advanced theories with present and next generation experimental plasmas.

Participants

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Robert Stark
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Jim Turner
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(and several others)

REPORT OF WORKING GROUP III:

NEW APPLICATION WITH ADVANCED FUELS

Several new applications for advanced fuel fusion reactors were identified by this group with the understanding that the primary objective was to discuss non-electric uses to which these reactors might be put. The applications included:

Synthetic Fuel Production
Chemical Non-Fuel Processing
Industrial Process Heat
Coherent Radiation Power Transmission

The specific features of advanced reactors which are important to these applications are the potential for reactor operation in a virtually neutron-free environment (e.g., with p-^{11}B fuel) which would avoid chemical product contamination by radioactivity, high structural integrity of reactor walls and blankets because of lowered neutron damage and the opportunity for hands-on maintenance. The high bremsstrahlung radiation and coherent synchrotron output were the two other features deemed important to these applications (e.g., p-^{11}B with electron temperatures of approximately 100 to 150 keV, and D-^{3}He with temperatures of 35 to 40 keV).

Synthetic fuel production could be accomplished by high-temperature thermal cracking of water and carbon dioxide followed by separation of species to form hydrogen and hydrocarbons such as methane, methanol, and other fuels. Ultraviolet photolysis of water could be achieved for similar ends with appropriate reactor wall windows. Also, direct plasma interaction with the chemical reactants is possible when the plasma fuels involve only non-radioactive species, for example, in the case of p-^{11}B fuel.

The possibility of obtaining collimated, coherent synchrotron radiation with appropriate plasma heating geometries was also suggested by Dr. H. Sahlin, visiting the group at its first meeting. Such radiation might be capable of transmitting power over some distance and also could be focused for more intense energy deposition. Radiation peaked at a particular wavelength was judged to be more valuable than broadband radiation such as bremsstrahlung from an equilibrium plasma, if the wavelength lent itself to resonant chemical interaction such as selective dissociation. No specific cases were discussed, but clearly there are many current examples of this idea in non-thermal fuel cracking and in isotope separation chemistry.

Only the thermal applications appear to be shared with DT reactor technology, but the relaxation of constraints placed on structural integrity by neutron damage may allow higher temperatures in an advanced-fuel reactor.

Basic R&D required in this area concerns primarily the identification and characterization of high-temperature chemistry and/or photolytic chemistry suited to the energy output conditions of advanced reactors and the development of reactor blanket concepts for carrying out chemical processing. It was noted both in our working group as well as in the talk by B. Eastlund that chemical topping cycles might be feasible where the waste heat from a high-temperature chemical process (e.g., $T \gtrsim 2000^{\circ}F$) could be used to drive a conventional steam bottoming cycle to produce electric power. The coupling of these two cycles also represents an area worthy of research and development. Specific problems in blanket design which need attention include the trade-off between first-wall transparency (e.g., to X-rays or UV photons) and material strength to withstand thermal and pressure stresses, material compatability with chemical feedstocks and products, and blanket pressures required versus the volume of blanket which can be used for processing.

A brief discussion of the climate desired for initiating these new ideas as topics for investigation considered questions such as the best way to involve utilities and their customers, in the planning and decisions needed to support advanced application research. Public acceptance was judged to be of primary importance and a route to achieving this would be through the utilities with staffs who had become well infomred on the options offered by advanced fuel reactor and the pace at which these reactors and their applications could be brought on line. Specific selling points appeared to be the fact that production of synthetic gas would be an important product for an early off-line application of an advanced fuel fusion reactor. Off-line attributes allow a "learning period" during which utilities can explore their new technology without endangering the power system reliability. Yet, at the same time a product will be created which can be used for peaking power in gas turbines or fuel cells.

Participants

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J. Fillo, Raporteur

Report of Working Group IV: New Energy Conversion

Section I: Summary

- Six energy conversion techniques were identified as applicable to advanced fuel fusion reactors:
    1. Direct Collection Topping
    2. High Temperature Gas Turbine with Organic Bottoming Cycle
    3. Thermal Enchanced Electrolysis
    4. Potassium Topping Cycle with Steam Bottoming Cycle
    5. Energy Exchanger (ALA Hertzberg)
    6. Magnetohydrodynamic Conversion

  The use of each of these and conventional conversion was considered for advanced fuel cycles and D-T. A summary of the Group's conclusions is presented in tabular form in Section III.

- Brief descriptions of each conversion scheme are presented in Section II.

- The research recommendations are
    1) Incorporate these concepts into integrated fusion reactor designs. Blanket designs, in particular, affect and are affected by the conversion system.
    2) Connect T_e for the various fuel cycles to x-ray spectra and relate to wall absorption properties.
    3) Conversion techniques numbers 2 and 4 are better understood, at present, and thus require less research.

- The overall conclusion was that while new, high efficient conversion cycles are advantageous to high Q systems, they are <u>critically important</u> for low Q advanced fuel cycles.

- Conversion efficiency alone is not the only important parameter in choosing between conversion possibilities. Comparative total system studies which take into account such factors as capital costs and operating expenses would help narrow the field. At this point, however, all the possibilities listed deserve consideration.

Section II: Relations Between Conversion Systems and Particular Fuels

Some Comments on the Table 1 (next page)

- The table is intended to answer the following questions: Assuming that there were fusion devices which could utilize the particular fuel cycle and characterized by either high or low Q (the ratio of output energy to input energy), is the particular conversion system applicable? If not, why not? If so, what efficiency might be expected and of approximately what fraction of the output energy?

- The efficiencies given are intended to be "hard" numbers, i.e. achievable rather than theoretical values (but in some cases rest on unproven technology).

- While particular fuel cycles are considered, particular machines are not. Therefore, care should be taken when interpreting the table in the context of a particular fusion device, e.g. Tokamak. This is one reason why total system studies incorporating these conversion systems are needed.

- The coupling of a particular converter to a particular machine and blanket design may not be straightforward. Again, system studies are called for.
- In the D-T case for those conversion systems which require a high temperature (>1000°C) only ~2/3 of the fusion energy (the fraction carried by neutrons) be utilized. This is because with a wall of conventional material the wall temperature would be around 500°C and only the neutrons could reach a high temperature blanket (which must also breed tritium).

Table 1

	D-T		p-^{11}B		Catalyzed-D				D-^3He		^3He-^3He (d)
					Low Q		High Q				
	Low Q	High Q	Low Q	High Q (d)	#1	#2	#1	#2	Low Q	High Q	Low Q
Direct Collection Topping Cycle	$\eta \geq 50\%$ of leaking particle power	Not Needed	?	?	$\eta \geq 50\%$ of leaking charged particles	$\eta > 50\%$ of leaking charged particles	$\eta \geq 50\%$ of leaking particles	$\eta \geq 50\%$ of leaking particles	O.K.	O.K.	Marginal
High Temp. Gas Turbine with organic Bottoming Cycle	$\eta \leq 55\%$ (a) of 2/3	$\eta \leq 55\%$ (a) of 2/3	η Too Low	$\eta \leq 55\%$ of fusion power	$\eta \leq 55\%$ of all	$\eta \leq 55\%$ of all	$\eta \leq 55\%$ of all	$\eta \leq 55\%$ of all	T_e (c) Too Low	T_e (c) Too Low	η Too Low
Therma. Enhanced Electrolysis	$\eta \approx$ (a) 65-70% of all	$\eta \approx$ (a) 65-70% of 2/3	$\eta =$ 65-70% of all	$\eta =$ 65-70% of all	T Too Low	$\eta =$ 65-70% of all	T Too Low	$\eta =$ 65-70% of all	T_e (c) Too Low	T_e (c) Too Low	$\eta =$ 65-70% of all
Potassium Topping Cycle with Steam Bottoming Cycle	$\eta \approx 50\%$ (a) of 2/3	$\eta \approx 50\%$ (a) of all	η Too Low	$\eta \approx 50\%$ of all	$\eta \sim 50\%$ of all	$\eta \approx 50\%$ of all	$\eta \approx 50\%$ of all	$\eta \approx 50\%$ of all	T_e (c) Too Low	T_e (c) Too Low	η Too Low
Energy Exchanger (ALA Hertzberg)	$\eta \leq 70\%$ (a) of 2/3	$\eta \leq 70\%$ (a) of 2/3	$\eta \leq 70\%$ (b) of all	$\eta \leq 70\%$ of all	T Too Low	$\eta \leq 70\%$ of all	T Too Low	$\eta \leq 70\%$ of all	T_e (c) Too Low	T_e (c) Too Low	$\eta \leq 70\%$ (b) of all
Magnetohydrodynamic Conversion	$\eta \leq 60\%$ (a) of 2/3	$\eta \leq 60\%$ (a) of 2/3	Marginal	$\eta \leq 60\%$ of all	T Too Low	$\eta \leq 60\%$ of all	T Too Low	$\eta \leq 60\%$ of all	T_e (c) Too Low	T_e (c) Too Low	Marginal
Conventional Steam Cycle	η Too Low	O.K.	η Too Low	O.K.	O.K.	O.K.	O.K.	O.K.	O.K.	O.K.	η Too Low

η = Conversion Efficiency
T_e = Electron Temperature
#1 - Using a D-T-like blanket
#2 - Using a High-temperature wall/blanket

(a) Only about 2/3 of the energy can be obtained at high temperature since wall must be relatively cool. The other 1/3 can be converted using a conventional thermal cycle.
(b) Assumes a Beryllium (or similar) first wall with no energy deposition there.
(c) Assumes X-rays don't reach blanket.
(d) Some vocal few at the conference contended ignited p-^{11}B possible, but no one aruged for ignited ^3He-^3He.

- In the Catalyzed D case, two general blanket designs were considered:
    1) A D-T-like blanket with a cool (~500°C) wall and hot (>1000°C) interior blanket
    2) A hot wall/blanket similar to that in "Blanket and Magnets for Catalyzed D and D-^{3}He Reactors," J. Powell, J. Fillo, and J. Usher, reported in these Proceedings.

  In the first case, less than 1/3 of the fusion energy, i.e. the neutron energy, can reach the high temperature blanket. Thus, those conversion processes requiring high temperatures were deemed not usable since most of the energy appears at too low a temperature.

- Another way that a high temperature blanket can be achieved with a cool wall is that the x-ray radiation can carry its energy to the blanket A Low-Z wall such as beryllium is required. Also, required is a hard x-ray spectrum implying a high T_e.

Section III: Brief Descriptions of the Conversion Techniques

1. Direct Collection Topping (Direct Energy Conversion)*

The energy carried out of the reactor in the form of charged particles (fuel ions, fusion products, electrons) is guided to a set of electrodes, where deceleration in an electrical field converts this kinetic energy into electrical energy. A diverter connects the plasma container to the direct converter. Practical efficiencies are approximately 50% or somewhat higher under special circumstances where the particle energy distributions are nonthermal. Great care should be exercised in the design to minimize radiation and use fuel cycles which have a high fraction of charged particle output.

Direct energy conversion for neutral beams will be important in beam driven reactors. **The divertor is an important systems** consideration for a direct collection.

2. High Temperature Gas Turbine/Organic Bottoming Cycles

High temperature gas turbine closed cycles use an inert gas working fluid in a Brayton power cycle, to produce electric power from a high temperature thermal input. Typically, the high pressure working fluid (e.g. He at ~ 50 atm and a 1000°C)temperature expands through a turbine to lower pressure (e.g., 20 atm). The sensible heat of the expanded gas is then regeneratively exchanged to preheat the increasing high pressure gas before it passes through the high temperature heat source (e.g. the fusion blanket). After leaving the regenerative heat exchanger at relatively low temperature (i.e., ~ 200°C) the reject heat from the cycle is used as a heat inject to a low temperature Rankine power cycle using an organic working fluid (e.g., one of the freons). The cool He gas working fluid is then compressed back to its high pressure level by a compressor that is mechanically coupled to the turbine. For a heat source temperature of ~ 1000°C, the combined cycle efficiency (heat input to electrical output) will be on the order of 55% (45% for the gas turbine cycle and 10% for the organic bottoming cycle).

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*See A. S. Blum and R. W. Moir, "Direct Energy Conversion and Neutral Beam Injection for Catalyzed D and D-^{3}He Tokamak Reactors," these proceedings.

3. Thermal-Enhanced Electrolysis (Thermoelectro Chemistry)

Thermal-enhanced electrolysis is a means of utilizing the high temperatures potentially available in fusion reactors to produce hydrogen. The hydrogen can then be used as a fuel in a variety of ways, e.g. for electrical peaking units, in cars and other transportation systems, or for home heating. In this process, temperatures of 1000°C would be used to provide a significant fraction of the energy required for hydrogen production. This type of operation is actually a high temperature fuel cell, with solid-electrolyte membranes, working backwards. Steam is injected on one of the electrodes, the cathode, the water dissociates, and leaves H_2 on the side where the contact first occurs, $+O^{--}$. The O^{--} diffuses through the electrode, made of a conducting oxide (ZrO_2 for example). On the anode side, the O^{--} becomes O_2. The entry gas is steam mixed with hydrogen and the exit gas is mostly hydrogen mixed with steam.

4. Potassium Topping Cycle with Steam Bottoming Cycle.

In this binary cycle, the high temperature heat obtainable from a fusion reactor is used to boil potassium at ~830°C (2 atm). The potassium is expanded through a turbine and condensed (at say ~600°C and ~.003 atm) generating steam (~565°C and 272 atm (400 psi)) which is then expanded through a conventional steam turbine. Clearly, the efficiency of this binary cycle is Carnot limited, but it can achieve higher efficiency than a conventional steam cycle alone because it more fully utilizes the high temperature potentially available from fusion reactors. Because of work in the nuclear-electrical space-power program, the basic techniques and material problems are well understood.

5. The Energy Exchanger (ala Hertzberg)

The energy exchanger is a device which permits the extraction of expansion work from a very high temperature gas (2000-3500°K) of relatively high molecular weight into compression work of a low molecular weight gas which can then be conveniently re-expanded in a conventional turbine. The essence of the device is its ability to withstand the temperature since it is alternately exposed to the very high temperature gas followed by the low temperature gas such that the average wall temperature can be maintained at a level reasonable for conventional materials and structural limitations. Thus, it makes it possible to take full advantage of the capability of radiation output as a volume heater to heat a gas higher than the temperature of the wall container so that the full benefits of high thermal efficiency cycles and their synergistic effect on reducing the circulating power fraction and in turn the required Lawson criteria can be obtained.

6. Magnetohydrodynamic Conversion (MHD)

The MHD cycle is one form of the general class of closed Brayton cycles. Unlike a gas turbine cycle, where a high temperature mechanical turbine is used. The MHD cycle uses a ionized gas moving at high velocity through a transverse magnetic field to generate electricity. The other components of the cycle (i.e. regenerative heat exchanger, compressor, and bottoming cycle, if used) are similar to those for the gas turbine cycle (see description of the gas turbine cycle). Very high heat source temperatures are required on the order of 2300°C, to achieve adequate equilibrium ionization in the MHD generator. The gas working fluid (typically He or A at a few atm) is seeded with ~1% of an easily ionizable alkali metal (e.g. potassium or cesium) to achieve the needed ionization levels. The seed will largely condense at the low temperature point in the cycle. It must be collected and reinjected into the hot working fluid leaving the blanket. Unlike the gas turbine cycle, metal heat exchangers are not practical for regenerative heat exchange, and pebble bed, intermittent flow recuperators, similar to those developed for open cycle MHD, must be used. The cycle efficiency for MHD should be ~60%.

One reference has been identified which could serve as a starting point for one seeking more information about a particular conversion process:

1. "Direct Collection Topping," George Miley, Direct Energy Conversion, ANS, 1976, Ch. 3.
2. "High Temperature Gas Turbine," Miley, op. cit., p. 282-7
3. "Thermal Enhanced Electrolysis," B. M. Abraham and F. Schreiner, Indus. and Eng. Chem. Fundamentals, 13(4):305.
4. "Potassium Topping Cycle with Steam Bottoming Cycle," Miley, op. cit., p. 287-90
5. "Energy Exchanger," A. Hertzberg and R. Taussig, "New Energy Conversion Concepts," these Proceedings
6. "Magnetohydrodynamic Conversion," Miley, op. cit., p. 240-294

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The following persons at one time or another participated in the discussion of Working Group IV:

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GENERAL REMARKS CONCERNING JOINT EPRI/UNIVERSITY OF ILLINOIS CONFERENCE ON ADVANCED FUSION

June 27-28, 1977

E. J. STEEVE

Commonwealth Edison Company

As an observer from the electric utility industry, I was impressed by the active full two day program, both the formal presentations and the lively, interesting, and meaningful discussions.

The meeting erased the doubt I had initially about the value of discussing the subject of advanced fusion techniques even before the basic concept had yet to be demonstrated as practical. It is imperative that all the possibilities for harnessing this virtually unlimited energy resource be fully investigated, discussed, and resolved before committing huge sums of capital to a final commercial demonstration.

Repeating the agenda was excellent, but if I had to suggest an improvement for a future session, I believe a prepared discussion on "Electric utility operating economics and how it applies to fusion reactor design" would be of great benefit to the participants. The subject should cover the basic fundamentals of utility financing and also the planning principles for generating capacity additions. This suggestion was prompted by questions and remarks made by the participants during the conference which indicated a need for this type information.

OBSERVATIONS ON THE ADVANCED FUSION FUEL E.P.R.I. CONFERENCE

June 27-28, 1977

D. W. KERST
PROFESSOR OF PHYSICS
UNIVERSITY OF WISCONSIN

The possibility that a 50-50 D-T fusion reactor would not be a practical choice among other non-fusion energy sources was emphasized by C. P. Ashworth of Pacific Gas and Electric Company. His paper sharpened the motivation for the discussions of "advanced fuels" which he indicated may offer the only worthwhile energy sources which fusion can contribute.

Perhaps the clearest example of a fusion system with minimum neutron effects and direct conversion possibilities closest to practicality for industry would be the deuterium rich D-T or the D-D producer reactor with satellite D-^{3}He reactors running off of the ^{3}He from the remote D-D reactor. But the cross sections for these reactions force us to large systems because of lower power density and higher magnetic fields due to higher temperatures. But the large size for D-D producers may be tolerable since its location need not be near much of the load.

To a physicist, the difficulty of making and containing a high density and such a high temperature pure plasma is so great that there is hesitation in looking beyond to what was thought to be the next generation (advanced fuels) after D-T. When more experience with the "advanced" schemes is gained, it should be easier to tolerate some of the ideas associated with them. For example: the cavalier extraction of plasma for direct conversion by flux bundle removal is alarming when in the past carefully trimmed toroidal containment fields were made to achieve circular symmetry. But we may learn how to do this and in the process not only to "direct convert" but also to purge the ashes in a steady state continuous reactor. Direct conversion techniques need more widespread attention. Or, as another example: the idea of internal ring structures to confine with low internal field, and thus with minimal synchrotron radiation, must be contemplated for quite a while before the serious possibilities become evident.

The suggestions made for raising the temperatures used in heat engines will allow us to save more energy even without direct conversion. Some desirable fuels with minimal neutron production could then become more practical. Such improvement in at engines seems very important to develop for general reasons and because advanced fuels run at high temperatures emitting much of their plasma energy in X-rays, high temperature X-ray heated walls and an associated high efficiency heat engine cycle become important in particular.

High technology possibilities must be shown sufficiently manageable for the reliability needed by the industry. The electrical utility industry can make authoritative contributions if E.P.R.I. can objectively and effectively stimulate the pursuit of these problems, which appear quite important at the present stage of fusion development.

CHIEF IMPRESSIONS FROM THE EPRI MEETING ON ADVANCED-FUEL FUSION

June 27-28, 1977

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Neutron-Free Reactions. It appears the p-^{11}B may fade in favor of catalyzed p-^{6}Li as the "ideal" fusion fuel. When the ^{3}He reaction product is completely burned, then p-^{6}Li has approximately the same reactivity as p-^{11}B, but has the advantage of reduced bremsstrahlung loss. Another advantage is the contribution to fusion reactivity of ^{6}Li-^{6}Li reactions (although some of these produce low-energy neutrons). On the other hand, ^{6}Li cannot be regarded as an abundant fuel.

Cross Sections. More accurate cross sections are needed at energies above 100 keV, but it cannot be claimed that such information is urgently required.

Deuterium Fuel. Perhaps the greatest environmental advantage of D-D and D-^{3}He relative to D-T is that the elimination of the tritium-breeding requirement results in an enormous flexibility in blanket construction for reducing long term activation. In particular, the recent Brookhaven designs have incredibly low activation levels. The second great advantage is the large reduction in tritium inventory, since the blanket can be tritium-free. If D-D and/or D-^{3}He reactors with minimum-activity blankets can be implemented, the motivation for pursuing the neutron-free reactions may not be overwhelming.

Energy Conversion. The large increment in capital cost associated with direct energy conversion may hardly be worthwhile, inasmuch as 50 to 90% of the energy from ignited advanced-fuel plasmas emerges in the form of radiation or neutrons. It would seem that the development of high-efficiency thermal cycles is a more profitable investment.

Tokamaks. Low-beta tokamak plasmas with specially tailored temperature and density profiles can reach ignition in catalyzed-D or D-^{3}He, with acceptable fusion power densities and reasonable reactor size. I would claim that the tokamak is closer than any other magnetic confinement device to reaching ignition in catalyzed-D, just as it is closest with regard to D-T. A catalyzed-D ignition test reactor might be an appropriate next step after a D-T EPR.

OBSERVATIONS ON THE ADVANCED FUSION FUEL E.P.R.I. CONFERENCE

June 27-28, 1977

A. T. MENSE, N. A. UCKAN AND J. R. McNALLY, JR.
Oak Ridge National Laboratory
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The meeting was certainly worthwhile from the point of view of providing a wider audience some information on the aspects of fusion power from "Advanced Fuels." It provided a forum for a healthy exchange of views, and meetings on this topical area are called for on at least some periodic basis.

The only real criticism was that more experimental people should attend the meetings in order to clearly enunciate how difficult it really is to attain these "high quality" plasma environments needed for the practical implementation of advanced fuel cycles. Proof of principle is one thing--proof of economic feasibility quite another.

It is our belief that much more interchange should take place between those interested in advanced fuels (utilities primarily) and those laboratories (university, government, or industrial) where something can be done about it.

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May 24, 1977

To: Tentative Participants, EPRI Review Meeting on Advanced Fuel Fusion, June 27-28, 1977, Commonwealth Edison Building, Chicago, Illinois.

From: G. H. Miley, Chairman
Nuclear Engineering Program

Subject: Background material - 1973 EEI Report on Development of Advanced Fuel Fusion

To provide some "food for thought" prior to the meeting, I am enclosing a copy of an old report to EEI on Advanced Fuel Fusion.* I presented this to several review groups at that time and, as could be expected, the whole subject was very controversial. Some exploratory studies were subsequently launched by EPRI, but a program of the type envisioned in the EEI report has not been possible due to (1) doubts in the fusion community about whether or not this is a proper direction, and (2) lack of funds for studies not related to early achievement of a demonstration of D-T energy breakeven. Consequently, I feel that this report still provides some important background for our June meeting. (If you cannot attend the meeting yourself, please see that this material is passed on to the proper person.)

In closing, I would observe that the attitude towards advanced fuels has changed remarkably in the past year or so. Advanced fuels have been discussed in several papers at the last three APS meetings I have attended. The ERDA program plan (ERDA-76/110/1) for fusion development, issued last year, devotes a page to subject (attached for your information). None of this happened earlier. The reason, I believe, is simple. <u>If fusion is to be viewed as a competitive, inexhaustible energy source, the ultimate goal must be advanced fuel fusion. The question is, how do we plan to achieve this goal</u>. In view of the importance of this question, the EPRI

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*Due to a lack of space, this report is not included in the EPRI Meeting Proceedings, however, an outline is contained in the paper "Comments on Advanced-Fuels and Workshop Objectives," by G. Miley that appears here.

review meeting is, in my (prejudiced) personal opinion, timely and important. This will be the first comprehensive meeting on the subject since the one organized in 1973 to provide background for the EEI report. Hope to see you at Chicago!

GHM:cs
Enclosures

Fusion Power by Magnetic Confinement--Program Plan *

July 1976

C. Fuel Cycles

First generation fusion reactors are expected to use deuterium and tritium as fuel. Several environmental drawbacks are, however, commonly attributed to DT fusion power. First, it produces substantial amounts of neutrons that result in induced radioactivity within the reactor structure, and it requires the handling of the radioisotope tritium. Second, only about 20% of the fusion energy yield appears in the form of charged particles, which limits the extent to which direct energy conversion techniques might be applied. Finally, the use of DT fusion power depends on lithium resources, which are less abundant than deuterium resources.

These drawbacks of DT fusion power have led to the proposal of alternatives for longer term application--for example, fusion power reactors based only on deuterium. Such systems are expected to (1) reduce the production of high energy neutrons and also the need to handle tritium; (2) produce more fusion power in the form of charged particles; and (3) be independent of lithium resources for tritium breeding.

It has also been suggested that materials with slightly higher atomic numbers (like lithium, beryllium, and boron) be used as fusion fuels to provide power that is essentially free of neutrons and tritium and that release all of their energy in the form of charged particles.

Although such alternatives to DT fusion power are attractive, there is an important scientific caveat. To derive useful amounts of power from nuclear fusion, it will be necessary to confine a suitably dense plasma at fusion temperatures (10^8 °K) for a specific length of time. This fundamental aspect of fusion power is expressible in terms of the product of the plasma density, n, and the energy confinement time, τ, required for fusion power breakeven (i.e., the condition for which the fusion power

*Excerpted from ERDA - 76/110/1.

release equals the power input necessary to heat and confine the plasma). The required product, $n\tau$, depends on the fusion fuel and is primarily a function of the plasma temperature. Of all the fusion fuels under current consideration, the deuterium-tritium fuel mixture requires the lowest value of $n\tau$ by at least an order of magnitude and the lowerst fusion temperatures by at least a factor of 5. When the plasma requirements for significant power generation are compared with the anticipated plasma performance of current approaches to fusion power, it is apparent that fusion power must initially be based on a deuterium-tritium fuel economy. However, the eventual use of alternate fuel cycles remains an important ultimate goal and consequently attention will be given to identifying concepts which may permit their ultimate use.

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on

Advanced-Fuel Fusion

June 27-28, 1977

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